JRC Scientific and Technical Reports

2nd Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop

WWER – WS2

Summary

EUR 23719 EN -

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JRC 50028

EUR 23719 EN
ISSN 1018-5593

Luxembourg: Office for Official Publications of the European Communities

Printed in the Netherlands

EUROPEAN COMMISSION
DIRECTORATE-GENERAL JRC
JOINT RESEARCH CENTRE
Institute for Energy

2nd Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop
WWER – WS2
Amsterdam, 26-28 November 2008

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. 1.1).

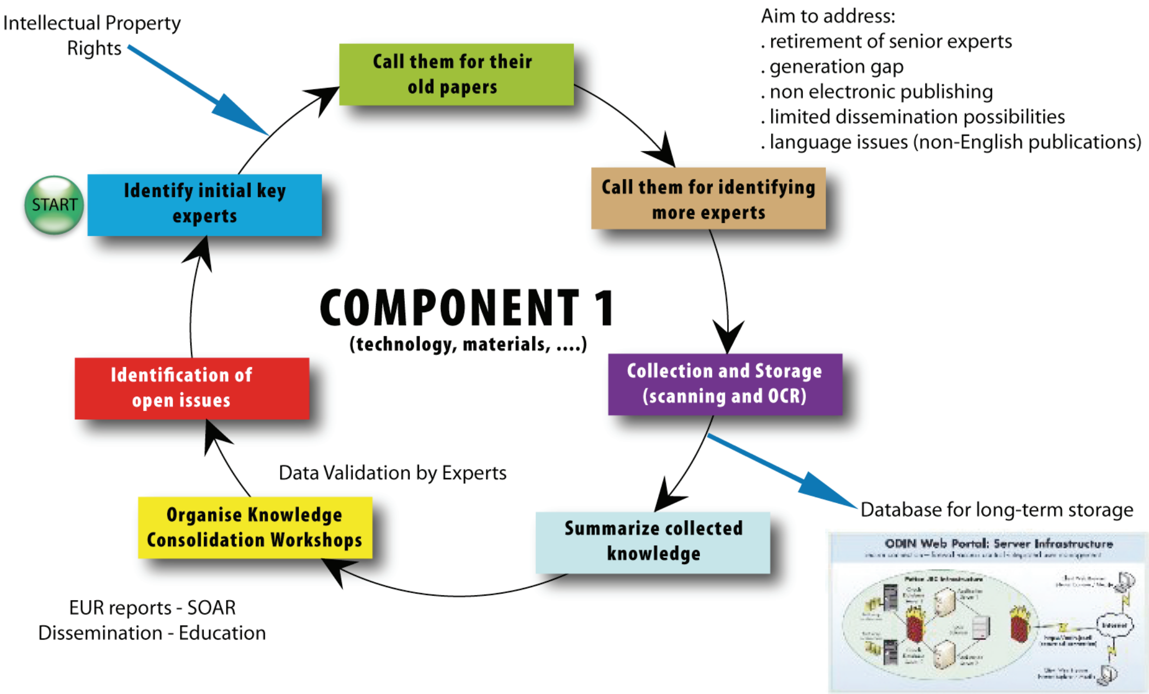


Figure 1.1: Nuclear Knowledge Consolidation Method

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 second time for the present pilot study for consolidating and preserving WWER RPV safety related literature, which is part of a wider Nuclear Knowledge Preservation and Consolidation activity in the Nuclear Design Safety unit of the Institute for Energy. The WWER type reactors were widely built and used, mainly throughout Russia and the Eastern European countries (annex3). The WWER RPV was chosen due to the urgency regarding loss of knowledge and due to the proactive attitude of the involved experts and scientists.

2 Scope

Several reviewers received between 10-100 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 first workshop the predefined fields of expertise in WWER RPV Embrittlement were discussed and redefined as described in figure 2.1.

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

Figure 2.1: Subdivision of WWER RPV Embrittlement Expert Fields

In the below chapters the reviews per expert field are summarized.

3 Start-of-Life Toughness

It is well know that neutron irradiation affects the properties of reactor pressure vessel material. The methodologies to assess the embrittlement for LWR (PWR, BWR and WWER types) are all based on the shift induced by irradiation of a reference transition temperature. This shift is obviously defined as the transition temperature (RT_{NDT} or Tk) for the irradiated material minus the transition temperature for the un-irradiated material plus a margin therm. This shift is measured by charpy impact tests and applied to the fracture toughness reference curves.

The main objective of this paper is to review the available literature on fracture toughness data of non-irradiated material, which is start-of-life fracture toughness for WWER reactor pressure vessels.

Nowadays, there are in operation [IAEA 2008] 6 WWER 440/230 units, 16 WWER 440/213 units and 20 WWER 1000 units. For WWER-440 RPV the steel selected was a Cr-Mo-V steel while for the WWER-1000 a Ni-Cr-Mo-V was used. The designation for the steel type used in the early systems (WWER-440) is 15Kh2MFA. For systems of the same size but built after 1980, the specification 15Kh2MFAA is applicable. This specification places special restrictions on phosphorus ands copper content. A nickel modification of the 15Kn2MFAA composition, designated 15Kh2NMFAA is used for the higher power systems (WWER-1000).

The Russian approach applies material specific fracture toughness reference curves based upon an “eye ball” lower bound of the KIC tests on large test specimens of the non-irradiated materials. Each material has different curves for 1) normal operation, 2) operational occurrences and hydraulic tests and 3) emergency operations. The fracture toughness reference curves, in terms of an effective temperature (T-Tk) where Tk is the critical brittleness temperature, for emergency situations are, see Figure 3.1:

WWER 440 Base material (15Kh2MFA and 15Kh2MFAA)

$$[K_I]_3=35+45\ \exp(0.02(T-Tk))$$

WWER 1000 Base material (15Kh2NMFA and 15Kh2NMFAA)

$$[K_I]_3=74+11\ \exp(0.0385(T-Tk))$$

WWER 440 and WWER 1000 welds

$$[K_I]_3=35+53\ \exp(0.0217(T-Tk))$$

The following temperature generalised dependence of $[K_{IC}]$ can be used for RPV PTS analysis:

$$[K_{IC}] = 26+36\ \exp\ [0.02\ (T-Tk)];\ [K_{IC}] \leq 200\ \text{MPa m}^{1/2}$$

This dependence corresponds to specimen thickness of 150mm and Pf = 0.05.

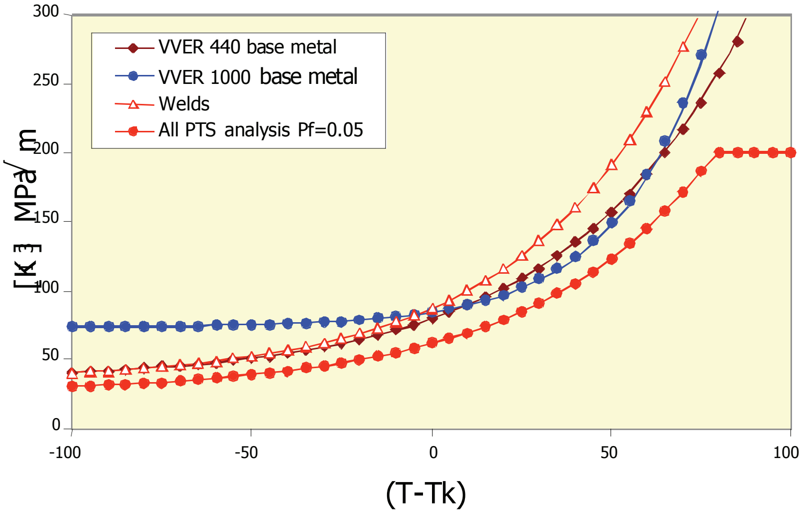


Figure 3.1 Russian approach reference fracture toughness curves for emergency situations

These curves are included on the Code PNAE G-7-002-86. Recently it has been proposed new reference KI-curves for PNAE G-7-002 Code. They are all more conservative than the older ones, except for WWER-440 base metal.

The following discussion tries to evaluate the goodness of these curves by experimental data

3.1 Consolidated Conclusions

Summarising the available references, it can be said that for WWER reactor pressure vessel steel a consistent fracture toughness data base in the start of life conditions exists (Prometey database). Different ways of analysing this data base can divided in two tendencies. One is followed by Russian researches and is based on the Tk reference transition temperature and new references fracture toughness curves are suggested to replace the curves of PNAE. The other way is included on the VERLIFE procedure and it is based on the Master Curve concept, using the RTT0 reference transition temperature.

In any case, if the approach of initial condition + shift due to irradiation + margin is applied, it is evident that the SOL toughness is needed, but if the approach is based on irradiated results + fluence projections, the initial state is not needed and the uncertainties are reduced. So if irradiated fracture toughness specimens exist, with the application of the VERLIFE lower bound, the initial state of the RPV is not needed.

3.2 Open Issues

Regarding crack arrest data, more experimental data is needed in order to establish a crack arrest fracture toughness reference curve for WWER RPVs.

For non-homogeneous material the standardized "Master Curve" does not describe the fracture toughness. The use of the SINTAP procedure, bimodal "master Curve" or maximum likelihood (MML) estimation of T_0 should be clarified and in some way standardized.

3.3 Reviewed papers and summaries

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

Brumovsky M.,"WWER Reactor pressure vessel integrity assessment comparison and experimental verification"- IAEA Specialist's Meeting on The Integrity of Pressure Components of Reactor Systems, Paks, Hungary, 215-29 May 1992.

Brumovsky M., "Comparison of PWR and WWER codes for reactor pressure vessel life assessment"- Workshop on Internacional Practices on Reactor Pressure Vessel Integrity Assessment", 26-30 August 1996, Rez, Czech Republic.

Brumovsky M., "Unified Procedure for Lifetime Assessment of Components and Piping in WWER Nuclear Power Plants (VERLIFE)"- Final report – EU Contract No: FIKS-CT-2001-20198

Brumovsky M., "VERLIFE Unified"- Procedure, ATHENA Workshop, AMES Thematic Network EURATOM FP5, October 25-27, 2004, Rome, Italy

English C., Jendrich U., Karpunin N., Lomaki S., Prosvirin A., Schulz H., Wallin H., "Licensing related assessment of RPV embrittlement of WWER 440/230 Units"- IAEA Specialist Meeting on Irradiation Embrittlement and Mitigation, 26-29 April 1999

Guinovart J., Langer R., Lidbury D., Wardle G., Wallin K.., Houssin B., "Unified reference fracture toughness design curves for RPV steels" – EC DG XI Contract B7-5002/97/000809/MAR/C2, 2000

Joly P, Houssin B., Gauthier J.P., Pelli R.., Wardle G., Langer R., Kastner B., "Compendium of pressure vessel steels and weldments properties and data required for structural integrity analysis" - EC DG XI Contract B7-6340/95/000877/MARC/C2, 1998

Timofeev B., Karzov G., Blumin A., Anikovsky V.,"Fracture toughness of 15X2MFA steel and its weldments", International Journal of Pressure Vessels and Piping 77 (2000) pp. 41-52.

Timofeev B., Karzov G., Blumin A., Smirnov V., "Determination of crack arrest toughness for Russian light water reactor pressure vessel materials"- International Journal of Pressure Vessels and Piping 77 (2000) pp. 519-529

Wallin K., Törrönen K., Ahlstrand R., Timofeev B., Rybin V., Nikolaev V., Morozov A., " Theory Based Statistical Interpretation of Fracture Toughness of Reactor Pressure Vessel Steel 15x2MfA and its Welds" – 10th International Conference on Structural Mechanics in Reactor Technology (SMIRT10) Anaheim, CA, USA August 14-18, 1989

Karzov G., Timofeev B., Chernaenko T. " Database of RPV materials for brittle fracture assessment of NPP equipment"- 3º International Conference on Strength, Durability and Stability of Materials and Structures. SDSMS'03, Klaipeda (Lithuania), 17-19 Sep 2003

Planman T.,Wallin R., Rintamaa R., "Evaluating crack arrest fracture toughness from Charpy impact testing"- 14th International Conference on Structural Mechanics in T. Planman, K. Reactor Technology (SMIRT14) Lyon, France, August 17-22, 1997

A large number of fracture toughness results are presented by Finish and Russian researchers in SMIRT 10 conference [Wallin 1989]. The materials used consisted of five different heats of the 15Kh2MFA base material and two different welds, one submerged arc weld and one electroslag weld. All these data are statistical treated, size normalized, and the Master Curve is derived, see figure 3.1. As can be seen in this figure, the fracture toughness of the 15Kh2MFA and welds can be described by the Master Curve.

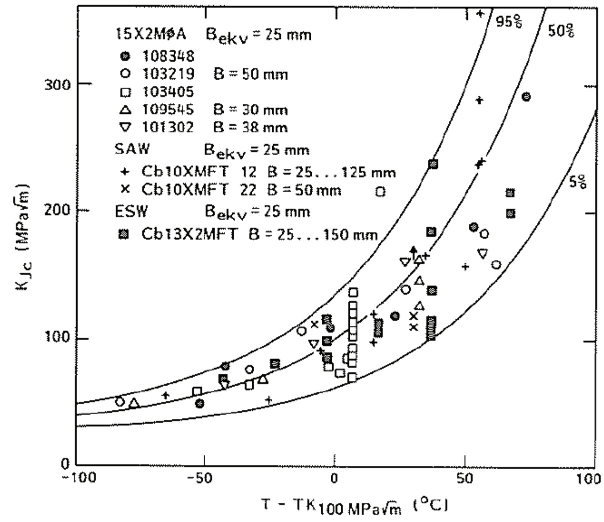


Figure 3.1: Normalized (25 mm) temperature dependence and scatter for WWER-440 materials [Wallin 1989].

This database was later checked and updated by Siemens and VTT assisted by Russian researchers in the frame of different EC-DG XI contracts [Joly 1998 and Guinovart 2000]. A totally of 803 fracture toughness test results were collected of WWER 440 and WWER 1000 RPV materials. The results show that all the fracture toughness reference curves from the PNAE G-7-002-86 should be revised. Some of these results were published by English for WWER 440 Weld metal in a summary of the project TACIS/RF/TS/01-C "Licensing related assessment of reactor vessel embrittlement", see figure 3.2 [English 1999].

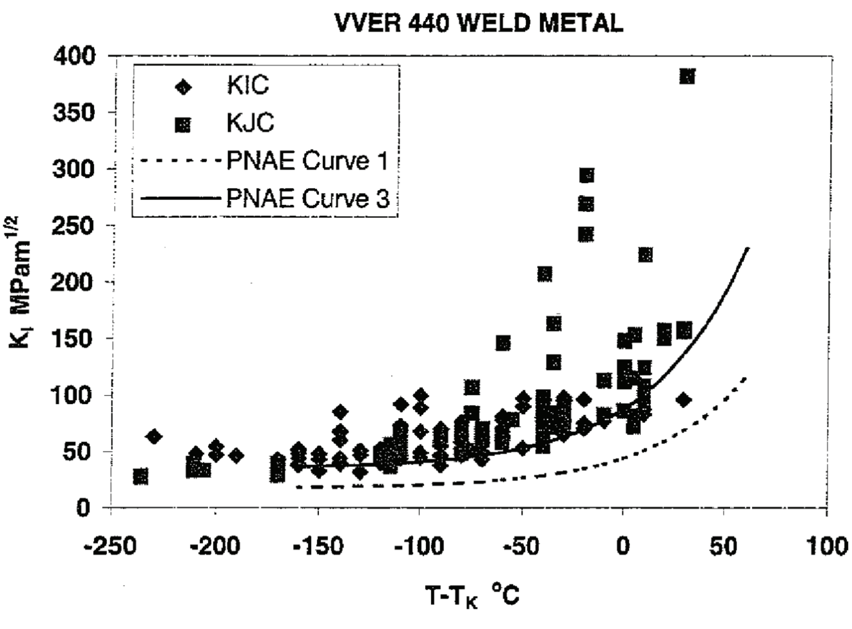


Figure 3.2 Fracture toughness data and corresponding reference curves from the Russian norms [English 1999]

Regarding WWER-440 RPV materials, the extended database includes fracture toughness test results of 15X2MFA steel (10 heats), 15X2MFAA steel (six heats) as well as their welds, produced by submerged arc welding with Cb-10XMFT wire under AH-42 flux (seven welded test samples) and with Cb-10XMFTY wire under AH-42M flux (five welded test samples), with electroslag wire of the type Cb-13X2MFT by using OF-6 flux (three welded test samples) and by manual arc welding with H-6 and H-3 electrodes (one welded test sample for each electrode type) [Timofeev 2000 and Karzov 2003]. This database also included the heat used for the round robin experiment performed by 8 laboratories from Russia and Ukraine.

The comparison of the experimental data for the united database, containing K_{IC} (seven heats) and K_{JC} (three heats) test results with the reference curve three, showed that some data are located below this curve. The authors derived the envelope curve of the base material results (231 tests of 10 heats of 15X2MFA), see Figure 3.3. This envelope is expressed as:

K_{IC}=27+38 exp(0.02(T-Tk))

The authors suggest this dependence to be used for the comparison with other database in particular for pure 15X2MFAA steels and general database for welds.

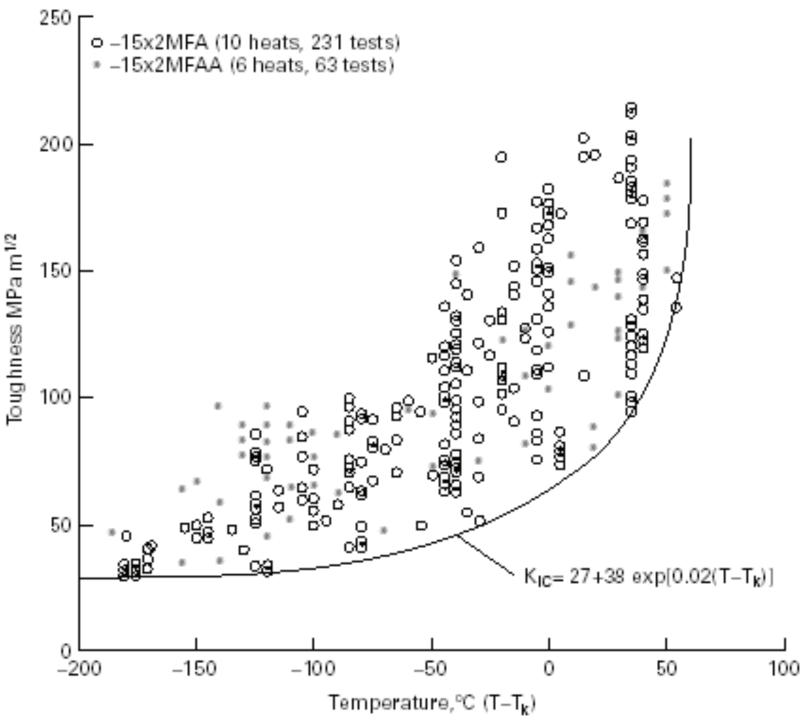


Figure 3.3 Comparison of experimental results for 15Kh2MFA nd 15Kh2MFAA steels [Timofeev 2000a]

In summary, this big database was used mainly to demonstrate that the PNAE specific curves can be non conservatives and to verify the applicability of the Master Curve to WWER RPV steels

Brumovsky [Brumovsky 1992] presented an experimental validation of lower bound curves for WWER RPV steels, 15Kh2MFA and 15Kh2NMFA and their welds, see Figure 3.4. The author recommended to create an international database of all fracture toughness data (static, dynamic and arrest) generated by testing specimens of different thickness from different manufacturers.

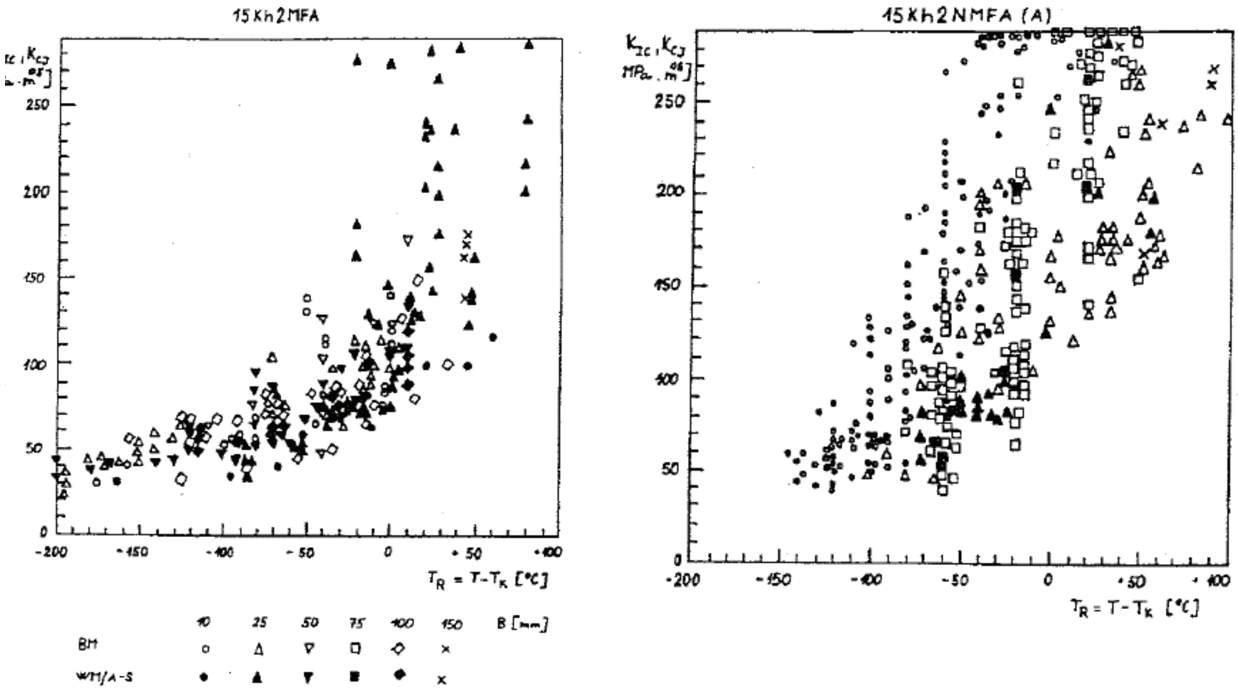
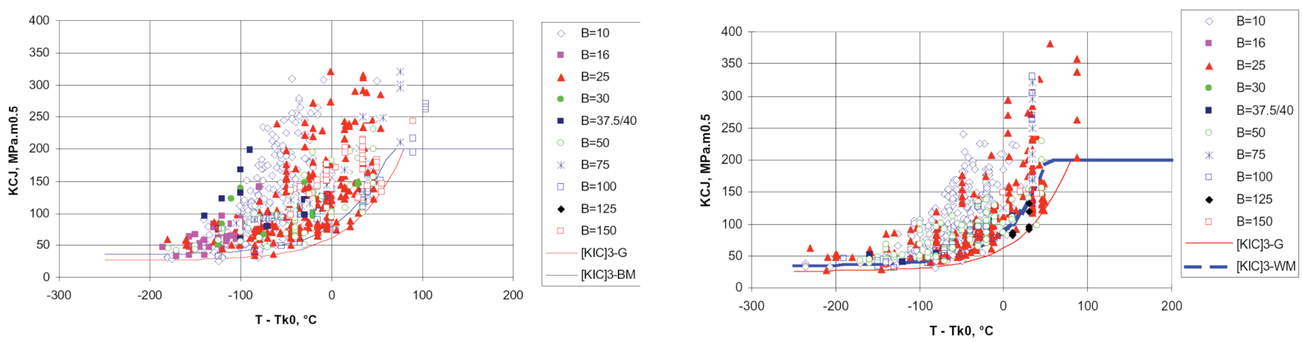


Figure 3.4 Experimental fracture toughness data [Brumovsky 1992]

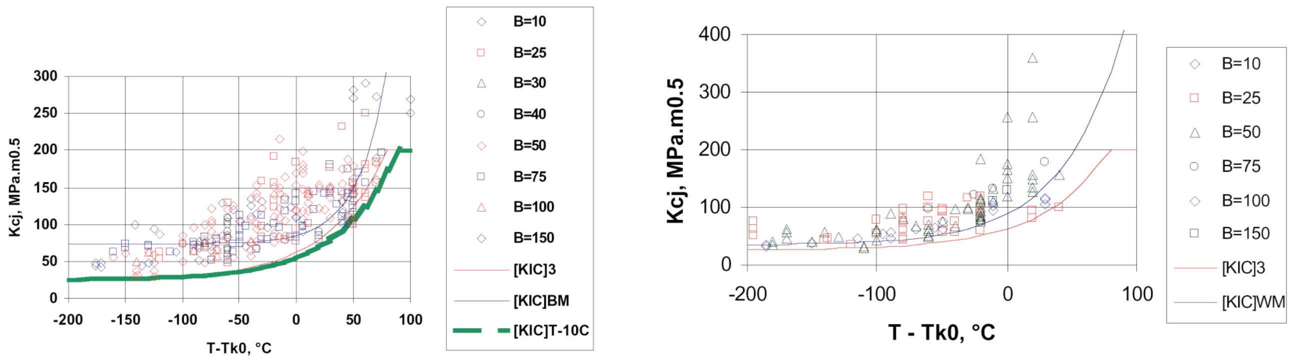
Within the 5th FWP concerted action VERLIFE, the Soviet fracture toughness reference curves were also verified. Main goal of the project was a preparation, evaluation and mutual agreement of a "Unified procedure for Lifetime Assessment of Components and Piping in WWER Type Nuclear Power Plants" [Brumovsky 2004]. This procedure should be based on former Soviet rules and codes and would be usable in nuclear power plants in Finland, Czech Republic, Slovak Republic and Hungary for Periodic Safety Reports and Components/Plant Life Management. More than 1,200 data for WWER-440 RPV materials and more than 700 data for WWER-1000 RPV materials from different WWER countries were collected within this project. The fracture toughness data collected within this project can be seen in 3.5 for WWER-440 RPV steels and in Figure 3.6 for WWER-1000, with the Soviet fracture toughness reference curves



15Kh2MFA Base metal

15Kh2MFA weld

Figure 3.5 Fracture toughness data for WWER-440 reactor pressure vessel correlated with transition temperature Tk0 (B = specimen thickness, [KIC]3 = generic design fracture toughness curve) [Brumovsky 2004]



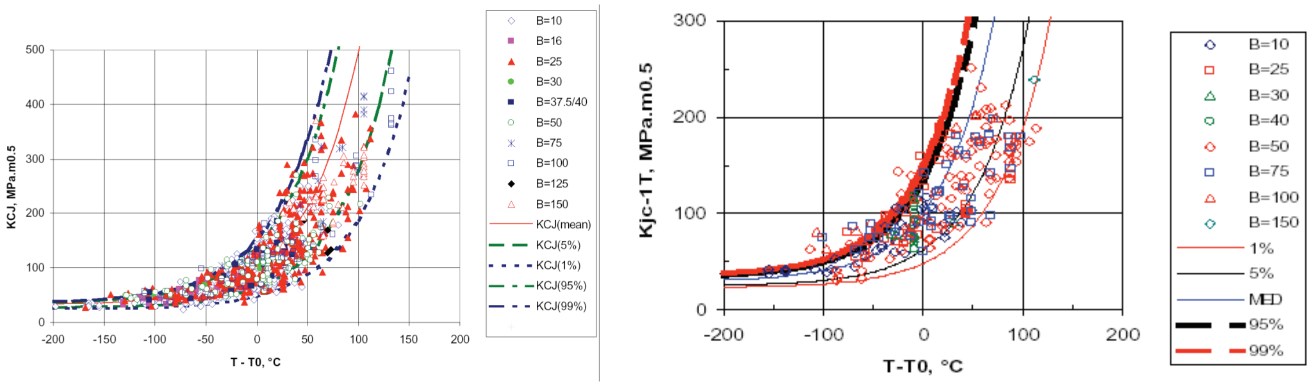
15Kh2NMFA Base metal

15Kh2NMFA weld

Figure 3.6 Fracture toughness data for WWER-1000 reactor pressure vessel correlated with transition temperature Tk_0 (B = specimen thickness, [KIC]3 = generic design fracture toughness curve) [Brumovsky 2004]

Analysis of this database again showed that the specifics Soviet fracture toughness reference curves are not conservative. Only generic curves are reliable as they cover practically all experimental data.

Within VERLIFE the use of the Master Curve was also checked, see Figure 3.7.



WWER 440 base metals and welds

WWER 1000 base metals

Figure 3.7 Master curves for WWER RPV material from VERLIFE [Brumovsky 2004]

One of the recommendations from VERLIFE procedure is a new lower bound of fracture toughness data based on the Master Curve [Brumovsky 2003]. This lower bound is

$K_{JC}^{5\%}=25.2+36.6\exp(0.019(T-RT_{T0}))$ where RT_{T0} is defined as $T0+\sigma_{T0}$.

The positive temperature gradient and attenuation of neutron radiation across the pressure vessel wall provides a mechanism by which initiated cracks may be arrested before they propagate through the wall. So beside initiation fracture toughness K_{IC}, crack arrest fracture toughness K_{Ia} it is necessary to perform a integrity assessment of the RPV.

In the ASME code crack arrest reference fracture toughness curve is included and used to determine the pressure-temperature limit curve of the vessel. This type of reference curve is not included in the Soviet Code. A suggested dependence of crack arrest fracture toughness is proposed by Timofeev [Timofeev 2000b], but the authors pointed out that to construct a robust reference temperature dependence of KIa it is necessary to carry out additional investigations.

Wallin et al [Planman 1997] have proposed a crack arrest master curve identical to the crack initiation master curve concept except that the reference temperature T0 is replaced by the crack arrest reference temperature TKIa, see Figure 3.8. This reference temperature can be determined by the analysis of the forces of a instrumented charpy test.

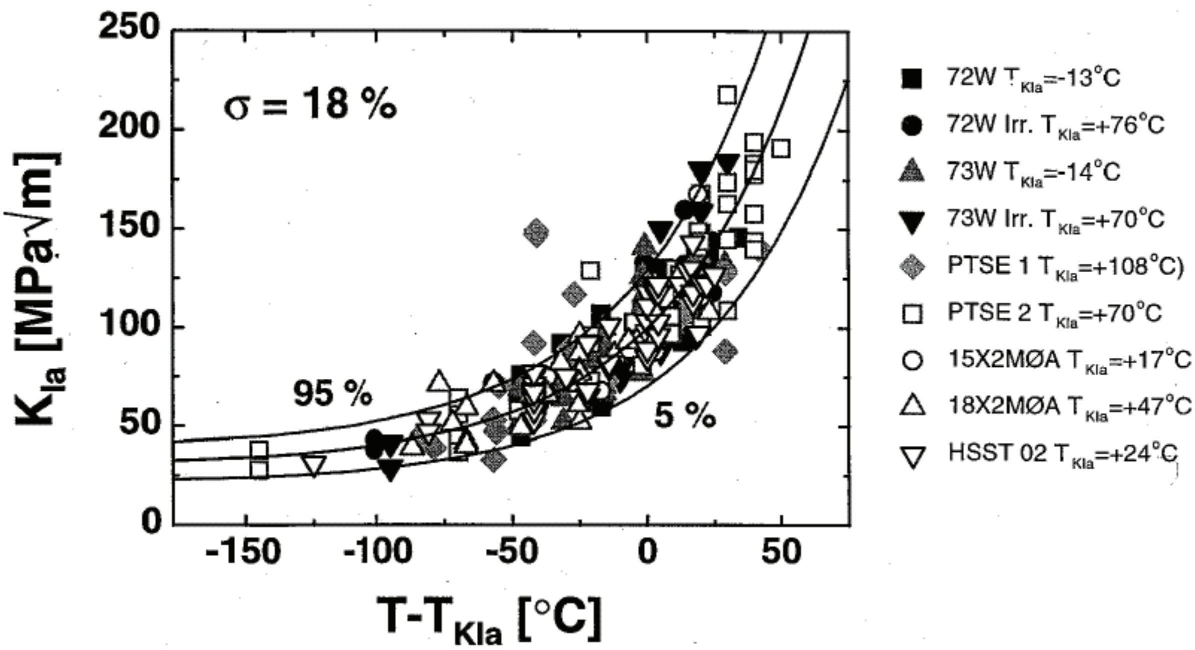


Figure 3.8 Crack arrest data for WWER-440 and WWER-1000 base metals [Planman 1997]

4 Irradiation Shift Prediction

For convenience, the papers reviewed have been grouped into sub-categories as shown in Table 4.1. In the pilot scheme, the irradiation shift papers would have been in the modelling folder. Table 4.1 shows how the subcategories relate. Of course papers may overlap sub-category boundaries, and may also be appropriate to other folders.

Irradiation shift sub-category	Papers in irradiation shift sub-category	Modelling sub-category used in 2007
Irradiation data and modelling	6	Irradiation damage modelling
Irradiation damage evaluation and techniques	8	Irradiation damage modelling
Re-irradiation embrittlement	1	Irradiation damage modelling
(Wallin) Master Curve	9	Fracture toughness modelling
Margolin/Prometey Unified Curve	2	Fracture toughness modelling
Upper shelf toughness	2	No papers in this sub-category
Probabilistic approach	2	No papers in this sub-category
Sub-category not applicable to irradiation shift		Pressurized thermal shock modelling
Sub-category not applicable to irradiation shift		Severe accident modelling

Table 4.1: Sub-categories

Irradiation damage and modelling

The papers reviewed reflect the healthy position, noted last year, of the state of the art in developing shift prediction models for WWER steels. Physical understanding, rather than empirical fitting, is being increasingly used to increase the robustness of the models, and to justify the approach taken, for example the augmentation of surveillance data with research data.

Some issues, however, remain to be fully resolved. The first is the effect of phosphorus in WWER steels. More work is needed if understanding is to be completed, and this may be particularly important to the effect of annealing and re-irradiation after annealing. Nickel also remains an issue. Nickel and manganese effects are currently of considerable interest for RPVs with MnMoNi steels and there could be considerable benefit in collaborative programmes involving both WWER and MnMoNi steels experts. At one time IGRDM had a separate WWER steels technical area; discussions of these steels are no longer separated since there is now a good prospect of greater commonality in modelling. One area worth further investigation in this regard is the 20% difference observed between Charpy and toughness shift for WWER-440/WWER-213C base and weld materials (Paper 3). For MnNiMo steels this difference is observed for base but not weld. Resolution of this may also help understanding of Master Curve shape issues – is the shape the same for all materials, or just nearly so? There are many other issues, most are, in principle, common to WWER and MnMoNi steels; as noted last time, there is potential for technical surprises to emerge (the problem of “unknown unknowns”).

Finally, there are no papers on the development of physical modelling of irradiation damage, such as has taken place in the REVE and EC PERFECT projects (and will be in the EC PERFORM60 project). Although of little practical applications value at present, such models eliminate unknown unknowns, provide the only legitimate means of extrapolating from current materials and irradiation conditions, provide a framework for the development of understanding and the development and maintenance of expertise, and are vitally important for the longer-term future.

Irradiation damage evaluation and techniques

The issue of data quality is addressed in Papers 10, 13 and 14. It is a problem for NPP generally that the designers and operators of older NPP did not anticipate that issues might emerge late in life putting a premium on high accuracy data. This is understandable (there were more unknown unknowns then) but the cost of retrospectively improving the data is high. No doubt there is much more that could be done for other data, but the problem in extending the work will be the cost. A step towards justifying (or not) the cost would be to estimate uncertainties for all data and the impact of these on the uncertainties on the shift models.

Some papers address the problem from the other direction – improving the evaluation techniques, and another describes a direct approach to validate the predictions, the extraction of test samples from the outside wall of an operational RPV. Which approach, data improvement, evaluation method improvement or direct validation, is best, will depend on costs and other practical issues.

One paper describes the simulation of irradiation-embrittlement by changing heat-treatment or alloy content. This may have value in some circumstances, but could give potentially misleading results in others.

Re-irradiation embrittlement

There was only one paper in this review. This was a presentation that demonstrated the maturity of work in this area for WWER-440 RPV steels. Work is ongoing in this area; it has obvious value for practical applications, and also for understanding of irradiation phenomena.

Fracture toughness

In the previous review it was noted that most of the toughness papers were about the Margolin/Prometey Unified Curve model. This time the balance is in favour of the Master Curve. This is now a mature technology, which has delivered considerable benefits. The continuing work reported in these papers revolves around relatively minor (in the sense that they will not undermine the key concepts) issues; the difficult regulatory issues; and applications to WWER plant.

Several of the above papers address remaining issues. One they do not address is whether the Master Curve is an adequate description of toughness, or should the Unified Curve, of which the Master Curve is a special case, be used? This is a question that should be resolved. Work is also needed on the extreme lower tail of the fracture toughness distribution for use in probabilistic assessments.

Upper shelf toughness did not feature in the previous set of papers. One paper presents a ductile fracture approach; another provides a physically-based model that relates to Master Curve shape.

Probabilistic modelling

It was noted in the last report that the conditions for a (real or postulated) defect in a vessel are more complex than is normally represented in an experimental programme.

One paper addresses this issue in the case of pressurized thermal shock in the context of a probabilistic approach. One paper described the PASCAL probabilistic fracture mechanics code being developed by JAERI. In a sense, neither of these is directly appropriate to the topic "irradiation shift". However, it is important to develop models in the context of understanding of the applications and, in particular, to bear in mind that integrity evaluation techniques extend operational boundaries. It is not only important to estimate statistical uncertainties accurately, but also to estimate potential bias between the mean prediction and the true value.

Other issues

Progress has been made since these papers were written, especially for the older ones. However, a number of the older papers contain useful insights as to how we got to where we are now and/or useful data. Older papers are therefore worth re-visiting, and perhaps in some cases re-evaluating with the benefit of later knowledge. For example, one paper notes a non-monotonic effect of neutron attenuation; was this due to data scatter or inhomogeneity in materials, or could it be attributed to some physical effect which is not yet fully emerged?

4.1 Consolidated Conclusions

The following general conclusions can be drawn from the review:

Irradiation damage and modelling

As noted at the first workshop, the situation is healthy:

- Physical understanding (rather than emprical fitting) is being increasingly used:
  - to develop shift prediction models for WWER steels
  - to justify the use of research data to augment surveillance data
- Understanding is being increasingly underpinned by microstructural observations

Irradiation damage evaluation and techniques

- A lot of good work has been done to retrospectively establish improve the quality and reliability of the data, validate predictions and develop evaluation procedures

Re-irradiation embrittlement

- Mature area

Transition regime fracture toughness

Compared to 1st workshop much more on Master Curve than Unified Curve

- Master Curve
  - Mature, highly successful, technology
  - Applications to WWER steels are being explored, with success
  - Introducing Master Curve into US codes was not straightforward
- Unified Curve
  - Well-developed (as reported last time)
  - More general (MC a special case of UC) but more complex mathematically

Probabilistic modelling

Progress is being made

- Codes

- Complex loadings (welcome development)

4.2 Open Issues

The general open issues, which are related to economic reasons and not to safety, are:

Irradiation damage and modelling

- Rôle(s) of phosphorus are not fully resolved
  - GB embrittlement vs precipitation
  - Site competition on boundaries
- Rôles of nickel and manganese (and interactions between these) in WWER steels
  - Same for MnNiMo steels – area for possible collaboration?
- Charpy and toughness shift correlations for plate and weld – difference between MnMoNi and WWER-440 steel behaviour?
- Problem of "unknown unknowns" produces need for multi-scale physical modelling (as in PERFORM60)

Irradiation damage evaluation and techniques

- Need to report uncertainty estimates on data (a general problem!)
- Use of heat treatment to simulate irradiation damage – when is it valid?

Re-irradiation embrittlement

- Precise rôle of phosphorus, and its behaviour during annealing and re-irradiation
- Annealing studies data help improve understanding of the mechanisms of irradiation damage generally

Transition regime fracture toughness

Compared to 1st workshop much more on Master Curve than Unified Curve

- Master Curve
  - There are still a number of small issues, but not clear how important these are for applications
  - Direct use of MC (use of T_0 not RT_{T0}) is still a problem for US-based codes
- Unified Curve
  - Is a more general model needed/justified
  - Use of thermally embrittled material - can it be justifiably used to simulate effects of irradiation?

Probabilistic modelling

- Tails of distributions – initiation versus arrest toughness
- Importance of realistic models and uncertainty estimates
- Importance of ensuring there are no unknown unknowns

4.3 Reviewed papers and summaries

The summaries of the individual papers are given below in order of sub-category and year of publication.

Sub-Category 1 – Irradiation Data and Modelling

Tsykanov V.A., Golavanov V.N., Krasnoselov V.A., Kolesova T.N., Kozlov D.V., Prokhorov V.I., Karzov G.P., Filimonov G.N.-"Estimation of radiation embrittlement of steel 15Cr2MoΦAA containing 0.75% of nickel and corrosion-resistant cladding after an exposure in facility "KORPUS"",- ca. 1990

This paper gives a report of the irradiation of one cast of a 0.17%C, 0.4% Mo, 3% Cr, 0.8% Ni, 0.7% Mo steel, and associated stainless steel cladding material in the KORPUS research reactor facility. The experiment is well documented (though the translation is not always easy to follow), but the raw test results themselves are not tabulated. An interesting observation was that the effect of neutron attenuation was not monotonic (although there was an overall reduction of shift from inside to outside and the relatively small shifts and experimental scatter may have confounded the intermediate results). The steel (which contained low levels of P and Cu) had good radiation resistance (at 9.5 x 10^{19} n/cm^2, E>0.5MeV). The cladding remained ductile after irradiation (in the temperature range -100 to 200 ^{o}C).

Amayev A.D., Kryukov A.M., Levit V.I. and Sokolov M.A.,-" Radiation Stability of WWER-440 Vessel Materials" - published in "Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels: An International Review (Fourth Volume). ASTM STP 1170, Lendell E Steele Ed. ASTM 1993, pp 9-29

The comprehensive paper provides a useful background to WWER-440 embrittlement issues and describes the results of investigations in surveillance and research programmes for the steels involved and subsequent modelling. These included an assessment of the effect of direct exposure to the coolant – no effect was found at 2 x 10^{19} n/cm^2, E>0.5MeV, with an irradiation temperature of 250 ^{o}C. This observation allowed all results, encapsulated samples and exposed to coolant samples, to be pooled.

The irradiations were carried out at fluxes of 4 x 10^{11} and 4 x 10^{12} n/cm^2 at 270 ^{o}C to up to a fluence of about 7 x 10^{20} (all neutron exposure values are for E>0.5MeV.) From the results the authors deduced that four mechanisms of irradiation damage were operating: an independent P mechanism; a P-Cu interaction mechanism; an 'other' mechanism, which could not be associated with composition; and an independent Cu mechanism. These were present in different proportions depending on chemical composition and flux. Regression equations were developed for these mechanisms, though for practical purposes simpler models could also be used. Other observations were: beyond about 4 x 10^{20}, there was a distinct increase in the rate of embrittlement (for the non-Cu, non-P contributions) above the (fluence) 0.3 dependency in the simpler models; that higher irradiation shift was observed at the lower flux; and that the P effect was not due to segregation.

Brumovsky M, Novosad P., Falcnik M.,. Vacek M., and. Malec J. "Re-evaluation of Irradiation Embrittlement of Surveillance Specimens" - published in Nucleon 3, 1998

The paper discusses issues connected with WWER-440/WWER-213C surveillance programmes, in particular the issue of fracture toughness, compared with Charpy shifts. Broken Charpy specimens were reconstituted to provide directly comparable fracture toughness data for base material and welds. It was found that, for both materials, the toughness shift was about 20% greater than the Charpy shift (the figure illustrating this is missing from the paper). However, the absolute difference in shifts, for end of life fluences, was not great in comparison with experimental scatter.

These new results showed smaller shift differences than the original assessment in which it had been necessary to normalize results from the separate Charpy and toughness specimens to the same fluence. The authors believed that this could be due to flux effects.

Pechenkin V.A., Konobeev Yu.V.,Stepanov I.A., and Nicolaev Yu. A., "On the mechanism of irradiation embrittlement enhancement in reactor pressure vessel steels at high neutron fluences" - published in "The effects of radiation on materials: 21st international symposium", ASTM STP 1447, M.L. Grossbeck et al eds., ASTM 2004, pp 138 - 148.

The authors point out that the (then) current approach to embrittlement prediction for WWER440 steels was based on an empirical/statistical approach. This was of course not reliably extrapolable outside the database and had indeed been shown to be give non-conservative predictions for high dose surveillance capsule results. Growing knowledge of embrittlement mechanisms suggested that, while the effect of copper saturates at relatively lower fluences, additional embrittlement can then occur due to segregation of phosphorus to grain boundaries.

They then describe the development of a physically-based model to predict this effect. This was based on a modified McLean approach. Because of the scatter in the values of key constants, such as the activation energies reported in the literature, two sets of values were used. Both sets showed that segregation (at a given time) was greater at surveillance specimen fluxes than at RPV wall fluxes. Both sets gave quantitatively similar grain boundary phosphorus segregation at high temperature; at RPV operating temperatures they showed similar trends.

A phosphorus embrittlement (shift) model was produced using the GB phosphorus segregation model by considering known mechanisms and relationships between segregation and grain boundary fracture stress. In the overall model, the phosphorus segregation shift was added to the existing shift model at fluences beyond an empirically-determined (but physically justified) threshold. The phosphorus contribution to shift was calibrated from thermal ageing data. Model predictions agreed well with previous surveillance data; a new experimental result, developed in this work, was consistent with model predictions that the rate of phosphorus embrittlement would reduce with fluence. The model could take into account pre-irradiation phosphorus segregation but this was not tested by the data available.

The authors pointed out, that further work was needed, in particular to take into account the effect of irradiation-induced sinks on segregation kinetics.

Debarberis L., Acosta B., Sevini F., Kryukov A., Gillemot F., Valo M., Nikolaev A., Brumovsky M., "Role of nickel in a semi-mechanistic analytical model for radiation embrittlement of model alloys" - published in the International Journal of Nuclear Materials, 336 (2005) 210-216.

This paper extends the three damage component model, discussed in several Debarberis et al. papers, to include parameters for the effect of nickel. The analysis was based on data on model alloys irradiated in the HFR at Petten and in the Kola NPP. Both irradiations were carried out at 270 ^{o}C, in the former to about 7 x 10^{18} n/cm^2; in the latter to about 65 x 10^{18} n/cm^2. The fluxes are not given in the paper (though references are given to further details).

A large effect of nickel was observed for both the matrix damage and the precipitation damage components (the effect of nickel on the phosphorus component was not clear). The model was extended to parameterize the effect of nickel. Since there were only two fluences, this had to be done using a simple linear model, but this substantially reduced the residuals to the fit compared with the previous version of the model. It was shown that the new model enabled a better analysis of re-embrittlement after annealing.

Nikolaev Yu A., "Radiation embrittlement of Cr-Ni-Mo and Cr-Mo steels" - published in the Journal of ASTM International, Vol 4, No 8 (Received June 2006).

This detailed paper discusses the development of physically-based models for WWER-440 and WWER-1000 steels. The paper notes that models (including the regulatory model) based on accelerated irradiations may be substantially non-conservative and that damage may depend on microstructure, weld being more sensitive than plate of the same chemical composition.

For WWER-440 steels, the surveillance data are quite limited and all have similar phosphorus and copper content. It was therefore necessary to consider also research programme results from irradiations carried out in surveillance channels. Based on the available understanding, including that from observations of the fine-scale microstructure, a four-term model was constructed. The models were carefully fitted to the data using a number of statistical parameters and plots to optimize the model parameters. The model fitted the data well.

For WWER-1000 steels, there were more surveillance data available and it was not necessary to use research programme data for modelling. For base material, which has low variations in nickel content, a relatively simple model was effective. For weld, nickel and manganese could be seen to have a significant effect on irradiation sensitivity. The final model included these elements as well as silicon, an effect of which emerged from residual analysis, and had been observed in work elsewhere.

<u>Sub-category 2 – Irradiation damage evaluation and techniques</u>

Popov A A., RogovM .F., Dragunov U.G. and Parshutin E.V., "I mprovement of methods to evaluate brittle failure resistance of the WWER reactor pressure vessels" - no reference, date uncertain

The report discusses an analysis of methods to determine the toughness transition temperature of reactor materials. It consists mainly of figures and tables and is drawn from a number of sources. It refers to the development of a project to pursue improvements to this area.

Kohopaa J., Valo M. and Ahlstrand R "Evaluation of the Radiation-Damage from the reactor pressure vessel at Loviisa 1 using samples taken from the outer surface" - published in the international workshop on WWER-440 reactor Pressure-Vessel embrittlement and annealing, 29-31 March 1994, Zavazna Poruda, Slovak Republic

This report discusses the testing of samples cut from the Loviisa 1 RPV to verify that the surveillance program specimens represent the actual base material in the highest neutron flux region.

The samples were extracted by mechanical sawing and varied in thickness between 6 and 9 mm with a mean diameter of about 100mm. Small 4 x 3 x 27mm KLST specimens were tested to obtain shift values. Taking into account corrections for the difference in fluence and specimen size these agreed well with the values from the actual surveillance data for base material for full-size Charpy specimens. There were no indications of a neutron flux effect on the outer surface of the vessel. Tensile testing was also done.

Chemical analysis results were in reasonable agreement with previous data except in the case of copper where the values from the samples and the surveillance specimens were systematically lower than the manufacturer's data.

The microstructure was generally as expected but there were indications that the manufacturing heat treatment was monitored by measuring the furnace rather than the actual forging temperature leading to more extended tempering treatments. Irradiation was found, using FEGSTEM, to increase the amount of grain boundary segregation, but that in the RPV samples was not higher than that in the surveillance specimens; fluences however were relatively low.

Fluences derived from the samples using 54Mn activity confirmed currently used fluence estimates.

The work showed that the operational life of the RPV was not limited by irradiation embrittlement of the base material.

Pelli R, Planman T, Rintamaa R., Karzov G. and Timofeev B., "Simulation of irradiation embrittlement of welded pressure-vessel material corresponding to the end of WWER-440 reactor service life" - presented at the IAEA specialist meeting on irradiation embrittlement and mitigation, 23-26 October 1995, Espoo, Finland

The report is in the form of a series of slides and figures with no text. It discusses embrittlement limits including that of cladding and a range of methods for simulating end of life properties, for example by changing heat treatment or alloy chemical content to give higher transition temperatures. Welds were seen to be a particular problem in this respect. The tables and figures give useful and interesting data in respect of achieving the desired objectives.

Apostolov et al T., "Analyses for evaluation of reactor pressure vessel metal state and lifetime at Kozloduy NPP Unit 1"- published in BGNS Transactions, Vol 1, No1, 1996, p30-35

Samples from weld 4 of the Kozloduy NPP Unit 1 had been examined to obtain phosphorus and copper contents. The paper details the methods and the results.

Calculations had been also made on the neutron fluences and induced activities using a range of codes, and these were compared with measurements of the activities on the samples, as well as on detectors irradiated in the reactor. Measurements and calculations agreed well for four of the six samples but in two cases there was a thirty percent difference with the measured activity being lower than that calculated. In the case of the detectors there was a significant deviation for only one of the nine cases; this involved a ^{93}Niobium detector.

The measurements and estimates used were used to calculate the irradiation embrittlement of the RPV. The paper discusses means to limit re-irradiation embrittlement during the next cycle of operation, to avoid exceeding an unacceptable value.

Falcnik M, Novosad P, Kytka K. and Brumovsky M-" Procedure for a determination of irradiation embrittlement trend curves"- presented at the IAEA specialist's meeting on irradiation effects and mitigation, Vladimir, Russia, 15-19 September, 1997

The paper discusses procedures for redevelopment of irradiation embrittlement trend curves. Charpy and fracture toughness data were obtained on base weld and heat affected zone material for WWER steels.

The results from the surveillance specimen program were described reasonably well by a three parameter Weibull model; however a bimodal distribution was apparent in irradiated material.

Transition temperature shifts were the same regardless of whether mean or lower bound curves were used, and we're not significantly affected by the use of a log normal rather than a Weibull distribution.

It was also demonstrated that the trend curve exponent was approximately equal to 0.5, and that the Master Curve could be used to obtain data from the limited number of samples. An issue not resolved was the effect of the fluence measurement error on the transition temperature shift accuracy; greater accuracy of fluence measurement (5%) was desirable, but not yet achievable.

Mitev Ml., Belousov S. and Ilieva K – "Neutron and gamma Monte Carlo calculations for the determination of radiation damage of WWER-1000 reactor vessel", abstract only, published in BgNS Transactions, vol 9, No 1, 2004, p266

The short abstract indicates that the paper discusses Monte Carlo calculations of neutron and gamma spectra for WWER-1000 vessels. Damage is estimated in terms of dpa, and gamma damage is compared against neutron damage.

A Ballesteros et al.-"Consolidation of the scientific and technological expertise to assess the reliability of reactor pressure vessel embrittlement prediction in particular for the arctic plant area (COBRA)"- published in Nuclear Engineering and Design 235 (205) 411-419

A problem with WWER-440 surveillance programs is that the operating conditions are not well known. This paper presents work in the COBRA project to establish values of irradiation temperature and neutron fluence. In this project capsules containing the state of the art dosimetry and temperature monitors were irradiated in the Unit 3 of the Kola NPP. Thermocouples were also installed. In addition, models were developed to give better estimates of gamma heating in locations where there were no direct measurements.

The thermocouple readings showed that there was a small, about 5^{o}C difference between the mean inlet water temperature and the surveillance specimen temperatures. Melting temperature monitor results were inconsistent between the two chains and within chains. They also contradicted the thermocouple readings.

The flux measurements were consistent with previous data and the different detectors were consistent with each other within 10%, except in the case of niobium, where the difference reached about 20%.

Results of the 2D modeling for various cases showed that capsule overheating would be expected to be less than 10°C and these calculations were consistent with data. A satisfactory 3D model was also developed.

<u>Sub-category 3 – Re-irradiation embrittlement</u>

YU Erak D., Kevorkyan Yu. R.,Chernobaeva A.A.,Shtrombakh Ya.I."Prediction of re-irradiation embrittlement for WWER-440 reactor pressure vessel steel materials" - slides *presented at the AMES 1st Biennial conference on through-life toughness prediction in reactor steels, Hungary, February 2006.*

The presentation discusses the problems of WWER-440/230 RPVs arising from the high embrittlement rates, the lack of information on some important variables and the absence of an accurate re-irradiation embrittlement model. The lateral shift model initially specified was shown to be conservative so further data were sought from research programmes, including accelerated irradiations. Embrittlement on re-irradiation occurred at a slower rate than on virgin material. Atom probe studies and fracture tests showed that, due to over-ageing, copper does not play a significant role after annealing, but phosphorus does. Consequently the lateral shift model is not appropriate.

In work done by Amaev et al, a flux effect had been observed for embrittlement of unirradiated materials containing copper. The copper/flux effect would not apply to irradiated and annealed materials and a re-irradiation model involving phosphorus and fluence only was proposed. This fitted the data reasonably well, but further work (the PRIMAVERA project) was needed to extend the re-irradiation database and develop the model.

Sub-category 4 – Master Curve

Sokolov M.A.., McCabe D.E., Alexander D.J. and Nanstad R.K., "Applicability of the fracture toughness Master Curve to irradiated reactor pressure vessel steels" - no reference given (date about 1998)

This paper is undated, but appears to be one of the relatively early papers investigating the applicability of the Master Curve for irradiated reactor pressure vessels steels. It comprises an extended summary together with a number of view graphs. The issues addressed are the equivalence of Charpy and toughness shift, whether irradiation affects the shape of the Master Curve, the adequacy of the statistical basis of the Master Curve, the use of small specimens, and the influence of intergranular fracture.

It was found that the Master Curve derived from pre cracked Charpy specimens represented large linear elastic toughness data from HSST-02 well. More work was however needed to develop adjustments for small specimen thickness to take into account their width to thickness ratio. The available data did not suggest a Master Curve shape change with radiation; this observation for Charpy shifts of up to about 250°C. Charpy and toughness shifts were found to be the same for weld metals, but the toughness shift for base materials was 12°C greater than the Charpy shift. No data were yet available on the issue of intergranular fracture and its influence on Master Curve.

Rosinski S.T. and Server W. l., "Application of the Master Curve in the ASME code"- published in the International Journal of Pressure Vessels and Piping 77 (2000) 591-598

This paper presents work in connection with the development of ASME codes to allow use of the Master Curve for estimation of RPV toughness through the RT_{T0} parameter instead of RT_{NDT} zero. ASME Code Case N629 allows this for both on irradiated and irradiated RPV materials. The technical basis for this was being revised to reflect the application to irradiated materials and is the subject of this paper.

The paper describes the limitations of the original RT_{NDT} approach and the basis of the new one. It describes how the functional equivalence between RT_{T0} and RT_{NDT} was to be achieved so that the intent of the ASME Code is maintained. The applicability of the Master Curve methodology was demonstrated by assessing a larger range of unirradiated data. It was shown that the results normalized to a 1T specimen size generally fell in relation to the Code Case curve. It was also shown that the new methodology, in effect, is such that for all cases there is a minimum margin of 10°C. In that sense the use of RT_{T0} was considered to be more conservative than that based on RT_{NDT} for which the margins varied significantly.

Approximately 850 irradiated toughness values were available for examination and less than 1% of these data were not bounded by the Code Case curve. This was notionally better than the unirradiated case; however it was pointed out that in for the unirradiated tests there was a greater number of larger specimens, which would be more likely to exhibit lower toughness values. But the main point was that the definition of RT_{T0} appeared equally applicable to irradiated as well as unirradiated steels.

Serrano M, Perosanz F.J. and Lapena J. "Direct measurement of reactor pressure vessel steels fracture toughness: Master Curve concept and instrumented Charpy-V test" - published in the International Journal of Pressure Vessels and Piping 77 (2000) 605-612

This paper presents a study to assess the use of Charpy V-notch specimens to obtain fracture toughness on irradiated data, and thus avoid the need to use fracture toughness versus Charpy correlations with their associated uncertainties. The material used for the investigation was the IAEA reference material JRQ in the unirradiated condition. Charpy V-notch tests were compared with PCCV and ½T compact tension tests. The Charpy tests were instrumented in order to better estimate the onset of crack growth. Details are given of the methods used to obtain and analyze the data, including taking into account the issues associated with instrumented Charpy measurements. The method of obtaining the Master Curve T_0 from the fracture toughness tests is also described in detail.

It was shown that the Master Curve, determined by testing small specimens, is valid but some toughness values were outside of the tolerance band and it was recommended that more work should be done. It was shown that toughness could be estimated by instrumented impact testing of standard V-notched Charpy specimens and that these gave results in accordance with static fracture toughness specimens in spite of the dynamic nature of the tests.

Viehrig H.W., Boehmert J., Dzugan J. and Richter H. -"Master Curve evaluation of irradiated Russian WWER type reactor pressure vessel steels - published in "The effects of radiation on materials: 20th international symposium", ASTM STP 1405, S T Rozinski et al eds., ASTM 2001

A paper presents results of the joint German-Russian research program to investigate the influence of deleterious elements on the embrittlement of the WWER type RPV steels due to neutron irradiation. These were then used to evaluate the validity of the Russian predictive formula. In addition the annealing behaviour of different WWER steels was investigated. The emphasis of the investigations was on the application of the Master Curve approach.

Overall about 800 specimens from 24 different heats of the WWER steels were irradiated, including both weld and base materials. Irradiations were carried out in high flux locations in the Rheinsberg reactor at 255°C.

Charpy V-notch, pre-cracked Charpy SENB and 1T-CT specimens were irradiated. Metallographic investigations were carried out to characterize the basic microstructures. Results were compared with predictions using the standard equations, and the effects of annealing were also shown.

For base materials, the transition temperature shifts for future irradiation could generally be conservatively estimated by the predictive formulae, using a maximum irradiation embrittlement coefficient. For the WWER weld material, shifts were greater than prediction. Recovery in upper shelf energy was greater than that in transition temperature, and some cases of over-recovery were observed. The detailed behaviour could not be explained by the chemical composition or microstructural differences and nano-scale microstructure examinations were planned to clarify the observed observations.

Wallin K., Planman T, ValoM and Rintamaa R, "Applicability of miniature bend specimens to determine the Master Curve reference temperature T_0" published in Engineering Fracture Mechanics 68(2001) 1265-1296

This paper investigates the applicability of miniature three-point bend specimens to determine the Master Curve transition temperature, and also compares T_0 estimates from three point bend and CT specimens. The comprehensive test program involved ten materials, three of which were tested in both the unirradiated and irradiated states, and four different specimen cross sections ranging in size from 10 x 10 to 3 x 4 mm. Results are presented in a series of plots. The Master Curve analysis generally followed ASTM E1921-97, buts made use of a more sophisticated fitting method suggested for a revision of the standard. Some of the departures from the standard were noted and explained.

With the exception of one anomaly, the results for the different sizes showed overlapping scatter bands. In addition to the data generated in the program some other data were taken from the literature for the overall analysis.

It was concluded that miniature bend specimens are applicable to determine the Master Curve transition temperature, T_0, though, to achieve the same accuracy, more specimens are required as size decreases; 30 or 40 specimens in the case of the smallest sized specimens (3.3 x 3.3 mm) investigated, compared with 7 specimens for 10 x 10 Charpy-size specimens. Even so, the saving in material was about a factor of three (though the most efficient size was 5 x 5 mm).

From the comparison of compact tension and three point bend specimens, it was concluded that the former gave a T_0 on average 8° C higher than the latter.

Kirk M. and Mitchell M. "Potential roles for the Master Curve in regulatory application" - published in the International Journal of Pressure Vessels and Piping 78(2001) 111-123

This paper presents a USNRC perspective of the use of the Master Curve approach to replace the RT_{NDT} approach. It demonstrates the need and benefits of improved toughness estimation and then considers the technical issues to be resolved in the use of the Master Curve for end of life of toughness estimation. The seven issues are: curve shape, small specimen testing, implicit margin on the bounding curve (does the RT_{T0} method provide the margins implicit in currently used approaches), irradiation damage projection, uncertainty procedures(whether current guidelines can be applied to Master Curve); loading rate effects; and crack front length effects.

These issues are discussed comprehensively and with reference to ongoing work in these areas. The paper concludes that acceptance of Master Curve technology as a viable alternative to existing techniques depends on technical research progress in the coming years. Two general themes for such work were emphasized. First was a need for a systematic framework to address uncertainty questions for Master Curve assessments; it was suggested that a new uncertainty framework may be needed. The second was the issue of better understanding of the physical basis of the Master Curve approach. Better understanding was needed because it would be difficult to generate the empirical data to resolve the issues, and therefore there would have to be more dependence on theoretical or physically based solutions.

Server W., Rosinski S.,Lott R, Kim C. and Weakland D." Application of Master Curve fracture toughness for reactor pressure vessel integrity assessment in the USA"- published in the International Journal of Pressure Vessels and Piping 79 (2002) 701-713

This paper describes in detail the application of the Master Curve approach to the Beaver Valley Nuclear Power Station Unit 1. This unit has a potential problem with reaching extended end of life due to the projected shift in RT_{NDT} approaching the pressurized thermal shock screening criterion. Potentially the situation could be ameliorated using Master Curve-based (RT_{T0}) approach, which had been accepted by NRC in the case of the Kewaunee reactor.

There were however any number of issues to be resolved including such factors as margins forward predictions off the shift and bias terms for specimen size and geometry. These were discussed in detail with specific reference to the plant in question. Important differences between the Kewaunee and Beaver Valley cases were also treated.

The evaluation suggested that the Master Curve methodology would give an RT_{PTS} value lower than the current limit. In the longer term a supplemental surveillance capsule would enables direct measurement of fracture toughness values for extended end of life fluences and thus reduce the margins required in the current evaluation.

Brumovsky M.,"Check of Master Curve application to embrittled RPVs of WWER type reactors" - published in the Vessels and Piping 79 (2002) 715-721

The paper first compares the ASME approach to defining K_{IC} and K_{IR} and the equivalent approach for WWER steels. It then investigates the potential for using the Master Curve for WWER steels. Although some data for the latter were included in the original Master Curve analysis, confirmation was needed using a large database. It was shown that, for the 1200 data available, the Master Curve well represented the temperature dependence of the median fracture toughness and the distribution of the data. The slope of the curve did not change with irradiation, and the specimen size correction also appeared to be applicable to WWER-440 steels. Next steps would be to revise the Czech ASI Codes for WWER pressure components to harmonize these with PWR approaches as much as possible. In addition the EC FP5 VERLIFE programme, involving Czech, Slovak, Finnish, Hungarian and Bulgarian collaboration would develop a unified procedure for lifetime assessment for WWER component and piping.

Ballesteros A.,Bros J.,Brumovsky M.,"Applications of the Master Curve approach to irradiated steels" - published in the proceedings of ICONE12, twelfth international conference on nuclear engineering, April 25-29, Arlington, Virginia, USA

This is a short paper describing the ASME, the Master Curve and the ASME Code Cases N-629 and N-631. It describes the open issues relating to the Master Curve approach. These include application outside the valid temperature range, application to materials failing by grain boundary fracture, application to inhomogeneous materials, choice of safety margins, and lack of Master Curve based trend curves for irradiated steels. In

Spain the three year CUPRIVA project was investigating Master Curve application, with Santa Maria de Garoña and Ascó participating as pilot plants. Reconstituted specimens were being used to establish fracture toughness. Preliminary results showed that the ASME approach is highly conservative.

Sub-category 5 – Margolin/Prometey Unified Curve

Margolin, Gulenko, Nikolaev and Ryadkov "Problems for prediction of the temperature dependence of fracture toughness for highly embrittled RPV steels"- published in International Journal of Pressure Vessels and Piping 80-(2003) 817-829 and 82 (2005) 679-686

There are a total of five slides. The first compares results for WWER-1000 steel base metals in an embrittled state (produced by a special heat treatment) analyzed on the basis of the Master Curve approach and the Prometey model. The latter provided a better description of the shallow curve shape. The second slide describes the Unified Curve concept and provides more examples in comparison with the Master Curve. In the fifth slide it has shown that, as T_0 increases, the Unified Curve predicts toughness increasingly better than the Master Curve.

Margolin B.Z., Gulenko A.G., Nikolaev V.A. and Ryadkov L.N.,"Prediction of the dependence of KJC(T) on the neutron fluence for RPV steels on the basis of the Unified Curve concept"- published in International Journal of Pressure Vessels and Piping (2005) 679-686

In the Unified Curve approach, the shape of the curve depends on the parameter Ω, which decreases with embrittlement. In order to determine the effect of irradiation on the toughness of a material it is necessary to know the dependence of Ω on fluence. Purpose of this paper is to obtain that dependence. It is shown that Ω is related to a Weibull parameter describing the probability of finding in each unit cell carbide with minimum strength less than σ_d. Through this parameter, and constants dependent on irradiation temperature, neutron spectrum and chemical composition, Ω could be related to fluence. The results were compared to Master Curve predictions, for which fracture toughness curve is invariant with embrittlement. The difference between the two Is quite small for low values of transition temperature but become significant as transition temperatures increase above about 50°C. The prediction results and data from the Dukanovy and Bohunice NPPs were in good agreement.

Sub-category 6 – upper shelf toughness

Margolin B. Z., Kostylev V.I., Ilyin A.V. and Minkin A.I., "Simulation of JT curves for reactor pressure vessel steels on the basis of a ductile fracture model".

The paper proposes a procedure for prediction of JR curves for RPV steels in the unirradiated and irradiated states. Material representing WWER-1000 base material was investigated in these two states, the embrittled state being achieved through a special heat treatment. The ductile fracture model, which had been previously published by Margolin et al, was described and then the procedures used for determining the parameters for this model. This required measurements from cylindrical tensile specimens. The third part of the procedure was to perform an FEM analysis of the stress and strain fields near the stationary and growing crack tips. Results of the predictions were compared to measurement and were in good agreement.

Mark EricksonKirk and Marjorie EricksonKirk "An upper shelf fracture toughness Master Curve for ferritic steels"- published in 83 (2006) 571-583

The Master Curve does not quantify toughness on the upper shelf and the objective of this paper was to examine whether to the upper shelf toughness of different ferritic steels exhibit a common dependence in the same way as they do in the transition region.

A database of over 1000 upper shelf fracture toughness data was compiled. Details are given of the database and the methodology used for model development. In particular consideration was given to physical process is of ductile fracture in formulating the model.

It was found that the temperature of dependence of fracture toughness was consistent across all the (ferritic) steels examined and was of the same form as the temperature dependence of the flow stress. The reason for this commonality was that the upper shelf fracture toughness depends directly on the energy required to move dislocations through the matrix and this depends, within the range of practical interest to nuclear vessels, only on the lattice atom vibration.

Sub-category 7 – probabilistic modelling

Li Y., Kato D., Shibata k. and Onizawa K. "Improvements to a probabilistic fracture mechanics code for evaluating the integrity of an RPV under transient loading"- published in International Journal of Pressure Vessels and Piping 78(2001) 271 -282

This paper describes the development at JAERI of a probabilistic fracture mechanics analysis code, PASCAL. The paper outlines the main features of PASCAL and describes improvements to it. These include: improved stress intensity factor calculations; better means of dealing with overlay cladding; and optimization of the sampling and cell dividing procedures. Work to improve the code is being continued.

Margolin B.Z., Kostylev V.I. and Keim E., "Prediction of brittle fracture of RPV steels under complex loading on the basis of a local probabilistic approach"- published in International Journal of Pressure Vessels and Piping 81 (2004) 949-959

The paper describes the development of the Prometey model for use in the case of non isothermal and non-monotonic loading, for example warm pre-stressing conditions in pressurized thermal shock. The model allowed calculations to be performed to assess warm pre-shock effects on fracture toughness; these were confirmed by experimental data.

4.4 Further references

Odette G.R., Yamamoto T. and Klingensmith T. "On the effect of dose rate on irradiation hardening of RPV steels", Phil Mag 85, Nos 4-7, 01 Feb – 01 Mar 2005, 779-797.

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Roberts S.G., Noronha S.J., Wilkinson A. J., Hirsch P.B.," Modelling the initiation of cleavage fracture of ferritic steels", Acta Materialia 50 (2002) 1229-1244

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5 Property-Property Correlation

A collection of around 100 papers on property-property correlations for irradiated reactor pressure vessel steels has been reviewed in order to have a clear picture of the advances achieved so far in this field, and to determine existing gaps and open issues. It allows identifying future research activities needed. The review performed focus in three main areas. Namely, 1) non-destructive examination of the embrittlement condition, 2) fracture and Charpy toughness determination, including specimens size and constrain issues, and 3) embrittlement trend curves and results from samples cut out from operated WWER RPVs. Accordingly, the evaluation report is structured in three parts, but with a common objective: to gain insight into property-to property correlations.

During the discussions the use of the Master Curve Approach at intergranular fracture was discussed and concluded, that it would not work.

5.1 Consolidated Conclusions

<u>Part1: NDE</u>

In several papers the TEM and PAS techniques have been chosen to characterise microstructural changes in surveillance specimens of WWER type RPV steels, and RPV steels irradiated in research reactor. The results obtained in several experiments indicate that AC (Angular correlation) Positron Annihilation Spectroscopy is an effective technique for the evaluation of microstructure changes and, in combination with other spectroscopic methods (Mössbauer spectroscopy, Transmission electron microscopy, Neutron diffraction, Internal friction, etc.) can contribute to an increase of NPP operational safety. Interpretations of the results obtained from angular correlation PAS technique are in agreement with the findings from the Mössbauer effect experiments or PAS lifetime measurements on RPV steels reported in other papers not included in the ODIN database.

The results obtained with the electric properties based techniques (STEAM and REAM) are very promising. It is confirmed that there exists a relationship between the change of the Seebeck coefficient and the changes in the mechanical properties due to neutron irradiation. The application of the resistivity measurements is also promising but needs improvements to the experimental set-up. The results obtained from the measurement of the thermoelectric voltage on several model alloys with nickel content ranging from 0.7 to 2 wt% and different copper concentrations are encouraging, showing that the STEAM non-destructive method has very high potential for studying the copper precipitation contribution in the radiation embrittlement phenomenon.

The magnetic-hardness testing technology, which is based on the simultaneous use of results of kinetic-hardness and magnetic methods, has been used in Russia. Preliminary investigations on the possibility to measure magnetic properties through cladding have given positive results. The measurements were carried out directly on the surface of base metal, and also through two different cladding layers with thickness 3÷4mm and 8÷9mm.

Atom probe tomography of 15Kh2MFA Cr-Mo-V steel surveillance specimen investigation has determined that changes in mechanical properties correlate with the presence of spherical and cylindrical manganese-, silicon-, copper-, phosphorus- and carbon- enriched features in the matrix of irradiated base and weld materials. There were significant increases in the yield and ultimate tensile strengths, and reduction in the elongation and reduction of area.

Where other techniques fail to produce evidence for any effect, the results reported by K. van Ouytsel and his co-workers show that the internal friction technique is sensitive to the small changes induced by thermal ageing. Amplitude-dependent internal friction allows the determination of the yield stress of the material. For steels in the unirradiated condition, the results compare excellently with static tensile test results.

The SANS measurements confirmed continuous growth in volume fraction of irradiation-induced precipitates with increasing fluence. This growth is probably accompanied by increasing concentration of solute atoms, as indicated by the changing ratio of magnetic to nuclear scattering cross-sections. The variation of apparent volume fractions as observed by SANS is closely correlated with the change of ductility transition temperature, which demonstrates that SANS can be used for reliable assessment of radiation embrittlement in RPV steels.

Part 2: Master Curve

Around 30 papers have been reviewed and can be divided by the following subjects: Specimen size /constraint issues; Fracture toughness reference curves and Charpy/toughness/yield shifts

- Regarding constraint issues and specimen size, the use of small specimens for fracture toughness and charpy tests is validated.
- A general consensus exists in applying the “Master Curve” for WWER RPV materials up to a moderate fluence. Some author pointed out that for high irradiated material is not a reliable technique.
- New reference fracture tougness curves are suggested to be included in Russian Codes.
- In general charpy and fracture toughness shifts are comparable up to fluences near EOL. However, the correlation between T_k and T_o is quite poor showing a great scatter.

Part 3: Embrittlement Trend Curves

Large efforts have been done in past for the collection and creation of new test data especially from surveillance programmes.

While in the case of WWER-440 type RPVs, Standard surveillance programme finished and were fully tested and results evaluated, in many reactors there are several types of Supplementary programmes with the aim to obtain additional data with low lead factor irradiation, data on cladding properties as well as of IA and IAI regimes.

In the case of WWER-1000 reactors large effort is given to the testing and reevaluation/reconstitution of survellance specimens from Standard programmes to obtain more data with higher reliability based on new neutron dosimetry and calculations. These programmes are still running and will be running for a long time to collect necessary information on radiation embrittlement of RPV materials especially for long term irradiation, e.g. at least to design EOL fluence.

5.2 Open Issues

Part1: NDE

Most of the discussed techniques are still under development. Even though that the tests in laboratory have been successful and the results are promising, it is needed further research in order to apply such techniques in real components to assess and monitor ageing. There exists also a need of put in common experiences and results, to validate and qualify NDT to characterise materials ageing.

Positron annihilation, neutron diffraction, etc. are well known techniques with reduced portability, and can be better developed as reference methods for precise laboratory measurements.

The possibility to combine techniques needs to be further studied for effectiveness.

The qualification of the NDT techniques needs to be undertaken in order to prove reliability and capabilities. In addition, the generation of guidelines and standards for application need to be developed at European level in order to promote the use of such techniques to the advantage of citizen safety and European industrial competitiveness.

There is also a need to detect and quantify damage mechanisms that lead to reduce fracture toughness and hence to reduced flaw tolerance due to in-service embrittlement phenomena. A non-destructive determination of the embrittlement state would reduce the amount of material used for destructive tests, and would benefit surveillance programmes having an insufficient amount of available test material. Such NDT capabilities would provide substantial early warning of component deterioration and enable utilities to optimise their operating and maintenance practices, resulting in reduced costs and increased asset utilisation.

The ENIQ methodology could be adopted to provide a formal route to validating NDT techniques for materials degradation monitoring. However, it is clear that techniques for monitoring materials degradation in nuclear power plants are much less well developed and widely used as the standard NDT procedures and systems validated using the ENIQ methodology. The immediate future requirement for industrial validation of such techniques should focus on the generation of the practical evidence of NDT capability, optimised design of NDT systems and procedures, and the preparation of strong technical justification for the claimed capabilities.

Part 2: Master Curve

From the last 10 years a great number of papers are published related to constrain effect on fracture toughness. This constraint effect can be seen as a specimen size effect, specimen’s geometry effect and type of loading effect. From the point of view of correlate fracture toughness obtained by different bend specimens, the statistical size correction proposed by the “Master Curve” is valid, but some controversy exits when a comparison between bend and compact specimens are made. This effect of geometry on fracture toughness can be evaluated analytical and experimentally. A review and comparison of the existing constraint correction models should clarify this issue. The effect of shallow crack and biaxial loading was also treated by several authors on the past and some generalised model is needed.

Regarding the dependence of fracture toughness with the temperature, some issues are still unresolved. The degree of embrittlement up to the Master Curve can be applied and if this degree of embrittlement is relevant for existing operating plants should be made. In this line, the Unified Curve proposed by Margolin seems a useful technique, but the application needs more sophisticated information.

Crack arrest toughness reference curves are not included on the Russian codes and generation of new data is needed. Also to establish a good correlation between crack arrest and initiation more experimental data are needed.

More data from model steels & realistic welds are needed to construct a solid correlation formulation between the transition temperature shift versus yield strength increase due to irradiation.

<u>Part 3: Embrittlement Trend Curves</u>

WWER-440:

- Trend curves with sufficient reliability

WWER-1000:

- Not sufficient large database
- problem with standard surveillance programme, data evaluation, spectral energy effect
- probable effect of thermal ageing
- synergistic effect of Ni, Mn, Si, etc.(?) + ΔTT
- few data for EOL fluence and Plex
- „LATE BLOOMING EFFECT“ for high F, high Ni (thresholds)

Samples:

- Only surface samples, no information about properties through thickness
- Mostly unknown initial properties
- Correlation between standard charpy and subsized specimens – absolute values
- Information about current state, no shift
- Fracture toughness tests – MASTER CURVE (?)

Trepans from decommissioned RPVs:

- Mostly unknown initial properties
- Information about current state, no shift
- Found some nonhomogeneity in chemical composition (distribution of Cu and P)

5.3 Reviewed papers and summaries

<u>Part 1: NDE</u>

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HASCIK J., SLUGEN V, LIPKA J., KUPCA L., HINCA R., TOTH I., GRÖNE R., UVACIK P., "Progress in investigation of WWER-440 reactor pressure vessel steel by gamma and mossbauer spectroscopy",

LIPKA J., HAŠČÍK J., GRÖNE R., SLUGEŇ V., VITAZEK R., HINCA R., TOTH I., KUPČA L.," Some aspects os experimental investigation of the RPV material properties", Proceedings of the International Conference on Nuclear Option, Opatija, Croatia, 1996

SLUGEŇ V., HAŠČÍK J., LIPKA J., KUPČA L., MIGLIERINI M., TOTH I., GRÖNE R, VITAZEK K., "Mossbauer and electron-positron annihilation study on nuclear reactor steel", Proceedings of the Annual meeting on nuclear technology (1995)

MILLER M.K.., RUSSELL K.F., KOCIK J., KEILOVA E., "Atom probe tomography of 15Kh2MFA Cr-Mo-V steel surveillance specimens", Micron 32 (2001) 749±755

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SLUGEŇ V., "Defects Investigation in Neutron Irradiated Reactor Steels by Positron Annihialtion, Transsactions" of the 17th international conference on Structural Mechanics in Reactor Technology (SMIRT 17), Prague, Czech Republic, August 17-22, 2003

The assessment of material ageing requires normally extensive destructive testing by using different kind of testing and samples. A non-destructive determination of the embrittlement state would extend the usefulness of the surveillance material by reducing the material used for destructive studies and would benefit reactor pressure vessel surveillance programmes having an insufficient amount of available test material, and ultimately allowing tests to be performed directly on the component to evaluate. Such non-destructive techniques (NDT) capabilities would provide substantial early warning of component deterioration and enable utilities to optimise their operating and maintenance practices, resulting in reduced costs and increased asset utilisation.

The collection of papers reviewed includes also around 30 papers related to the use of non-destructive techniques to qualitative and quantitative measure radiation embrittlement and hardening, and thermal ageing in some cases. The NDT measurements are correlated with the mechanical properties of the pressure vessel steels used in the investigations. The following techniques are considered by the authors:

- Positron annihilation (PAS)
- Mössbaouer spectroscopy (MS)
- Seebeck and Thomson effects (STEAM)
- Magnetic methods
- Kinetic hardness method
- Transmission Electron Microscopy (TEM)
- Atom probe tomography
- Internal friction
- Small-angle neutron scattering (SANS)

Several papers combine different techniques since the interpretation of results using a single technique is neither easy nor straightforward in case of study complex systems like steels. Due to this fact, the combination of various techniques is very fruitful for better understanding of all the mechanisms of microstructural changes affected by irradiation. In particular, Positron Annihilation and Mössbaouer Spectroscopy seem to be powerful combination mainly with Transmission Electron Microscopy.

Most of the discussed techniques are still under development. Even though the tests in laboratory have been successful and the results are promising, further research is needed in order to apply such techniques to assess and monitor ageing in real components. There exists also a need of putting in common experiences and results, to validate and qualify NDT to characterise materials ageing.

Part 2: Master Curve

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[4] *HOLZMANN M, DLOUHY I,. BRUMOVSKY M., (1999) "Measurement of fracture toughness transition behaviour of Cr-Ni-Mo-V pressure vessel steel using pre-cracked Charpy specimens", International Journal of Pressure Vessels and Piping Volume 76, Issue 9, , August 1999, Pages 591-598.*

[5] *BALART M.J., KNOTT J.F.,(2006) " Effects of geometry and flow properties on the fracture toughness of a C-Mn reactor pressure vessel steel in the lower shelf region", International Journal of Pressure Vessels and Piping Volume 83, Issue 3, , March 2006, Pages 205-215.*

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[8] *BORIS MARGOLIN, EUGENE RIVKIN, GEORGE KARZOV, VICTOR KOSTYLEV, ALEXANDER GULENKO, (2003)" New Approaches for Evaluation of Brittle Strength of Reactor Pressure Vessels Transactions" of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17) Prague, Czech Republic, August 17 –22, 2003.*

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[14] *BORIS MARGOLIN, ALEXANDER GULENKO AND VICTORIA SHVETSOVA, (2003)" Fracture Toughness Prediction for RPV Steels with Various Degree of Embrittlement Transactions" of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17) Prague, Czech Republic, August 17 –22, 2003.*

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[16] *VIEHRIG H-W., MURASOV M, MARC SCIBETTA, WALLIN K., (2006)" Application of the Master Curve Approach on WWER-440 Reactor Pressure Vessel Steel of the Nuclear Power Plant Greifswald Unit 8", 1st Biennial Conference on Through Life Toughness Prediction in Reactor Steels Hotel Helios, Hévíz, Hungary, 6th to 8th February 2006 AMES.*

[17] *VALO, M., WALLIN, K., LUCON, E., KYTKA, M., BRUMOVSKY, M., ACOSTA, B., DEBARBERIS, L., KOHOPÄÄ, J., GILLEMOT, F., & HORVATH, M. (2005). "The Euratom 5th Framework Programme Project FRAME (fracture mechanics based embrittlement); description of the project and first results". Nuclear Engineering and Design 235(2-4), 445-455.*

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PWR materials"1st Biennial Conference on Through Life Toughness Prediction in Reactor Steels Hotel Helios, Hévíz, Hungary, 6th to 8th February 2006 AMES.

The papers can be subdivided into three main areas:

- Specimen size/constraint issues
- Fracture toughness reference curves
- Charpy/toughness/yield shifts

Three papers from VTT investigators deal with the use of small specimens to characterize impact and fracture toughness curves. Two approaches for correlating the Charpy impact temperatures determined with different size specimens is applied by Valo **[1]**. The better correlation is found with the VTT formula T(KLST)-T(ISO-V)-35ºC.

Regarding fracture toughness tests, VTT researchers point out that there are no size effect on T_0 value when bend specimens are compared **[1-3]**, see figure 5.1. The size of the bend specimens range from 3x4 up to 10x10. The suggested optimum geometry is 5x5 mm^2 **[3]**. On the contrary, Holzmann **[4]** pointed out that to determine T_0 the pre-cracked Charpy specimen are valid but a constraint correction is needed to produce fracture toughness data equivalent to 1T SENB specimens. The constraint correction procedure suggested is based on the toughness scaling model proposed by Dodds and co-workers.

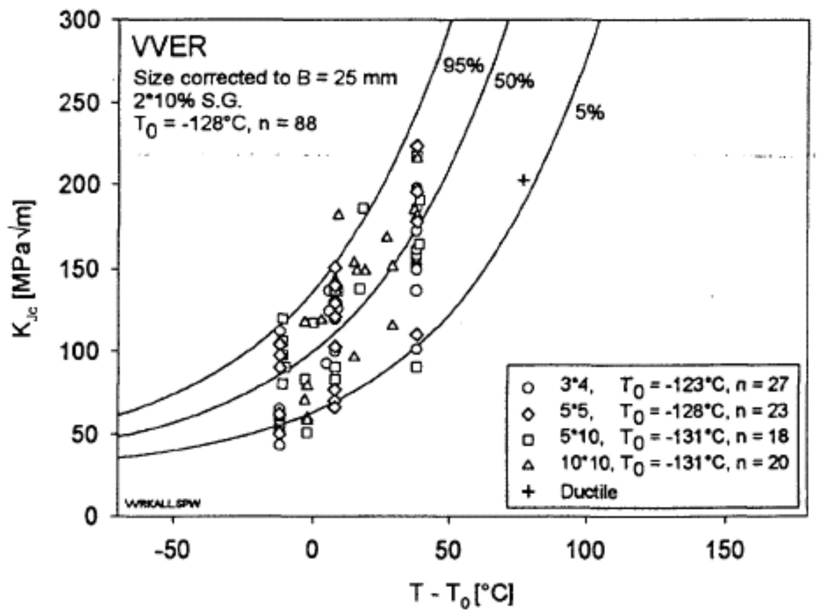


Figure 5.1: Effect of specimen size on T_0 [2-3]

An analysis of constraint loss using T-stress is presented by Wallin [2-3] and a bias value, which is defined as the difference between T_0 value obtained by testing bend and compact specimens, of 8ºC is suggested. A comparison between bend and compact specimens is also presented by Balart **[5]**.This paper describes the results of a fracture toughness test programme carried out on a C-Mn steel plate for two different specimen geometries (10 mm thickness precracked Charpy and 25 mm thickness compact tension) in the lower shelf region of the temperature/fracture-toughness curve. Although the Master Curve methodology predicts no size effects on the lower shelf, size effects were observed as a combination of specimen geometry and constraint effect.

Bolton et al **[6]** presents the results of nearly 400 toughness tests of pre-cracked Charpy specimens removed from four submerged-arc welds in a decommissioned Magnox RPV at Trawsfynydd. The judgement made by the authors is that the start-of life properties of side-grooved Charpy geometry specimens are the same as those of plane-sided 25 mm

Compact Tension specimens. The authors pointed out that the lower bound toughness is independent of specimen size. This prediction is also demonstrated by the ESIS European database.

Two papers deals to the effect of biaxial loading on fracture toughness **[7, 8]**. Both of the papers reach a similar conclusion that is the biaxial effect is only pronounced in situation when the small scale yielding condition is loosed. Margolin [8] also proposes some formulas which allow taking into account the shallow crack effect.

Other group of papers deals with fracture toughness reference curves. Timofeev et al presented a fracture toughness database that include K_{IC} of 10GN2MFA steel and its weldments **[9]**, 15X2MFA steel, 15X2MFAA steel as well as their welds **[10, 11]** and K_{Ia} values for 15X2MFA and 15X2NMFA steels and their welds, produced by submerged arc welding (SAW) in the as-produced condition (BOL) and after embrittlement simulated by heat treatment, corresponding to the material state at the end of life (EOL) **[12]**. In the case of K_{IC} data, the obtained information is a reliable basis for the construction of reference fracture toughness temperature dependences for these materials. In the case of crack arrest data the experimental data accumulated allow construction of an envelope for WWER-440 RPV manufacture. Regarding crack arrest toughness, another paper of Bass included a new dependence of K_{Ia} with the temperature as a result of the analysis of a database **[13]**.

Two papers from Margolin present the so call Prometey model for fracture toughness. A comparison with the Master Curve is made for a WWER-1000 material in the as received, thermally aged and irradiated states **[14]**, and the main conclusion is that the slope of the $K_{IC}(T)$ curve for the embrittled steel is less that the predicted by the Master Curve. If the $K_{IC}(T)$ dependence is governed by the $\sigma_Y(T)$ dependence, as it is suggested by the author, two main conclusions are made: first that as the degree of embrittlement increases, the $K_{IC}(T)$ curve shape changes and, secondly, that for high embrittlement, K_{IC} is independent on temperature practically. The author also proposes a variation of the Weibull exponent b with the degree of embrittlement. In the other paper related to the Prometey approach a correction for ductile crack growth is presented **[15]**, that is for non irradiated material the correction due to crack growth is not necessary. The correction for ductile crack growth depends on degree of material embrittlement.

Finally, the application of the "Master Curve for WWER-440 RPV steels is presented by Viehrig **[16].**K_{Jc} values were measured on Charpy size SE(B) specimens of two WWER-440 base metal shells of the RPV of Greifswald Unit 8. The results of the standard Master Curve analyses show that significantly more than 5% of the K_{Jc} values lie below the fracture toughness curve for 5% fracture probability. The application of the SINTAP lower tail analysis did not also lead to reference temperatures, whose indexed curve for 20% fracture probability does not envelop 80% of the values. The optimum method was found to be the random MML approach.

The last group of papers reviewed area related to Charpy/toughness/yield shifts. FRAME project results are presented by Valo **[17]**. The aim of FRAME was to start systematic development of Master Curve based embrittlement monitoring. First estimate for a model, which describes the dependence of embrittlement on the copper, phosphorus and nickel contents of the material, is given.

Comparison of transition temperature shifts determined from impact Charpy V-notch toughness as well as from static and dynamic fracture toughness tests are given by several papers. Valo **[18]** concluded that the fluence dependence of the measured transition temperature shift in a Charpy-V impact test and in a dynamic fracture toughness test are equal for seven series A533B1 steel. Previous experiments with the same material demonstrated that static fracture toughness shift was higher than the shift in the Charpy-V impact test. On the other hand, Brumovsky **[19]** said that the higher is transition temperature shift, the higher is the difference between static and dynamic transition temperature shift, see figure 5.2. These differences can be neglected up to WWER-440 EOL design neutron fluence.

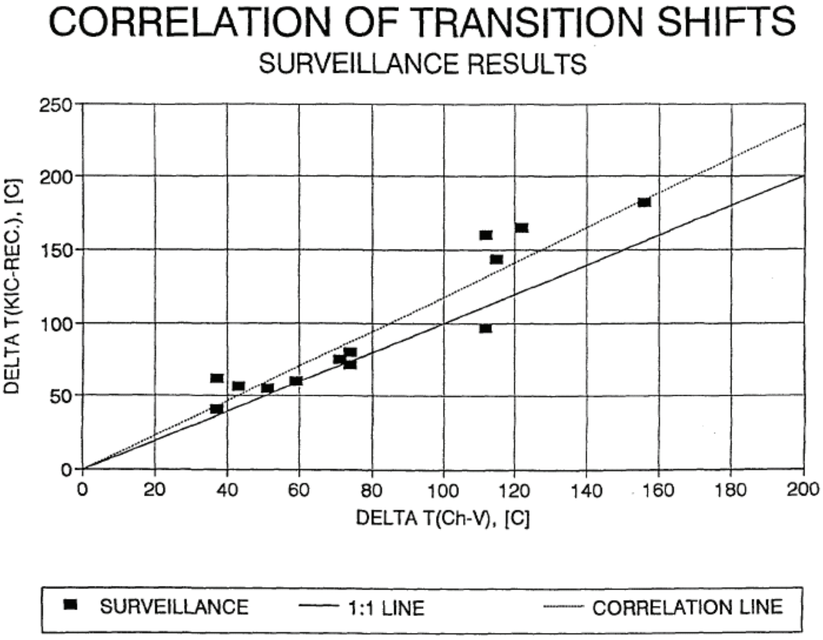


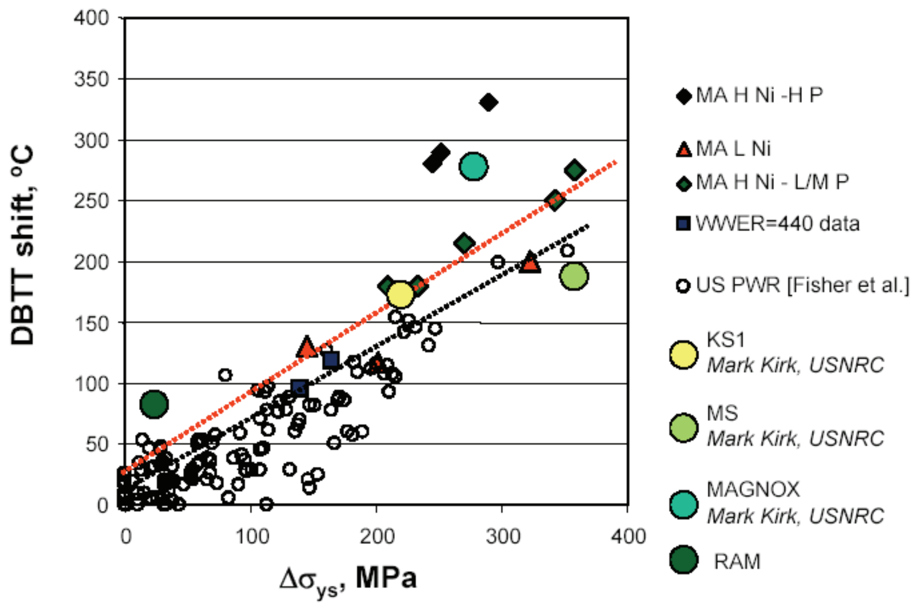
Figure 5.2: Correlation between static fracture toughness and Charpy notch impact toughness transition temperature shifts for surveillance specimens [19]

Nanstad et al **[20]** tried to correlate the recovery by annealing of USE and ΔT41J, and it seem that much greater recoveries occur for USE that for ΔT_{41J}. They also compare the annealing effects on fracture toughness with those on CVN toughness, and they conclude that the post-annealed recovery of the fracture toughness transition temperature appears to be somewhat less that that for the Charpy impact toughness. More data are needed to confirm that tendency.

Golovanov **[21]**, compare the techniques included in the valid national codes of Russia and France on assessment of irradiation embrittlement. The objective of the work was to study the effects of specimens cut-off technique (including the orientation of specimen longitudinal axis, notch orientation, place of specimen cut-off, etc) and determination technique of T_c (critical brittle temperature) and ΔT_F (shift in critical brittle temperature after irradiation). The differences area noticed when determined the properties of base metal. On the other hand the differences in calculating the shifts are small.

A general review of the effects of neutron irradiation and the methodologies of assessment these effect for LWR reactor (in the USA and the former USSR) are presented by Server et al **[22]**.The transition temperature shift can be correlated to change in yield strength: $\Delta T = C\Delta\sigma_y$. for US reactors the C value is 0.65ºC/MPa for welds and 0.5ºC/MPa for plates. Suggestions of fluence rate effect have been made, but surveillance conditions in the USA show limited rate effects. In the former USSR, while the rate differences may reach an order of magnitude, investigators believe temperature variations can be so overwhelming as to cover any flux effect.

Another correlation between the transition temperature shift and the change in yield strength is presented by Debarberis **[23]**, see figure 5.3. FRAME tensile results on model alloys are presented and a correlation with DBTT shift is suggested. For alloys with low Ni & high Ni with low P similarities to PWR/WWER materials where found. For high Ni & high P alloys the behaviour is different: higher correlation (moderate hardness increase), and no improvement by normalisation to initial σ. More data from model steels & realistic welds are needed.



<u>Figure 5.3:</u> correlation between the transition temperature shift and the change in yield strength [23]

The current provisions for determination of the temperature shift of the reference static fracture toughness curve due to irradiation of reactor pressure vessel steels are based on the assumption that they are the same as for the Charpy shift. Within the activities of the Heavy-Section Steel Irradiation Program (ORNL) this assumption was evaluated relative to data reported in open literature [Sokolov 2000a]. A database was assembled from information reported in the literature regarding radiation-induced shifts of static fracture toughness and Charpy impact toughness. Application of the master curve concept was used to determine shifts of fracture toughness curves; the hyperbolic tangent function was used to fit Charpy data. For weld metals, on average, the Charpy transition temperature shift at 41J is the same as the shift of fracture toughness, with 95 % confidence intervals of about ± 26ºC but for base metals, on average, the fracture toughness shift is 16% greater than the Charpy 41J temperature shift, with 95% confidence intervals of about ±36ºC. In other paper from the same authors [Sokolov 2000b], they notice that fracture toughness shifts are slightly higher that the Charpy shifts for Plate 02 and Weld 73W. Contrarily, fracture toughness shifts for WWER-440 Weld 502 were slightly less that Charpy shifts. The observed differences are within the scatter reported in the literature. It is interesting to point out that the correlation between the radiation induced hardening and transition temperature shifts for the WWER-1000 and WWER-440 welds is in good agreement with the large Power Reactor Embrittlement Data Base (PR-EDB), see figure 5.4.

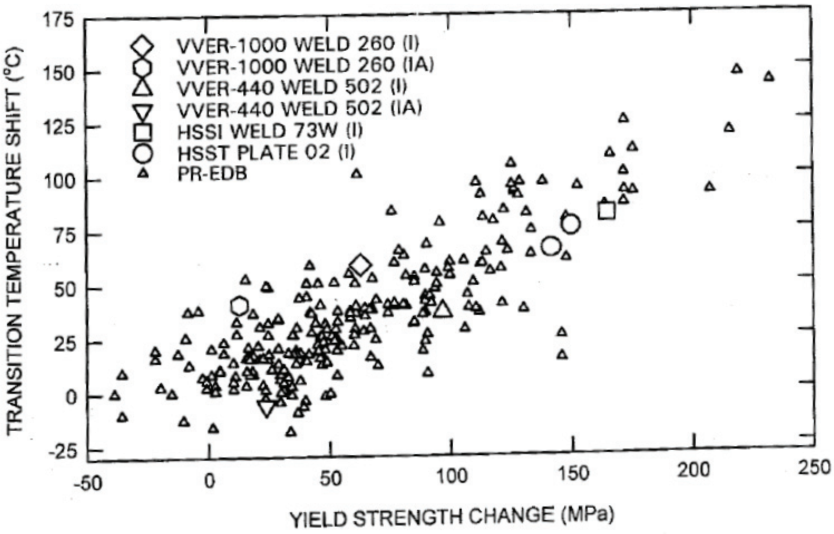


Figure 5.4: Transition temperature shift versus change in yield strength [Sokolov 2000b]

Other authors also reported much higher fracture toughness shift than Charpy shifts, but attention should be paid on these conclusions because there are many factors to be taking into account as the differences in flux between Charpy specimens and fracture toughness specimens [Ozsvald 1999] and pre-cracking procedures [Ahlstrand 1993].

Some author tried to correlate the fracture toughness transition temperature T_0 with the T_k value determined according the Russian code [English 1999, Brumovsky 2002]. In general the relation is quite poor, see figure 5.5 and the big scatter means that, for some materials T_k may be strongly conservative, while for others, the conservatism is not so large. For Western type RPV steels a similar lineal correlation can also be found between T_o and T_{41J}, but the scatter is less than for WWER materials [Sokolov 2000a].

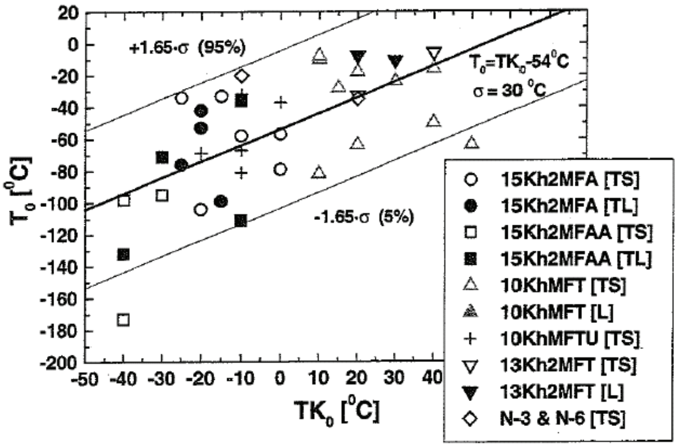


Figure 5.5: T_0 value obtained for WWER-440 materials as a function of Tk_0 [English 1999]

Part 3: Embrittlement Trend Curves

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All papers can be divided into the following groups:

- WWER RPV material characterization
- Material irradiation embrittlement assessment
- Material microstructure
- RPV surveillance programmes
- Comparison of fracture toughness tests
- Sampling from RPVs and Correlation between standard and subsize impact specimens
- RPV integrity assessment

And additionally

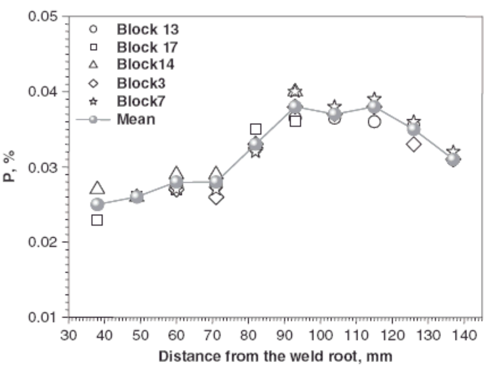
- Trend curves

<u>WWER RPV material characterization</u>

Chernobaeva A.et al. [1] describes work carried out for a proper characterisation of initial properties of a material selected for the project "PRIMAVERA" – this project aims to develop a physically validated prediction model for transition temperature shift dependence of WWER-440 weld metals vs. neutron fluence after reactor pressure vessel annealing (re-embrittlement).

Work was carried out in two directions – in study of distribution of chemical elements (C, Si, Cr, Ni, Mo, Mn, Cu, P, S, V) and of mechanical properties (T_k, Rp0.2) together with the analysis of the minimum required number of specimens for reliable determination of transition temperature value in base metal and weld metal.

Chemical analysis found relatively homogenous content of all elements through the weld, while P content varied between 0.022 to 0.042 mass % as it is seen in the Figure below



The weld metal mechanical properties evaluated by transition temperature values using standard Charpy specimens are very homogenous and not correlated to the chemical composition or to the main cutting directions. The only factor which has a measurable influence on the transition temperature as well as on yield strength is the distance from the fusion boundary to the base metal while the influence of other elements is insignificant. A transition temperature increase of ~ 9 to 12 °C is observed in the zone in the vicinity of the fusion boundary (at a distance of ~ 9 to 10 mm from the fusion line). The maximum variation in the transition temperature over the circumferential direction is 8 °C.

Regarding minimum number of specimens to tested to obtain a reliable transition temperature, it was found that fourteen specimens are sufficient for a correct evaluation of weld metal transition temperature while more tests are required for base metal (satisfactory notch quality and best practice testing must be ensured).

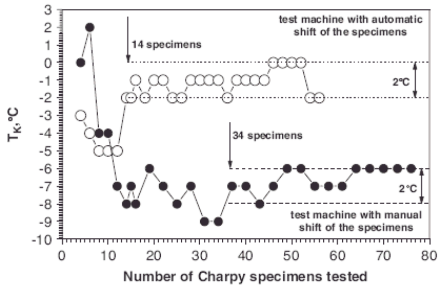


Fig. 9. T_K dependence on number of specimens tested, effect of the testing procedure.

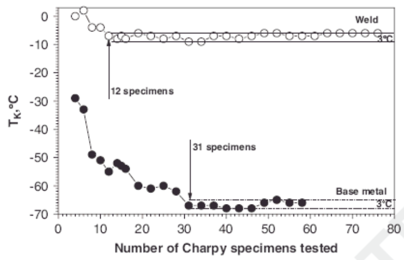


Fig. 10. T_K dependence on number of specimens tested, for weld and base metals.

Vodenicharov and Kamenova [2] studied surveillance specimens from Kozloduy WWER-1000 reactor pressure vessel weld material with nickel content 1.70 mass %. Notch impact tests were performed with instrumented pendulum thus an additional analysis of force-displacement diagram permits to gain additional information on the influence of neutron irradiation on crack initiation, propagation and crack arrest.

Testing the first two sets of specimens showed that trend line of weld metal transition temperature is lower than predicted by Russian Code and also lower than some other Russian plants – these results were obtained for neutron fluence up to $1.2x10^{23}$ m^{-2} (En > 0.5 MeV) that represents approx. 25 % of design EOL fluence.

Analysis of force-displacement diagrams showed that general yield force and maximum force are increasing by irradiation while temperature dependencies of all characteristics temperatures (T_I – ductile crack initiation, T_N – ductility temperature and T_0 – temperature of USE onset).

Vasilchenko et al. [3] studied the effect of side grooving and small specimen sizes on fracture toughness values of weld metal 12-Kh2N2MAA for WWER-1000 and WWER-1500 units. They discussed also the effect of pre-cracking load and determination of J-integral values.

Tests confirmed that use of small size specimens (0.5 CT) can result in valid fracture toughness values comparable with test results of 2 CT and 4 CT specimens when size correction is applied. In the same time, conception of “Master curve” was confirmed for this type of material. Morever, experimental results based on “Master curve” showed higher values of fracture toughness than in accordance with the “Unified Curve” corrected for the vessel wall thickness of 150 mm.

Use of side grooves can be applied for determination of fracture toughness in ductile region but in brittle region their use is incorrect as some decreased values can be obtained.

Authors also recommended a re-evaluation of the Russian standard GOST 25.506-85 for fracture toughness testing taking into account experimental results and other international standards (ASTM, ESIS).

Grigoryev et al. [4] performed analysis of properties of WWER-1000 RPVs of Russia manufacturing to be able to calculate RPV failure probability. The following properties were collected into databases and analysed: chemical analysis (Cu, P content), mechanical properties (yield strength, ultimate tensile strength, elongation and reduction of area), transition temperature, fracture toughness (including "Master curve" approach) and defect sizes, all in base as well as in weld metals.

The following distributions were tested: normal, lognormal, Weibull, exponential and gamma using different criteria. As a result, the following distributions were found as reliable:

RPV part	Characteristics	Distribution law
Beltline region base metals	Ni content	Lognormal
	P content	Weibull
	Cu content	Normal
	Transition temperature, T_{k0}	Not found
Beltline region weld metals	Ni content	Weibull
	P content	Lognormal
	Cu content	Lognormal
	Transition temperature, T_{k0}	Not found

Lognormal distribution of defects is taken on the bases of generalized results from analysis of defect sizes in WWER-440 RPVs.

For all other properties, it is recommended to use normal distribution.

Material irradiation embrittlement assessment

Amaev et al. [5] gives results of the complex investigation of irradiation embrittlement and hardening of WWER-440 RPV materials carried out in Russia. Irradiation was performed in surveillance channels of operating WWER type reactors at temperatures of 250 and 270 °C up to fluences of $7x10^{24}$ m^{-2} (En > 0.5 MeV). Materials of commercial RPVs as well as other materials were irradiated and tested to obtained trend curves for different chemical composition of materials. Regularities in the influence of copper and phosphorus contents as well as low and high neutron flux have been investigated.

Main results can be summarized as follows:

It has been found that in irradiation of specimens of bith the base metal, 15Kh2MFA steel, and the weld metal for along time up to a neutron fluence of $7x10^{24}$ m^{-2} (En > 0.5 MeV) (which is more than three times that of the design EOL fluence of WWER-440 RPV), saturation of the radiation embrittlement does not occur.

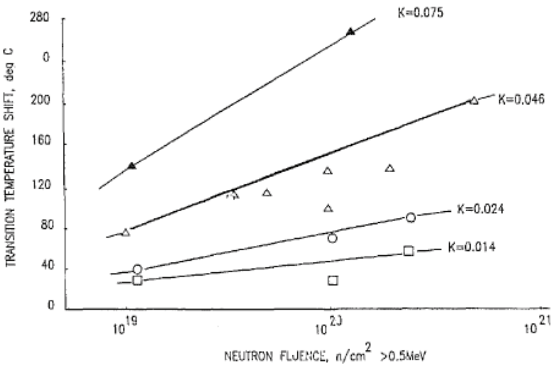


FIG. 8—*Dependence of shift in transition temperature at an irradiation temperature of 270°C as a function of fast neutron fluence for base and weld metal as characterized by different values of the "chemical factor"* (K). K = *%P + 0.1%Cu.*

The regularities in the influence of Cu and P on the radiation embrittlement have been investigated. Using a correlation-regression analysis, the dependence of the shift in the transition temperature of the base metal and weld metal on the P and Cu contents in the steel have been found. It is shown that the radiation embrittlement of the weld metal is the result of at least four different processes. Three of these processes are due to the individual contributions of P, Cu and P-Cu interaction. The fourth process is probably connected with the accumulation of radiation defects and possibly other impurities in the lattice. However, for practical reasons, such equation would not be very convenient. As content of Cu and P are the main contributors to the radiation embrittlement in these materials, some simplifications were proposed: unification of power exponent of neutron fluence to 1/3 and including single effects of Cu and P contents lead to the following formula:

$$\Delta T_k^{270\,°C} = [609\ (\ \%P + 0.1\ \%\ Cu) - 2\ ].\ (F/10^{22})^{1/3},\ °C$$

Good illustration of the applicability of this formula is given in the following graph:

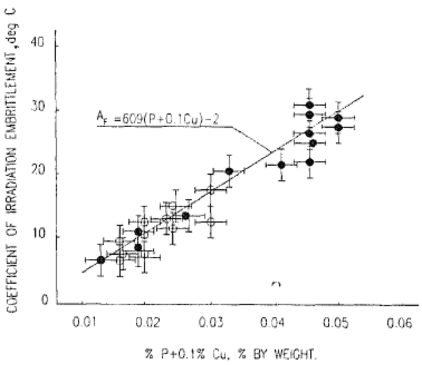


FIG. 5—*Coefficient of radiation embrittlement* A_F *as a function of P and Cu content in base metal* (○) *and its weld metal* (●) *at an irradiation temperature of 270°C.*

Nevertheless, it was also pointed out that, at neutron fluences higher than $4x10^{24}$ m^{-2} (En > 0.5 MeV), a noticeable increase in the radiation hardening and embrittlement occurs, which would appear to be due to the processes of radiation damage of the steel determined directly by radiation defects (i.e. formation of radiation defect clusters, dislocation loops, or small voids).

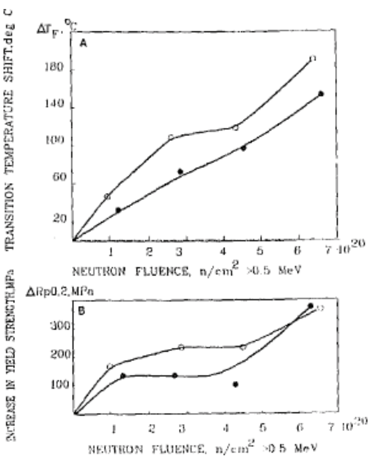


FIG. 9—*Increase in transition temperature and yield strength of sets of surveillance specimens of base (○) and weld metal (●), (Kol'skaya NPP unit 3).*

Effect of neutron flux (flux rate) was also detected, even though results were obtained only for relatively small fluences, i.e. in the range of 1 to $5x10^{24}$ m^{-2}(En > 0.5 MeV): lower flux ($4x10^{15}$ $m^{-2}.s^{-1}$) results in higher radiation embrittlement (characterized by a coefficient A_F) than for a larger flux ($4x10^{16}$ $m^{-2}.s^{-1}$). In this fluence range, value of coefficient A_F for lower flux is decreasing with the fluence increase but still higher than for a higher flux.

Debarberis et al. [6] proposed a semi-mechanistic model of irradiation embrittlement for WWER-440 RPV materials based on the statistical analysis of available data and physical description of potential damaging mechanisms. Three major damage contributions considered are:

- direct matrix damage due to neutron bombardment is assumed to be a square root dependence on fluence¨for a given material and a given temperature,
- Cu-rich precipitates – copper together with other elements lead tio precipitation mechanism of nanoparticles also inducing matrix hardening and embrittlement. Such precipitation mechanisms continues until saturation depending on the available amount of precipitants, Cu concentration in particular
- Other segregates can be also formed both proportionally to the matrix damage and attracted into the Cu precipitates. Diffusion of segregates plays also a role. A simple model was proposed to describe this additionally contribution based on "logistic" shape type of function describing a process of gradual increase then a rapid saturation
- The effect of the various embrittlement mechabnisms was considered to be additive to the total damage expressed in terms of transition temperature shift:

The proposed model is:

$$\Delta T_k^{270\,°C} = a(Cu)\,[a - \exp(-bF)] + c(P) \text{ x } [0.5 + 0.5 \tanh (F - F_0)/d] + kF^n, \quad °C$$

This model was used forasnalysis of experimental data and fitting the model.

This model has many advantages, mainly:

- it is suited with analysed data of WWER-440 materials mainly after annealing,
- it permits better verification of the hypothesis on relative importance of different mechanisms at the different embrittlement phases, supporting the hypothesis that P is leading re-embrittlement parameter after annealing and that Cu has only a marginal effect,
- it is possible to predict re-embrittlement rate after annealing.

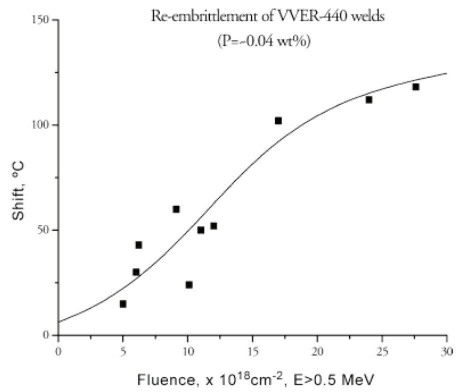


Fig. 3. Dose dependence of transition temperature shift on fluence in the initial stage of re-irradiation.

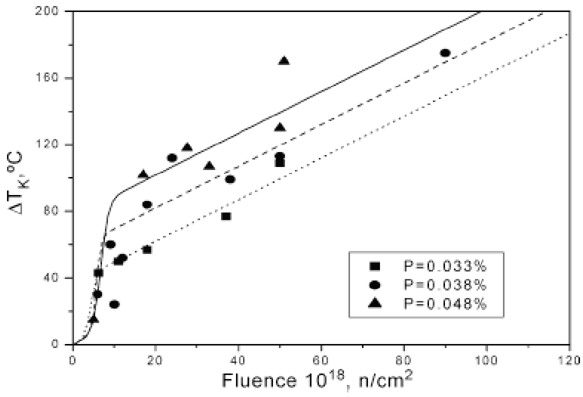


Fig. 4. The results of one variant calculated.

DEBARBERIS L. et al. [7] reported on results from parametric study of irradiation embrittlement of a large set of model alloys and their comparison with behaviour of commercial steels. 32 different model alloys with different content of Ni, P and Cu were irradiated in experimental reactor. Transition temperature shifts were then compared with different predictive formulae for commercial RPV steels for PWR and WWER type RPV materials.

Comparison of these results with formulae showed that such model alloys, even if the results of their irradiation embrittlement are not immediately transferable to commercial RPV steels, are very suitable for study and the results obtained can be the basis to understand mechanisms and better focus irradiation experiments on commercial RPV steels.

For model alloys with very low content of nickel, chemistry factors of existing prediction formulae containing copper and phosphorus well correlate to the obtained transition temperature shifts; the Russian Guide formula in particular.

The presence of nickel always shows a clear negative influence in increasing transition temperature shifts. Modification of the Russian Guide chemistry factor AF could be obtained in relatively simple manner in order to take into account nickel content:

$$CF = A_F = [P + 0.07\ Cu]x[Ni - 0.1]^a$$

where a is a constant parameter.

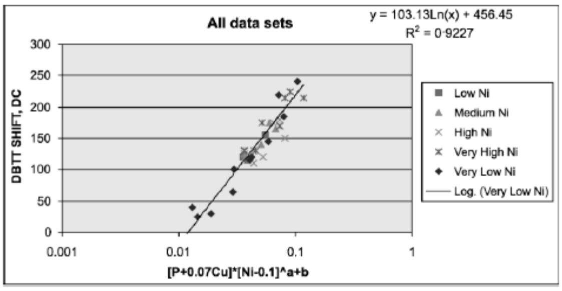


Fig. 4. Correlation between modified CF and DBTT.

The effect of nickel, for low and middle copper alloys, seems to become even more marked from nickel contents above 1.2 mass %; this is in agreement with data obtained on WWER commercial steels.

The presented data support the models of embrittlement based on the fact that nickel increases the phosphorus effects like diffusion and segregation.

Transition temperature shifts of low copper data are not separated by those obtained for very low copper; the effect of copper is visible from a threshold level of approximately 0.1 mass %. This is also in agreement with observations done on commercial steels.

So many similitude in behaviour are shown by the model alloy study results when compared to available commercial steel results and knowledge. Such similitude is encouraging the continuation of the study on model alloys, especially when slight effects of such parameters should have to be studied.

DEBARBERIS L. et al. [8] perform comparison of radiation embrittlement of PWR and WWER type of RPV steels. Results from tested welds of both types of steels were made using semi-mechanistic model of radiation damage with further analysis of these data. Comparison of characterical constants of the model serves as a basis. The following model was applied:

$$\mathrm{DBTT}_{\mathrm{shift}} = a \cdot \Phi^{0.5} + b \cdot \left[1 - \mathrm{e}^{-\Phi/\Phi_{\mathrm{sat}}}\right] + \frac{c}{2} \cdot \left[1 + \tanh\left(\frac{\Phi - \Phi_{\mathrm{start}}}{d}\right) - c_0\right]$$

While for the analysis, six parameters are required for the proposed model: a, b, Φ_{sat}, c, Φ_{start}, and d. Parameter b depends mainly on the Cu kontent and c on the P kontent ($b = b_1 . (Cu-0.05)$;, $c = c_1.P$).

The following values of parameters were obtained:

Summary of semi-mechanistic model parameters

Parameter	Modelling	PWR welds	WWER-440 welds	Difference	Remarks
a	Matrix damage rate	5.5	6.8	~20%	Higher in WWER due to lower operational temperature
b_1	Effect of copper at saturation	400	480	~17%	Slightly higher for WWER-440 due to the lower operational temperature
c_1	Effect of phosphorus at saturation	2500	3000	~17%	Idem
$\Phi_{\mathrm{sat}}\times10^{18}$	Start of saturation of precipitation	8	55	~85%	Much higher for WWER-440 due to lower temperature and different material structure
$\Phi_{\mathrm{start}}\times10^{18}$	Start of saturation of segregation	9	100	~91	Much higher for WWER-440 due to lower temperature and different material structure

In spite of the encountered difficulties to compare both PWR and WWER weld types some important conclusions were produced:

- Matrix damage rates are basically equal for both weld types, the observed differences can be explained by different irradiation temperatures. The same matrix damage can be used for both families of materials.
- Precipitation and segregation coefficients are quite similar for different welds.
- Fluences at which segregation and precipitation effects start to saturate are higher for WWER-440 than for PWR welds due to the lower irradiation temperature.

In principle, if the different irradiation temperature is taking into account, radiation stability of WWER-440 materials is in the higher fluence range greater than foreseen for PWR.

VODENICHAROV S. et al. [9] describes activities connected with the assessment of neutron-induced embrittlement of materials for 4 units of WWER-440 in NPP Kozloduy. The paper describes main parameters for such assessment – chemical composition of weldments, calculation of neutron fluences, changes in construction of individual units, annealing of RPVs, in-service inspection.

Based on calculations of radiation embrittlement and/or from testing samples cut off from some RPVs, transition temperatures and their trends were determined. Lateral shifts was applied for re-embrittlement rate after annealing.

VODENICHAROV S. et al. [10] gives information on all six RPVs in NPP Kozloduy but detailes are given only for weld of Unit 1(WWER-440) and partially of Unit 5 (WWER-1000). Radiation embrittlement is defined by the shift of transition temperature from notch impact tests. Additionally, in Unit 1 weld (15Kh2MFA type steels) analysis of the distribution of phosphorus through the thickness of samples cut from the inner surface of the weld showed that P content was 0.046 %, but some scatter of data were also found.

Nevertheless, Auger spectroscopies showed that P enriched zones are rarely registered on fracture areas containing small size facets. The shape of P depth distribution curve in these zones is more representative for the case of phase precipitation than for a case of P segregation on grain boundaries.

In the same time, no typical intercrystalline fracture developed in Auger specimens impactly fracture at -120 °C.

Microstructural study of the material from Unit 5 (15Kh2NMFA type steel) showed that no significant changes were found after irradiation up to $5x10^{18}$ cm^{-2} except for a small number of defects of dislocation loops type and some increase in rounded precipitates number in grain volume.

VALO M. et al. [11] describes the material research programmes as a support for annealing of the Unit 1 in Loviisa NPP. For the programmes, mostly archive materials from the Standard surveillance programmes were used after reconstitution of the broken specimens. Additionally, some tailored weld material was used. Besides standard Charpy V-notch specimens, sub-size KLST type specimens were also irradiated, annealed and tested. Several irradiation-annealing-irradiation regimes were applied, up to three times. Study of the effect of different annealing temperatures and times was also performed – finally, standard regime 475 °C-100 h was confirmed.

The annealing studies performed allow the following conclusions:

- toughness properties of Loviisa RPV materials recover well during annealing, its residual shift does not depend on the annealing parameter within a large range of parameters,

- no evidence of temper embrittlement was found in the annealing studies,
- the KLST specimens are capable of monitoring transition temperature shifts in irradiation and annealing for weld metal – their shifts are comparable with shifts of Charpy V-notch specimens
- this relation does not work well in base metals,
- good agreement was also found in correlation between T_{KIa} and T_{F4kN} transition temperatures.

Material microstructure

MILLER M.K. et al. [12] used atom probe analysis for study of microstructure of low-copper WWER-440 surveillance samples after large neutron fluences.

These experimental results clearly demonstrate that low copper pressure vessel steels are susceptible to embrittlement due to neutron irradiation at high fluences. In addition, they suggest that post-irradiation annealing treatments may not remove the susceptibility of the steel to re-embrittlement during further neutron irradiation. Therefore, the small quantity of copper remaining in the matrix would produce little re-embrittlement on subsequent irradiation.In low copper materials, annealing treatments are likely to redistribute the manganese and silicon and recover of the mechanical properties. In both type of materials, the manganese and silicon can segregate to dislocations and produce embrittlement on subsequent irradiation.

The observed changes in mechanical properties correlate with the presence of manganese-, silicon-, copper-, phosphorus- and carbon-decorated dislocations and other features in the matrix of the neutron-irradiated base and weld materials. The results also indicate that there is an additional mechanism of embrittlement during neutron irradiation that manifests itself at high fluences.

B.PAREIGE et al. [13] applied three-dimensional atom probe to study the solute distribution and the early stage of precipitation in metallic alloys. In the paper, comparison of microstructural changes in steels and model alloys was made after irradiation by neutrons, electrons and ions.

In the WWER RPV steels, in IAI regime, no neutron-induced Cu-Si-Ni-Mn-P-enriched clusters were observed in the matrix in contrast to the case in the I state. Considering the irradiation condition (additional fluence of 1.5 x $10^{23}m^{-2}$) and results reported in the literature, a number density of 3 x $10^{23}m^{-3}$ of these clusters could have been expected. This indicates that re-irradiation does not appear to promote the formation of new-Cu-Si-Ni-Mn-P complex clusters or a low number density (below $10^{22}m^{-3}$.

The comparison of the three microstructures suggests that from the htree “mechanisms” at the origin of the embrittlement, the formation of copper enriched clusters may be neglected in the re-embrittlement process.

Ion irradiation of Fe-0.1 at.%Cu and Fe-0.3 at.%CU shows that for constant ion irradiation conditions, the characteristics of the copper precipitates (size and composition) are similar. In the case of enhanced-precipitation (nucleation and growth) of copper atoms under these irradiation conditions a factor 2 on the v alue of the size of the particles could have been expected between these two different alloys. This suggests that the production of point defects due to displacement cascades influence the evaolution of the microstructures in Fe-Cu alloys. These irradiation conditions (low Fe-ion energy, low primary-knocked-atom energy) and the atomic scale description of the microstructures may be directly compared to numerical simulation.

Before electron irradiation of JRQ steel, the solute atoms in the ferritic matrix were homogenously distributed. In irradiated specimens, a decrease of the Cu, Ni and Mn contents was observed and correlated with the presence of tiny clusters as well as phosphorus segregation. The composition of these Cu clusters was: 22±7 at.% Cu, 10±5 % Ni, 3±3 % Mn (balance is Fe). Comparison of behaviour of JRQ steel with Fe-Cu alloys may indicate that enhanced precipitation under electron irradiation takes place when copper content in solid solution is above 0.12 at. %. In addition, a wide range of solute atoms is present in the JRQ steel which may affect the precipitation process and will also change the macro- and microstructure of the materials.

Further study is necessary.

BONCHEV Ts. et al. [14] applied atomic spectroscopy for chemical analysis for study of the weld metal of Kozloduy NPP Unit 1 with high content of phosphorus. The method was applied for in-situ measurements directly on the vessel and compared with results obtained from scraps taken from similar positions.

Detailed process of preparation, training and validation of the methods is described in details. Details of technology of the measurements, like requirements for the heads with measuring device mounted on the Universal Maintenance Cabin, then limitations of the method and complications with measurements, e.g. effect of radiation etc. are also described.

Obtained results are in a good agreement with standard chemical analysis from scraps but it shows that results strongly dependent on the position in the weld, e.g. measurements within upper 2 mm shows large differences:

- surface (after grinding of 0.5 to 0.7 mm) is P=0.100±0.010 %,
- the point for 0.7 mm is P=0.052±0.011 %,
- the point for 1.4 mm is P=0.032 %,
- the point for 2 mm is P=0.034±0.003 %.

Thus, these points were situated in different weld layers – large non-homogeneity of the phosphorus distribution within the weld is the most important conclusion.

<u>RPV surveillance programmes</u>

KUPCA L. [15] describes in detail three different surveillance specimens programmes for Slovak WWER-440/V-213C type reactors:

- Standard surveillance specimen programme (SSSP) for V-2 reactors (Bohunice NPP, Units 3 and 4),
- Extended surveillance specimen programme (ESSP) for the same reactors,
- Modified surveillance specimen programme (MSSP) for NPP Mochovce.

ESSP and MSSP have been designed using experience from operation and testing and deficiencies from the SSSP, operated for 5 years in V-2 reactors. New programmes are characterized by new design of containers (with specimen inserts and neutron monitors and irradiation temperature monitors) and chains (with bellows instead of chain rings).

The paper contains results from chemical composition of surveillance specimens of Units 3 and 4 of NPP Bohunice as well as Units 1 and 2 of NPP Mochovce. Moreover, initial mechanical properties (tensile and critical temperature of brittleness) and the same results after 5 years of irradiation in surveillance position (but without appropriate fluences) are also given.

KUPCA L., BENO P. [16] (not complete) describes also several surveillance specimens programmes for Slovak WWER-440/V-213C type reactor pressure vessels. Content of the paper is practically identical with the paper [15].

Comparison of fracture toughness tests

BRUMOVSKY M. et al. [17] provides a comparison of the shifts in static and dynamic transition temperatures in WWER reactor pressure vessel materials. Materials of 15Kh2MFA and 15Kh24NMFA (base and weld metals) were studied in initial and irradiated conditions – shifts in critical temperature of brittleness determined from Charpy V-notch impact tests and shifts in static and dynamic fracture toughness transition temperature, $T_{100MPa.m0.5}$.

Discussion is performed on the paradox between material testing within specimen surveillance programmes and fracture mechanics assessment of reactor pressure vessel integrity: impact tests with V-notch specimens vs. static tests with fatigue cracks.

In the programme, reconstitution technique was applied on broken Charpy specimens and static and dynamic fracture toughness test specimens were prepared and tested. Thus, direct comparison of both transition temperature shifts was made.

Comparison of both shifts resulted in the following correlations:

$\Delta T_{100MPa.m0.5} \approx 1.2\ \Delta T_{CHV}$ for 15Kh2MFA type of steel,

$\Delta T_{100MPa.m0.5} \approx 1.4\ \Delta T_{CHV}$ for 15Kh2NMFA type of steel.

In the same time, no difference was found between shifts in Charpy impact transition shifts and dynamic fracture toughness transition temperature shifts.

Difference between $\Delta T_{100MPa.m0.5}$ and ΔT_{CHV} for EOL fluence is practically neglectable – within scatter of data.

Programme also showed that reconstitution technique is very useful and effective method for comparative study of different type of testing.

BRUMOVSKY M. et al. [18] describes the construction of irradiation embrittlement trend curves from results of surveillance specimen tests. A standard procedure for a determination of them for WWER materials is influenced by a reliable determination of neutron fluences and by the scatter in results, especially of static fracture toughness in transition region.

For optimization of a procedure for a determination of irradiation embrittlement curves, the following types of static fracture toughness temperature dependences and transition temperatures were calculated and compared:

- transition temperature and shift evaluated at level of 100 $MPa.m^{0.5}$ from mean regression lines using logarithmic-normal distribution,
- transition temperature and shift evaluated at level of 100 $MPa.m^{0.5}$ from lower-bound curve using logarithmic-normal distribution,
- transition temperature and shift evaluated at level of 100 $MPa.m^{0.5}$ from mean regression lines using Weibull´s distribution,
- transition temperature and shift evaluated at level of 100 MPa.m0.5 from lower-bound curve using Weibull´s distribution.

Because of limited amount of samples which are used for static fracture toughness tests within surveillance programme it was useful to use also reference transition temperature T0 using “Master curve” approach.

Similar analysis was also performed for this approach:

- reference temperature T0 and its shift calculated from mean line using “Master curve” approach,

- reference temperature T0 and its shift from lower-bound line using “Master curve” approach.

Comparison of the effect of the accuracy in neutron fluence determination showed that they should have been as low as ± 5 % which is still far from reality.

As a conclusion, it was found:

- transition temperature evaluation of static fracture toughness values measured within standard surveillance specimen programme using lograthmic-normal and Weibull´s regression models has proved that irradiation transition temperature shifts are practically identical regardless of statistical method used.
- in the case of limited amount of specimens, it is possible to use “Master curve” approach,
- trend curve can be applied in the form of

$$\Delta T = A_F \,.\, (F/F_0)^n$$

with the exponent n close to ½.

<u>Sampling from RPVs and Correlation between standard and subsize impact specimens</u>

VALO M, AHLSTRAND R. [19] describes results obtained within the study of boat samples taken from RPV Greifswald NPP Unit 2 – before and after annealing. Both base and weld metals were cut off from the inner surface of the uncladded vessel. Because there are no data about the initial, unirradiated condition, only recovery shift, i.e. its residual value could be tested. The measured value are compared with Loviisa Unit 1 data and also with boat samples data from Kozloduy Unit 2 and Novovoronezh Unit 4 data.

Residual values of the shift were found larger (25 °C) than those in the Russian norm that is underestimating.

Transition temperature recovery is only about 75 % and depends on the neutron fluence due to the constant value of the residual shift.

Upper shelf energy was restored for more than 100 %, as also in other projects. In the same time, upper shelf energy of Greifswald 2 is lower than in Loviiisa 1.

The correlation between transition temperature shift and neutron fluence needs some reanalysis, as exponent for low fluence is lower than 1/3.

If subsize impact specimens are used for safety assessment, than it is necessary to create not only correlation between Charpy size and subsize impact specimens (i.e. for their shifts and also for their absolute values, if initial values do not exist), but also correlation between impact transition temperature shifts and static fracture toughness data.

STROMBAKH Ya. [20] discussed the behaviour of re-irradiated materials taken from operating units, i.e. from boat samples.

Main problems with WWER-440/V-230 type RPVs are as follows:

- not enough data on damage mechanisms for this 1st generation RPVs, mainly due to high content of impurities,
- extremely high trend curves of radiation embrittlement due to high P and Cu contents,
- lack of initial mechanical properties and chemical composition of weld metals (including no archive materials),
- lack of surveillance specimens for 15 RPVs

- due to high content of impurities, embrittlement reached its maximum allowable value and RPVs had to be annealed –no specimens for annealing recovery and re-embrittlement rate.

Thus, cutting off templates – boat samples was the only way how to obtain at least some information about RPV materials. In this case, only subsize impact specimens could be manufactured and tested.

Correlation between standard and subsize impact specimens was checked on specimens from decommissioned RPV of Novovoronezh Unit 2 in irradiated state and good agreement was obtained.

In the same time, it was found that the normative formula for radiation embrittlement of WWER-440 surveillance welds underestimates the shifts – thus a correction coefficient due to the chemistry factors was proposed to add, i.e.

to change existing normative formula:

$$\Delta T = 800(\%P + 0.07\ \%\ Cu).\ (F/F_0)^{1/3}$$

to corrected:

$$\Delta T = 750(\%P + 0.063\ \%\ Cu).\ (F/F_0)^{1/3}$$

And for a 95 % conservative estimation:

$$\Delta T^{95\%} = (5.7 + 800(\%P + 0.07\ \%\ Cu).\ (F/F_0)^{1/3}$$

It was also observed that with the decreasing neutron flux, the radiation embrittlement is increasing.

Using templates/boat samples is a very effective and the only way to estimate radiation embrittlement of such RPVs.

Lateral shift for re-irradiation embrittlement is conservative for all tested materials, thus it is recommended for estimation of RPV condition after annealing and during re-irradiation.

PLATONOV P. et al. [21] analysed results obtained for templates from Kozloduy-1 RPV. Chemical analysis, determination of neutron fluence and mechanical properties were also performed as well as some microctructural study using electron scanning microscope and transmission electron microscope.

Chemical analysis confirmed high content of P (with scatter 0.041 and 0.051 mass %) between and medium content of Cu (0.10 mass %).

Determination of fast neutron fluence based on ^{54}Mn activity allowed to compare calculation with experimental results – agreement for last two cycles was quite good.

Subsize impact specimens (5mm x 5 mm) were machined from templates and tested for determination of transition temperature. Moreover, ABIT (automated ball indentation testing) method was applied for determination of material yield strength.

Estimation of the effect of annealing was performed too, and with the use of an annealing temperature of 560 °C-2 h for an estimation of initial transition temperature was used. It was found that weld No.4 in Unit 1 behaves much better than it was expected and thus the second annealing of the vessel was not recommended and not necessary.

KUPCA L. [22] reported about preliminary results from testing of scoop specimens taken from RPVs of NPP V-1 I Jaslovske Bohunice (WWER-440/V-230 type cladded RPVs).

Scraps from the outer surface of the vessel in the position of weld No.4 were taken and chemical analysis performed. On the basis of this analyses, estimation of the initial transition temperature of the weld metal as well as of its shift due to radiation embrittlement (using normative formula) was calculated.

Relatively good agreement was found between calculated initial transition temperature and KCV values at 20 °C from the RPV test passport.

Microctructure of the weld showed to a good quality within the scoop samples.

KRYUKOV A. et al. [23] reported about investigation of samples taken from Kozloduy unit 2 RPV. Measurement of neutron fluence as well as hardness, tensile properties and impact transition temperatures (on subsize specimens) on a set of samples was performed. Chemical analysis of P and Cu contents were also made – P contents was found in a relatively large interval – between 0.0162 to 0.0373 mass %, while Cu was between 0.155 and 0.183 mass %.

Transition temperature on subsize specimens was defined for constant relation between energy in fully ductile fracture (determined at instrumented impact test) and USE,

$$A_{pi}/USE = constant$$

i.e.

$$USE(10x10)/USE(5x5) = 8.1$$

$$USE(10x10)/USE(3x4) = 22.3.$$

From this criterion, the following correlation were found:

$$DBTT(10x10) = DBTT(5x5) + 50\ °C,\ \sigma = 21\ °C$$

$$DBTT(10x10) = DBTT(3x4) + 65\ °C,\ \sigma = 24\ °C.$$

Transition temperatures for unirradiated, irradiated and annealed base and weld materials were calculated and compared with experiments with a good agreement.

Tensile tests showed to a sufficient ductility even after irradiation, annealing resulted in decrease of yield strength by 120 MPa.

Efficiency of the annealing was found good – for weld metal not less than 85 % in the shift.

KOROLEV YU. N. et al. [24] (not full paper) performed assessment of irradiation response of weld samples from Novovoronezh Unit 3 and 4 RPVs.

Subsize specimens (5 x 5) were cut from NV Units 3 and 4 and tested especially to detect re-embrittlement rate. Subsize specimen results were also compared with full size Charpy ones cut from NV Unit 2.

Based on many test results, the following correlation between standard and subsize specimen results was proposed:

$$TT(10x10) = 47 + 1.04\ TT(5x5),\ \sigma = 20\ °C$$

which shows that there no 1:1 relation between these temperatures.

KOROLEV YU N.et al. [25] reported probably about the same study programme as in [24].

The previous formula was corrected in the following way, based on detailed regression analysis using the following criterion:

$$KCV_k/USE = const.$$

where KCV_k is the criterion for absorbed energy,

then the regression line is:

Tk(10x10) = 45 + 0,96 Tk(5x5), σ = 20 °C

And 95 % confidence line:

Tk(10x10) = 52 + Tk(5x5), for Tk $\leq$ 0 °C,

Tk(10x10) = 52 + 1.085 Tk(5x5), for Tk $\geq$ 0 °C.

Using such correlations, good agreement was found for estimated materials.

KOROLEV YU.N. et al. [26] give a wider survey about results from samples cut from WWER-440 RPVs of operated units.

Again, correlation between transition temperatures from standard Charpy specimens and subsize 5x5 specimens was studied, using samples from operated units as well as from trepans of Novovoronezh Unit 2 decommissioned NPP. Based on thse tests, the previous correlations were practically confirmed. Main correlations are shown in the following graphs.

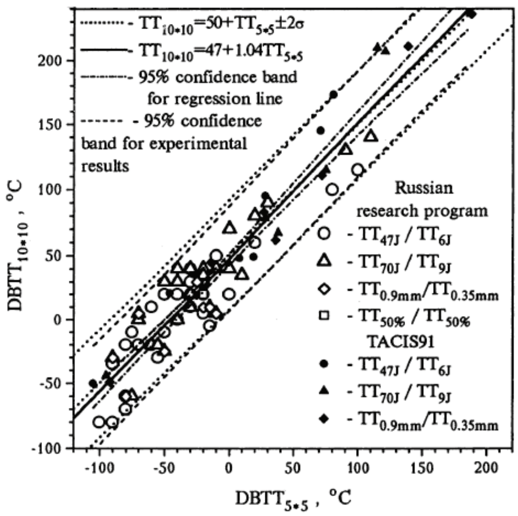


Fig. 3. Comparison of test results of standard Charpy and subsize impact specimens.

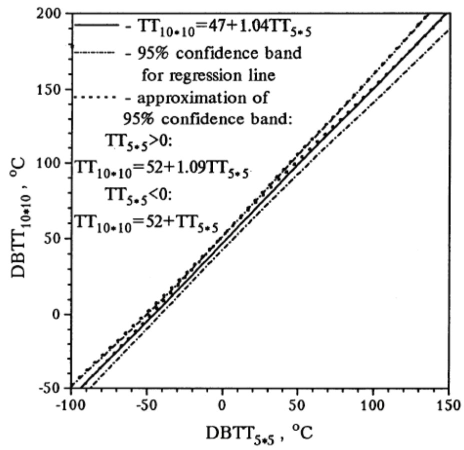


Fig. 4. Conservative approach for correlation of Charpy and subsize specimens.

$$T_K^{10\times10} = 52 + T_K^{5\times5}\ (T_K^{5\times5} \leq 0),\quad °C \qquad (8)$$

$$T_K^{10\times10} = 52 + 1.09T_K^{5\times5}\ (T_K^{5\times5} \geq 0),\quad °C \qquad (9)$$

As a main result from this study, these 95 % confidence lines were taken as the most important for further assessment of RPV lifetime.

KOHOPAA Y. et al. [27] referred about evaluation of radiation damage in Loviisa Unit 1 RPV using samples from the outer surface.

From two such samples, chemical composition, metallography and study of microstructure as well as impact toughness testing on 3x4x27 mm and tensile specimens (2x1 mm in section) was made.

Testing of base metal showed that there is a good agreement between this material and surveillance specimens and prediction of the transition temperature shift can be well assessed. For comparison of different specimen sizes, transition temperatures from subsize specimens were increased by + 65 °C.

Chemical composition was found correlated well with the RPV passport.

Some increased value of segregations was found on grain boundaries but not larger than in surveillance specimens.

RPV Integrity assessment

SCHMIDT J. et al. [28] showed results from the integrity analysis of Kozloduy Unit 1 RPV.

Integrity assessment contained especially detailed PTS analysis when typical postulated semielliptical surface defect 10x60 mm (noncladded vessel) was chosen and all calculations were performed for it, including calculations of PTS with cold plumes. For these PTS analysis conservative inputs for thermal hydraulics were given to obtain most conservative results. It was found that maximum allowable transition temperature for 10 mm deep defect was T_k^a = 178 °C, while for 230 mm deep defect it is only T_k^a = 144 °C.

Material properties were based on testing samples taking from inner RPV surface and was found that current T_F = 91.5 °C which is sufficiently lower value.

Due to the fact that material properties were based only on one point (i.e. one neutron fluence), it would be necessary to repeat the process to obtain more reliable data and a trend curve.

KEIM E. et al. [29] compared lifetime assessment based on PTS analysis using Western (PWR-KTA) and Eastern (WWER-Russian norm PNAEG) approaches.

Evaluation of a large set of different PTS regimes (mostly of LOCA type) were analysed for WWER-1000/V-320 type RPV using conservative inputs for thermal hydraulics including cold plumes existence.

Integrity evaluation was performed in accordance with KTA 3201.2 using material properties (specific design fracture toughness curve for weld metals) from Russian PNAEG. Postulated defect was chosen as a semielliptical sub-clad crack with dimensions 10x60 mm. Finite elements calculations in elastic-plastic region was performed with application of tangent method for crack path diagram. Resulted T_k^a = 114 °C was found; the deepest point of the crack was found as the critical point in this calculation.

HOSNEDL P. et al. [30] reported about the experimental programme that was prepared to confirm the PTS calculations on one side and experimental verification of these calculations on large scale specimens on the other side.

Large scale specimen (with section of 140 x 600 mm) on large testing machine ZZ 8000 with a capacity in tension up to 80 MN was prepared with a surface defect of postulated size, i.e. 35x200 mm (0.25 ot the wall thickness) from an artificial aged WWER-440 RPV material (to an embrittlement close to the EOL). PTS modelling was prepared by loading the specimen by tension to a nominal stress during operation, heated up to operating temperature and then to cool down by a flow of cold water. During the test, mechanical loading, temperature through thickness, deformation, COD as well as acoustic emission monitoring was performed together with simultaneous calculations of stress and temperature and SIF fields.

During the experiments no crack initiation was observed on fracture surface which was confirmed after the final fracture of the specimen in cold region.

STROMBAKH Ya. [31] discussed the situation and bases for the potential life extension of WWER-1000 RPVs.

The paper summarized main problems with the existing Standard surveillance programmes for V-320 type reactors and main activities/programmes carried out for the improvement of the situation – in neutron dosimetry and material testing. Some results from these programmes are also shown with the discussion of their interpretation and possible use.

The following proposal was made depending on the nickel content the welds:

Ni content in weld No.4	possibilities
1.70 - 1.88	ANNEALING – for PLEX?
	Control of radiation damage during design operation, Annealing for PLEX
1.57 - 1.64	Control of radiation damage during operation and re-qualification for PLEX
1.10 - 1.21	EOL lifetime assured Re-qualification for PLEX

The paper also discussed the problems with the proper and reliable trend curves for the weld metal with high nickel content due to the lack of data especially for high (close to EOL) fluences and necessity for modification of the surveillance specimen programme to reach high fluences.

5.4 Further References

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6 Annealing and Re-irradiation

The reactor pressure vessel (RPV) is non removable part of nuclear power plant. Post-irradiation annealing is the most effective way of RPV materials radiation embrittlement mitigation. The use of post-irradiation annealing allows managing the lifetime of nuclear power plant as long as it is economically feasible.

Up to now, post-irradiation annealing has been used as the plant life management (PLiM) method only for WWER-440 RPVs. Fourteen RPVs were successfully annealed in the period of 1987-1996. Five of them are under operation now. The remaining part was shut down most of all not because of technical reasons.

This summary was prepared on the review of 50 papers. Most of the papers (46) were received from the ODIN data base and contents the results, concerning Post-irradiation Annealing and Re-irradiation.

Several important aspects of RPV annealing are listed below:

- The study of the mechanical properties recovery due to annealing;
- The development of annealing regime;
- The validation of the annealing efficiency for standard regime;
- The technical problems concerning of the annealing;
- Licensing problems for the annealing;
- Re-irradiation behavior;
- Surveillance program for the post annealing operation;
- Assessment of the RPV state;

The summaries of the papers were split into above mentioned part.

6.1 Consolidated Conclusions

The following general conclusions can be drawn from the review:

- The overpowering part of results in the papers is in good agreement.
- The regime of post-irradiation annealing for WWER-440 RPVs has been developed.
- Post-irradiation annealing has been successfully performed for 14 RPVs.
- All the technical and licensing tasks have been solved.
- The re-embrittlement models are developed.
- The monitoring of the RPV materials re-embrittlement is carried out in two different ways for cladded and uncladded RPVs:
    - wide program of re-irradiation in surveillance channels of WWER-440 RPVs;
    - cutting out of templates from the inner surface of RPVs, for assessment of the current state, additional re-irradiation of the templates in surveillance channels of WWER-400s and special programs for re-irradiation.

6.2 Open Issues

The general open issues are:

- development of post-irradiation annealing for WWER-1000 materials
- physically based model for re-irradiation

6.3 Reviewed papers and summaries

In Figure 6.1 the time series analyses of the papers reviewed is presented. It indicates that most part of the papers is from the conference proceeding (36 from 50 papers). The papers published in journals are available for the experts usually 3 years later then from Conference proceeding. It means that conferences are very important for the exchanging of the information. In area of RPV radiation embrittlement the most successful conference was IAEA Specialists Meeting on Irradiation Effects and Mitigation. It is necessary to resume this conference after 4 years of break.

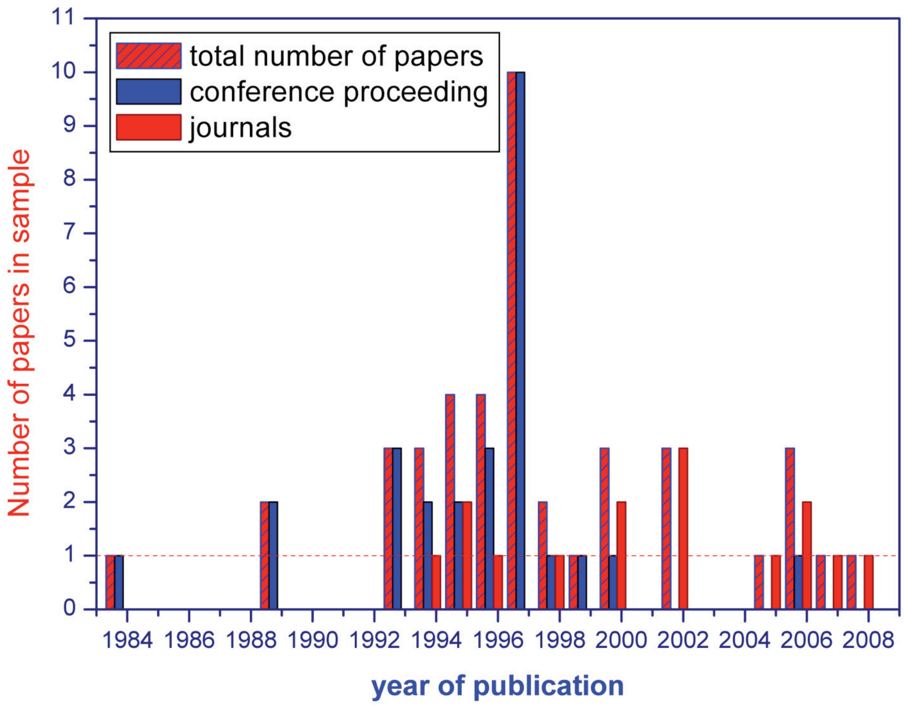

Fig. 6.1 Time series of the papers reviewed

The distribution paper according to organization is given in the Figure 6.2. It shows that it is possible not all papers are collected.

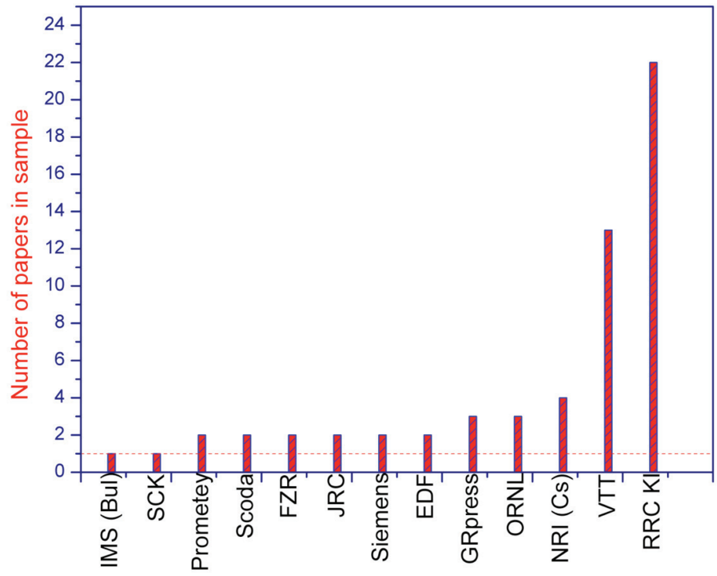

Fig. 6.2 Distribution paper according to organization

The study of the mechanical properties recovery due to annealing

The range of chemical elements in materials studied is given in the table 6.1.

Chemical elements	Range of contents, % wt
Ni	0.05 ÷2.45
Mn	0.44 ÷1.61
P	0.005 ÷0.055
Cu	0.03 ÷0.99

Table 6.1 The range of contents of chemical elements in materials studied

Effect of temperature of annealing on WWER RPV materials mechanical properties recovery

The most detailed study of effect of temperature of annealing on recovery of transition temperature of irradiated Charpy specimens from WWER-440 RPV steels has been done in paper *Amaev et al, 1993, 1994*. It was shown that temperature of pos-irradiation annealing has to be higher then 420^{o}C for effective recovery of the transition temperature of irradiated RPV materials.

The results, published in paper *Nanstad et al, 1997* confirm previous conclusions. Annealing at 434^{o}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 454^{o}C/168h.

The same summary was obtained in the paper *Kohopaa et al, 2000.* The effectiveness of annealing increases follows the line of regimes: 400- 450-475^{o}C/100h.

The study of post-irradiation annealing of WWER-1000 RPV materials has been done in work *Vacek et al, 1993*. After 1 hour annealing of material WWER-1000 type the recovery of yield stress ($R_{p0.2}$) increase when temperature of annealing (T_{ann}) increase from 300 to ~500^{o}C, but does not reach full recovery even at 600^{o}C. Increase of the temperature 450-600^{o}C (1 h) does not effect of the recovery efficiency.

The study, published in paper *Nikolaev et al, 1995* shown that after 72 hours annealing residual embrittlement of WWER-1000 materials depends on temperature: the best efficiency was received at 460 and 490^{o}C.

The summary of the results concerning the effect of temperature obtained in all paper is given in figure 6.3.

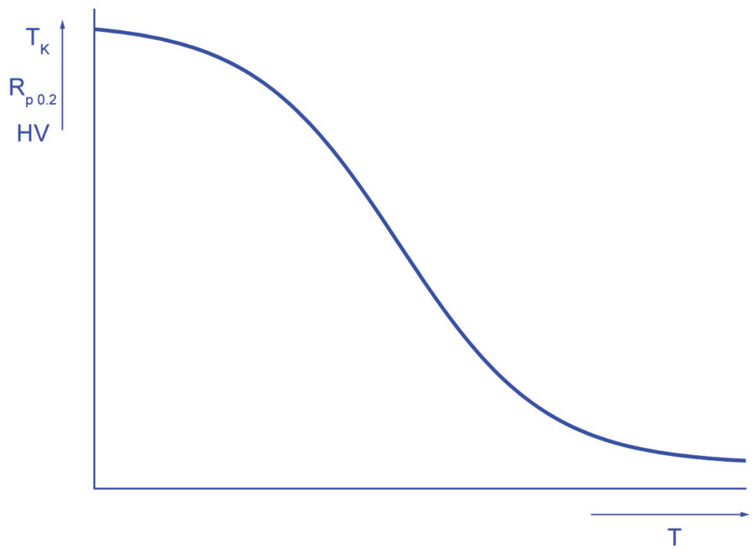

Fig. 6.3 Effect of temperature of annealing on irradiated RPV material properties recovery

Effect of duration of annealing on WWER RPV materials mechanical properties recovery

The most detailed study of effect of duration of annealing on recovery of transition temperature of irradiated Charpy specimens from WWER-440 RPV steels has been done in paper ***Amaev et al,1993, 1944***. The most intense reduction of micro hardness occurs during first hours, and then process slows down and stabilizes at the level, characteristic of the given T_{ann}. It was shown in paper ***Popp et al, 1989*** that 24 hours annealing is enough for the effective recovery of WWER -440 materials at 425 and 450^{o}C.

Increasing of annealing time of material WWER-1000 type at 425^{o}C demonstrate increasing recovery of $R_{p0.2}$ from 50% after 1 hour annealing to ~100% recovery after 50 hours of annealing (***Vacek et al, 1993***).

The summary of the results concerning the effect of duration of annealing obtained in all paper is given in figure 6.4.

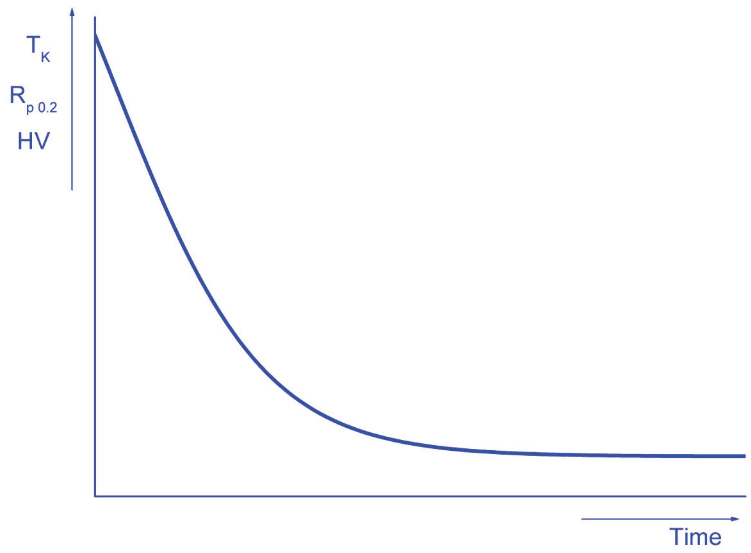

Fig. 6.4 Effect of duration of annealing on irradiated RPV material properties recovery

Effect of chemical composition on WWER RPV materials mechanical properties recovery due to annealing

The conclusion obtained in the paper *Amaev et al,1993, 1994* is following: The higher phosphorus the higher residual embrittlement. This effect is clearer at 340 and 420^{o}C and less at 460^{o}C. (Fig. 6.5)

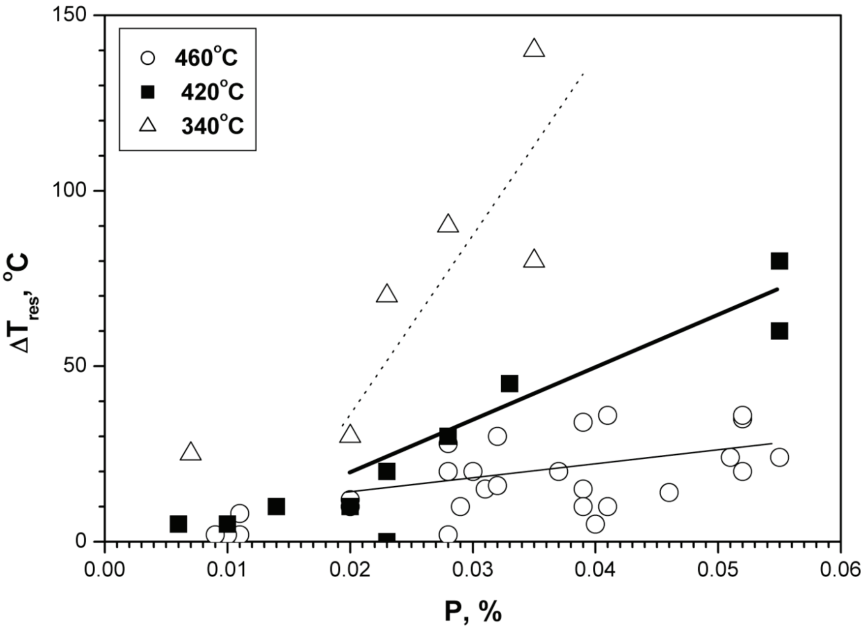

Fig. 6.5 Residual embrittlement of WWER-440 steels after post-irradiation annealing at different temperatures

The opposite conclusion was obtained in the paper *Vacek et al, 1993*: Recovery of hardness (HV) of irradiated RPV steels due to annealing depends on Cu content.

The statistical analysis of chemical composition of commercial WWER-440 steels, used in the study *Amaev et al, 1993, 1994* shown that correlation between Cu and P is about 0.6. It means that it is not possible to separate effect of phosphorus from the effect of Cu. It the paper *Valo, Debarberis et al 2008* the study of model materials has allowed to separate effect of Cu and P, because the single variable experiments have been done in this work. The following conclusions can be drawn:

- Residual embrittlement after annealing does not depend on phosphorus content for the model alloys. The results are confirmed by the research data of WWER-440 steels.

- The higher the copper content, the higher the residual embrittlement after annealing for the model alloys. The effect is evident, when copper content is above 0,25-0,30 mass% and higher. The effect is not observed for typical WWER-440 RPV steels, where copper content is varying in a much lower range; 0.10-0.24 mass %. (Fig. 6.6)

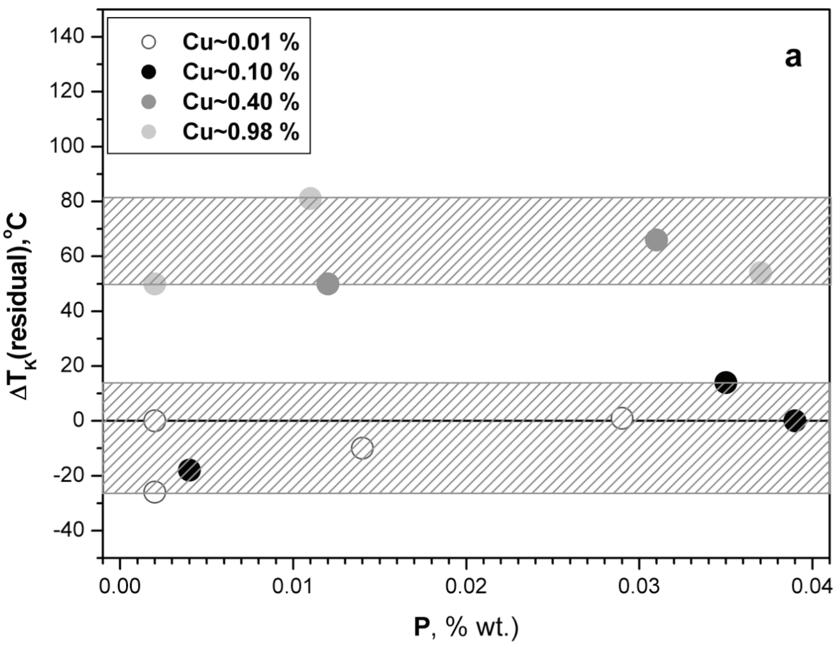

Fig. 6.6 Residual embrittlement of model materials after post-irradiation annealing at 475°C during 100 hours.

The results obtained are very well compared with similar data for WWER-440 RPV materials (Fig. 6.7).

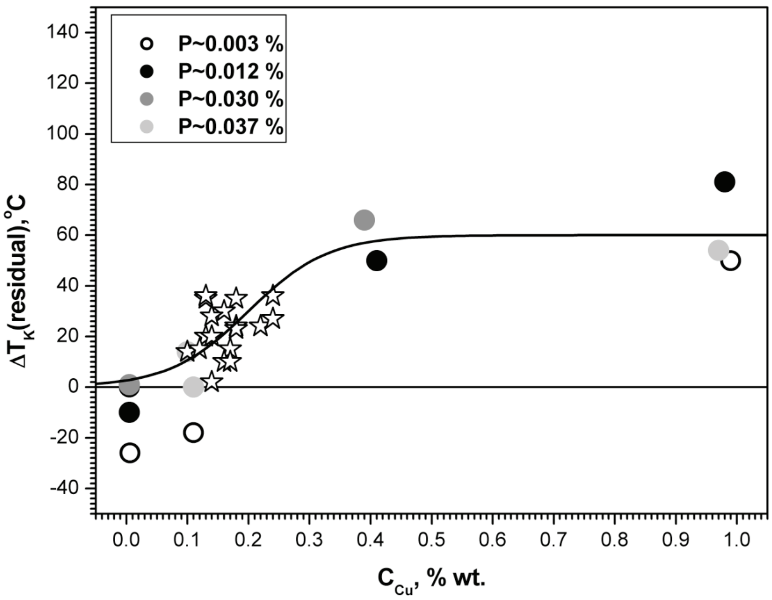

Fig. 6.7 Residual embrittlement of model materials and WWER-440 steels after post irradiation annealing at 475°C during 100 hours

The paper *Nikolaeva et al, 1994* pointed that formation of phosphorus segregation under post irradiation annealing can effect residual embrittlement.

The study of effect of annealing on irradiated WWER-1000 steels properties has been done in the work *Yu Nikolaev et al 1995*. Eleven WWER-1000 steels irradiated in wide range of fluence ($3 \div 344 \times 10^{18} cm^{-2}$, E>0.5 MeV) (weld (w) and base metal(f)) have been annealed during 72 hours at 400, 460, 490°C. There is no correlation between *Ni* and *Mn* contents, *Ni* and *P* contents, *Ni*, *Mn*, *P* contents and fluence. It means that it is possible to make analysis of chemical composition effect on ΔT_K (residual). The effect of different parameters are shown in the pictures 2.6÷2.10.

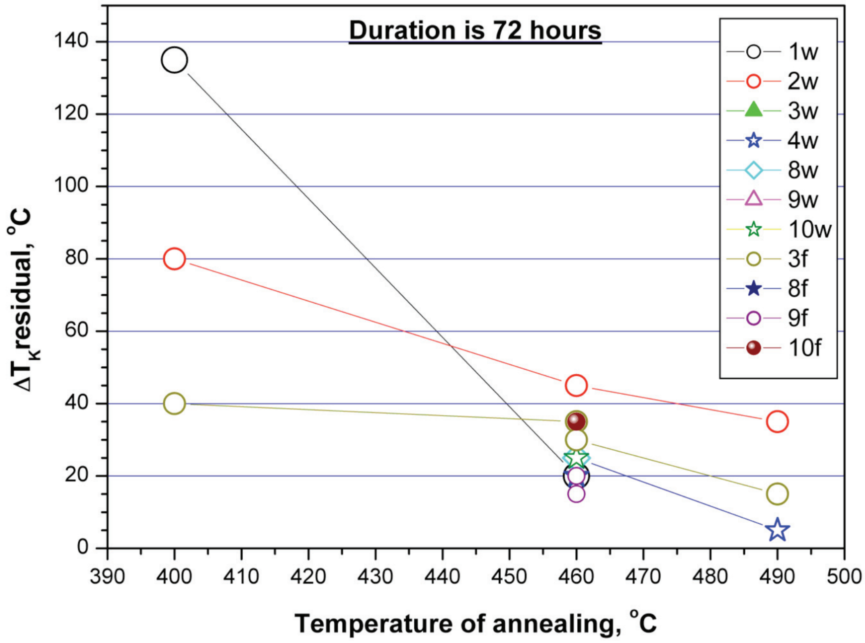

Fig. 6.8 Residual embrittlement of WWER-1000 steels (weld (w) and base metal(f)) after post-irradiation annealing during 72 hours (effect of temperature)

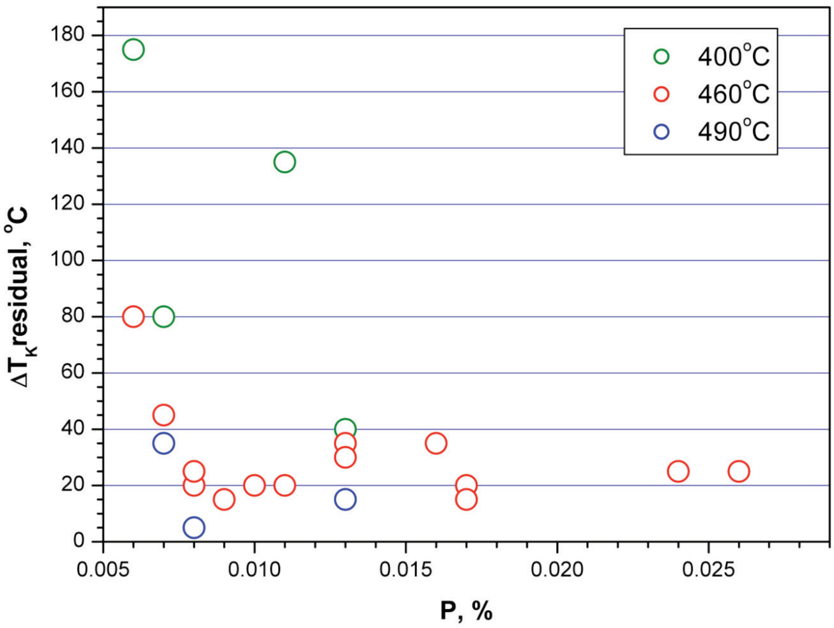

Fig. 6.9 Residual embrittlement of WWER-1000 steels after 72 hours post-irradiation annealing at different temperatures (effect of phosphorus)

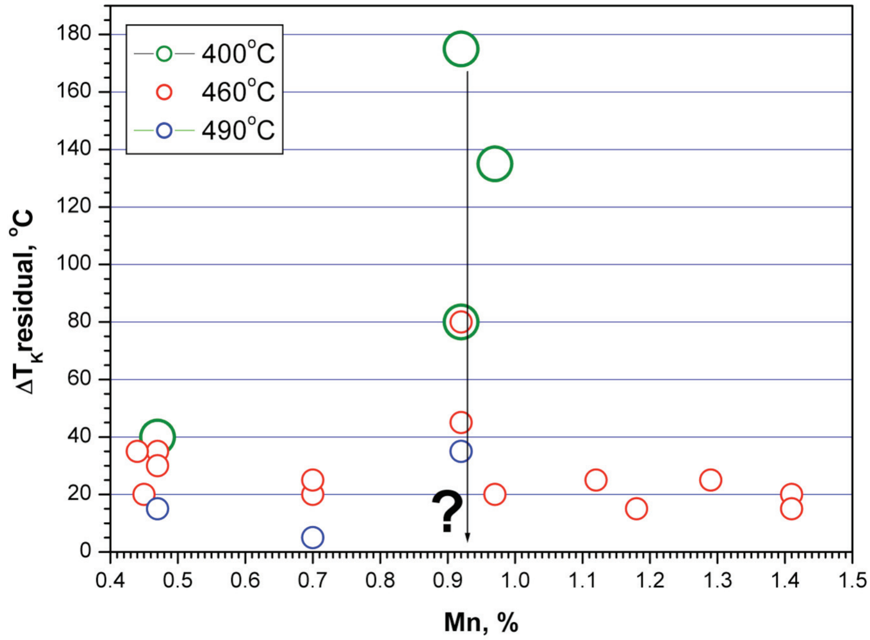

Fig. 6.10 Residual embrittlement of WWER-1000 steels after 72 hours post-irradiation annealing at different temperatures (effect of Mn)

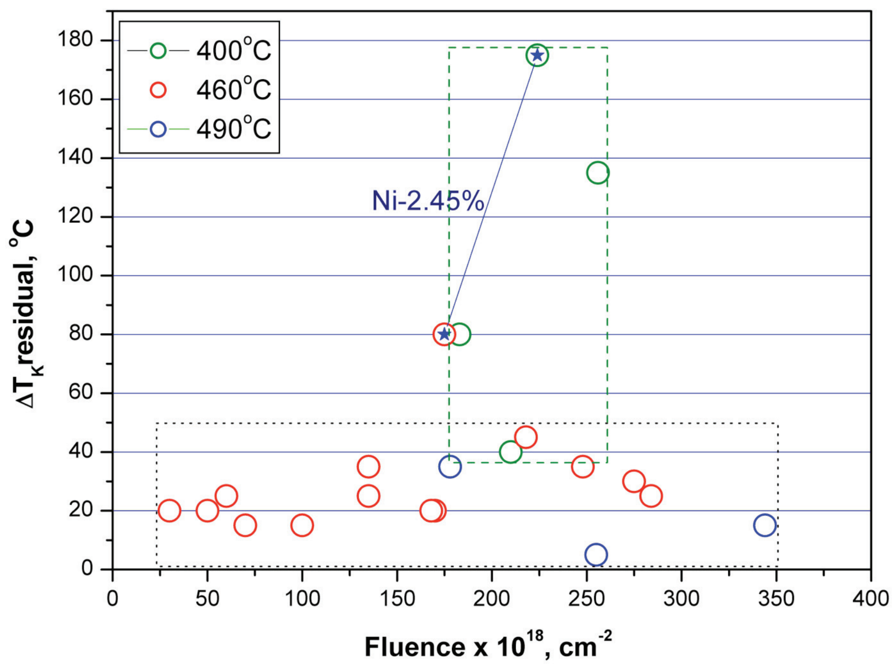

Fig. 6.11 Residual embrittlement of WWER-1000 steels after 72 hours post-irradiation annealing at different temperatures (effect of fluence)

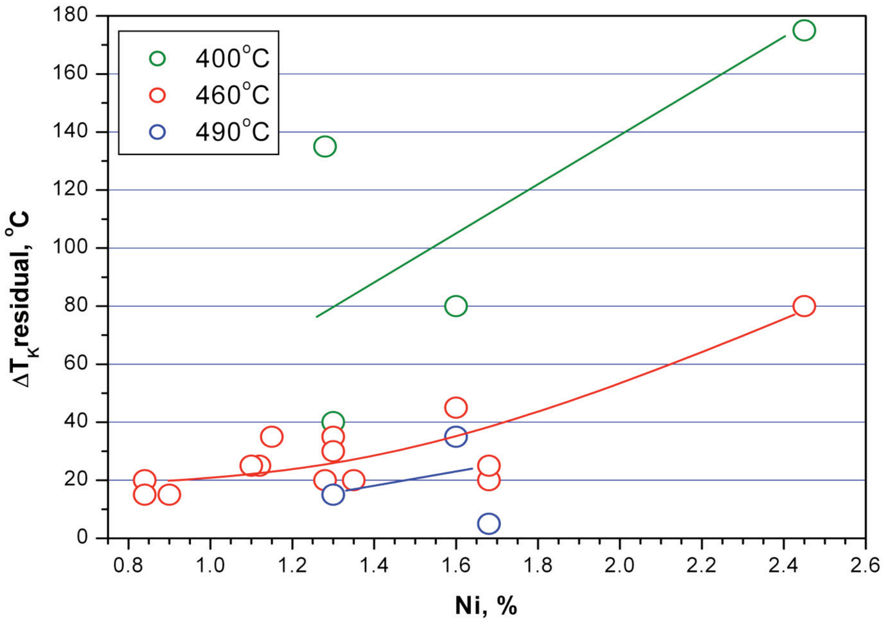

Fig. 6.12 Residual embrittlement of WWER-1000 steels after 72 hours post-irradiation annealing at different temperatures (effect of Ni)

The data presented at the figures 2.8-6.12 shown following:

- There is clear effect of temperature: the higher temperature, the better recovery.
- No effect of phosphorus and manganese.
- Effect of fluence at 400°C temperature of annealing (during 72 hours).
- Slight effect of Ni at 490°C temperature of annealing (during 72 hours).
- Effect of Ni at 400 and 460°C temperature of annealing (during 72 hours).

We can plot all the data in coordinates: ΔT_K(residual) - Mn×Ni×F, where Mn and Ni are contents of manganese and nickel correspondingly; F is fluence. (Fig. 6.13)

There is no effect at 490°C annealing and there is clear effect at 400 and 460°C annealing. It can mean that 72 hours of annealing is not enough for the temperatures 400 and 460°C.

Effect of fluence on WWER RPV materials mechanical properties recovery due to annealing

No effect of fluence on T_K(residual) was obtained in the works *Amaev et al, 1993, 1994* (Fig. 6.14). Slight effect of fluence was shown in the paper *Bumovsky et al, 1995*.

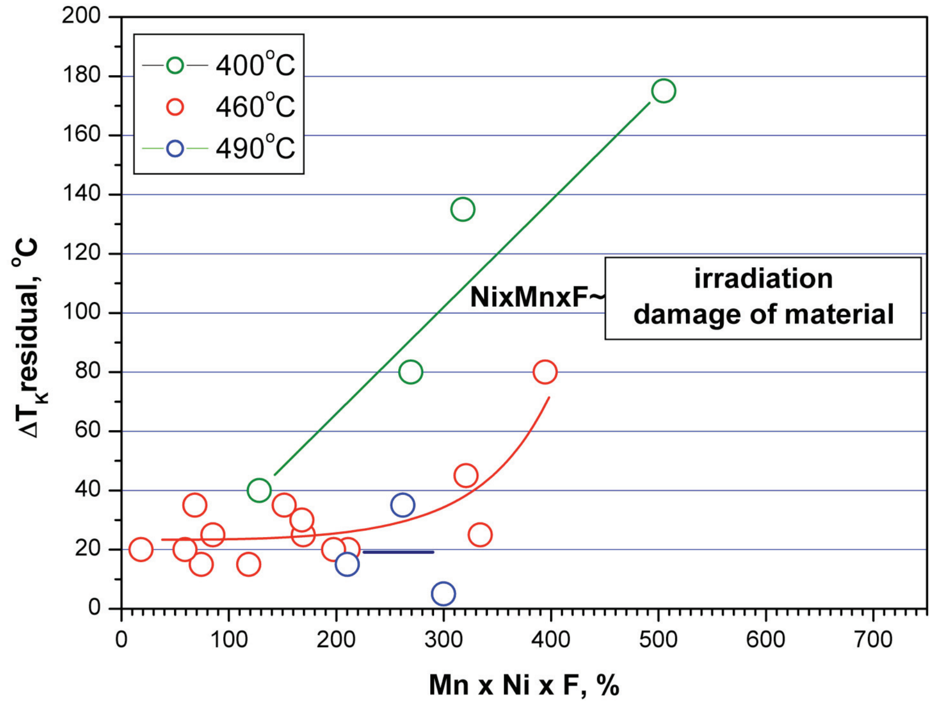

Fig. 6.13 Residual embrittlement of WWER-1000 steels after 72 hours post-irradiation annealing versus (Mn×Ni×F)

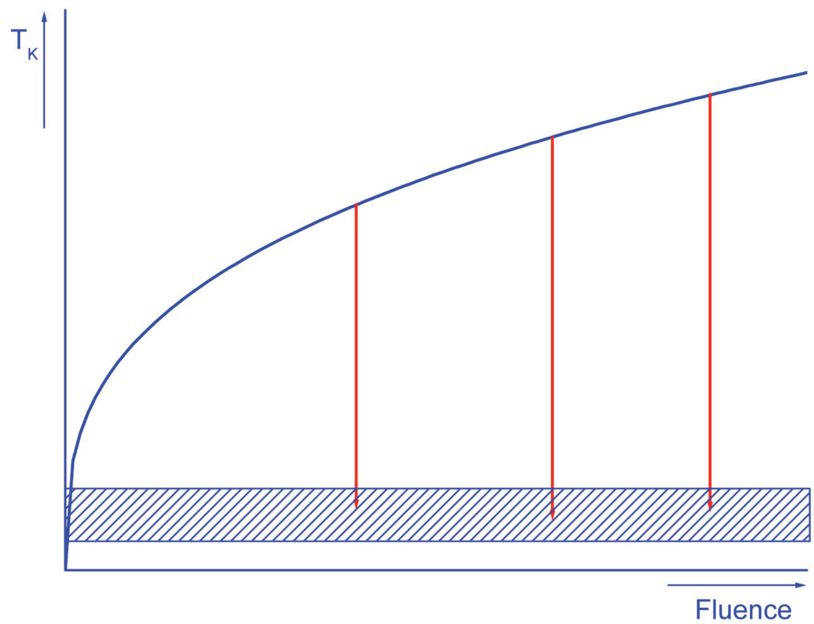

Fig. 6.14 No effect of fluence on T_K(residual)

Analysis of the data from the paper ***Nikolaev et al, 1995*** shows that there is effect of fluence on residual embrittlement WWER-1000 steels: The higher fluence the higher residual embrittlement after annealing 400°C/72 h (Fig. 6.15).

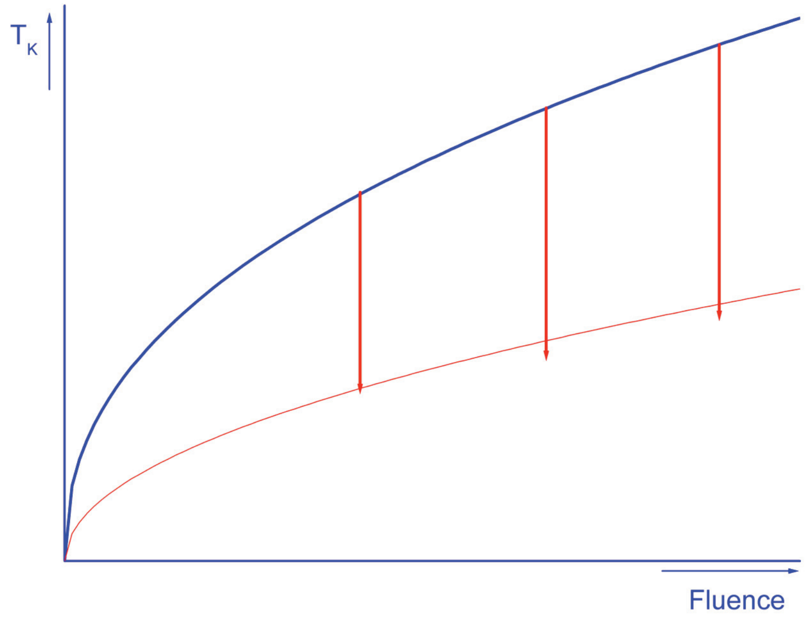

Fig. 6.15 Effect of fluence on T_K(residual).

Microstructure evolution due to post-irradiation annealing

There are several papers concerning the microstructure evolution due to post-irradiation annealing. In the paper *Gurovich et al, 1997* it was shown that the density of rounded precipitates, dislocation loops and disk shaped precipitates occurred under irradiation in WWER-440 steels decrease. I accordance with *Pariege et al* under post-irradiation annealing of WWER-440 weld the density of CRPs decreases, size of precipitates increases. After annealing only pure Cu precipitates were observed.

The formation of phosphorus segregation under post irradiation annealing can effect residual embrittlement of RPV materials (*Nikolaeva et al, 1994*). The inter-granular fracture slightly increase after post-irradiation annealing of WWER-440 steels (*Kuleshova et al, 1997*).

After 24 h of annealing of irradiated WWER-1000 weld and base metal all the irradiation induced precipitates are dissolved (*Miller et al, 2007*). The nanoclusters were present after post irradiation annealing for 2 h but not after 24 h (Fig 6.16 and 6.17).

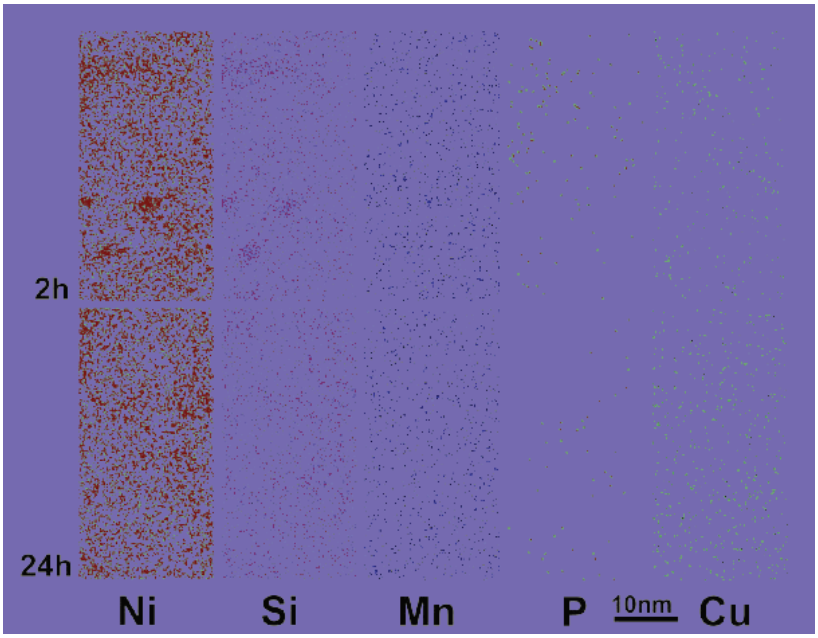

Fig. 6.16 Atom maps of the highest fluence base metal (fluence = 11.5 x 10^{23} m^{-2} (E > 0.5MeV)) after the post irradiation annealing treatments of 2 and 24 h at 450 °C. *(orig. Fig. 14.9)*

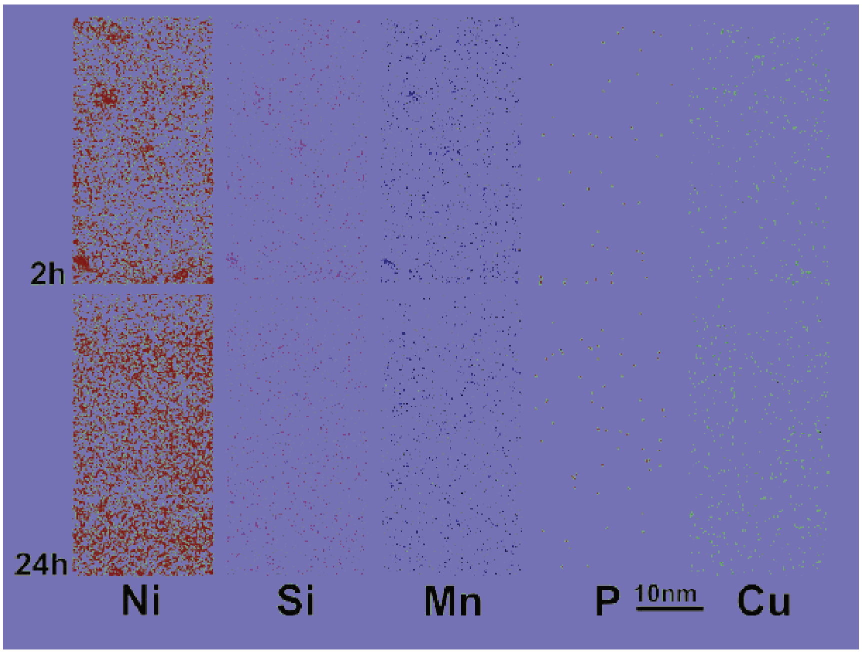

Fig. 6.17 Atom maps of the highest fluence weld metal highest fluence weld metal (fluence = 14.9 x 10^{23} m^{-2} (E > 0.5 MeV)) after the post irradiation annealing treatments of 2 and 24 h at 450 °C. *(orig. Fig. 11.5)*

The development of annealing regime

All the papers which give the information of recovery of radiation embrittlement due to annealing include no information or discussion about the way how to use the results of experiment to receive a parameters of annealing regime for RPVs.

Validation of annealing efficiency

Residual embrittlement (ΔTK(residual)) assessment

After 150 h annealing of WWER-440 steels at 460°C residual embrittlement is not higher then 40°C. The medium value is 20°C (***Amaev et al, 1993, 1994***). After annealing of WWER-440 steels at 475°C/168h residual embrittlement is 5 - 46°C. The medium value is 30°C (***Brumovsky et al, 1995***). After annealing of WWER-440 steels at 475°C/100h residual embrittlement 14-18°C (***Kohopaa et al, 2000***); 1°C (***Lucon et al, 2002***) and 25°C (***Valo et al, 1997***).

Respond of different parameters (T_K, USE, T_0, $R_{p0.2}$, Hv) on post-irradiation annealing

After annealing of WWER-440 steels at 475°C/168h residual Rp0.2, is about 50 MPa (*Brumovsky et al, 1995*); almost full recovery of $R_{p0.2}$ (*Lucon et al, 2002*). Annealing at 400°C leads to full recovery of Rp0.2 of WWER-1000 steels (*Nikolaev et al, 1995*).

Post-irradiation annealing of WWER steels coarse over recovery of USE (*Brumovsky et al, 1995, Kohopaa et al, 2000*, *Lucon et al, 2002, Nanstad et al, 1997*).

After annealing of WWER-440 steels at 475°C/168h residual of T_0 is about 20°C (*Brumovsky et al, 1995*) and 7-17°C (*Kohopaa et al, 2000*); T_0 residual is about 12°C (*Lucon et al, 2002*).

Technical problems concerning of the annealing

The old type of WWER-440 RPVs has several features (*Brumovsky et al*):

- Relatively small diameter of vessel. →It means end of fife fluence (EOL) fluence is high ~2×10^{20} cm^{-2} (E.0.5 MeV).
- Not enough information about materials, as en example Cu and P content and T_{K0}.
- The high rate of embrittlement.
- No surveillance programs.

As a way of mitigation the annealing was chosen. 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.

Regime is 475°C/100h; Heating and cooling rate ≤20°C/h; Total time is 9-11 day. 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 impossibility to test of real materials from RPV (*Ahlstrand et al*).

Annealing temperature should be in a window: enough high for the recovery of the properties and lower then temperature, which can effect the others components, such as primary piping, support structure and biological shielding, concrete and so on.

In accordance with *Brynda et al* there is concern about the temper embrittlement during annealing and it is necessary to control temperature during annealing several points.

The detailed review of the procedure of "wet' and "dry" annealing has been performed in paper *Mager, Dragunov et al*.

Licensing problems for the annealing

The papers *Brumovsky et al* describe wide supporting program had been done by Scoda and NRI for annealing. The program included the following items:

- Design of the furnace.
- Calculation of thermal and stress fields in reactor.
- Design of annealing of weld № 4 and adjusted aria.
- It was decided to add additional ring to the furnace to decrease temperature gradient.
- The special demonstration experiment had been done with vessel with natural dementias.
- The program of irradiation and annealing specimens has been done.
- Templates form outer surface had been cut to check chemical composition.

The papers studies *Ahlstrand et al* and *Rantala et al, 1995* contain the detail description of work, done for the licensing of the RPV annealing. The work has started in 1980. Then the lager systematic investigation program from Loviisa surveillance specimens had been done. Another part of licensing program was carried out by Moch-Otshig in Russia. This program included study of cladding.

Mager, Dragunov et al,

The paper *Mager, Dragunov et al* also gives deep information of procedure of post-irradiation annealing of RPVs. Before and after annealing the vessel was nondestructively inspected. The special procedure of dynamic hardness testing was developed and special device was constricted. The small specimens from inner surface has been cut and studied.

Re-irradiation behavior

Several papers include the result of re-irradiation behavior study.

V. Nikolaev et al, 1984:	The rate of the embrittlement depends on the residual embrittlement after annealing. The development of the principle of "lateral shift" has been done in this work.
Amaev et al, 1993:	The rate of the embrittlement depends on the residual embrittlement after annealing. The three models of re-irradiation has been proposed: lateral shift, conservative and vertical models. The lateral shift is the most adequate model.
Amaev et al, 1994:	The re-embrittlement depends on phosphorus content.
Kohopaa et al, 2000:	The new re-embrittlement model had been proposed.
Lucon et al, 2002:	The rate of re-embrittlement is lower then rate of embrittlement.
Nanstad et al, 1997:	The rate of re-embrittlement is following lateral shift for US and Russian steels.
Valo et al, 2006	Re-irradiation behaviors follows well the lateral shift model based on best fit irradiation behavior

Several models of re-irradiation embrittlement are shown below.

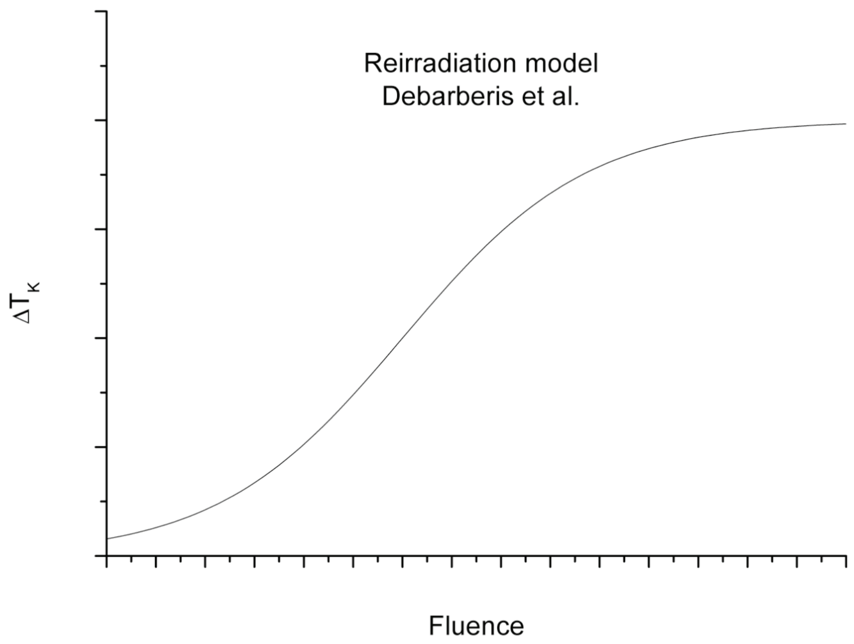

Fig. 6.18 Re-irradiation model from paper Debareris et al.

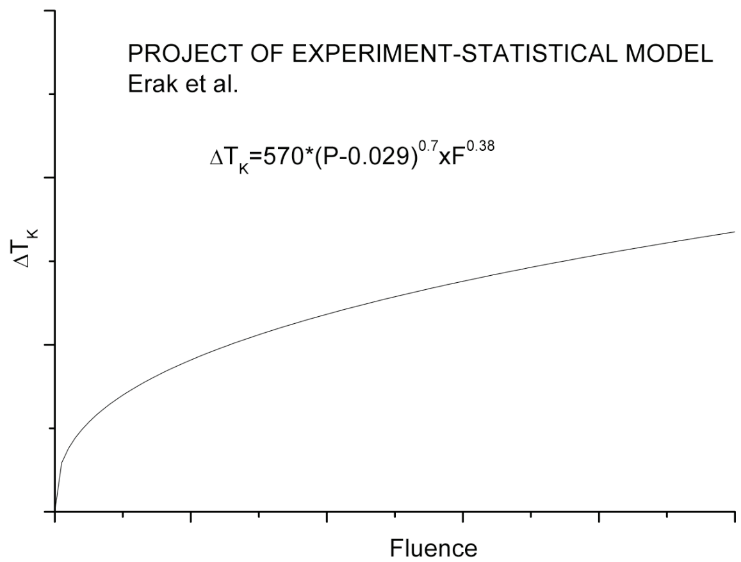

Fig. 6.19 Re-irradiation model from paper Erak et al.

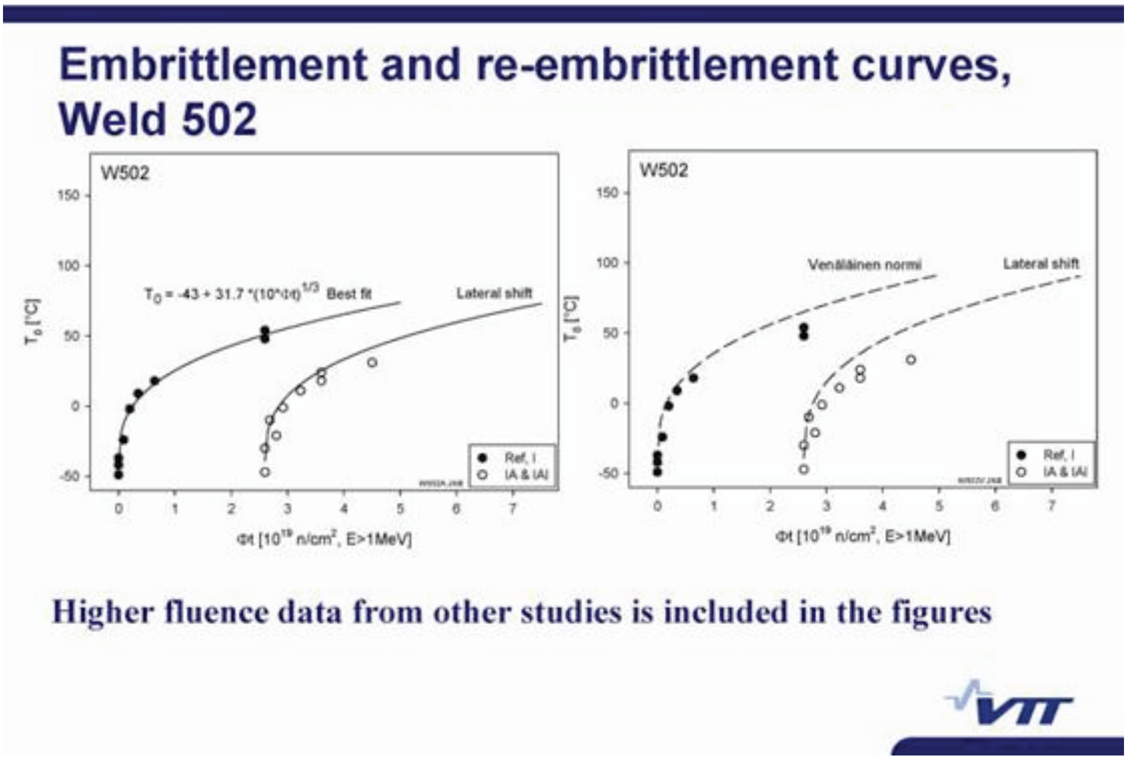

Fig. 6.20 Re-irradiation model from paper M. Valo et al.

The analytic view of re-irradiation model from the paper ***Shtrombakh and Nikolaev Yu., 2007*** is following: $TTR = 1230C_P + 207\ 0.04 + 1.91 + 210C_P - 0.02F0.63$.

If $C_P < 0.020$ % , then $C_P = 0.020$ %, where C_P is phosphorus content.

Surveillance program for the post annealing operation

There are two different principles of the monitoring of the RPV materials re-embrittlement. One is for cladded RPVs described in paper ***Ahlstrand, 1995*** and practically used for Loviisa 1:

- The new surveillance program for monitoring of re-irradiation of weld № 4 consists of 300 specimens.

Another ideology is used for uncladded RPV:

- For annealed RPVs without cladding the material of temples, cut out from inner surface of RPVs at regular intervals are used direct after cutting for assessment of current stage and after additional irradiation in surveillance channels of WWER-440 for the forecast (Shtrombakh, 2000).

Assessment of the RPV state

Ahlstrand et al. 1995	The small samples from inner wall at the core region of NPS Graifswald 2 were cut before and after annealing. The 3×4 Charpy specimens testing, chemical analyses, fluence determination and micro-hardness had been done. It was concluded, that correlation between sub-size and standard Charpy specimens need improvement and confirmation.
Kohopaa et al. 1994	The small samples from outer wall at the core region of Loviisa 1 were cut out in 1993 from 2 forging above and below weld № 4. The 3×4 Charpy specimens were tested and used 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. The lifetime of Loviisa-1 is not limited by base metal irradiation embrittlement.
Kryukov et al. 1995	The small samples from inner wall at the core region of Kozloduy 2 were cut out in 1992 after 16 year of operation from the weld № 4 and forging. The 3×4 and 5×5 Charpy specimens were tested. The chemical composition was obtained. The actual T_K after irradiation and post-irradiation annealing were determined, and the reconstruction of the T_{K0} was assessed. Correlation between sub size and standard Charpy specimens testing was determined.
Valo et al. 1995	The results obtained from 3×4 Charpy specimens from Greifswald 2 were compared with surveillance data of Loviisa 1. The data from Greifswald 2 do not point directly on dose rate effect, but it is not excluded. It was concluded that fracture toughness behavior should be checked in surveillance program.
Kryukov et al, 1996, Korolev et al, 1998, Shtrombakh, 2000	The correlation between sub-size and standard Charpy specimens has been developed then improved. The assessment of Kozlody-2 before and after annealing has been done using the temples. The assessment of Russian annealed NPPs are performed using the study of temples and re-irradiation of part of temples in surveillance channels of WWERs-440.
Valo et al. 1998?	The technique of manufacturing mini Charpy specimens from standard Charpy specimens was introduced. 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 RPV materials transition temperature shift.
Lucon et al. 2002	Standard and 3×4 Charpy specimens from WWER-440 weld were irradiated, annealed and re-irradiated in research reactor. Correlation between sub-size and standard Charpy specimens testing was analyzed. It was concluded that sub-size specimens underestimate T_K in comparison with standard Charpy specimens testing.

7 Material Factors

The number of papers collected from the last year is much larger in the field of materials than the number of papers collected for the previous meeting. Consequently several new aspects can be found in the papers. The papers can be grouping according to the topics of them as it follows:

- Chemistry effects (effect of alloying and polluting materials (14)
- Ni effect (thermal ageing) (6)
- Microstucture effect (2)
- Boron effect (1)
- Chemistry formulae use (11)
- Late blooming? (4)
- NDE Testing (2)

Effect of alloying and polluting elements

The present codes and standards consider only few alloying and polluting elements, namely the Russian code considers the radiation sensitivity depedence on P and Cu content, the western codes considers the effect of Cu and Ni.

The present researches shown that several other elements affect the radiation sensitivity:

$$\Delta T_F = \Phi^n * Ni^a * Mn^b * Si^c * S^d * P^e * Cu^f * Mo^g$$

This type formula (with 10 constant) cannot be used in the everyday evaluation of the properties of structural materials and several important factors are still missing to describe the real material embrittlement behaviour. Different authors made efforts to study the effect of the individual elements on the radiation embrittlement shift. They discovered that the different alloying and polluting element have synergetic effects (e.g. Manganese and Nickel). The study of irradiation embrittlement using model alloys and the analyses of different databases are the information tools for study of the synergetic effects.

Boron and Molibdenium are the new elements of which effects on radiation embrittlement is considered. No paper dealing with the effect of Vanadium which is an important alloying element in the WWER materials showing high radiation stability.

Ni effect (thermal ageing)

Papers dealing with the effect have two main statement: Nickel and Manganise has synergetic effects that is if the weight of them together is over 1.5-2% than the radiation embrittlement rate of the material will be increased.

Other group of the papers discuss the comparison between the radiation embrittlement and thermal ageing of the high nickel content steels. A set of steel with different nickel content have been irradiated at 290°C and the same steels have been thermally aged at 500°C. In both cases the embrittlement shifts are linearly increased with the increasing nickel content showing the during irradiation radiation assisted thermal ageing is a major contributor of the embrittlement. Ni-Mn synergetic effect is still an open issue, few data exists to guess the trend of the effect and micromechanical understanding of it is still missing.

Microstructure effects

Few papers consider the effect of the microstructure. The grain size is smaller at the surface of the forgings and plates than in the middle, consequently there is a transition temperature difference between the surface and middle section. This bias decreases during irradiation, but it doesn't disappear even at irradiation as high as the RPV EOL fluence of the commercial reactors.

Chemistry formulae

Most of the formulae describing the radiation embrittlement shift of the RPV materials are using fix exponent over the fluence. Several researches proved that the exponent is not independent from the material and changing between 0.2 and 0.9. This verify the formulae used in VERLIFE procedure where the exponent is not fixed but obtained by function fitting for the surveillance results.

Late blooming

Four papers showing data deviating from the simple exponential law used to describe the transition temperature shift in the function of fluence. The clean steels after certain fluence level showing increased embrittlement rate. This may show the start of a new embrittlement mechanism. Late blooming is an open issue of which need to be studied in the future in the frame of Long Operation Life programs.

NDE Testing

Two papers show that the thermoelectric power measurement provide results of which can be correlated with the embrittlement rate (transition temperature shift). No clear evidence which micromechanical change causes the effect.

7.1 Consolidated Conclusions

A simple exponential law in chemical formulae does not describe exactly the embrittlement rate in the function of the fluence. Fitted exponent on surveillance data are the best solution.

7.2 Open Issues

Late blooming or deviation from the trend curves exist in some cases. Further evidence is required to decide if these cases are really verified late bloming or the deviation is only resulting from a testing error or another unknown effect.

Boron and Molibdenium are presently considered elements which could cause effects on radiation embrittlement. No paper is existing dealing with the effect of Vanadium which is an important alloying element in the WWER materials showing high radiation stability.

7.3 Reviewed papers and summaries

Williams T. J., "The effect of nickel, manganese and copper on the Irradiation sensitivity of low alloy steel welds", published in International Journal of Pressure Vessels and Piping 81 (2004) 657–665

The paper deals with the comparison of the radiation degradation behaviour of the East and West European RPV steels. As earlier Mr. M. Davies have shown the steels and welds easily can be categorized as Eastern and Western steels, but the difference between the different steels are in the chemical composition, production technology and in the environmental effects (irradiation fluence, flux, spectra, temperature). The paper compares the irradiation hardness shifts of different steels up to 1 dpa fluence. Effect of chemical composition is discussed considering Manganese, Copper, Nickel, effects. To compare the different irradiations "irradiation temperature factor" is introduced. The paper concludes that practically no difference between Eastern and Western steels, in both cases the composition and the environmental factors determine the embrittlement rate.

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

The paper recognized basic mechanism of radiation damage for primary embrittlement of steels and welds is due to copper precipitation.

It introduced an equation for DBTT shift to account for saturation occurrence. To determine the effects of Cu as well correlation to Ni and P contents 21 different chemical alloying steel, model alloys, have been tested and irradiated. Those model alloys cover a wide spectrum of Ni, Cu, and P content. DBTT shifts was measured to show the embrittlement effect. As it is expected large shifts have been experienced in case of high cooper content. Thermoelectric properties based on the Seebeck and Thomson effect measured too. The Relative Seebeck Coefficient (RSC) applied to characterize the irradiation effect. There is a clear dependence of the RSC with Ni content. The RSC values increases after irradiation for high Cu alloy (~0,4wt%) with Ni content ranging from 0.7 to 2 wt%. The same RSC values doesn't affected by the irradiation in case of low Cu alloys (~0,1wt%).

P content does not affect the RSC value. The paper concluded that RSC values measures radiation embrittlement indirectly by measuring the decrease of Cu content available for precipitation.

Amayev A.D., Kryukov, Levit V. I. and Sokolov M.A., "Radiation Stability of WWER-440 Vessel Materials", published in Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels: An International Review (Fourth Volume), ASTM STP 1170, ASTM, Philadelphia, 1993, p. 9-29.

The life limiting ageing for the RPV material is radiation embrittlement. It limits the service life of the NPP. WWER steels manufactured on the early period produced steels with hig P content and unknown Cu contents. The paper presents testing results of surveillance sets irradiated in commercial reactors and research reactor testing of WWER440 materials. The radiation embrittlement was characterized by ΔTF (transition temperature shift) using full and sub size Charpy impact testing. Some specimens irradiated in direct contact with the coolant, shown no bias compared with the specimens irradiated in surveillance channels. The measured data have been compared with the Russian formulae where the ΔTF dependent on the P, Cu content and the fluence of fast neutrons. P and Cu synergetic effect is considered too. The analyses of the available database shows different relation between the fluence and the irradiation caused shift in case of high and low flux. The P contribution to the shift is similar for high and low neutron fluxes however Cu contributes differently according to high or low neutron flux. The contribution of radiation defects and other impurities is independent from the flux.

Sometimes at high level irradiation the slope of the transition shift curve show increases instead of saturation.

Debarberi L., Sevini F. ,Acosta B., Kryukov A., Nikolaev Y., Amaev A.D. and Valo M "Irradiation embrittlement of model alloys and Commercial steels:analysis of similitude behaviors", published in International Journal of Pressure Vessels and Piping, 79, 2002, p. 637–642

The understanding of the irradiation embrittlement is a key issue for lifetime assessment. The paper starts tracking down the mechanisms involved on DPA. It highlights the importance of chemical composition, the use of the chemistry factor to trace trend curves and formulas for transition temperature shift calculations based on deleterious elements contents in the steels as provided in the US Regulatory Guide. Model alloys provided by Russian Kurchatov Institute with different chemical composition of the deleterious elements, P, Cu and Ni. The range of alloying and polluting elements represents the steels used in RPV-s. The samples were irradiated to ~ 0.00829 DPA (~5.14.10^{22} nm^{-2}) in the LYRA rig at JRC-HFR (high fluence research reactor of the Joint Research Center at Petten). The material was characterized by Charpy test, hardness and STEAM (thermoelectric power testing). The irradiated specimens tested at VTT. Increases of deleterious elements content increases the DBTT shifts. For middle P content (~0.012) Ni content increases the DBTT shifts. The DBTT shifts obtained from the data and calculated by Russian Guide from the chemistry shown good correlation. The paper introduces a modified CF (chemistry factor) and shows it's correlation with the DBTT shift. Finally it presents the DBTT shift data for NPP, the data obtained on model alloys and the similitude of model alloys behavior and commercial steels.

Böhmert J.,. Ulbricht A., Kryukov A., Nikolaev Y., Erak D., "Composition Effects on the Radiation Embrittlement of Iron Alloys", published in: Effects of Radiation on Materials 20^{th} International Symposium, ASTM STP 1405, 2001 p.383

Eight different iron based model alloys have been irradiated in the surveillance channels of two WWER reactors. Low and high percent of Cu, P, Ni, Si, and Mn varied in the alloys.

Impact and tensile tests performed on the irradiated materials to check the embrittlement rate. Two fluence levels used: $1*10^{19}$ and $8*10^{19}$ n/cm^2 E>0.5 MeV. Increasing P and Cu content significantly increased the transition temperature shift, in the case of Ni significant effect on the mechanical properties observed only over 1.5%.

A set of the specimens have been annealed after irradiation at 475°C 100 hours.

The results verified the use of the Russian code. Interesting results is that the increasing phosphorus also increased the hardness shift.

The irradiation produces inhomogeneities (matrix defects, precipitates, segregations). The size of the inhomogeneities are independent from the irradiation fluence (at least after a threshold value), but increasing with increasing nickel content. The results have shown, that the nickel increases the irradiation sensitivity but the results didn't confirm the statement, that this effect exist only over 1.5% nickel. Finally analyzes of the annealed samples shown that in the case of high Cu or Ni content the recovery is only partial.

Davies L.M., "Nuclear Power Plant Life Management with Particular Reference to the Reactor Pressure Vessel", published in The definitions of Plant Life, Operational Life, Design Life

Periodic Safety Review developed in the paper in order to differentiate between the terms and to show their significance in terms of Plant Life Management. Assessment of integrity of RPV involves procedures is shown. It discusses the role of degradation mechanisms in the material properties.

Davies M., Kryukov A., English C., Nikolaev Y., "EAST/WEST STEELS FOR REACTOR PRESSURE VESSELS", published in IAEA Specialists Meeting, Vladimir, Russia, September 1997

The report is presented in three parts. Part 1 is a comparison between the irradiation behavior of Western and Eastern steels. Part 2 explains irradiation embrittlement and Part 3 the role of chemical composition on irradiation sensitivity.

Part 1 – For comparison the fluence is normalized to E>1MeV and embrittlement trend curves are compared. The extent of the fluence scale reflects the End of Life Peak fluences. The curves are based on real material and predictions calculated by Russian Code for the Eastern steels and by the USNRC Regulatory Guide 1.99 Rev. 2 for the Western steels.

The shift of Charpy Transition Temperature is calculated with both approaches as well the chemistry factor that describes the deleterious elements. The US regulation considers the effects of Cu and Ni and the Russian code calculating the TT shift for Cu and P, as those elements increase the irradiation sensitivity on each type of older steels. In the case of both approaches the higher content of deleterious elements result higher chemistry factors. The USNRC Guide as well as the Russian Code includes all three elements: P, Cu, Ni. The effect of P is considered on a different way: in the eastern steels, namely P becomes the dominant sensitizing element while in the western steels copper is the dominant sensitizing element for irradiation embrittlement.

The irradiation ageing of western and eastern steels follows similar trends. Older steels with high content of deleterious elements show the greatest degradation. Annealing recovers irradiation damage equally on western and eastern steels.

Part 2 – Summaries the status of understanding of mechanisms involved on irradiation embrittlement of western steels. The three basic mechanisms the Cu-rich precipitates, matrix damage and grain boundary segregation of elements such as P causing embrittlement. The last mechanism cause changes in the fracture stress but not in the yield stress that is resulting non-hardening embrittlement. Consequently to study and test fracture toughness and tensile properties are equally important to get correct information on the ageing state. The matrix damage comprises the point defect clusters and dislocation loops. The use of Pala and Doppler techniques considered to quantify matrix damage and defect clusters. Cu precipitations are known as barrier the dislocations. Thermal ageing may results similar effect than irradiation. Other alloying elements can associate Cu precipitates and form a new constituent precipitate. Others elements as P segregates to grain boundary as a result of irradiation thus intergranular cohesion strength is reduced and the failure mode is modified.

Part 3 – The differences of alloying elements and the deleterious elements of Western and Eastern steels promotes different mechanisms. The western RPV wall materials have Cr (~0.1- 0.14 wt. %) and eastern 2-3%. The Ni content can be the same for most steels but it is markedly higher for the WWER-1000 maerials, sometimes it exceeds 1.5%. Analysis on radiation sensitivity show the most deleterious element for western steel is Cu and for eastern (WWER-440 type) is the P. Generally the neutron flux in the western RPV-s is lower than in WWER-RPV-s. For both steels the synergy among the Cu, Ni and P was observed during different researches. Several pictures and graphs are given showing measurements leading to the above remarks and conclusions. During the annealing process the degree of recovery depends on the P and Ni contents.

Debarberis L., Törrönen K.T., Acosta B., Sevini F., Kryukov A., Nikolaev Y., "Experimental Studies of Copper, Phosphorus and Nickel effect on RPV Model Alloys at two different fluence"s; a Joint Project between JRC Petten and Kurchatov Institute, Moscow, published in

Different Ni, Cu and P contents model alloys were produced and irradiated through a joint project by the IAM, presently IE, JRC and the Kurchatov Institute. Those 33 model alloys represents the existing variation on the chemical composition of RPV steels. The model alloy was irradiated in HFR, Petten and Russian NPP-s. The paper presents analysis on Charpy samples of the 7 model alloys which irradiation was completed. The analysis confirm the appropriated use of the chemistry factor, accounting for Cu and P, at Russian Code with different Ni content to predict TT shift. For high Ni content steels a chemistry factor extended to comprise this element would improve the code.

Karzov G.P., Filimonov G.N., "Improvement of Reactor Pressure Vessel Materials for WWER Reactors of New Generation of High Safety and Service Life, I.V. Gorinin", published in Proc of Honorary Conference on 100 years anniversary of the birthday of acad Kurchatov and Alexandrov.

The first generation of the WWER steels (WWER-440/230 reactors) was sensitive for radiation embrittlement. Clean steel have been developed for the second generation to increase the radiation toughness and decrease the embrittlement rate. The effect of several alloying and polluting elements (Cu, P, Ni, Sb, Sn, Pb, As, Al etc.) and the microstructure considered. Figure shows the segregations produced at the manufacturing of the steel ingots.

As a results of the development of the new generation of the 15H2MFA steel Master curve predicted.

"P Kamenova T., Vodenicharov S., Momchilova E. and Gaidarova V.," phosphorous content and distribution in the metal of RPV weld4", published in Report IAEA, 1997, Sofia, Bulgaria, p. 66

The paper investigates P concentration and distribution in weld material. Samples were cut from the weld seam.

Different wet chemical standards to determine P had been compared, and the EN 10184 AC methodology selected. The average content of P was 0.046 %. Samples were tested by impact fracture. The character of the fracture was investigated by fractography, it was transgranular and it shown typical transcrystal properties. The identification of morphological features has been made by Auger and Scanning electron microscope. P enriched zones in facets along secondary cracks edge, O peak in facets near the inner surface of RPV wall discovered. The P enriched layer has limited thickness, 10-20 nm. The shape of the depth profile is more representative for phase precipitation than P segregation on grain boundaries. The proposed P distribution is suitable to predict the mechanical properties, i.e. toughness.

Kryukov A, "The state of the art of WWER type RPV: radiation embrittlement and mitigation", published in SPECIALISTS MEETING ON IRRADIATION EFFECTS AND MITIGATION, Vladimir, Russia, 15-19 September, 1997

The paper summarizes the results obtained in the TACIS 91 program.

The different types of WWER RPV-s have different problems. Some of the WWER 440-V230 RPV (old generation) have high phosphorus and copper content in the welds, lack of surveillance program and 9 vessels are not cladded. Few difficulties occur with the second generation WWER-440 V213 reactors, and the high nickel content of some weld of the WWER-1000 reactors also discussed.

Most surveillance and research data are showing the good correlations between transition temperature shift (ΔT_k) measured and calculated accordance with Russian Guide. The paper describes the distribution of the irradiation assisted phosphorus precipitations near the grain boundary, the dependence of DBTT shift (increases) on nickel content.

After the annealing of the first generation vessels the re-embrittlement rate is generally different from the first embrittlement rate. The paper discusses the use and comparison of Lateral or Conservative model.

During the test program efforts was made to find good correlation between standard and subsize Charpy specimens.

The paper recommends further work on the following tasks for WWER-440 life management: to elaborate new Codes on the modern database, to justify the model for re-embrittlement (after annealing) prediction and to create International Data Base on Aging Management and Life Extension (IAEA).

Morozov A.M., Nikolaev V.A., Jurcseko EV., "Effect of alloying and impurities elements on radiation embrittlement of WWER-1000 reactor pressure vessel materials", The paper presented: in proc. of the Honorary Conference on 100 anniversary of academic Kurcsatov and Aleksandrov.

The paper discusses the effect of alloying and polluting elements. It states that the 1/3 exponent in the formulae used to predict the embrittlement rate from the chemical composition is not valid if the nickel content higher as 1.5%. This case no saturation observed and the value of the exponents can be as high as 0.8.

Ni, Mn, Si, S, Mo, P, Cu, are the elements affect the radiation embrittlement sensitivity.

Kolaev A.V., Badanin V.I., "Boron Effect on Radiation embrittlement of low-alloyed steel", presented in Atomnaja Energija, vol 41. 1976 december p.422.

The 15H2MFA RPV steel has been alloyed with 0.04% boron. The boron content considerably increased the radiation embrittlement rate.

Nikolaev V.A., Badanin A., "Impurity elements influence on embrittlement of ferrite pearlite steel after neutron irradiation and thermal ageing", presented in Metallü 1975 vol 2. p. 126.

The paper compares the degradation effect of 4 different Ni alloyed 15H2MFA steels. Linear correlation have been found between 350°C irradiation and at 500°C 500 hours thermal ageing in the function of the nickel content. This proves the theory, that Nickel in the low alloyed steels causes thermal ageing, and the low temperature irradiation accelerates the thermal ageing.

Vacek M., Novosad P. and Havel R., "Radiation Embrittlement and Annealing Recovery of CrNiMov Pressure Vessel Steels With Different Copper and Phosphorus Content", published on Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels: An International Review (Fourth Volume), ASTM STP 1170, ASTM, Philadelphia, 1993, p. 172-182.

The paper deals with degradation dependence on chemical composition of the CrNiMoV steel. Specimens were taken from different heats having different chemical composition. Tensile, hardness, Charpy impact energy and dynamic fracture toughness transition temperature measured on unirradiated and irradiated specimens.

The transition temperature shift ΔTT_{41J} strongly increased with increasing Cu content and the effect of P was less significant.

The shift of impact energy and fracture toughness of all samples showed a linear relationship. The paper also presented studies on postirradiation annealing. It used a relation formula to provide the tensile property and fits it to the experimental data. Recovery of hardness was observed for all temperatures from 350 to 600°C. Cu has not effect on annealing response and P slightly decrease recovery.

8 Environmental Factors

A collection of papers on Environmental Factors influencing on the reactor pressure vessel (RPV) materials have been reviewed with the aim to track the evolution of this active field of research in the last decade. The revised papers present only part of the studies in the area. Concerning the task to recognize the leading organizations in the field, to have a clear picture of the advances achieved so far and, to determine the existing gaps and open issues which allow identifying future research activities needed, it is needed to revise all papers in the area. For example the questions regarding the reactor dosimetry and determination of neutron fluence is presented by some papers of the INRNE. The research activities needed are taken from a review paper prepared after a special workshop dedicated to the state of the art and future necessities. The conclusions are based on 38 papers, collected and electronically stored.

The environment factors of radiation damage of RPV metal discussed in the papers are:

- neutron fluence – fluence factor,
- fluence rate (fast neutron flux) – fluence rate factor,
- gamma flux (fast neutrons, thermal neutrons and gamma ray) – gamma rays factor,
- chemical composition of coolant (water) – chemical factor,
- irradiation temperature – temperature factor.

8.1 Consolidated Conclusions

- The influence of different environment factors (neutron fluence, fluence rate, gamma rays, chemical composition and temperature) is difficult to be separated due to the complicate conditions. That is why the investigations try to fit an empirical presentation of the factors' dependenceл
- The papers devoted to the metal properties changes under irradiation are not well completed with data of the sources of uncertainties related with the neutron field parameters, chemical composition, temperature as well as estimations of their contribution in the total uncertainty.
- The irradiations are performed in different neutron fields without possibility to vary the parameters, only the fluence could vary and be defined well.
- The Semi-mechanistic model seems to give satisfactory description of the DBTT shift dependence on fluence rate, temperature, Cu-precipitation and P-segregation. More recent data are needed to verify the model. The surveillance data used for verification have to be reviewed and re-evaluated.
- The determination of the neutron irradiation conditions (neutron flux, spectrum) of irradiation at NPPs RPVs has to be presented more in detail for comparative purposes. Since a semi-mechanistic model for DBTTshift takes into account the fluence rate the limits of its application has to be demonstrated.
- It is recommended the uncertainty of neutron fluence determination of the sets with Charpy specimens, and the uncertainty of embrittlement/damage parameters based on the neutron fluence to be included in the Database of surveillance specimen results data for WWER-1000 type RPVs. This will be favourable for a comparability of the data from different NPPs.
- The neutron fluence adjustment procedure based on the induced activity of used neutron monitors has to be a continuous task. This method has to be harmonized

applying sharing of experiences, inter-comparisons, joint experiments and calculations.

- Establishing a link between damage production and the resultant change in mechanical properties is very useful for LT prognosis.
- The calculations of the neutron flux responses for the vessel surveillance, especially in locations behind the vessel are recommended to be done by the appropriate BGL library.
- Harmonization of calculation and measurement methods for use in WWER, PWR and BWR design and structural integrity assessments is recommended.

8.2 Open Issues

- There is not presented enough information on validation of RPV and surveillance specimens' neutron fluence. There are not presented uncertainty data of neutron field parameters and their contribution to the damage evaluation results.
- Trepans taken out from decommissioned NPPs are a good material base for material properties examination and neutron fluence determination using the measured induced activies data. Based on all these data more appropriate correlation between metal behavior and neutron irradiation could be established.
- To overcome the insufficiencies of the surveillance methodology, improvements based on enhanced calculational reactor dosimetry tools, innovation of experimental methods and approaches as well as creation of new benchmarks are needed.
- To be formulated the knowledge and skills necessary regarding radiation damage of GenIV RPV materials is recommended.

8.3 Reviewed papers and summaries

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Amaev-014, Amayev A.D., Kryukov A.M., Levit V.I., Sokolov M.A., "Radiation stability of WWER-440 vessel materials", An international review (Fourth Volume). Philadelphia, PA (United States), ASTM STP 1170, American Society for Testing and Materials. 1993, p. 9-29

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Neutron fluence – fluence factor

- Experimental studies

Main conclusions based on the results of experimental studies devoted to the fluence factor causing radiation embrittlement of the WWER RPV metal, are presented below.

Experimental study (Amaev-014) of radiation embrittlement of the WWER-440 RPV base and weld metal show:

- the coolant at irradiation temperature of 270 ^{0}C does not lead to any additional increase in the irradiation transition temperature
- saturation of radiation embrittlement does not occur for irradiation of specimens of base and weld metal with neutrons, E> 0.5MeV, to 7 x 10^{20} n/cm^2 (3 times of radiation LT).
- The radiation embrittlement of the weld metal is determined mainly by: the individual contributions of P, Cu, and P-Cu interaction, and the direct accumulation of radiation deffects.
- The increasing of radiation hardening and radiation embrittlement of the RPV metal at neutron fluence higher than 4 x 10^{20} n/cm^2 is explained by the radiation defects (formation of clusters, dislocation loops, small voids) produced.
- The radiation embrittlement is stronger at a lower neutron flux within the range of fluences from 1 to 5 x 10^{19} n/cm^2 .

The transition from ductile to brittle condition for cladding metal was found to be typical for a ferritic-perlitic steel (Badanin-002). This is as an answer to the question "Is there a favorable effect of the WWER RPV-440 steels cladding.

Results of experimental studies (Golovanov002) at KORPUS facility, RIAR, Dimitrovgrad, Russia show that the shift of the radiation embrittlement temperature Tk of WWER-1000 vessel base metal is less than normative change at a neutron fluence of 3.10^{19}<F< 11.10^{19}cm^{-2}.

The structural changes of surveillance specimens of the base (15Kh2MFA) and weld (Sv-10KhMFT) metals of the WWER-440/213 steels, irradiated by fast neutron fluence (E > 0.5 MeV) beyond design operation (from 8.66 x10^{20} n /m^2 up to the fluence of 8.66x10^{24} n /m^2), investigated by transmission electron-microscopy and fractographic studies (Kuleshova-020), show that there is an evolution in radiation-induced structural behavior with radiation dose increase, which causes a change in relative contribution of the mechanisms responsible for radiation embrittlement of RPV materials.

The irradiation causes essential increase in the yield strength and in critical brittleness temperature of the steel (Figs. 8.1 and 8.2). At fluences above (1–1.3) x10^{24} n/m^2 the character of yield strength dose dependence has no formal correlation with transition temperature shift in the range of fluences, where yield strength decrease and DBTT growth (especially for weld metal) are observed.

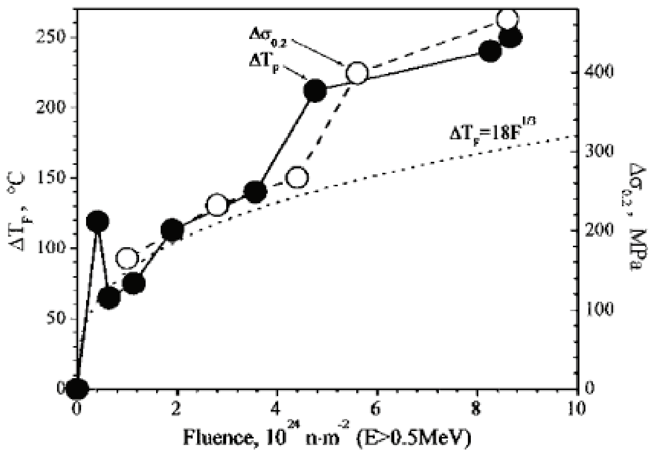

Fig. 8.1. Ductile-to-brittle transition temperature and yield strength of the base metal as a function of fast neutron fluence in comparison with the standard reference dependence specified for evaluation of radiation embrittlement for WWER-440 base metal (DTF = 18F1/3) (Kuleshova-020)

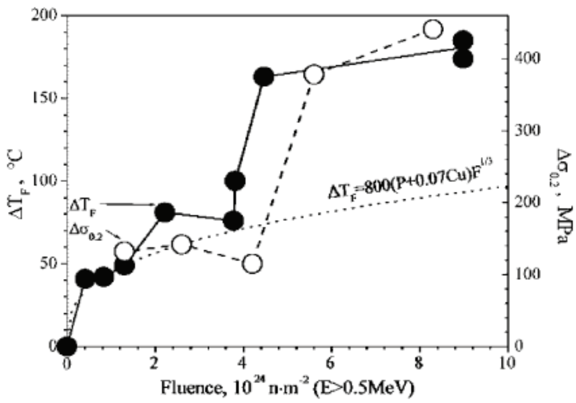

Fig. 8.2. Ductile-to-brittle transition temperature and yield strength of the weld metal as a function of fast neutron fluence in comparison with the standard reference dependence specified for evaluation of metal radiation embrittlement for WWER-440 weld metal (DTF = 800(P + 0.07Cu)F1/3) (Kuleshova-020)

The empirical model based on the Russian standard formula with accounting for the phosphorus accumulation on the grain boundariest is in a good consistency with experimental data is developed, Fig.8.3 (Kuleshova-020).

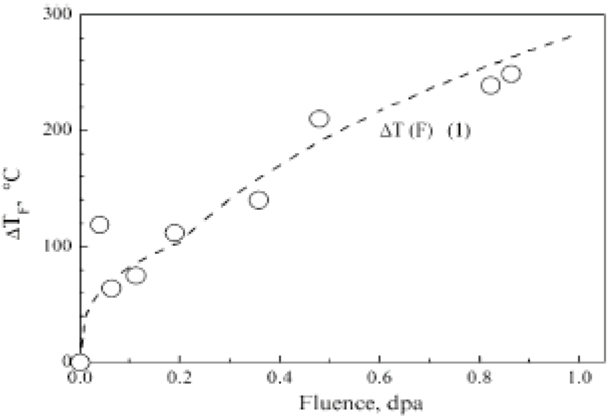

Fig. 8.3. Comparison of calculated DBTT dependence on damage dose for the base metal with the experimental data (Kuleshova-020)

The through-wall attenuation of toughness properties of the WWER and PWR RPV base and weld metal, is greater than expected. This conclusion is based on the results of experiments of specimens (Charpy V-notch impact, pre-cracked Charpy size static fracture toughness and instrumented hardness specimens) irradiation at RIIAR, Reactor RBT-6, irradiation facility KORPUS, Dimitrovgrad, Russia (Kytka-022).

Operational induced affections of the RPV metal microstructure after undergoing highest neutron irradiation exposure are observed on trepans taken out from the Novovoronesh NPP Unit-2 with WWER-440/230 RPV base metal and weld No. 4, after 16 effective full power years, $6.5x10^{19}$ n/cm^2 (Maussner-001). The extensive examination of the weld metal has shown the existence of dislocation loops ('blacks dots') with diameters <010 nm in an equal or slightly higher number density.

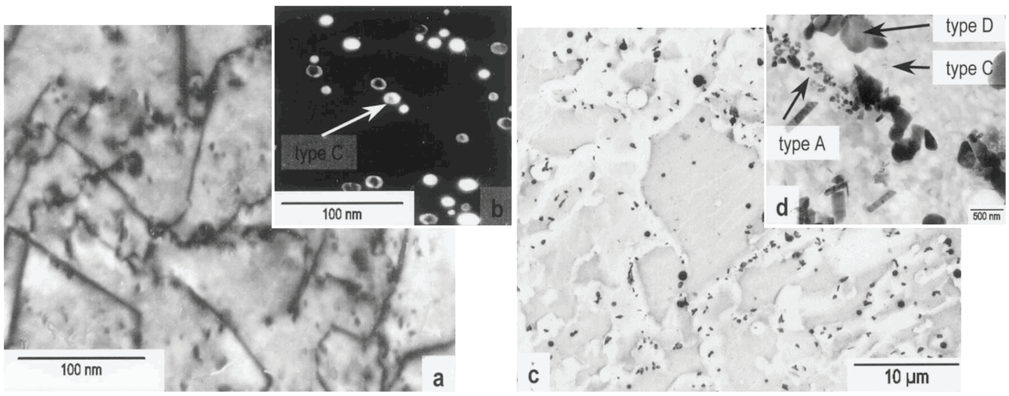

Fig. 8.4. (a) Microstructure of the irradiated weld metal, disk shaped fine carbides, fluence: 2.7_10^{19}:cm^2 (E>0.5 MeV). (b) Detail from Fig. 15a, dark field image obtained from MC reflection. (c) Microstructure of the irradiated weld metal, carbides, fluence: 2.7_10^{19}:cm^2 (E>0.5 MeV), extraction replica. (d) Detail from Fig. 15c, carbides of types A, C, and D in the inset. (Maussner-001)

Small quantity of copper (0.06 at.% Cu) remaining in the matrix of copper steels, used in a WWER 440 reactor would produce little re-embrittlement on subsequent irradiation (Miller-002).

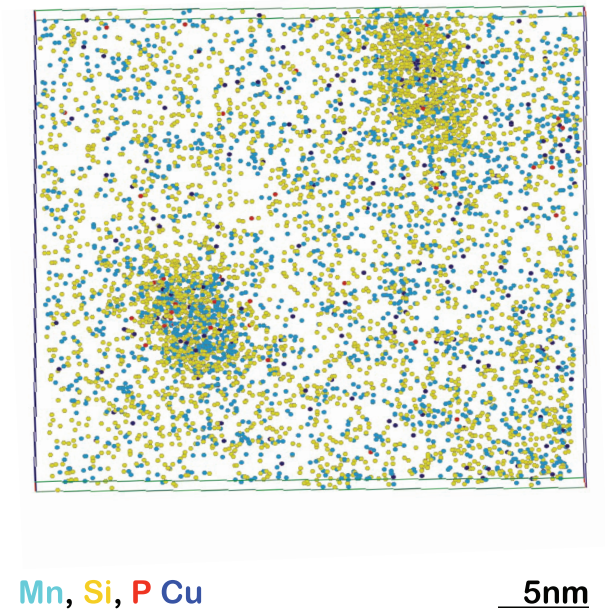

Fig. 8.5. Microstructure analysis: Atom maps of the solute distribution in the neutron-irradiated weld material. Two spherical Mn- and Si-enriched regions are evident. (Miller-002)

Relation between neutron fluence and damage effects

The relation between neutron fluence and damage effects is extremely important issue and is under investigation. Logic path recommended for assessing Eq. (3) is shown in Figure 8.6.

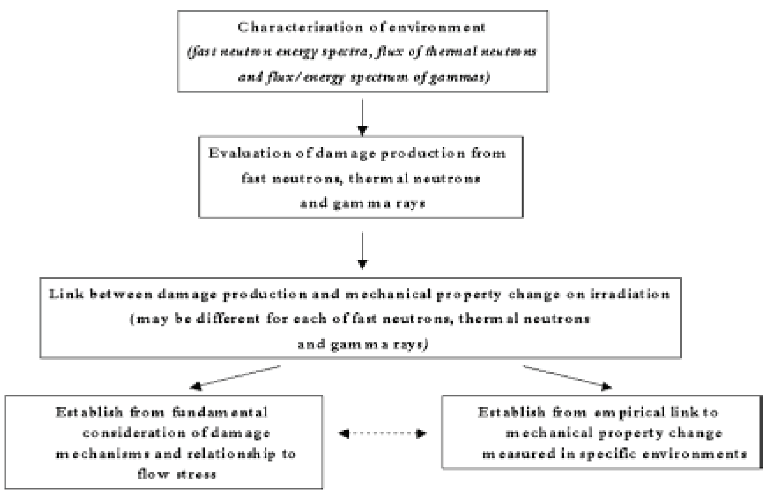

Fig. 8.6. Logic path adopted by TAGSI in assessing Eq. (3) (001Knott)

Embrittlement effects produced by neutrons result from atom displacements, which produce point defects and defect clusters. Following (001Knott) the total change in the fracture properties due to a combination of neutron and gamma displacements mechanisms of hardening, as characterised by a shift in transition temperature $\Delta T_{40\ J}$ is defined by the form

$$\Delta T_{40\ J} = B' + A^*F_T(D_F + k^*D_{Th} + k_\gamma D_\gamma)^{0.5} \qquad \text{(eq.3 of the paper)}$$

where B' and A* are constants fitted to the data, F_T is the correction factor to account for the dependence of embrittlement on the irradiation temperature, k* and k_γ are empirical factors which represent the effect of the thermal neutron dose, DTh, and gamma induced dpa dose D_γ, on mechanical properties relative to that of the fast neutron dose.

There is no internationally accepted practice on determination of the flux and energy spectrum of gamma rays.

Link between damage production and the resultant change in mechanical properties is presented in Fig 8.7 (001Knott).

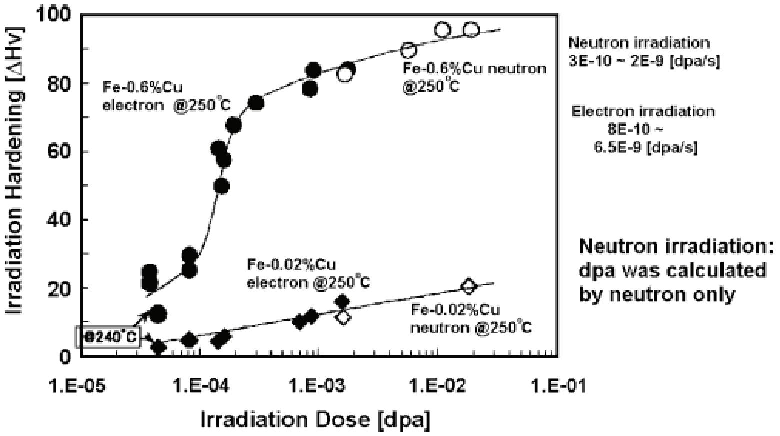

Fig. 8.7. Plot of the irradiation hardening versus irradiation dose (dpa) observed by Tobito et al. [21] in fast neutron or electron irradiated model Fe–Cu alloys (001Knott)

LT management based on neutron fluence

The neutron fluence data are used to predict the metal damage, to assess/verify safety margins for PLIM, to evaluate the remaining RPV lifetime, to support PLIM strategies (Ilieva-008 or Ilieva-008- ICRESH 2005). Appropriate low-leakage and low-fluence core loading patterns at Kozloduy NPP reactors are being determined so that RPV neutron exposure be minimized. The low-leakage loading scheme (loading scheme with 36 burned-up assemblies in the periphery of reactor core) reduces the neutron fluence with about 30 to 40% toward the standard loading. The low-fluence loading scheme is scheme with 36 dummy (steel) cassettes put in the reactor core periphery instead of fuel assemblies. It reduces the neutron exposure up to 70% toward the standard loading.

The current temperatute of radiation embrittlement of WWER-440/230 reactor vessel of all units 1 to 4 of Kozloduy NPP are determined based on Russian standard formula and calculated and validated RPV neutron fluence (Ilieva-new13_U1-Tk).

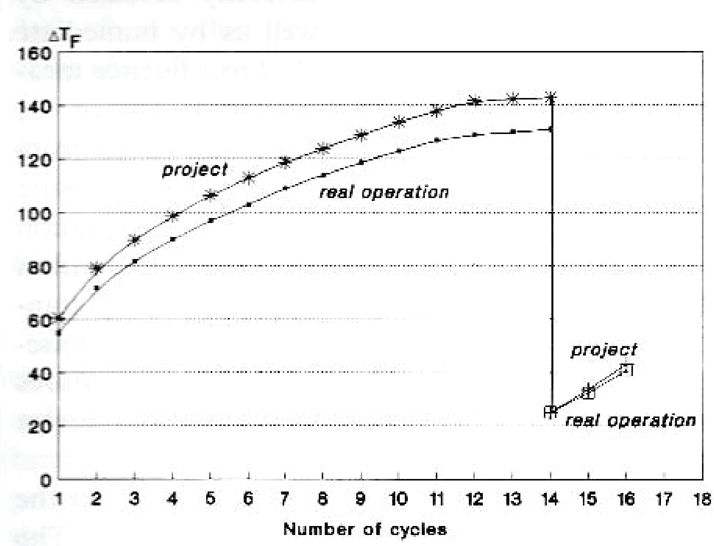

Fig. 8.8 Shift of the transition temperature due to neutron irradiation in maximum overload directions (30 deg) of the RPV weld 4 of Unit 1, Kozloduy. NPP (Ilieva-new13_U1-Tk).

Neutron fluence rate (neutron flux) – fluence rate factor

As nuclear power plants age, neutron embrittlement of reactor pressure vessels becomes a crucial consideration for continued safe plan operation. Most experimental irradiation studies are conducted in test reactors so as to simulate many years of reactor operation in the space of weeks or at most months. The validity of such accelerated results is under consideration and investigation. The fluence rate (flux) effect is a factor that contributes to the total uncertainty in the quantification of the RPV embrittlement level (Ballesteros-flux_effects). The effect of fluence rate (flux) on embrittlement behaviour is a complex phenomenon and it dependent on the alloy composition, the fluence and the irradiation temperature. In particular, copper, nickel and phosphorus contents of the steel are important variables. The fluence rate effects have been observed in both research and commercial reactors. A further impediment to testing for fluence rate effects is the general tie between fluence rate (flux) level and neutron spectrum. Decoupling these two factors experimentally is difficult if not impossible.

Observation, effects of fluence rate factor

The radiation embrittlement (flux effect) of the WWER-440 RPV materials is stronger at a lower neutron flux:

- within the range of fluence from 1 to 5 x 10^{19} n/cm^2 (in this case the neutron fluxes are 4x10^{11} n/cm^2.s, and 4 x 10^{12} n/cm^{2}s) (Amaev-014).
- at a same fluence, 8.10^{18} cm^{-2}, E>0.5 MeV, there is observed a significant difference in DBTT shift values produced for 2 fluence rates (fluxes) that differs each other 10 times, at the Kola NPP, Russian Federation and in the Rovno NPP, Ukraine (Debarberis-001, Debarberis-007) (Fig.8.9).

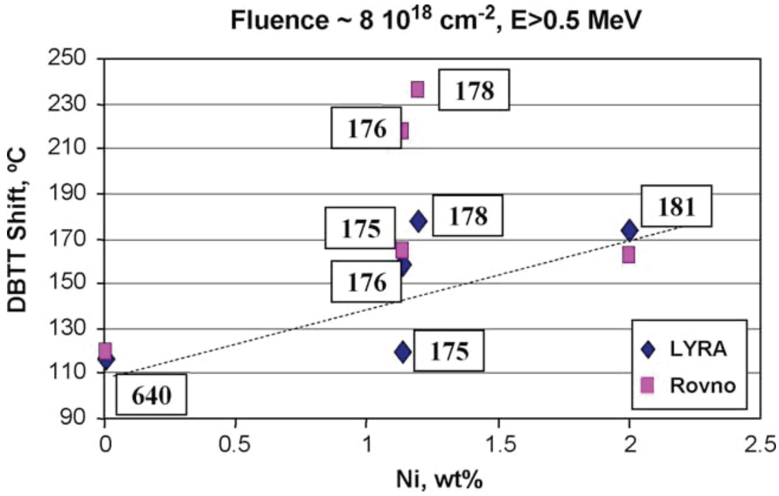

Fig. 8.9 Comparison of LYRA–ROVNO data sets: a significant difference in DBTT shift values produced for 2 fluence rates (fluxes) that differs each other 10 times (Debarberis-001)

- at radial fluence diference up to a factor 3, at the experiment in the power reactor WWER-440 of Rheinsberg NPP – 2, Charpy V-notch, SENB and CT specimens of Russian RPV base and weld metals. (Boehmert001) - Fig. 8.10 (Fig.8)

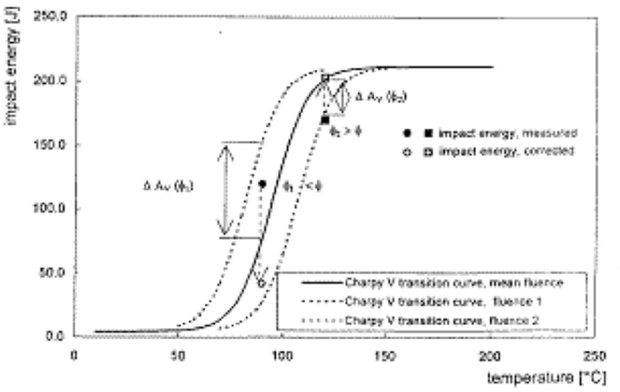

Fig. 8 Schematic representation of the correction of measured Charpy V impact energies of specimens irradiated up to different fast neutron fluencies

Fig. 8.10 Schematic representation of the correction of measured Charpy V impacte energies of specimens irradiated up to different fast neutron fluences (Boehmert-001)

The flux effects are strongest for high Co, Ni and P precipitation (Debarberis-001, Debarberis-007): Examples of dependence/no dependence on fluence rate are presented based on experimental studies of model alloys with varying levels of Co, Ni and P (at the LYRA facility in the High Flux Reactor at JRC-IE at Petten).

Experimental study of radiation embrittlement of the WWER-440 metal samples show that for steel with low Ei<0,018% impurities (Ei=%P+0,07%Cu) content the transition temperature shift (Fig. 4, Nikolaev V-001,) does not depend nor on neutron energy variation (spectral index varies within 0.84 - 1.9) nor on flux (within 10^{17}-10^{16} n/m^2) (Nikolaev V-001, comparison of research reactor BBP-M experimental data results with experimental surveillance set data of Russian Novovoronez NPP 4, Kolskaia NPP 3&4, Rovno NPP 1, Finland Loviisa NPP, Bulgarian Kozloduy NPP 2, Czech Bohunice NPP 3 and Ducovane NPPs 2-3-4, and German Greiswald NPP).

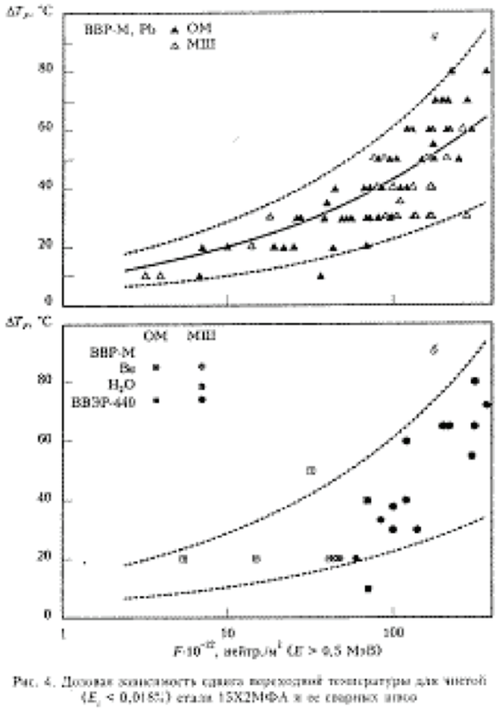

Figure 8.11 (Fig 4). The transition temperature shift for steel with low impurities content (Nikolaev V-001)

But for steels with higher impurities Ei, P>0.020 or Ei > 0.045 (as base metal 15Х2МФА and welds (АН-42, Cb-10ХМФТ)) the radiation embrittlement increases with spectral index increasing (Fig. 8.12 (Fig 5), Nikolaev V-001).

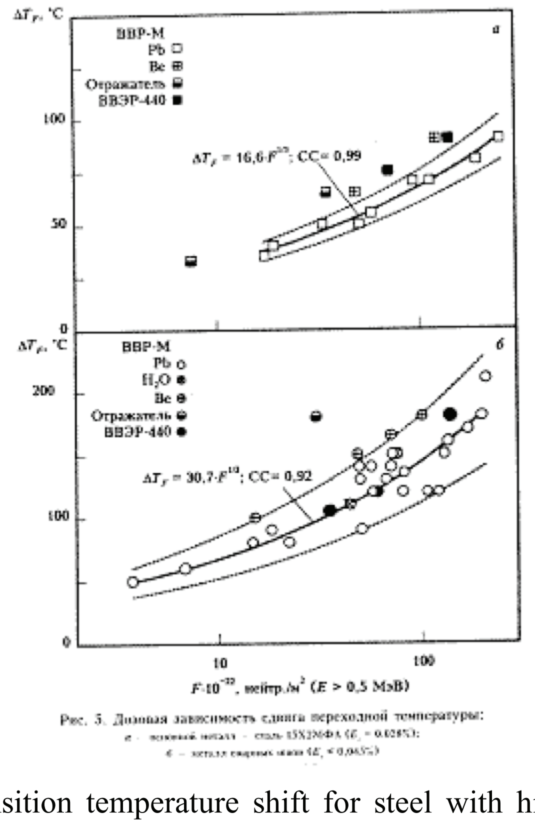

Figure 8.12. The transition temperature shift for steel with higher impurities content (Nikolaev V-001)

Description of the fluence rate factor: empirical and semi empirical

A semi-mechanistic irradiation shift model is developed for DBTT shift. It takes into account the fluence rate.

The dependence of the DBTT shift on the fluence rate, Cu-precipitation and P-segregation, and temperature is represented by so called semi-mechanistic model (Debarberis-001, Acosta-020). The temperature effect is represented by the right multiplicative term to the Semi-mechanistic model.

$$DBTT_{shift} = \left\{ a \cdot \Phi^{0.5} + b \cdot \mathrm{Cu} \cdot \left[1 - e^{-\Phi/\Phi_{sat}}\right] + \frac{c \cdot \mathrm{P}}{2} \cdot \left[1 + \tanh\left(\frac{\Phi - \Phi_{start}}{d}\right)\right]\right\} \cdot \left[\frac{F}{e^{\left(-\frac{E}{kT}\right)}}\right]$$

The fluence rate effect is taken into account by the term FF.

$$DBTT_{shift}^{LF} = DBTT_{shift}^{HF} + FF$$

When $\Phi_{start}^{LF} \approx \Phi_{start}^{HF}$

$$FF = b_1 \cdot \mathrm{Cu} \cdot \left[e^{-\Phi/\Phi_{sat}^{HF}} - e^{-\Phi/\Phi_{sat}^{LF}}\right]$$

The comparison of the DBTT shift evaluated by the semi-mechanistic model and experimentally determined is a good substantiation of the improved Semi-mechanistic model.

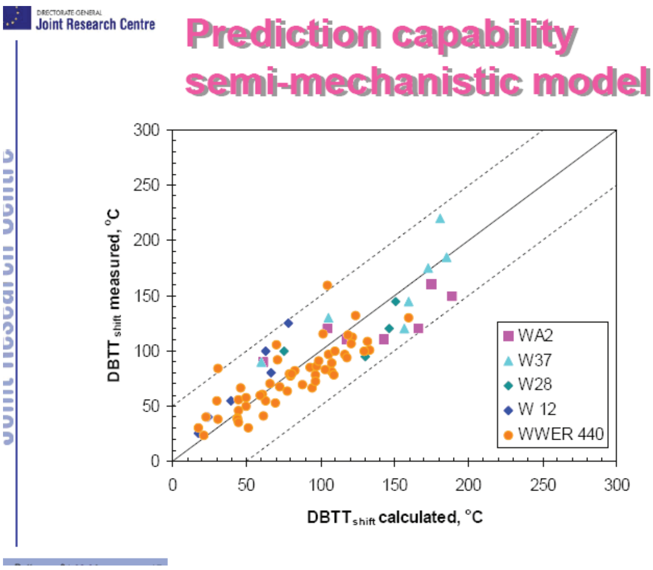

Fig 8.13. Comparison of the DBTT shift determined by the semi-mechanistic model and experimentally (Acosta-020)

The results of experimental studies (Debarberis-007) of model alloys carried out at different fluence in the LYRA facility in the High Flux Reactor at JRC-IE at Petten, in the Kola NPP, Russian Federation and in the Rovno NPP, Ukraine were used for the semi-mechanistic model verifying and development.

The proposed model confirms the observations that fluence rate dependence mainly occurs at intermediate fluences (far from saturation) and for high Cu and P contents,

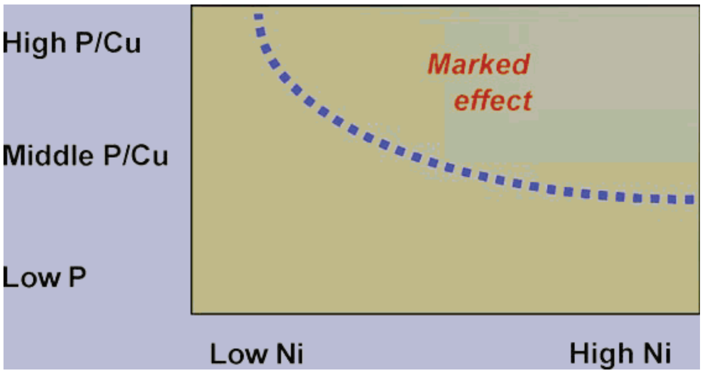

Fig.8.14. Observed field for fluence rate effects on model alloys (Debarberis-001)

Comparison of the LYRA data and Rovno data demonstrates that fluence rate is responsible for a marked effect for certain model alloys, in particular:

- No effects are observed for model alloys with low nickel content, or for alloys with high nickel content in combination with low phosphorus content.
- On the contrary, marked fluence rate effects are observed for model alloys with high nickel contents in combination with phosphorus.
- The Semi-mechanistic model, based on the nickel-phosphorus synergism, could be used for prediction of neutron embrittlement.

Neutron spectra influence

A further impediment to testing for fluence rate effects is the general tie between fluence rate (flux) level and neutron spectrum. Decoupling these two factors experimentally is difficult if not impossible (Ballesteros-flux_effects).

The use of the displacement per atom (DPA) units for WWER steel damage evaluation seems more general and adequate measure since it takes into account the neutron spectrum (Ilieva-004_Lomakin, Ilieva-new12_Voloshenko)

Gamma flux (fast neutrons, thermal neutrons and gamma ray) – gamma rays factor: Mechanical property change in environments with high fluxes of gammas and thermal neutrons

The total change in the fracture properties due to a combination of neutron and gamma displacements mechanisms of hardening, as characterised by a shift in transition temperature $\Delta T_{40\,J}$ is defined by the form

$$\Delta T_{40\,J} = B' + A^* F_T (D_F + k^* D_{Th} + k_\gamma D_\gamma)^{0.5} \qquad \text{(eq.3, 001Knott)}$$

where B' and A* are constants fitted to the data, F_T is the correction factor to account for the dependence of embrittlement on the irradiation temperature, k* and k_γ are empirical factors which represent the effect of the thermal neutron dose, DTh, and gamma induced dpa dose, $D_{\gamma,}$ on mechanical properties relative to that of the fast neutron dose.

The ratio of gamma quanta damage dose to that caused by the neutrons is below 7% at location just next to the WWER-1000 reactor vessel. This results is based on neutron and gamma transport calculations for the WWER-1000 reactor vessel mock-up created at the reactor LR-0 at NRI/Rez, Czech Republic (Ilieva-009, Ilieva-010).

Chemical composition of coolant (water) – chemical factor

It is shown that the influence of the coolant at irradiation temperature of 270^0C does not lead to any additional increase in the irradiation transition temperature of the investigated materials (Amaev-014).

Join review (Buckthorpe-002) by the UK, Russia, Czech Republic and EC on current codes and standards regarding corrosion fatigue and stress corrosion cracking in steels for LWR, BWR and WWER, review of LWR and WWER plant experience and R&D results shows that more data and a clear understanding are required in order to write code provisions particularly in the area of high cycle fatigue. Harmonization of calculation methods for use in WWER, PWR and BWR design and structural integrity assessments is recommended.

Study (Buckthorpe-002) of the effect of reactor water environment factor shows that one or two parameters, the oxygen content of the water and the sulphur content of the steel, can be so limited that corrosion fatigue effects become minimal.

Irradiation temperature – temperature factor

The influence of temperature factor is taken into account in the Semi-mechanistic model of the DBTT shift (Acosta-020). More works on verification are needed.

The cladding metal fracture toughness in the whole temperature range (–196 to 20 ^{0}C) is higher than would be in case of heat-resistant ferritic-perlitic steel (Badanin-002).

Neutron fluence calculation

The collected papers represent only part of the activities of INRNE on WWERs' neutron fluence determination. There are many other papers devoted to the same issue reporting the experience of the countries operating WWERs type of reactors. They are published in some journals as Nuclear Science & Engineering, Nuclear Engineering & Design, Progress in Nuclear Energy, local country' journals etc. as well as in the proceedings of the International Symposium on Reactor Dosimetry (ISRD) held on every three years which is the unique world forum on RD.

The main conclusions presented below are based on a state of the art review of Reactor Dosimetry used for reactor pressure vessel irradiation damage assessment and lifetime evaluation of the Russian type WWER reactors "Reactor Dosimetry for WWERs RPV Lifetime Assessment", Krassimira Ilieva, Sergey Belousov, Antonio Ballesteros, Bohumil Osmera and Sergey Zaritsky, accepted for publication in*:* Progress in Nuclear Energy, PNUCENE-D-07-00043, Elsevier Editorial, 2008, In print

The radiation monitoring is an approach for non-destructive determination of the neutron exposure and prediction of the radiation damage of RPV and its internals and claddings, and therefore for planning ways for improving both operation and plant life extension. The reactor dosimetry, through calculations and measurements, provides a good enough description of the neutron field parameters.

The main difficulties regarding the neutron fluence determination are indirectly related to the surveillance methodology:

- no direct measurements on RPV and internals;
- shortcomings of surveillance assemblies' design and location,

- relatively short half life (312 days) of the radio-nuclide ^{54}Mn which is available in the metal;
- construction design uncertainties.

To overcome the insufficiencies of the surveillance methodology, improvements based on enhanced calculational reactor dosimetry tools, innovation of experimental methods and approaches as well as creation of new benchmarks are needed.

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.

Harmonization of calculation methods for use in WWER, PWR and BWR design is recommended.

The reactor dosimetry improvements will reduce the neutron fluence uncertainty and thus substantiate the extension of NPP lifetime.

These conclusions are summarized within common EC and IAEA projects by specialists in Reactor Dosimetry from countries operating WWER reactors such as Russia, Ukraine, Czech Republic, Finland, Hungary, Slovakia and Bulgaria, together with specialists from Western European countries such as France, Spain, Germany, Belgium, the Netherlands, and UK, operating PWR and BWR type reactors.

Neutron fluence calculation methodology

Reactor Dosimetry is an important field since it provides neutron fluence data that are used for the evaluation of the material irradiation damage, and therefore it is a crucial input for the safety assessment of any nuclear reactor. The reactor dosimetry is needed to study material's stability, to allow prediction of damage, to quantify material damage, to assess/verify safety margins for Plant Life Management (PLIM). The PLIM needs reliable estimation of radiation field parameters including their uncertainty for reduction of the conservatism of material damage assessment and RPV lifetime estimations. The importance of these issues requires that the increasing of the accuracy and reliability of the RPV fluence determination be a continuous task.

The referred publications represent the of the neutron fluence determination methodology developed and continuously improved at the Institute for Nuclear Research and Nuclear Energy of Bulgarian Academy of Sciences (INRNE), Sofia, Bulgaria (Ilieva-new6_YUNSC). It present the objectives of Surveillance program, calculation and verification of RPV neutron fluence, RPV benchmarks, and RPV LT prognosis.

Calculational modeling of neutron detectors' activity accounting for local power variations is developed for neutron fluence verification and assessment (Ilieva-new1_NSE_Modeling). This modeling is being applied from 1994 for all neutron fluence calculations for Kozloduy NPP units 1-6.

Problem oriented neutron cross section libraries BGL440 and BGL1000 appropriate for neutron fluence determination on the reactor vessel for WWER-440 and WWER-1000 respectively is generated. The calculation of the neutron flux responses for the vessel surveillance, especially in locations behind the vessel is recommended to be done by the appropriate BGL library (Ilieva-new4_BGL, Ilieva-new4_BGL_libr_paper, Ilieva-005, Ilieva-006).

Following the results of neutron detectors measurements an uncertainty of 10% is proposed to be the requirement for the neutron exposure monitoring by ex-vessel detectors as for RPV neutron fluence verification so for neutron flux and radiation damage dose determination for each surveillance sample (Ilieva-004_Lomakin).

There are series of studies and improvements of the neutron fluence methodology done during the years of experience as:

- a pin-wise presentation of the neutron source which describes more correctly the source negative radial gradient and decreases the fluence evaluation by 10% for WWER-440 (Ilieva-new5_NSE_Testing) and by 20% for WWER-1000 (Ilieva-new9) in comparison with the assembly-wise source presentation;
- developed an interface software package for RPV neutron fluence calculation converting the hexagonal reactor core codes' output data to input data for the discrete ordinate transport codes, extended by capacity to process pin-wise reactor core presentation (Ilieva-new7 -software_package);
- an adjustment of the WWER-1000 RPV neutron fluence using data of the discrepancy between calculated and experimental activity data of ex-vessel detectors. Applied for Kozloduy NPP Unit 5 the adjustment decreases the RPV neutron fluence value with about 10% and reduces about twice the RPV fluence uncertainty (Ilieva-new8 -Sensitivity-WWER-1000);
- a method for diminishing (for the example by 67%) the differences between measured and calculated activities of neutron detectors based on the Least Squares Method is developed (Ilieva-011). The results could be applied for fluence adjustment.
- the uncertainty related with the multigroup approximation, used in fluence calculation is significantly lower than the reported experimental and calculational inconsistency (Ilieva-007);
- a special device for positioning of activation detectors behind reactor vessel was elaborated for neutron fluence verification (Ilieva-003).

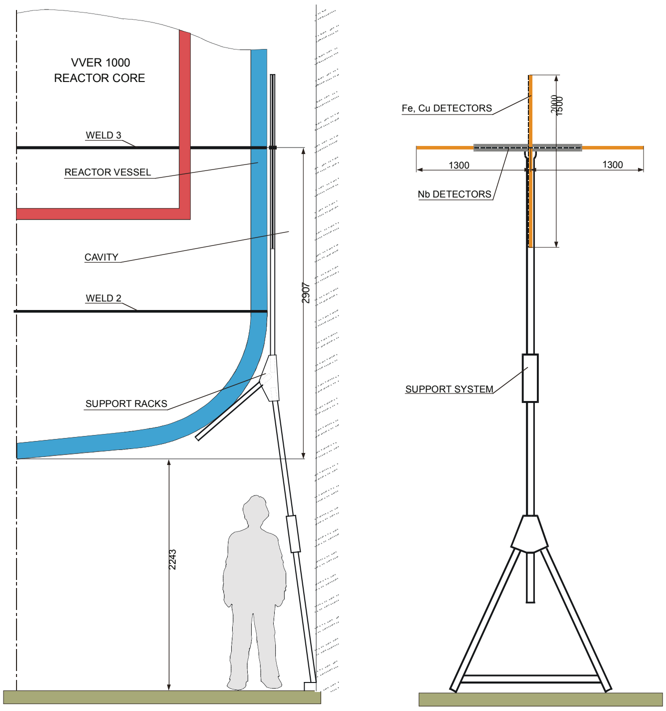

Fig 8.15. Device for positioning of activation detectors behind the RPV –additional information

Validation/verification of neutron fluence calculation

Reactor Dosimetry is an important field since it provides neutron fluence data that are used for the evaluation of the material irradiation damage, and therefore it is a crucial input for the safety assessment of any nuclear reactor. The validation of neutron fluence is a main requirement of the QA policy for safety operation of NPP. Measurement by activation detectors placed behind the vessel of WWER reactors is an unique possibility to verify/validate the calculated fluence on the vessel.

The verification of neutron fluence refers as the use of ex-vessel activation detectors on NPP so on RPV benchmarks.

Exceptional opportunity for direct verification of neutron fluence are the scraps and templates taken out from the inner wall of RPV. There are presented series of results:

- comparative analyses of calculated and experimental induced activity values of scraps from the inner wall of the Kozloduy NPP Unit 1 RPV taken out after 14th and after 17th cycle and detectors (iron, copper and niobium) irradiated during 17th cycle in the cavity behind the vessel. The consistency of the calculated and measured results is less than 9% (Ilieva-new2 RD);
- calculated and experimental induced Mn-54 activity values of the templates taken out after 18th cycle from the inner wall of the Kozloduy NPP Unit 1 RPV, and detectors (iron, copper and niobium) irradiated during 18th cycle in the cavity behind the vessel shown a consistency of the results for the inner wall of 5% and for the detectors activities behind the RPV of 15% (Ilieva-001, Ilieva-new3_IAEA_KNPP_U1).

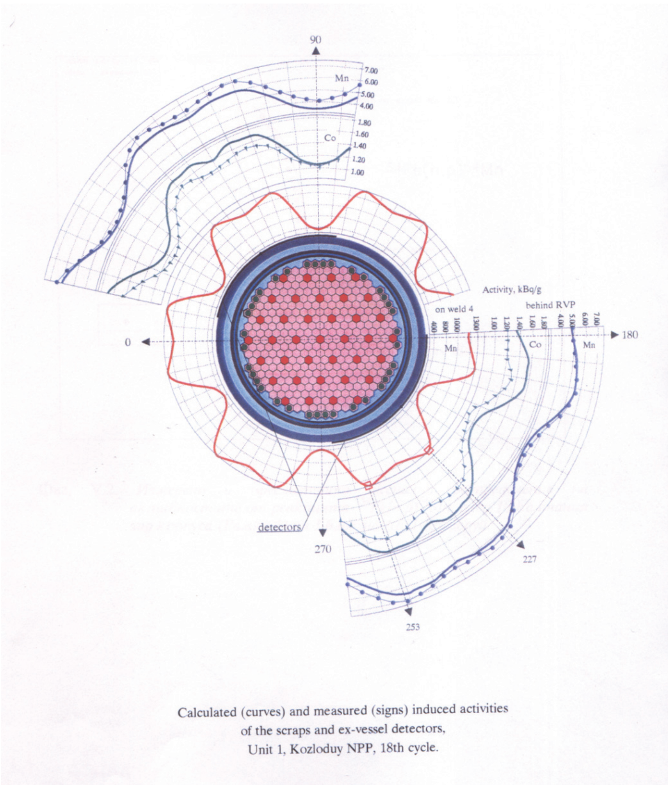

Figure 8.16 Calculated and measured induced activities of templates (scraps) and ex-vessel detectors, Unit 1 of Kozloduy NPP, 18th cycle (Ilieva-001)

Data of induced activities of sets of activation (Co, Nb, Fe, Ni, Cu, Ti and fission Np) neutron fluence monitors from the surveillance chains of WWER 440/213 RPV of the Dukovany NPP, Czech Republic irradiated during one, two or three cycles, were used for neutron fluence assessment (Novosad001).

Results from verification of neutron fluence for the WWER-1000 RPV of Kozloduy NPP Unit 6 after cycle 2 shown discrepancy less than 9% (Ilieva-003).

Calculation results for neutron fluence validated by the measured activity of threshold detectors irradiated in the vicinity of RPV support structure of Unit 4 Kozloduy NPP, are used to revise the design feature that the RPV support structure is not life-limiting component and no additional surveillance is required to justify the lifetime of the RPV support structure (Ilieva-new11-SS-Unit4-KNPP).

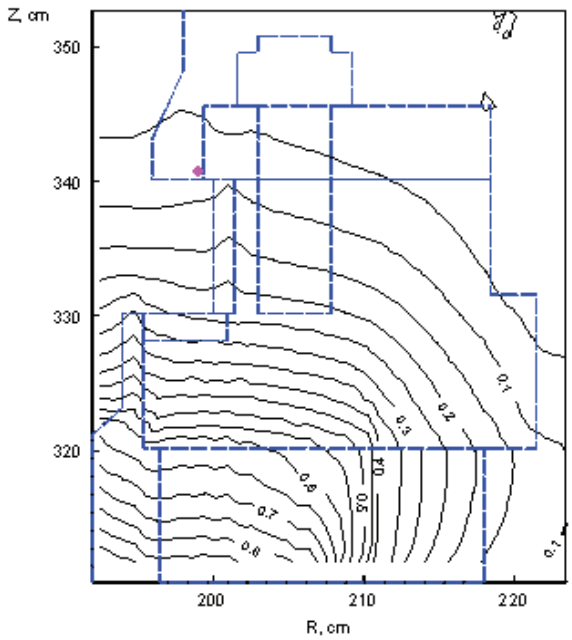

Fig. 8.17 Neutron fluence distribution in the area of the support structure normalized to the maximal value (in the down left corner) (Ilieva-new11-SS-Unit4-KNPP)

The WWER-1000 and WWER-440 RPVs' benchmarks developed at Czech LR-0 reactor, Rez, near Prague present a good base for calculation verification. Conformity analysis shows that the attenuation of neutron flux with energy above 0.5 MeV, through the WWER-440 RPV and its mock-up consists within the calculation uncertainty. The neutron flux attenuation through the WWER-1000 RPV and its mock-up consists also within the calculation uncertainty except the region behind RPV where the attenuation difference is due to the difference in the biological shielding materials (Ilieva-new10-conformity).

A good agreement between reference MCNP code and discrete ordinates DORT code results of DPA's caused by fast neutrons and gamma quanta is obtained for the simulator of WWER-1000 reactor vessel, so called WWER-1000 LR-0 benchmark (Ilieva-009).

8.4 Further References

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Neutron Induced Embrittlement of Reactor Pressure Vessels". September, 17-20, 2000, Varna, Bulgaria, CD-ROM

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Ilieva-RD-Review, Only summary, In print, "Reactor Dosimetry for WWERs RPV Lifetime, Editorial, 2008, In print

Sergey Zaritsky, Desislava Kirilova, Krassimira Ilieva, Sergey Belousov, "Progress in Nuclear Energy, PNUCENE-D-07-00043, Elsevier, Assessment

9 Mechanisms and Microstructural Evolution

This summary was prepared on the review of 32 papers. These papers were selected as most representatives for the area "Microstructure of RPV – WWER steels". Although there were much more interesting papers published during last 40 years, many of them were not appropriate archived or their content was repeated in other works.

9.1 Consolidated Conclusions

Beside these reviewed papers, new 12 papers (published in the last two years) were selected to complete this work.

Having in mind also the summary from first 10 papers (see annex 2) reviewed in 2007, following summary can be written.

The WWER reactors are most distributed type of nuclear power plants in the world. For illustration of this fact see Annex IV. The safe operation of these units as well as its effective lifetime management has to be based on deep understanding of processes going on during their operational loads and ageing. 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 40 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.

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. Comparison of results from destructive and non-destructive testing methods has to be based on deep theoretical knowledge about material microstructure. Reviewed papers were focused mostly 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.

9.2 Open Issues

The most important challenge for future is ensuring the safe operation behind the projected lifetime. Effective lifetime management based on 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 next deep study. Review papers can significantly help next generation of scientists. Huge amount of experimental result can be used very effectively by verification of different computer models directed to proper simulation of defects creation in RPV steels.

9.3 Reviewed papers and summaries

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3. *Alekseenko, N.N., Kuznecov, V.A., Nikolaev, V.A., Usatov, E.P., Fetisov, V.S.: "Hydrogen absorption and emrittlement of low alloyed steel under neutron irradiation in contact with water". In: Physical-chemical mechanics of materials N3, Naukova dumka, 1977 (in Russian)*
4. *Alekseenko, N.N., Kuznecov, V.A., Nikolaev, V.A., Usatov, E.P.:" Delayed fracture induced by hydrogenisation of neutron – irradiated steel 15Cr3MoV". In: Physical-chemical mechanics of materials N3, Naukova dumka, 1977 (in Russian)*
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9. *Debarberis, L. et al.: "Role of nickel in semi-mechanistic analytical model for radiational embrittlement of model alloys". In: Journal of Nuc. Mat., 336, 2005*
10. *Dragunov, Y. et al.: "Effect of irradiation on mechanical properties". In: Part of a report from about 1990.*
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12. *Gillemot, F., Oszwald, F., Gillemot, L., Uri, G., Pirfo, S.:" Synergetic effects of irradiation and thermal embrittlement". In: Proc. form IAEA symp., Vladimir, 1997*
13. *Gokhman, A., Bohmert, J., Ulbricht, A.:" Multi-component clustering of WWER-type RPV steels-thermomechancal aspects and impact on SANS". In: Journal of Nuc. Mat. 334, 2004*
14. *Gorynin, I., Timofeev, B., Chernaenko, T.: "Material properties degradation Assessment of the first generation WWER-440 RPV after prolonged operation". In: Proc. From SMIRT, 2003*
15. *Gorynin, I.:"Radiation induced effects of mechanical features of construction materials and observation methods". In: Proc. From 1st conf., Kiev, 1976 9IN (in Russian)*
16. *Kuleshova, E.A. et al.:" Microstructural evaluation of WWER-440 RPV steel under irradiation to neutron fluences beyoned the design operation period". In: Journal of Nuc. Mat, 342, 2005*
17. *Kryukov, A.:" The state of art of WWER type RPV: radiation embrittlement and mitigation". In: Proc. Of specialist meeting, Vladimir, 1997*
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29. *Slugen, V., Magula, V.: "The microstructural study of 15Kh2MFA and 15Kh2NMFA RPV steels using PAS, MS and TEM", In: Nucl. Eng. & Desg., 186, 1998*

30. *Pechenkin, VA., Korobeev, Y., Stepanov, I., Nikolaev, Y.: "On the mechanism of irradiation Embrittlement Enhanced in RPV steels at high neutron fluences". Proc. from 21st symp. ASTM STP, 2004.*

31. *Vodenicharov, S., Zlateva, G., Kamenova, T.: "Microstructure of the NPP Kozloduy unit 1 weld 4 metal". In: Nucl. Eng. & Desg., 191, 1999*

32. *Zlateva G., Vodenicharov, S., Kamenova, T.: "Microstructure of the metal of RPV weld". In: IAEA Report 1997.*

According to [1. Bischler]: Based on TEM results form irradiated Magnox reactor RPV weld samples – not only Cu precipitates, but also well distributed (Mn, Si) nitrides up to 50 nm were observed. Auger electron spectroscopy of refined graines (<20 micrometer) showed P and C to be main grain boundary segregants.

According to [2. Keilova]: Based on TEM - Only 2 types of Cr-rich carbide particles have been found i.e. M7C3, M23C6, in addition to vanadium carbide MC (M4C3) in studied 15KhMFAA steels.

According to [3. Alekseenko]: Positive contribution oh hydrogenisation – increase corrosion resistance at temperature <100°C.

According to [4. Alekseenko]: Positive contribution oh hydrogenisation – delayed fracture in 15Cr3MoV steels after neutron irradiation.

According to [5. Amaev]:

A) coolant temperature on level 270°C has not negative effect on neutron embrittlement of 15Kh2MFA steel

B) If the neutron fluence > $4E^{20}$n/cm^2 – noticeable increase of radiation hardening end embrittlement

C) Neutron fluence of $7E^{20}$ n/cm^2 is not enough for neutron embrittlement will be saturated.

D) Deep study of Cu and P content influence to the microstructure

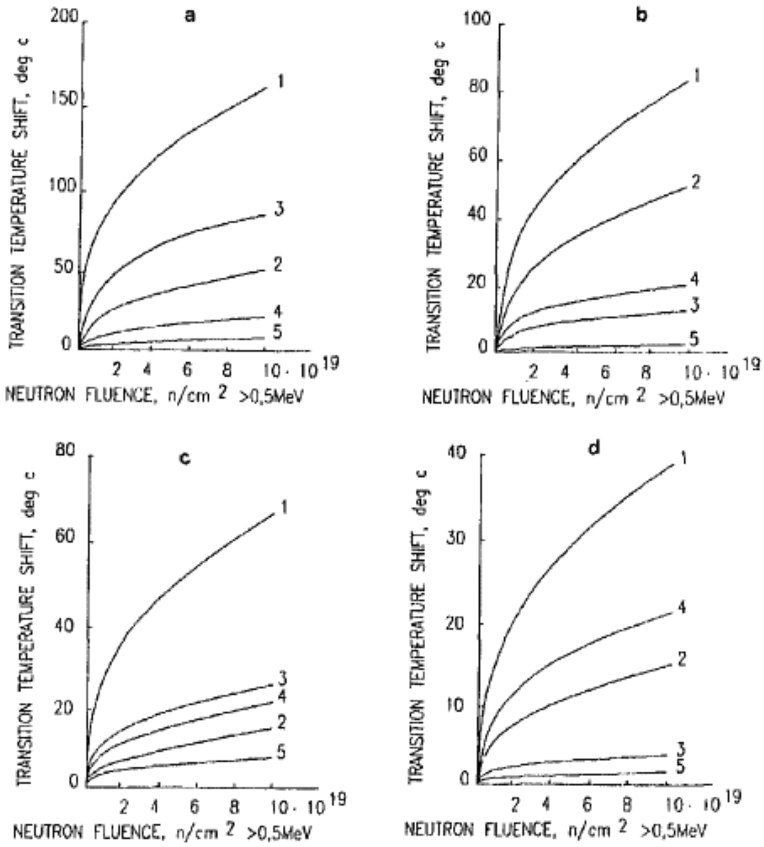


FIG. 2—*Dependence of shift in transition temperature of weld metal brittleness* (1) *and contributions to the shift from P* (2), *Cu* (5), *P-Cu interaction* (3), *and other mechanisms not including P or Cu* (4), *on fluence* F *of fast neutrons* (E > *0.5 MeV*) *at high neutron flux* ($\phi = 4 \times 10^{12}$ *n/cm²·s*). (a) *P = 0.035%, Cu = 0.22%.* (b) *P = 0.035%, Cu = 0.03%.* (c) *P = 0.01%, Cu = 0.22%.* (d) *P = 0.01%, Cu = 0.03%.*

According to [6. Boehmert01]: Based on SANS results from WWER-440 and WWER-1000 – The nickel-containing WWER-1000 steel is more sensitive to irradiation than WWER-440.

According to [7.Boehmert 04]: Types and composition of the materials as well as neutron fluxes hardly affect the characteristics of the features but change volume fraction of the defect. There is clear trend, the bigger volume fraction, the bigger increase the irradiation-induced hardness and strength and larger the shift of transition temperature. The correlations approximately follow square-root dependence.

According to [9. Debarberis]: Based on semi-mechanistic model – results were summarized in following pictures.

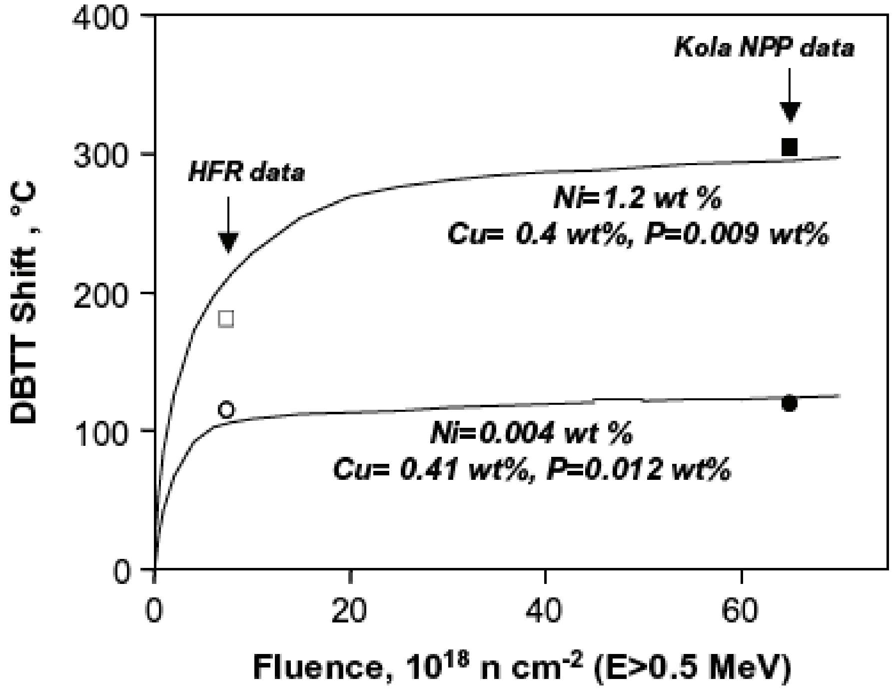


Fig. 7. Comparison of the fluence dependence of DBTT in alloys with different Ni contents.

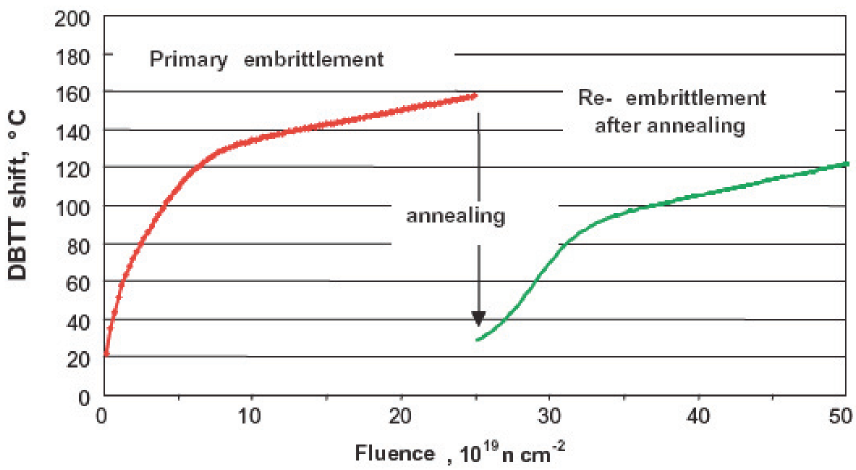


Fig. 11. Schematic of primary radiation embrittlement and re-embrittlement calculated with proposed model.

According to [10. Dragunov]: in the work of about 60 pages, the topic: “Effect of irradiation on mechanical properties” was deeply reported.

According to [11. Fedorov]: Electrical resistivity change in iron/copper alloy under neutron irradiation (1,77Cu) was studied.

According to [12. Gillemot]: Comprehensive study of thermal ageing effects was performed. Interesting achievement is summarized in the following 2 pictures:

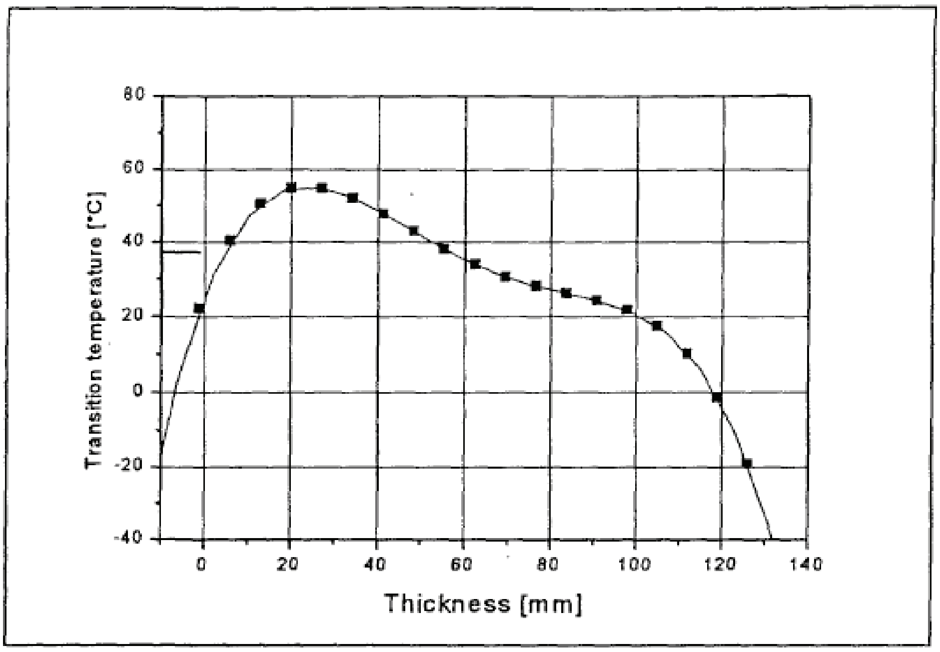


Fig.8. Realistic transition temperature distribution in the function of the wall thickness. This example is calculated from the data obtained on laboratory aged 15H2MFA steel (2000 hours 350 C) and irradiation data obtained on NPP surveillance specimens (2*1019 n/cm2 E>0,5 MeV).

Calculations have shown that in the case of a WWER-440 reactor the flux rate is about 5 times higher at the inner surface than at the outer one. It means that the embrittlement of the wall changes with the thickness as it is shown in Fig. 7.

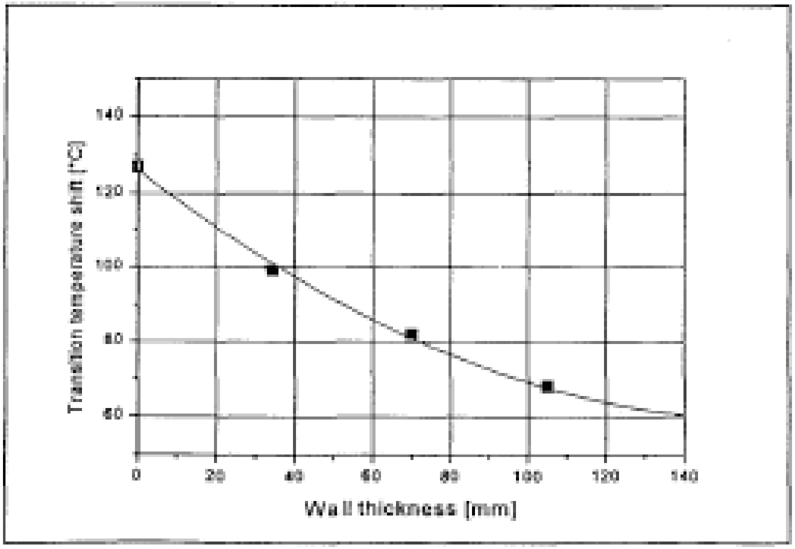


Fig.7 The transition temperature increase caused by irradiation in the function of the wall thickness. (Fluence at the inner wall is $2*10^{20}$ n/cm^2)

According to [13. Gokhman]: Based on SANS and APFIM – radiation raises the lattice defect concentration and, hence, changes the Gibbs energy system. This can be connected with the shift of the phase fields or even with the appearance of new phases.

According to [14. Gorynin]: After investigation of the 1st generation of WWER-440 steel, there is a statement of the producer that in lifetime projected for 30, all material features are O.K.

According to [15. Gorynin76]: Survey of radiation induced defects and their influence on RPV steels is presented in details.

According to [16.Kuleshova05]: Electron-microscopy and fractographic studies of the surveillance specimens from base and weld metal of WWER-440/V-213 RPV in the original state and after irradiation from $5E_{23}n/m^2$ up to overdesign values have been carried out. The maximum specimens irradiated time was 84 480 h. Radiation coalescence of cooper-enriched precipitates and extensive density increase of dislocation loops was observed and reported.

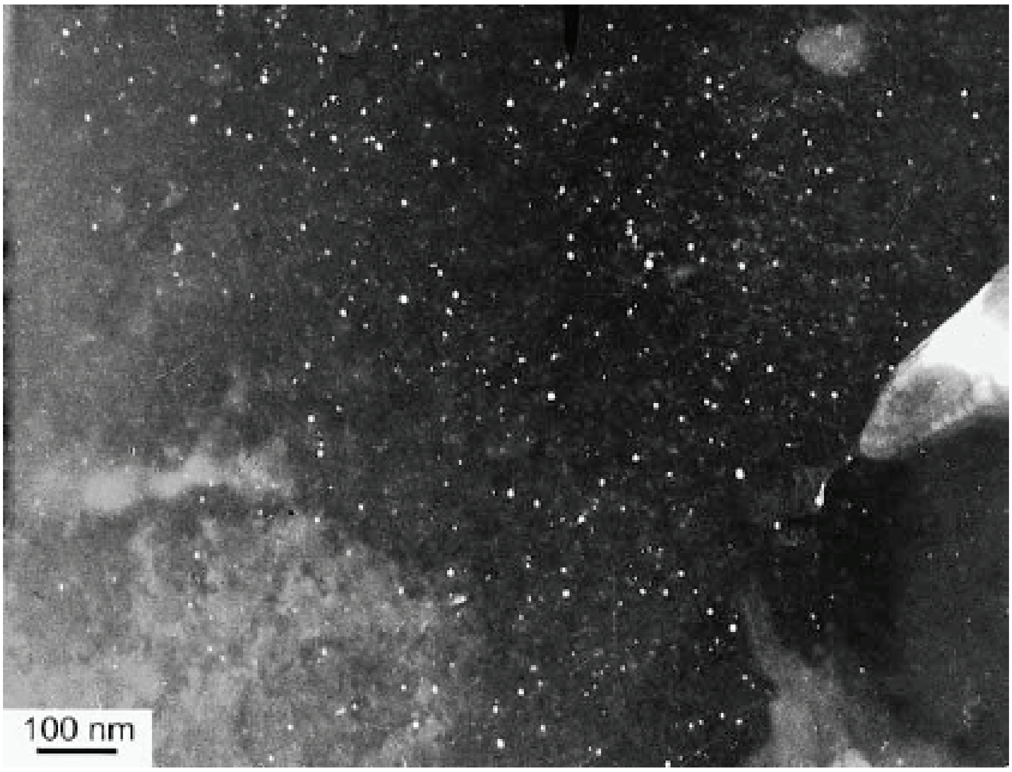


Fig. 9. Copper-enriched precipitates in the irradiated (fluence of 8.66×10^{24} n m^{-2}) base metal (×100 000).

According to [17. Kryukov97]: Comprehensive state of art in WWER RPV steel research. The Ni content effect was reported! A dramatic increase of transmission temperature shift was observed if the Ni content in RPV-steel goes over 1.3 % (see next 2 figures).

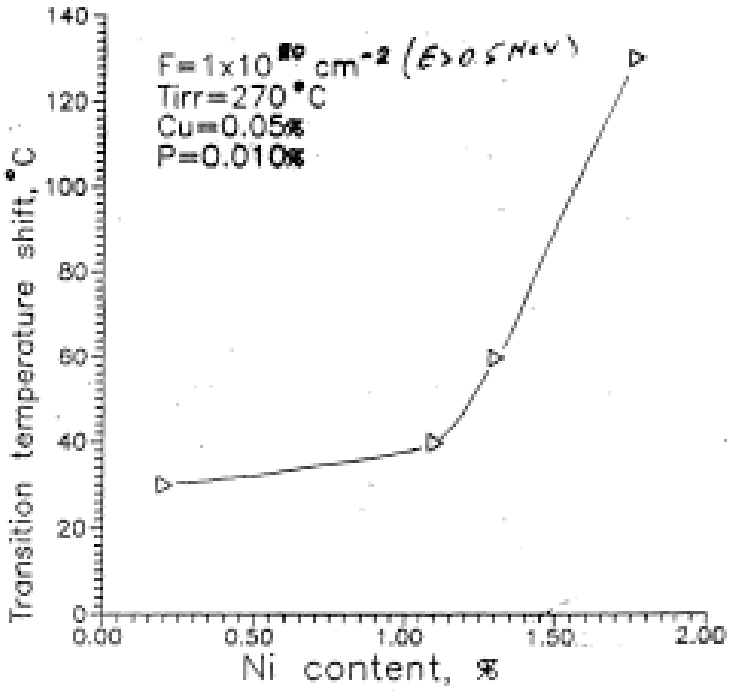
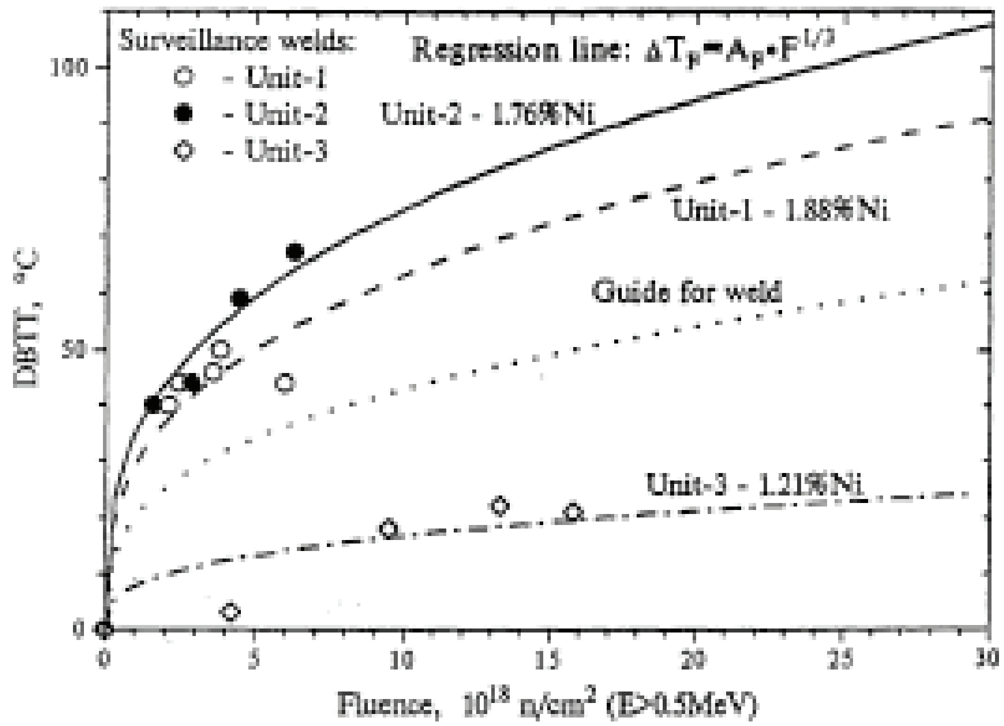


Fig.6 Transition temperature shift



Effect of nickel content on radiation stability of weld metal.

According to [18. Kuleshova02]: Irradiation of Russian RPV steels causes in WWER-440 steels occurrence of radiation defects – dislocation loops, increase in the density of disk-shaped precipitates and formation of many rounded precipitates. First type of precipitates is presumed to be vanadium carbides and the second are copper-enriched precipitates.

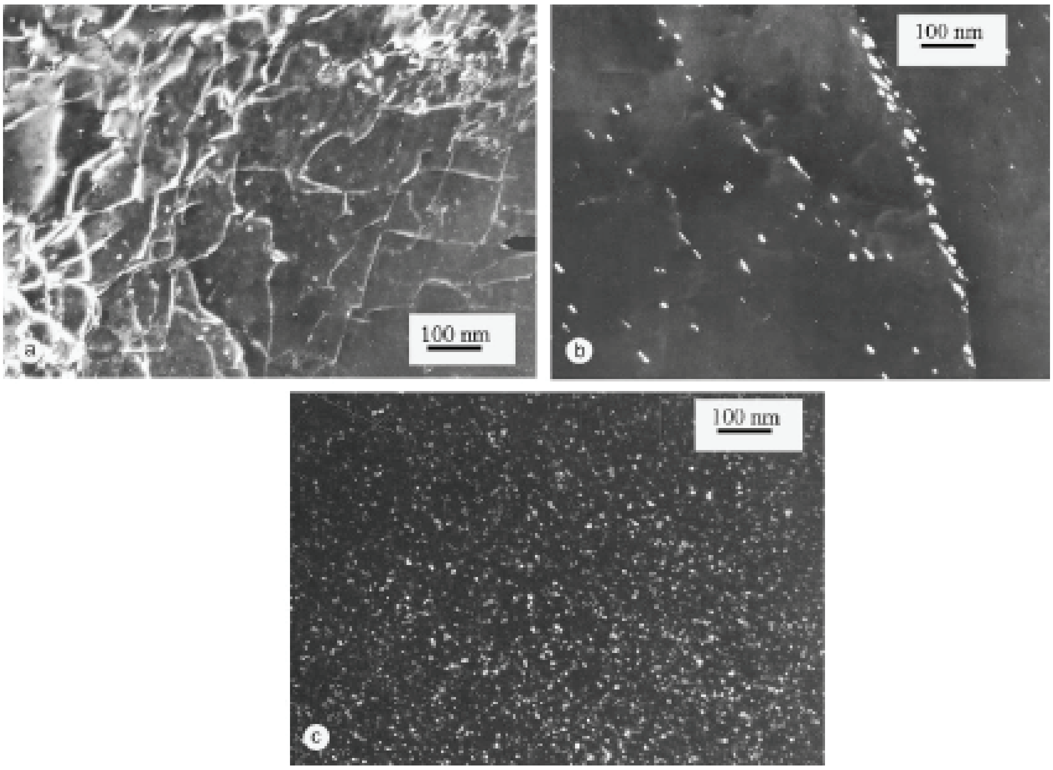


Fig. 4. Radiation-induced microstructural components of pressure vessel steels: (a) radiation defects—dislocation loops (black dots), (b) disk precipitates (carbides) inside and on boundaries of former austenite grains, (c) rounded precipitates (copper-enriched).

According to [19. Maussner]: Based on TEM results – irradiation causes increase of dislocation and induced precipitates and segregation of cooper in the carbides. These changes were observed more significantly in welds and HAZ than in base material.

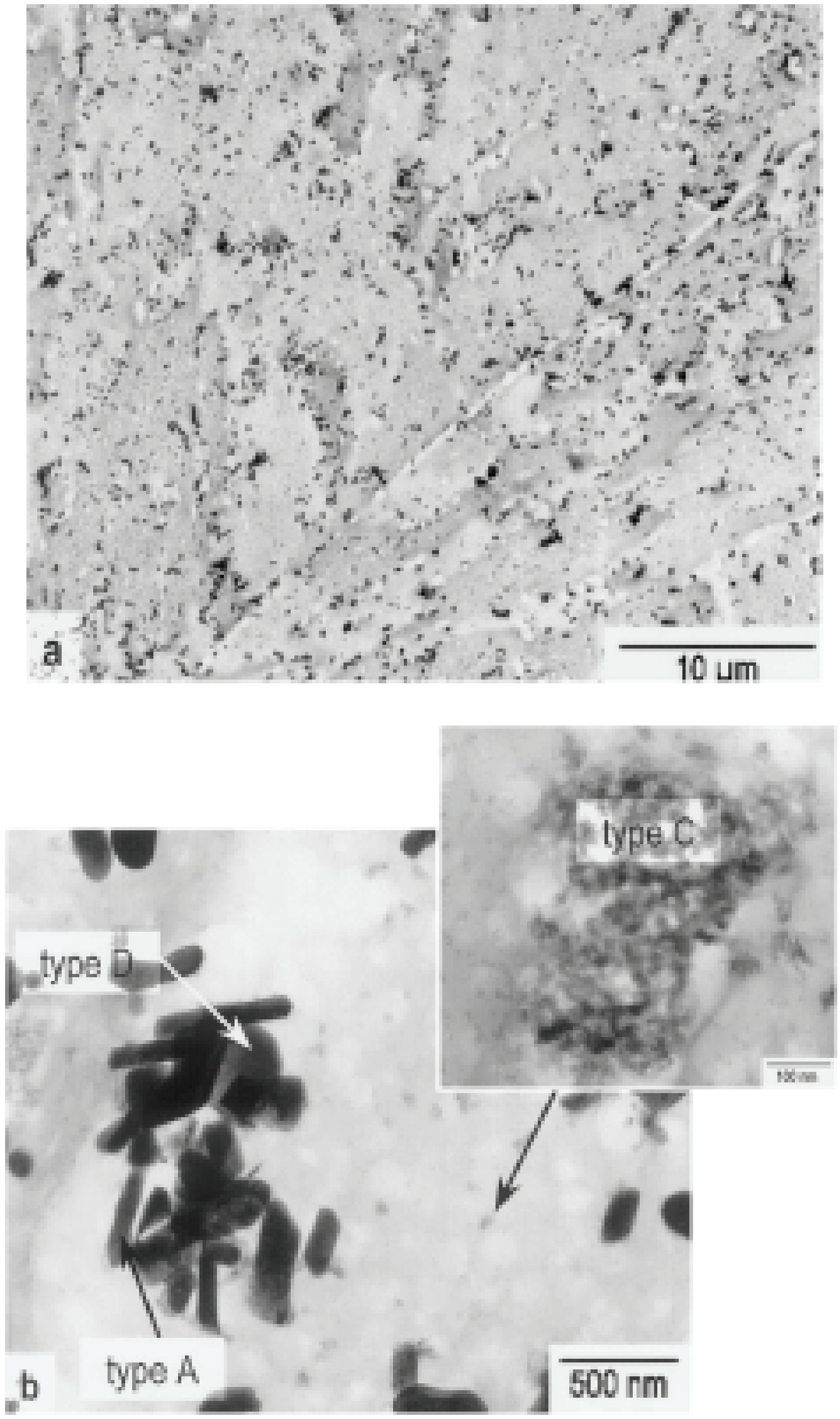


Fig. 8. (a) Microstructure of the irradiated base metal, fluence: $6.5 \times 10^{19}/\mathrm{cm}^2$ ($E > 0.5$ MeV), coarse carbides (types A – $M_{23}C_6$, D – M_6C), extraction replica. (b) Microstructure of the irradiated base metal, fluence: $6.5 \times 10^{19}/\mathrm{cm}^2$ ($E > 0.5$ MeV), fine MC carbides (type C), extraction replica.

According to [20. Morozov]: Change in the dislocation density more than an order does not make an essential effect on concentration of complex radiation defects.

In the works [21.-23. Nikolaev]: Deep survey on the mechanisms of the soluted atoms influence on radiation embrittlement and hardening of iron-alloys was reported.

According to [24. Nikolaeva]: The temper increases the tendency of steel to phosphorous segregation at the grain boundary. This is displayed through the shift of ductile–to-brittle-transition temperature.

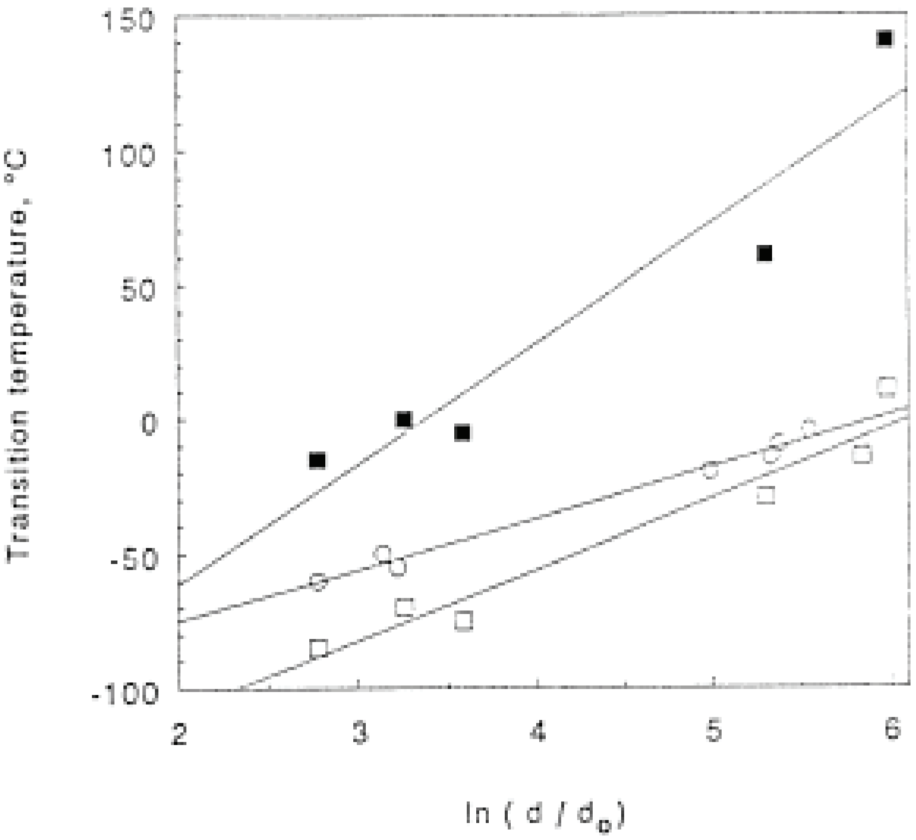


Fig. 5. Dependence of the ductile-to-brittle transition temperature (TT) on the inherited austenitic grain size (○ – heat 4 in "ductile" state, □ – heat 5 in "ductile" state, ■ – heat 5 in "brittle" state).

In the paper [25. Olshanskij], results from basic research was published.

According to [26. Platonov02]: Comparison of predicted and observed values of re-irradiation ductile-to-brittle transition temperature shifts for WWER-440 RPV core welds is provided. Three different modes of DBTT behaviour are presented and discussed.

According to [27. Platonov03]: The fast embrittlement to a certain degree, going on for first 1-2 years. The level is determined by P content only. Secondary increase of embrittlement is determined by flux. The higher flux, the earlier is the beginning of embrittlement and the greater is its rate. For more information see next picture.

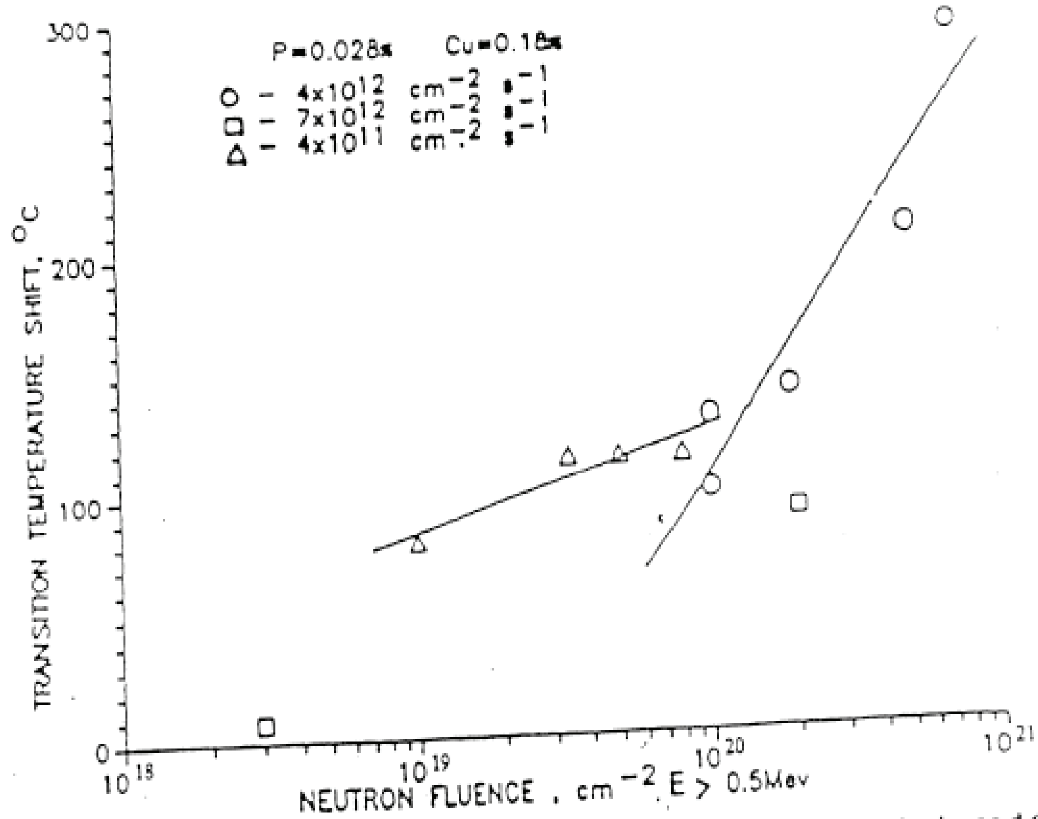


Fig.1 Radiation embrittlement of weld metal at irradiation temperature of 270°C.

According to [28. Slugen05]: Based on positron annihilation results - Effectiveness of annealing of irradiated 12Kh2NMFA (2houres at 475°C) is higher than at 15Kh2NMFA steel. On the other hand the 15Kh2MFA steel is more resistant to the irradiation up to high energy neutron fluence of about $1,25E^{24}$ cm^{-2} than weld steels as 12Kh2NMFA or Sv-10KhMFT.

According to [29. Slugen98]: It was confirmed that HAZ is the most sensitive place for thermal and neutron embritlement in reactor. Positron annihilation (PAS LT) results on the successive annealed HAZ specimens of WWER-1000 and WWER-440 steels showed the rapid increase in the vacancy-type defects formation in the temperature region 525-600ºC.

According to [30. Pechenkin]: The DBTT shift in specimens irradiated with the record neutron fluence of $8,3E^{20}$ n/cm^2 (E>0.5 MeV) has been measured and taken into account. It is shown that the dose dependence of the DBTT shift observed at the high fluences is in satisfactory agreement with that calculated on the basis of the model proposed, which takes into account the contribution of intergranular embrittlement induced by phosphorus segregation on the grain boundaries.

According to [31. Vodenicharov]: The carbides precipitates are distribute predominantly on the grain boundaries in the weld metal and at the periphery of bainite needles in the base metal and heat affected zone. These fine needle-like particles are not affected by annealing at 560°C.

According to [32. Zlateva]: Results similar to 32. Additionally, the annealing experiments showed that the stable parts of the black dots were particles and the smallest dots influenced by annealing were dislocation loops.

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10 PLEX Issues

In this study about 45 papers from ODIN DOMA database on WWER PLEX issue have been reviewed. Still about 12 papers shall be reviewed in order to complete the Section 8 papers on PLEX. Most of the papers reviewed are dealing with the most critical component, the RPV of the WWERs. This is expected since the RPV is considered an irreplaceable component, while other components and the piping can be replaced, if economically feasible. Cable and concrete ageing has not been covered in any of the reviewed papers. Accordingly this review concentrates on the RPV ageing and life management, mainly. The general summary given in the 1st NKP&C work shop is basically valid for this review as well and I will not repeat it here.

The neutron irradiation embrittlement of the RPV of WWERs has been the main topic in the papers. The whole scenario relating to the topic is covered very well and, of course, repeated many times in the different presentations. The activity of the European Networks have also been covered pretty well, especially AMES, which has dedicated all projects to neutron irradiation embrittlement and ageing of RPVs and Internals. Special attention should be given to papers produced by the legendary chairman of AMES, **Myrddin Davies**. When reading his papers one can get a very good view and understanding of the whole ageing problem on a global level and perspective, including energy-political aspects. It is suggested using his papers especially for educating young people in the sector.

It is a pleasure to read the old papers relating to activities of AMES. Even though AMES was criticised for not having political and funding power, it created very good ideas and innovations relating to PLEX and PLIM issues and many projects were completed based on AMES proposals. Projects were launched and completed as part of the EURATOM frame work programs, in kind projects and TACIS / PHARE. Some late TACIS and TAREG projects are still using ideas elaborated in AMES. Even the new “Network of Excellence” NULIFE is using AMES ideas at present. It is planned that AMES will continue as an umbrella project under NULIFE.

In perspective of the European SRA (Strategic Research Agenda) and SNE TP the PLIM and PLEX activities are under special focus. At present there are 152 LWRs in operation in Europe. The average age of this fleet is 25 years while the design life is 40 years in general. The annual electricity production is around 950 TWh. Extension of plant life by 10 years beyond design life 9500 TWh more clean electricity, 20 years 19000 TWh more clean electricity. 950 TWh of nuclear power reduces annually the CO2 emission by 650 Mtons CO2. The EC has in its SRA given limits on CO2 emissions. It will be extremely difficult to keep the limits if the nuclear power will phase out. The goal can be reached much easier by keeping the ageing NPP in operation beyond the design life. Phase out policy will cause big economical problems for countries applying such policy. Therefore PLEX can be seen as a most important subject in the very near future, and we need to make sure that the knowledge will be preserved for these important actions and decisions.

NULIFE is a new Network of Excellence, which commenced in 2006. It is a 5 years project funded partly by the EURATOM 6th FWP. The aim is at developing a European Entity, an International Association which will act as an entity providing technical support in PLIM and PLEX matters for especially NPP operators. The new entity will be financed by the members and mainly by the end users (NPP owners).

For evaluation of a safe life time for the RPV, information on plant lay-out (as built configuration), process description, operational parameters, emergency core cooling systems, thermal hydraulics, ISI, reactor physics and dynamics, neutron fluence, material toughness etc. is needed. Based on these information and thermal hydraulics and fracture mechanics calculations, stress intensity factors, KI, for cracks can be evaluated. The calculated KI will be compared to the critical stress intensity properties of the RPV material. Based on this evaluation the highest allowable critical temperature for the material, Tka can be determined. When evaluating the safe operation life time of the RPV we must look at these two parameters, KI and KIC. The cures can be shifted apart by different means and mitigation. For example the critical curve can be shifted back to original values by annealing or the shift can be slowed down by proper core loading management. On the other hand the calculated KI blow down curve can be "shifted to the left", to safe regions by several mitigation measures like heating of ECC water or adjusting the pressure of high pressure ECC pumps etc. By playing with these two parameters it is easy to analyse the integrity status of the RPV and make necessary decisions regarding PLEX.

10.1 Consolidated Conclusions

Based on the review the following conclusions could be made:

- The RPV is the most critical component when considering PLEX of WWER NPPs.
- Irradiation embritlement of the RPV has been a critical issue for the WWER NPPs, but the problem has been solved by proper measures and PLEX is already implemented successfully in a few operating WWER 440 units.
- Two TAREG projects (TAREG 2.01/00 and 2.01/03) on VVER RPV embrittlement and mitigation started in 2003 and 2005 and are still ongoing. The project results will be very important for improving the quality and validity of fracture toughness and irradiation embrittlement trend curves for the RPVs.
- When determining neutron irradiation embrittlement of the RPV of WWER 440 plants after annealing the Lateral Shift model seem to be conservative enough and has been adopted and accepted in general for PLEX purposes.
- WWER 1000 units have suffered from SCC (Stress Corrosion Cracking) of SG (Steam Generator) collectors. The problem has been solved by replacing problematic SGs and improving manufacturing procedures. For VVER 440 SGs erosion corrosion problems have been solved by replacing carbon steel feed water pipes by stainless steel.

10.2 Open Issues

- KIc trend curves for WWER RPV steels need upgrading
- Shape of the KIc curve needs to be agreed upon (Unified / Master curve)
- The shift in toughness curve due to neutron irradiation in WWER RPVs need upgrading (reconstitution)
- Neutron environment in WWER 1000 Surveillance position need upgrading and tuning
- Flux effect in WWER 440 NPPs could give non-conservative contribution in the PLIM /PLEX scenario
- WPS and K_{Ia} could be utilized to increase plant life

10.3 Reviewed papers and summaries

In the following the result of the 2nd review round is given paper by paper:

Davies 002.pdf, Myrddin Davies, "The Significance of plant life management", Oxford, UK

The paper describes plant life and plant life management as well as procedures for categorisation of components with examples of some key components. Examples of good practice and guidance are given for implementation of PLIM programs. Resent IAEA activities under the IWG-LMNP are described as well as some future expectations. The author underlines that the main purpose of the NPP is to generate energy for sale. He emphasises that the full scope of PLIM / PLEX include pre-operational activities like choice type of NPP, design, construction, commission, operation, decommissioning and the return of the site to a "green field". He describes the nuclear power in the world including plant type, capacity and number of operating plants relating to the age. The already classical life time management processes as well as key component selection criteria, identification and categorisation are also described as well as collecting of data and records.

Davies 003.pdf, "Introduction", Chapter 1, Davies L.M.

As the title says the paper is an introduction to a wider presentation of the PLIM issue. Accordingly the paper is an introduction to A IAEA TRS publication. The main topic of the paper is on RPV embrittlement and integrity as usual when describing critical issues of PLIM. It describes briefly the history of irradiation embrittlement of RPVs and includes the main components of the problematic. The paper describes the irradiation effects on mechanical properties, the influence of neutron fluence, irradiation effects trends, empirical modelling of radiation defects, mechanistic modelling, mitigation of radiation affects and integrity threats, current state of the art (probably in 1990's) and international programs.

Davies 004.pdf , " Davies L. M,. Van Duysen J. C., von Estorff U., Sycamor D."The AMES network and the Task Group on WWERs".

The paper describes the bases and history of the network "Ageing Materials Evaluation and Studies", AMES. The author was the founder of the legendary network. AMES was established in 1993. The main objectives were to provide information and understanding on neutron irradiation effects in reactor materials for supporting designers, operators, regulators and researchers and to generate projects in the area. The paper describes the structure, the organisation, objectives and particular emphasis is given to the work intended to perform for the working group on WWER's.

At the beginning 3 projects were nominated:

- Validation of surveillance practice and mitigation methods
- Effect of irradiation on RPV internals
- Significance of Phosphorus on toughness in steel during neutron irradiation

Project 1 was divided in 8 sub-groups. The paper 3 describes rather detailed the projects as well as the 8 sub-groups of project 1. It also discusses the irradiation embrittlement problems related to WWER 440 and 1000 RPVs.

AMES was in the beginning a network operated by the JRC-IAM in Petten. The chairman of the project was the author for a long time period in the beginning of the network. AMES created many important research projects relating to neutron irradiation embrittlement. AMES did not have own funding but many projects which were initiated by AMES were completed in EC funded frame work programs and as in kind projects. Many Tacis projects were also initiated based on AMES proposals. Even the bases for present ongoing TAREG projects were elaborated in the framework of AMES (Standard Description Sheets).

Debarberis 007.pdf, Debarderis L., von Estorff U., Crutzen S., Beers M., Stamm H., De Vries M.I., Tjoa G.L. "LYRA and other projects of RPV embrittlement Study and mitigation of the AMES Network",

Within the frame work AMES a number of experimental works on RPV embrittlement were carried out in the JRC-IAM in Petten. In order to conduct ahigh quality irradiation program an irradiation rig facility named LYRA was designed and built in IAM. The aim was at using the rig in irradiation experimenst in te HFR reactor in Petten. In this paper the LYRA irradiation rig, as well as other irradiation rigs are briefly described. The paper also describes the AMES network bases, objectives, organisation, activities as well as main projects. The paper describes especially modelling activities carried out in IAM giving insight in damage mechanisms, precipitations, grain boundary segregation, strengthening mechanism and ductile/brittle behaviour of RPV steels.

Debarberis 015.pdf, Debarberis L., Brumovsky M., "Preservation and management of knowledge on WWER RPVs",

The paper is a kind of a power Point presentation on the topic "preservation and knowledge management" detailing the basic needs for such activities. The paper describes the age of known experts in the sector as well as WWER experts. The paper describes the commencement of these NKP&C activities. In the field of RPV embrittlement and ageing a generational gap is appearing with regard to in depth knowledge of material behaviour and related neutron embrittlement issues because many of the experts in this field are retired or will soon do so. The NKP&C was initiated within the SAFELIFE action of the JRC IE in 2004. First the experts for these activities were nominated in 2005 (through AMES, IGRDM, IAEA etc.). The next steps are the Workshops, of which this one is the 2nd. Literature in the sector was collected and stored in the ODIN DOMA database in JRC IE. The results of the project will be useful also for young specialists of the new generation. The knowledge will be used both for PLIM and PLEX purposes as well as for design of new NPPs with long term design life time.

Debarberis 002.pdf, Bieth M., Debarberis L., Sevini F.,Kryukov A., "WWER RPV embrittlement projects relaed to the TACIS and PHARE Programs",

The paper describes very briefly the following Tacis and Phare projects related to RPV embrittlement:

- Phare 91, "Kozloduy unit 1 RPV boat sampling
- Tacis 1.1/91, "RPV Embrittlement"
- Tacis U1.02/92A, "Evaluation of RPV Embrittlement of South Ukraine NPP, including Embrittlement aspects"
- Tacis R2.09/94, "Integrity assessment of WWER 1000 RPV including Embrittlement aspects"
- Tacis R2.02/95, "Reactor Vessel Integrity assessment"
- Tacis R2.06/96, "Surveillance Program for WWER 1000 RPV"

- Tacis R6.01/96, “Ageing of RPV. Embrittlement analyses at the WWER 440 GreifswaldRPVs”
- Tacis R/TSO/WWER01C, “Licensing related assessment for design and operational safety of WWER-subtask C, reactor embrittlement”

The above projecs are described very briefly giving the ain elements. Therefore no detailed description relating to each single project will be given here, but reference is given to the paper itself. It was emphasized that despite efforts to solve the neutron embrittlement problems there are still a number of open issues to dig in. The main issue for WWER 440 is the lack of surveillance program and data for embrittlement and re-embrittlement after annealing for the 1st generation units. For the WWER 1000 RPVs a need for additional work to assess the neutron fluences for the surveillance position as well as the need for upgrading embrittlement trend curves by utilizing reconstitution was identified.

Debarberis 007.pdf, Debarberis L., Taylor N., Erikson A, Youtsos A, Bieth M., Sevini F., Toerroenen K., Guidez J., Weisshaeupl H. “An integrated view on PLIM; EC-JRC-IE project and programs”,

The paper is short, only 1.5 pages. Concern is given to safe and secure electricity supply in ageing NPPs in EC in the future since many NPPs are approaching the EOL design limit. Issues includes RPV embrittlement, embrittlement of internals and core shroud cracking, RPV head penetration cracking (CRDM), PCP integrity, steam generator degradation, electric cable and concrete ageing. In order to tackle the PLIM task an integration effort in JRC IE aiming atdefining a PLIM project, SAFELIFE, integrating the efforts done in the past and competencies of various European Networks. The structure of the SAFELIFE, which started in 2003, was shown in a describing figure.

Denis_Popp.pdf, Popp D., “The use of U.S. Licensing practices for WWERs”

The licensing process for NPP Temelin I&C and fuel designs were enhanced with introduction of USNRC practices. The USNRC Regulatory Guide 1.70 “Standard format and content guide for safety analyses reports” and NUREG 0800 “”Standard review plan for review of safety analyses reports for NPPs” were used for developing and reviewing Temelin licensing documents. The above documents can be used both for licensing an entire NPP and for major plant upgrading, such as I&C replacement. The application of the above guides provided many benefits serving plant designers as well asguidance fro the licensing of large upgrades and modifications. These standards have been adopted in many applications through the world and can be beneficial in the licensing of WWER as this Temelin activities showed in this project.

Dragunov 006.pdf, Dragunov Y., Kurakov Y., “The main objectives for works on lifetime management of reactor unit components”

The paper is a “power Point” presentation on the above topic. The main objectives on the PLIM works are:

- Develop regulations in the field of NPP component ageing and life time management
- Investigation of ageing processes
- Residual life evaluation taking into account the actual state of the NPP SSC, real loading conditions and cyscles as well as results from ISI

- Development and implementation of measures for maintaining and enhancing NPP safety level

The main ageing programs and R&D activities are briefly described in the paper. The activities related to PLIM of Novovoronesh units 3 and 4, Kola 1 and 2 and Balakovo 1 are briefly mentioned. The paper also describes the standard program for inspection of mechanical properties of NPP pipelines after 100 000 hours of operation. A special program is dedicated for the stem generators including: full scale testing, improvement of seals, water chemistry optimisation, blowdown system upgrading, corrosion investigation (collector and tubes probably), improvement of ISI and finally a program for evaluation of the life time of steam generator primary collector. Connection with Tacis projects are also described even though their contribution to modernisation of the mentioned plants is very small. This is a very professional paper, even if only viewgraphs were presented.

Golovanov 002.pdf, Golovanov V. N.,.Raeysky V.M., Koslov D., Krasnoselov D., Lichadeev V., Pimenov V., Prohorov V., "The Korpus facility in determination of residual life and validation of possible prolongation of the WWER 1000 vessel operation"

The paper describes conditions of experiments carried out at the KORPUS facility in NIIAR. The accuracy and homogeneity of the neutron fluence on the irradiated test specimens is of course much better than in the surveillance position of the WWER RPV (above the core). Base metal and weld metal specimens taken through the whole thickness of the forging was used in order to see the influence of distribution of alloying elements and impurities as well as macrostructure fluctuation. The results of the experiments showed that the toughness of the specimens irradiated in KORPUS was much better than the design trend curves would give. The paper is very short and brief and it is difficult to get a precise understanding of the experiments and results.

001Hurst.pdf, R.Hurst, N. Taylor, D. McGarry, B.R.Bass, R. Rintamaa, J. Wintle, "Evaluating the NESC-I test and the integrated approach to structural integrity assessment", Paper published in International Journal of Pressure Vessel and Piping 78 (2001) p. 213-234

NESC-I project objective is to validate the methodologies used to demonstrate safety in case of thermal shock PTS events on the RPV of NPP. The project included 34 worldwide organizations. A designed spinning cylinder test with hypothetical flaws simulated the aged reactor wall during a PTS loading event. The cylinder was cladded inside, the outer diameter was 1395 mm and total thickness was 175 mm. Clad was 8 mm thick 316 stainless steel. 14 different fatigue crack and chevron notch have been made into the cylinder. Some of them simulated surface defects, some simulated embedded cracks. Different subgroups worked on the different tasks including pre-test NDT inspection, material properties, structural analysis, instrumentation, evaluation and destructive examination. Several transition temperature characteristics parameters were obtained and compared. The cladding inhibited the cleavage fracture in the near surface region. The test results have been compared assessment code assessment. The RT_{NDT} temperature applied by codes was demonstrated to be a poorly suitable parameter for indexing fracture toughness data. The major source of uncertainty was the intrinsic variability in material toughness. The project produces a set of valuable interdisciplinary reports.

003Bieth.pdf, M. Bieth, C. Rieg, R. Ahlstrand" New TACIS regional projects on radiation embrittlement and integrity assessment of WWER reactor pressure vessels", Paper published in International Journal of Pressure Vessels and Piping 81 (2004) 677–682

TACIS/PHARE program was started in 1991; the projects financed by the European Commission provided important knowledge on neutron embrittlement effects on WWER reactor pressure vessel (RPV) materials. Two new projects are launched and the paper describes them: one on the WWER-440/213 type reactors one for the WWER-1000 type reactors. The aim of the projects is an extensive understanding of the RPV integrity assessment and to provide the Russian and Ukrainian operators conclusions on of acceptable safety margins and expected remaining lifetime of the plants.

Mitigation measures are addressed in the case of high Ni core weld of WWER 1000 reactor vessels and in case of some sensitive WWER 410/213 RPV materials.

003Kataoka.pdf, S. Kataoka, T. Otsuka "JAPEIC's Activity on Aging Issue Related to Neutron Irradiation of RPV/RV Materials", Paper published in International Journal of Pressure Vessels and Piping 77 (2000) 569–574

The paper introduces the Japanese PLIM project planned between1996-2005.

The Ministry of International Trade and Industry (MITI) is responsible for the safety and reliability of NPP in Japan. PLIM project has been started at by the Japan Power Engineering and Inspection Corporation (JAPEIC) and supervised by the ministry. The projects addressed the issues embrittlement of RPV steels, thermal aging of piping and surveillance specimens.

The Japanese uses a method to predict the drop or reduction of the USE following irradiation. The Japanese regulation related to irradiation embrittlement, JEAC4206 provides a method to predict the drop or reduction of the USE following irradiation, however, JEAC4206 does not provide a method to evaluate the RPV in case of a low USE value. Fracture toughness will be calculated from a correlation equation between the fracture toughness and USE too, not only from the Charpy transition temperature.

Test specimens irradiated in the OECD test reactor at Halden. Research is necessary due to difference among USA and Japanese steels.

The paper presents planned the reconstitution technology for surveillance specimens. Different welding technologies including friction welding is developed for reconstitution. The early Japanese surveillance specimens are cut from L (longitudinal) direction, since the present surveillance uses T (transverse) specimens. The purpose of the project is to produce T direction specimens from the remnants of the L direction Charpy specimens.

Bieth-002.pdf, M. Bieth, R. Ahlstrand, C. Rieg and P. Trampus, "Upgrading the operational safety of nuclear power plants through the TACIS nuclear safety assistance program", Paper published in the proceedings of ICONE12, 2004, Virginia USA

The paper presents the European Union's TACIS programme. The programme contributes to the improvement in the safety of aging reactors in the Commonwealth of Independent States (CIS).

The program purpose is transfer of technology and safety culture. The goal of one existing project is to understand the present RPV integrity assessments and address the open safety issues. The purpose of the second project is to implement leak-before-break theory into the safety analyses of WWER 1000 reactors. The knowledge gained by TACIS program initiated a project named SENUF linked to SAFELIFE action of JRC Institute for Energy. The paper describes the SENUF and highlights the on-site assistance in Armenia, Russia, Ukraine and Kazakhstan, as the information dissemination of TACIS under its umbrella.

Brynda-001.pdf, In Russian

Contri-001.pdf, P. Contri and T. Katona "Safety Aspects of Long-term Operation of Nuclear Power Plants", Paper published in the proceedings of SMIRT17, 2003, Prague, Czech paper D-02/1.

The paper deals with the safety aspects of the Long Term Operation (LTO) of ageing NPP-s. It aims to identify the technical aspects related to decisions. It presents models of LTO using the different concepts as PSR, PLIM+PSR, LR and FSAR. The paper shows the importance of design basis information for LTO. The organization chart of a full LTO project is described in this paper. Despite the different regulatory approaches, the LTO programs have similar technical content. Essential elements of the LTO program are the ageing management experience and the extrapolation of detected degradation on the planned operational lifespan. Finally it presents one of the most recent LTO applications at Paks NPP in Hungary. This LTO is in line with the IAE safety program.

Debarberis007.pdf, Debarberis L., von Estorff U., Crutzen S., Beers M., Stamm H., de Vries M.I., Tjoa G.L. "LYRA and other projects on RPV steel embrittlement Study and mitigation of the AMES Network", Paper published in Nuclear Engineering and Design 195 (2000) 217–226

The Institute for Advanced Materials of the Joint Research Centre (JRC-IAM) plays the role of operating agent and manager of the European networks, e.g. NESC, ENIQ, dealing with material behavior of structural components. AMES was set to bring together the European organizations RPV material assessment and research. AMES developed projects and produced 'state-of-the-art' reports on validation of surveillance practices, effects of irradiation on internals, phosphorus role on steel loss of toughness during irradiation and mitigation methods. The necessary facilities and tools for the studies include specific developed 'LYRA' rig at the HFR at Petten, Netherlands. Modeling activities and its fundamental mechanisms relevant to DBT phenomena were described in this paper too.

Debarberis006.pdf, Sevini F., Debarberis L., Taylor N., Gerard R., English C., Brumovsky M. "The AMES network in the 6th Framework Program", Paper published in Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17) Prague, Czech Republic, August 17 –22, 2003

AMES (Ageing Materials European Strategy) network was established in 1993 to gain expertise on nuclear reactor materials assessment and research on ageing management. Several projects were carried out relating to understand the influence of various embrittlement mechanisms (PISA); develop new techniques (AMES-NDT and GRETE);

improve the dosimetry aspects (AMESDOSIMETRY, MADAM and REDOS); improve surveillance and prediction of irradiated material fracture toughness (COBRA, FRAME). A project named ATHENA started in 2001 summarizes the obtained achievements and edit guidelines on important issues like the Master Curve, effect of chemical composition on embritllement rate in RPV steels, re-embrittlement models validation after WWER-440 annealing and open issues in embrittlement of WWER type reactors.

After 2003, the network is part of the PLIM initiative including others networks, i.e. NESC, ENIQ, NET and AMALIA. In September 2003 SAFELIFE network covers the PLIM issues embrace members of the established European networks AMES, NESC (for evaluation of structural components), ENIQ (for inspection qualification), NET (neutron evaluation techniques) and AMALIA (Assessment of Materials Ageing under the effect of Load and IASCC), members of European and international organizations. The SAFELIFE promoted by JRC will comprise actions and expert group development.

Gillemot003.pdf, , Gillemot F., Fekete T., Tatár L., Horvát M.," R&D Background for Life Management in Hungary", Paper published in the IAEA-CN-92/P6

Abstract on a paper dealing with research results obtained in the frame of PLIM in Hungary and the ongoing research to support plant life extensions for NPP Paks.

The topics are:

- consideration of the clad in elastic-plastic PTS analysis.
- application of less over-conservative trend curves, esecially the Master Curve
- extension of the analysis with crack arrest considerations
- increasing the understanding of vessel annealing, and elaborate the optimal annealing strategy.
- development nondestructive measurement methods of material degradation Several actions are available to extend the operational time of the WWER-440 V213 RPVs. It highlights the need of further studies on material ageing mechanisms.

Gillemot004.pdf, Gillemot F.,"National Report - Recent development in Life Management of the Pressurized Components", Paper presented at the Technical Committee Meeting of the International Working Group on Life Management of Nuclear Power Plants, Vienna, 1997.

The paper presents the recent developments on life management and safety on Hungary. It is divided in groups according to the type of actions: governmental, regulatory. The new Hungarian Safety Regulations are based on ASME and PNAE. The utility performed several safety enhanced actions and started the PLEX program. At the TSO-s research actions including the ageing studies, development of PTS methodology and RPV database started. The paper presented in detail the PTS methodology performed in the frame of the national project called AGNES for filling the gaps between the western and eastern safety codes. As a result of the preliminary PTS analyses a list of research actions has been elaborated. The ageing of cladding is a key factor for the WWER-440 type RPV's. The material of the RPV- (15Kh2MFa) seems to be tough against neutron and thermal ageing. The paper presented results on transition temperature in the function of the wall thickness of the thermally aged 15H2MFA steel.

Gillemot005.pdf, Gillemot F." PLANT LIFE MANAGEMENT IN HUNGARY", Paper presented at

The paper presents the PLIM program of Paks NPP, Hungary. The Paks NPP has four WWER-440 V-213 type units in operation, the first unit started operation in 1982. Several factors are affecting the PLIM in Hungary, including the effects of the political and economical changes within the country. The life management includes mitigation actions. Elaboration of a life management strategy for the whole plant has been started.. The papers presented the main components considered to be limiting the lifetime of the plant (RPV and steam generators) and the ageing mechanisms considered at the calculation of the lifetime. The ongoing preparation program establishes the basis for the future PLIM.

Gillemot007.pdf, , Elter J., Fekete T., Gillemot F., Oszwald F, Maróthy L., Rátkay S., "PTS ASSESSMENT -THE BASIS OF LIFE TIME EVALUATION AT NPP PAKS", Paper presented at IAEA Specialist's Meeting on "Pressurized thermal shock", Esztergom, Hungary, 1997

The paper presents the PTS analysis at NPP Paks, second generation of WWER-440/213 type. The analysis performed in the frame of the AGNES (Advanced General New Evaluation of Safety) project. The methodology developed used the experience of the PTS assessment at Loviisa (Finland), RPV Stade (Germany), RPV-s DOEL (Belgium); US practice and IAEA documentations. The paper presented the PTS methodology in detail including the transient selection, material database and K_{1c} reference curves. During the assessment the ASME- K_{IR} (Reference Fracture Toughness) curve was used. The purpose of the analyses was to verify the operational safety until the second periodic safety report (PSR). It concludes that the units 1-4 can be safely operated not only until the 24th operational year, but much longer. At the end of the paper a list of life management actions is presented.

Ilieva.pdf, Apostolov T., Ilieva K., Belousov S., Petrova T., Penev I., Taskaeva M., Taskaev E., Trifonov A., Prodanova R., Peneva Tz., Genchev T." Analyses for evaluation of reactor pressure vessel metal state and life time at Kozloduy NPP Unit 1", Paper published on BGNS Transactions, vol. 1, n.1, 1996, p. 30-35.

The paper presents the evaluation of the life time of the RPV Kozloduy NPP unit 1. The paper shows the results of chemical composition analysis, P and Cu content of the samples cut from weld material. It also presents the fluences measured and calculated after 17th cycle of operation. The T_k value is calculated using empirical relations for weld and base metals for the manufacturer and measured Cu and P content. It presented. The calculated shifts of the transition temperature caused by radiation and after annealing are presented. The paper concludes that the RPV satisfy the criteria for safety operation for 18 operational cycles.

Kamenova002.pdf, Vonenlcharov St.,Kamenova Tz.,Tzokov P.,Videnov A.,Petkov B.,KOZLODUY " NPP WWER - 440/230 REACTOR PRESSURE VESSEL RADIATION LIFE TIME", Paper published in

The aim of the work is to compare the RPV WWER440 NPP "Kozloduy" radiation life time, calculated by the different re-embrittlement rate laws after annealing using updated parameters describing neutron irradiation embrittlement and standard method for RPV integrity assessment

Mitigation efforts performed to extend the safe service life of the RPV. The paper presents a formula for calculation of critical temperature of embrittlement (T_{kf}), (ductile to brittle transition) according Russian standards. Chemical analyses of weld 4 for WWER440/230-s provided in the passport of the RPV-s are used in the formulae to predict T_{kf}. The paper provides the results for all units of Kozloduy.

The T_{kf} after annealing also presented and the effect of re-irradiation calculated using the "lateral (horizontal) shift" model. The re-irradiation embrittlement rate is calculated for all units of Kozloduy. Finally the lifetime assessment compares results of the conservative and horizontal embrittlement model. In all calculation real fluence results used and it extends the operational lifetime.

Karzov001.pdf, Karzov G., and Timofeev B.," Ageing of RPV and Primary Circuit Piping Materials forNPP with WWER during a Design Operation Time", Paper published on BGNS Transactions, vol. 1, n.1, 1996, p. 30-35.

The paper presents data of thermal ageing effect on mechanical properties of steels of NPP equipment, various types of WWER reactors, after a prolonged operation. The paper also provides a table with the main dimensions, operational parameters and material for NPP components. It provides also the estimated mechanical properties for the whole designed operation (30 years). The paper includes a overview of thermal ageing results of several references. These data are obtained for base metal and welded joints of reactor pressure vessels manufactured from 15Cr2MoV and 15Cr2NiMoV steels and piping of the primary circuit produced from 22K, 10MnNi2MoV, 16MnSi and 08Cr18Ni10Ti steels.

After 100000 hours of operation the primary coolant circuits of RBMK-1000 and WWER-440 reactors showed no degradation of the mechanical properties. The comparison of initial properties with the same properties after 100000 and 200000 hours operation are presented and results the same conclusion.

Karzov007.pdf, Anikovsky V.V., Karzov G.P. and Timofee B.T." PREDICTION OF SERVICE LIVE EXTENSION OF STRUCTURES OF NPPs CONTAINING THE RBMK REACTORS", Paper presented at

Abstract presented the tasks involved in NPP service life extension of RBMK reactor structures. The standard linear elastic fracture mechanics can't be used, since the wall thickness of the pipes are small and they made from low strengths steel. Due to thickness and non-defined stress the present norms are not usable. A new approach is proposed based on the construction of the fracture analysis diagram of the Pellini type.

Karzov009.pdf, Karzov G.P., "Basic Material Issues on Service Life Extension of NPPs with the WWER Reactors, Paper presented at

Abstract presents the work in the program to validate the lifetime extension of WWER-440/230 RPV units 3 and 4 at Novo-Voronezh NPP and units l and 2 at Kola NPP. The lifetime is limited by the shell metal of core zone or circumferential welds of the RPV. From the weld region mini Charpy samples were cut out. The DBTT was obtained on the specimens. On the base of the results the necessity of annealing can be considered. The paper presents different approaches as the use of Master curve, traditional Charpy approach etc.

Kryukov004.pdf, Kryukov A.," The state of the art of WWER type RPV: radiation embrittlement and mitigation", Paper was published in SPECIALISTS MEETING ON IRRADIATION EFFECTS AND MITIGATION, Vladimir, Russia, 15-19 September, 1997

The paper summarizes the results obtained in the TACIS 91 program.

The different types of WWER RPV-s have different problems. Some of the WWER 440-V230 RPV (old generation) have high phosphorus and copper content in the welds, lack of surveillance program and 9 vessels are not cladded. Few difficulties occur with the second generation WWER-440 V213 reactors, and the high nickel content of some weld of the WWER-1000 reactors also discussed.

Most surveillance and research data are showing the good correlations between transition temperature shift (ΔT_k) measured and calculated accordance with Russian Guide. The paper describes the distribution of the irradiation assisted phosphorus precipitations near the grain boundary, the dependence of DBTT shift (increases) on nickel content.

After the annealing of the first generation vessels the re-embrittlement rate is generally different from the first embrittlement rate. The paper discusses the use and comparison of Lateral or Conservative model.

During the test program efforts was made to find good correlation between standard and subsize Charpy specimens.

The paper recommends working on the following tasks for WWER-440 life management: to elaborate new Codes on the modern database, to justify the model for re-embrittlement (after annealing) prediction and to create International Data Base on Aging Management and Life Extension (IAEA).

Levit-001.pdf, Levit V.I." WWER-type nuclear reactor pressure vessel: Material radiation ageing issues and effect of thermal annealing as a mitigation method", Paper published on Ageing of materials and methods for the assessment of lifetimes of engineering plant, Penny (ed.), 1997, Rotterdam, ISBN 9054108746.

The paper describes the degradation of RPV materials during service termed radiation ageing (RA). RA mechanisms include radiation embrittlement (RE) effects, increase in DBTT, radiation hardening (RH) and radiation strengthening (RS). Diagrams summarize the DBTT shift (TTS) of WWER-type RPV materials in function of CU, Ni and Ni associated to Cu and P contents. The TTS calculated using USA, France and Russian equations considering chemical composition and fast neutron fluence. RH is mainly due to formation of small sized precipitations localized on vicinity of grain boundaries and grain bulk, they tend to grow in size and decrease the concentration in the bulk with increasing fluence. The recovery of the RPV materials during annealing is presented too.

The most important parameter for the degree of DBTT recovery is the difference between the annealing and irradiation temperatures. Higher temperature differences indicate a larger degree of DBTT recovery. Material singularities, chemical composition and produced defects will affect the annealing recovery rate.

The paper shows the recovery rate dependence on the annealing temperature. Formulae to calculate the residual transition temperature shift in the function of P content is elaborated.. The paper recommends to use the called lateral shift approach at re-irradiation to avoid conservatism on life assessment. Hardness measurement of inner or outer vessel wall is a good control method for measure the efficiency of annealing but further development on NDE methods is also recommended.

Nikolaev_YU-001.pdf, Nikolaev Yu.A.,Nikolaeva A.V.,Shtrombakh Ya.I.,"Radiation embrittlement of low-alloy steels", paper published on International Journal of Pressure Vessels and Piping 79, 2002, pp. 619–636.

In the first part the paper presents P, Cu and Ni effect on mechanical properties of irradiated material. Materials with different composition were exposed to neutron irradiation at 270–275 °C for 319 days. Fast neutron fluence was about ~ $8*10^{23}$ neutron/m^2. The neutron irradiation increases the yield and ultimate strength, transition temperature and reducing upper shelf of absorbed impact energy temperature dependence. Material with minimal phosphorous, copper and nickel contents shows the smallest radiation sensitivity. It was observed the Synergy effect between P and Ni has been observed and similar effect expected between Cu and Ni. The paper also presents results and considerations based on previous analysis referenced in the literature. It seems to be that molybdenum slightly reduces the diffusion and embrittlement rate, silicon increases them. Experimental database used to perform a statistical analyses and the result compared to the results calculated using the chemistry formulae of the in Russian Guide for WWER-440 and WWER-1000 reactor pressure vessel materials (base and weld). It is proved that the formulae for the WWER-1000 materials are not conservative in all cases and should be reconsidered.

Using the results of the statistical analysis of the database the paper proposes a new model for irradiation embrittlement ΔTF (transition temperature shift) with dependence on fluence and composition. Cu, P, Ni, and Mo are considered in the equations, but the correlation with the test results are still weak, showing that the chemical formulae without the production history of the materials give only a first guess. Large number of diagrams provided on the effect of the different alloying and polluting elements and on they synergism, unfortunately many of them are valid only for two compositions and for two fluence values.

For the WWER-440 welds clear dependence of radiation embrittlement on phosphorous and copper contents was found.

The paper also discusses the recovering of the mechanical properties by annealing. DBTT recovery rate of WWER-1000 RPV steels is much lower than WWER-440 RPV steels. It is found that the incomplete recovery is caused by to segregation of impurities at grain boundaries. The effectiveness of recovery depends on the Ni content and complete recovery was not achieved even after 490°C for Cr-Ni-Mo steels.

Oszvald001.pdf, Oszvald F.,” Research results from WWER-440”, Paper published on IAEA Specialist Meeting on "Irradiation Effects and Mitigation", Russia, 1997

At NPP Paks specimens sets have been irradiated in empty surveillance channels to prepare an extended surveillance program. The irradiated materials were: JRQ, JWQ (the IAEA reference steel and weld), a research heat of 15H2MFA base metal coded as CS, a weld coded as CSW and two model alloys from the IAEA CRP-3 program. The paper describes the encapsulation and the irradiation in a WWER-440 reactor. The CS had medium P and low Cu since CSW had high P and medium Cu. The samples were irradiated for 1 and 2 years and were investigated by mechanical testing. Yield strength increases substantially for JRQ material. CS was has a low sensitivity for embrittlement. After testing the CS steel specimens a hard layer on the surface of the specimens was observed by SEM. This layer was observed first in 1987, by testing the original surveillance specimens. It seems that the development of surface hard layer by irradiation is typical for V alloyed steels. The Charpy test shows low hardening response of the materials except in the case of JRQ and JWQ which presented high sensitivity to irradiation embrittlement. The same was observed by hardness measurements. For studying the post irradiation annealing effect broken halves of Charpy specimens treated at a temperature of 475 °C for 150 h and presents 100% recovery. The hardness of annealed specimens were same than obtained on the original unirradiated ones.. A surveillance weld metal tensile specimen irradiated for four years also have been annealed. Hardness distributions from surface to middle of the specimen were provided and showed that the hard layer has remained on the surface after annealing.

Oszvald002.pdf, , Elter J., Fekete T., Gillemot F., Oszwald F., Maróthy L. and Rátkay S.,”PTS ASSESSMENT -THE BASIS OF LIFE TIME EVALUATION AT NPP PAKS”, Paper published on IAEA Specialist Meeting on "Pressurized thermal shock”, Esztergom, Hungary, 1997

The paper presents results and evaluation of PTS study was performed on unit 3 of NPP in the frame of the AGNES project.

The main actions of the project were the selection of the transients, RPV geometry, thermodynamic calculations, material database and selection of hypothetical defects (3 models). In the calculation fracture mechanical integrity analysis, the fracture toughness (K_{1c}) and the crack arrest reference (K_{1a} or K_{IR}) curves are used. A work has been organized as first simplified conservative analyses performed, and if the results were not satisfying, the analyses continued with a more detailed less conservative method. Phase III take into account crack arrest, namely if the crack becomes stable before reaching 70% of the wall thickness the vessel integrity is not affected by the tested PTS case. The PTS calculations performed in the frame of the AGNES project have shown that NPP Paks units 1-4 can be safely operated at least until the next periodic safety report (PSR at the 24th operational year), even much further, but the purpose of the analyses was only to prove the safety until the next PSR. Finally the paper presented a short list of the life management actions of which may be used to reach at least 40-60 years of safe operation.

Pistora-001.pdf, Pistora V., Kral P., Evaluation of Pressurized Thermal Shocks for WWER 440/213 Reactor Pressure Vessel in NPP Dukovany, Paper published in the proceedings of SMIRT17, 2003, Prague, Czech

The paper presents the PTS evaluation of the NPP Dukovany. It has 4 units of WWER 440/213 reactors. Deterministic approach was used based on Czech standards and IAEA guidelines. The PTS analyses will determine the maximum allowable RPV material critical temperature of brittleness. The actual value of the RPV can be obtained by surveillance program. The work started by selection a list of transients: large breaks in main stream system (MSS), leaks from primary circuit through the pressurizer safety valve, spurious signals, LOCA and leaks from primary and secondary circuit. Altogether 50 thermal hydraulic (PH) transients selected and 33 structural analyses performed.

The thermal hydraulic calculations used conservative assumptions namely it used a 2D nodalisation model of the reactor down comer. The structural analyses was focused on the most embrittled weld in the beltline zone. For FEM calculations the SYSTUS and COSMOS/M codes were used. The defects were postulated according to IAEA guidelines. The paper presented some lessons learned with the PTS project. The final results of the PTS analyses are still to be concluded.

Platonov002.pdf Platonov P.A., Shtrombakh Ya.I., Kryukov A.M., Nikolaev Yu.A.," Evaluation of WWER-440 RPV metal condition and possibility of their lifetime extension", only abstract

Data from tests of WWER-440 RPV material show the equation to evaluate DBTT shift is not conservative and must be reconsidered. On studies of re-embrittlement after pos-irradiation annealing the lateral shift presented to be the most suitable to describe embrittlement kinetics. It concluded by affirming that the WWER-440, 230 and 213 RPV material conditions guarantee the design lifetime and permit extension.

Schmidt-001.pdf, Schmidt J.,Strombach J. and Gledatchev I. Reactor vessel integrity analysis of the Bulgarian NPP Kozloduy Unit 1,Paper published in the proceedings of PLIM + PLEX International conference, 1997, Prague, Czech

The paper presents the PTS analysis of the Bulgarian NPP Kozloduy Unit 1. The PTS calculation performed according to the German procedures. The critical situation will arise when during core cooling the injected water comes into contact with the RPV wall in the down comer. The selected transients were DN 20 break, main stream line break, lift up of SG collector lid, DN 100 break of pressurizer spray line and inadvertent opening of pressurizer safety valve. The thermal hydraulic (TH) calculations were performed by RELAP analyses. A conservative assumption within the TH analysis is the hot-zero-power condition. The most severe transient was the main steam line break with loss of offsite power and combined opening of 2 neighboring safety valves through whipping.

Fracture mechanics analysis was performed with 56N/mm^2 residual stress in the weld. The supposed minimum defect size was 10 mm (based on sensitivity of NDT). The T_{Ka} (allowed transition temperature) was calculated by FEM, for 10mm defect size the T_{Ka} =178°C . The material property was also obtained by sample testing. The T_k measured on mini Charpy specimens and its value compared to with the T_{ka} calculated by FEM. Using the Russian code the T_{KF} obtained from weld samples was 91.5°C. The paper shows that proper operator actions during PTS can improve situation and increase the margin between T_K and T_{Ka} . The operator actions are even more effective for LOCA cases.

Sehgal-001.pdf, Sehgal B.R., Theerthan A., Giri A., Karbojian A., Willschütz H.G., Kymäläinen O., Vandroux S., Bonnet J.M., Seiler J.M., Ikkonen K., Sairanen R., Bhandari S., Bürger M., Buck M., Widmann W., Dienstbier J., Techy Z., Kostka P., Taubner R., Theofanous T., Dinh T.N., " Assessment of reactor vessel integrity (ARVI)", Paper published in the Nuclear Engineering and Design 221 (2003), pp. 23–53,

The paper presents the cost-shared project ARVI (assessment of reactor vessel integrity). Nine organizations from Europe and USA participated. The objective of the ARVI Project is to resolve the safety issues that remain unresolved for the melt vessel-core interaction during a severe accident, i.e. to provide data and the models to assess (a) the feasibility of promulgating the in-vessel melt retention (IVMR) scheme in current and future plants or in its absence and (b) the time available before vessel failure. The description of these interactions, loads and vessel behavior requires integration of the sciences of thermal hydraulics, materials and metals phase change, it includes scale experiment and modeling. The major experimental project is work package 1 (WP1) EC-FOREVER in which data is obtained on melt pool natural heat convection and lower head creep and rupture. 1/10th scale model vessels with different carbon steel lower heads manufactured. Four vessels were tested using French 16MND5 and German 15MO3 steel. The paper presented the EC-FOREVER-2 test description in detail and results with analysis.

The WP2 is related to melt pool heat convection experiments in different facilities. All experiments are described in the paper. In the conclusion the paper presented that the so called BALI melt stratification experiments have shown that the metal layer focuses the heat flux transversely to the vessel wall. Similar results were obtained in the so called SIMECO integral experiments. Cooling of the metal layer at upper surface eliminates the focusing effect. Experiments performed for the stratification of the oxidic pool in the SIMECO and the COPO facilities show that the upwards/downwards heat flux ratio is drastically reduced. WP3 increased the analysis capabilities in the area of structural mechanics of creep failure of a PWR vessel, by using the ANSYS-Multiphysics, VTT PASULA and FRAMATOME SYTUS+ codes. WP4 dealt with modelling activities, namely, CHF (critical heat flux) modeling in gaps, code validation and the development of simple models, with input from previous work. WP5 made the assessment of an in-vessel melt retention scheme for WWER-440/213 plants in Hungary and the Czech Republic. In this assessment the models were performed by VESSEL and MVITA codes. The analyses with VESSEL and MVITA in VEIKI concentrated on the focusing effect of the metal layer that represents a challenge to the reactor vessel wall. The VESSEL code calculation with thin metal layer showed higher heat flux between the metal layer and vessel wall. The MVITA calculation gave lower heat fluxes. Some of the data, the insights and the models developed are being converted to simpler models, which are incorporated in the accident consequence analysis codes. Some part of the work could be used even for other industries.

Sevini-001.pdf, Sevini F., Debarberis L., Torronen K., Gerard R. and Davies L.M." Development of the AMES network throughout the 4th and 5th EURATOM framework programs", Paper published in International Journal of Pressure Vessels and Piping 81 (2004) 683–694

AMES (Ageing Materials European Strategy) network was established in 1993 to gain expertise on nuclear reactor materials assessment and research on ageing management. Several projects were carried out relating to understand the influence of various embrittlement mechanisms (PISA, model alloy actions); develop new techniques

(GRETE, RETROSPEC); improve the dosimetry aspects (REDOS); improve surveillance and prediction of irradiated material fracture toughness (FRAME). The new organization of AMES in dedicated task groups, together with the creation of the ATHENA thematic network, will make it possible to address in an effective way key ageing issues. During the 4th EURATOM framework program the projects REFEREE, RESQUE, MADAM and AMES-NDT were performed. The paper described the project and gave an overview of the outcome. REFERRE performed a detailed comparison of irradiation-induced shifts in DBTT measured by different test on 15X2MFA (WWERs), JRQ (irradiation sensible IAEA reference), 18MND5 (SG plate) and Bradwell weld materials were tested by the various participants. It presented that Irradiation-induced static and dynamic fracture toughness transition temperature shifts are similar, but there is a trend for the static shift to be larger than the dynamic shift. There is a need for a rigorous statistical analysis of the combined datasets from which correlations are produced. The RESQUE project was designed to evaluate two different reconstitution techniques, namely stud-welding and electron-beam welding. The use of long irradiated remnants of the surveillance specimens and application of reconstitution welding is a good and cheap way to get further information and enhance the plant safety. Electron beam welding is the best method to reconstitute the specimens, the loss of insert length is larger in the case of stud welding, but stud welding also can produce good quality and the investment is considerably less. The MADAM project deals with studies on damage indices and neutron dosimetry. The AMES-NDT was a round-robin test to evaluate different non-destructive testing and monitoring techniques which can be used to determine the presence of structural damage and degradation on nuclear materials. Within the 5th EURATOM framework programme a project named ATHENA will start in 2001. It will summarize the obtained achievements and edit guidelines on important issues like the use of Master Curve, effect of chemical composition on embrittlement rate in RPV steels, re-embrittlement models validation after WWER-440 annealing and open issues in embrittlement of WWER type reactors. The purpose of is PISA to improve the understanding of irradiation embrittlement, especially the role of phosphorus. COBRA is a part of Copernicus Program, this project will tackle the open issue given by the uncertainty in measurement of the correct irradiation temperature, FRAME concerned with fracture mechanics based trend curves for PWR and WWER RPV materials to validate the use of the Master Curve approach, GRETE follow-up of AMES-NDT, and REDOS follow-up of MADAM accurate determination and benchmarking of radiation field parameters, relevant for RPV monitoring.

The SAFELIFE promoted by JRC will comprise actions and expert group development on NPP Plant Life Management.

Vodenicharov001.pdf, In Russian

Vodenicharov002.pdf, Vodenicharov St.," REST LIFE TIME MANAGEMENT OF KOZLODUY NPPP UNIT 3 AND 4", Paper published in

The paper provides information on NPP Kozloduy Unit 3 and 4. It affirms that based on previous analysis the lateral law of re-irradiation should be applied for KNPP3 RPV radiation life time assessment. T_k calculations made according lateral embrittlement law show that the KNPP3 RPV integrity is proved practically up to 28th operational cycle. The T_{kf} shift for Unit 4 will be below safety limit beyond the design lifetime. Description of the about the surveillance program under development is also provided.

Vodenicharov004.pdf, Vodenicharov St., Kamenova Tz., Tzokov P., Videnov A. and Pekov B.," KOZLODUY" NPP WWER - 440/230 REACTOR PRESSURE VESSEL RADIATION LIFE TIME", Paper published in

The paper comments on the importance of predicting the embrittlement by empirical methods and re-embrittlement models. It presented the action for mitigation of embrittlement on NPP Kozloduy by installation of dummy elements to decrease the neutron fluence on RPV wall and to apply annealing to recover the RPV wall properties. The critical temperature of embrittlement or critical temperature of ductile to brittle transition T_{kf} calculated according to Russian standards is presented too. The chemical composition and T_{k0} are obtained from factory information and from tests for all the four units.

After annealing the T_{kf} is determined with the lateral (horizontal) shift model and presented for all 4 units. Finally the paper presents the lifetime assessment by conservative and lateral model with the design and with the real calculated fluences. A new method for RPV life time determination by reduction of the maximal allowed flaw is proposed by Gidropress. This method is applied now on Unit 1 and designed life time (30 years) is proved.

Vodenicharov-008.pdf, Vodenicharov St., Kamenova Tz., Tzokov P., Videnov A. and Petkov B.," KOZLODUY" NPP WWER - 440/230 REACTOR PRESSURE VESSEL RADIATION LIFE TIME", Paper published in Proc. of the Scientific-Technical Conference dedicated to the 20th anniversary of Kozloduy, 1994.

The paper considers the importance of predicting the embrittlement by empirical methods and by re-embrittlement models. It presents the action for mitigation of embrittlement on NPP Kozloduy e.g. installation of dummy elements to decrease the neutron loading on RPV wall or material property recovering using annealing. It presents the critical temperature of embrittlement or critical temperature of ductile to brittle transition T_{kf} calculated according to Russian standards. The paper gives the chemical composition and T_{k0} calculated either by factory data or tests performed on the four units.

After annealing the T_{kf} is determined with the lateral (horizontal) shift model. Finally the paper introduces the lifetime assessment by conservative and lateral model with using desig and real calculated fluences for 3 units.

A new method for RPV life time determination using maximum allowed flaw is proposed by Gidropress. This method is applied now on Unit 1 and designed life time (30 years) is proved.

Vodenicharov-016.pdf, Vodenicharov St." REST LIFE TIME MANAGEMENT OF KOZLODUY NPPP UNIT 3 AND 4", Paper published in the International Symposium on Nuclear Power Plant Life Management, Hungary, 2002

The paper provided basic information on the NPP Kozloduy Unit 3 and 4. Mitigation actions were performed to decrease neutron loading. Recovery annealing was applied to weld 4 of Unit 3.

It affirms that based on previous analysis the lateral re-embrittlement law should be applied for KNPP3 RPV radiation life time assessment. T_k calculations made according lateral re-embrittlement law show that the KNPP3 RPV integrity is proved practically up to 28th operational cycle. The T_{kf} shift for Unit 4 will be below the safety limit even beyond the design lifetime. A surveillance program is under development with Charpy and tensile samples placed into Unit 1 low and Unit 2 high flux channels. For Unit 3 material is planned to be irradiated, annealed and re-irradiated. For unit 4 the EOL fluence will be in the range of 8 to 14 x 10^{19} cm^{-2}.

10.4 Further References

none

11 Surveillance

In this part of knowledge preservation 30 papers were reviewed dealing with the surveillance specimen program application on the WEER-440 and WWER-1000 units. Authors prepared most of papers from the countries using WWER technology, such as Bulgaria, Czech Republic, Finland, Hungary, Slovak Republic, Russian Federation and Ukraine.

The main issues are:

- Upgrading the surveillance databases by upgraded neutron dose measurements,
- Acquisition of new impact test and toughness results on reconstituted surveillance specimens, including the evaluation of the 'Master Curve Approach',
- Validation tests of the shape of the fracture toughness curve and the base and weld metal and characterization of the cladding,
- Preparation of upgraded RPV integrity assessments, based on the latest approved methodology.
- Several papers are dealing with comparison of surveillance specimen results from both WWER-440 and 1000 units.
- There are discussed the motivations and reasons for surveillance specimen programs upgrading or modernization, new philosophy and approaches of modernized SSP, as well as demands from national regulatories.
- The mitigation of irradiation embrittlement of RPV`s through thermal annealing of older WWER-440 units were very successfully implemented in Bulgaria, Finland, Slovakia and Russia, with aim of planned LTO of these units.
- The efficiency of the RPV steel recovery was tested by implementation several modern procedures like boat sampling from RPV walls, instrumented hardness measurement, neutron diffraction, positron annihilation etc.
- The results of very detail analyses of irradiated samples from RPV materials by most modern methods application like FIM, APFIM and changes in magnetic parameters were published too.
- The aim of these sophisticated analyses was to understand the irradiation embrittlement phenomena.
- It was shown that radiation embrittlement of the weld metal is determined by the processes associated with the individual effects of copper, phosphorus, nickel, manganese and their joint effect
- The evaluation of RPV`s irradiation embrittlement trend during planned LTO is one of the most important issues too.
- Critical analyses of existing rules and codes followed to the activities to improve older or prepare new one, with aim to lower the over conservative approach by life-time evaluation.
- The history and present status of surveillance specimen programs in WWER countries are published in several papers.
- From this historical sight it is possible to see the continual effort both of researchers and regulatories to improve and modernize the surveillance specimen programs.
- Several theoretical models were presented mostly from Russian scientists, which may more or less describe the influence of impurities in RPV steels.

According to the PLIM and LTO of operated WWER units considering the ageing mechanisms are mostly evaluated:

- Irradiation embrittlement,
- Low cycle fatigue,
- Head cracking,
- The failure of the threads and thread housings,
- The state of the concrete supports,
- The ageing of the RPV internal structures.

11.1 Consolidated Conclusions

The monitoring systems of the NPP`s material degradation due to the irradiation, corrosion and thermal ageing processes must meet the requirements of:

- National Regulatory Authorities,
- recommendations of IAEA,
- internationally approved standards,
- safe and reliable operation of all WWER units.

11.2 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 if there is a great lack of experimental materials,
- the procedures for evaluation of power uprate influence,
- what are the properties of high irradiated heat affected zone around the RPV's welds and below the austenitic cladding,
- the procedures for evaluation of new fuel generation influence,
- how to test the irradiation embrittlement of reactor internals.

Analyzing and finding answers for these issues is a great challenge for future research projects and for new surveillance programs.

11.3 Reviewed papers and summaries

Ahlstrand R., Michel Bièth, Claude Rieg "Neutron embrittlement of WWER reactor pressure vessels—recent results, open issues and new developments", Published in: Nuclear Engineering and Design 230 (2004) 267–275

The procedures of verification of design plant lifetime and determination of the component and circuit condition are the topic of this paper.

Neutron embrittlement of reactor pressure vessels (RPV) is a crucial consideration for continued safe plant operation. Since 1991, the European Commission (EC) has financed a significant number of projects in this area, in particular through the TACIS and PHARE programs, the countries mainly concerned are Russia, Ukraine, Armenia, and Kazakhstan for the TACIS program, and Bulgaria, Czech Republic, Hungary, Slovak

Republic, Lithuania, Romania and Slovenia for the PHARE program. Recent confirmations of the irradiation temperatures of the surveillance specimens in the operating WWER nuclear power plants show that the surveillance specimens can be used further for the validation of the current and the expected neutron embrittlement. Two new TACIS projects are being launched, jointly with Russia and Ukraine, whose scope is:

- Upgrading the surveillance databases.
- Acquisition of new impact test and toughness results on reconstituted surveillance specimens, including the evaluation of the Master Curve Approach.
- Further validation tests of the shape of the fracture toughness curve and the base and weld metal and characterization of the cladding.
- Preparation of some selected upgraded RPV integrity assessments, with insights on the latest approved methodology.

The paper summarizes the major conclusions of the recent completed EC projects, reviews the remaining major open issues in the field of reliable determination of fracture toughness properties of the operating WWER RPV's, and details the scope of the new projects.

The results provided recently by the TACIS projects (namely SRR2/95 and R2.06/96) and some preliminary ones from the COBRA shared cost action have been recalled in this paper. The temperature registration carried out at Balakovo 1 showed a limited overheating of the surveillance specimens. Furthermore, the following actions were recommended:

- Neutron dose estimation upgrading by using refined calculation results systematically and 93mNb activity, measures for future surveillance sets.
- Complementary reconstituted specimens from broken ones for impact and static toughness tests, which will allow additional T_k and K_{1c} (T0) data to be obtained.

The possible change of shape of the fracture toughness curves for high-irradiated materials and the possible existence of any flux effect, which would require correction, may require some investigations in the near future.

For the WWER 440/213 RPV's an experimental program is being performed which will assess the real irradiation conditions (irradiation temperature) and provide for additional reference dosimetry data.

Ahlstrand, Jokineva H. and Kohopää J. "Radiation embrittlement of WWER-440 reactor vessel - Finnish experience", Nuclear Europe Worldscan 5-6/1995

The main results of the two WWER-440 Loviisa surveillance specimen program are presented. The plants Soviet-based design has been adapted to Finnish conditions and "Western" safety requirements. Because the water gap between the RPV wall and the core is small the neutron flux in the core region of the RPV wall is high. Due to the high neutron flux, the embrittlement of the RPV steel, especially the single circumferential core weld, has been the most serious problem of older WWER440 plants.

The Loviisa units were the first WWER440s with an official RPV surveillance program. First surveillance results showed irradiation embrittlement of the RPV to be much higher than expected. Since those first alarming results in 1980, a lot of work has been done to mitigate embrittlement of the RPV materials and to soften potential PTS (pressurized thermal shock) transient. The research has also aimed to increase the accuracy of the structural safety analyses.

To solve the problem of high embrittlement rate was implemented the core reduction by replacing peripheral fuel assemblies with dummy assemblies. The flux to the critical direction was reduced by 83%. Later, adopting the low-leakage core gave extra reduction of flux on the reactor wall.

Other measures implemented to reduce the frequency and thermal stresses of pressurized thermal shocks (PTS) were:

- increase of water temperature in the emergency core cooling (ECC) accumulators and tanks;
- modification of operation pressure temperature limits;
- decrease of high-pressure injection flow;
- decrease of the head of the high-pressure injection pumps and increase of pressurizer relief valve capacity to avoid potential opening of the safety valve in some PTS events;
- introduction of a signal from high primary system pressure to stop the high pressure pumps connected to the primary circuit;
- modification of protection signals relating to steam line break.

For the analyses of RPV`s structural safety in Finnish NPP`s, deterministic safety analyses are considered decisive by the national nuclear authority. A WWER-440-specific surveillance database was not available. So the material properties used in safety analyses have primarily been based on the plant-specific surveillance test results.

K_{1c} transition curves for irradiated material condition have been determined by two methods:

- K_{1c} values were measured directly with irradiated pre-cracked Charpy sized specimens using static three-point band testing;
- Using a standard reference K_{1c} -curve shifted by the transition temperature shift measured by Charpy impact testing. For the irradiated weld material, both methods gave practically the same brittle fracture transition curve.

The neutron fluence distribution on the vessel wall was calculated. For verification purposes, 23 material samples were ground from vessel inner surface and 15 samples from vessel outer surface. Also, dosimetry wires spanned the outer surface of the RPV in horizontal and vertical lines. The consistency with fluence calculations and activity measurements was excellent.

Beltline zones of the reactor vessels have been 100% non-destructively tested every eight years. The detection limit for a crack has been evaluated to be 5 x 10 mm with more than 90% probability. A safety factor of 3 was chosen and an elliptic crack of 15 mm x 30 mm postulated.

In addition to normal mechanical and thermal stresses, residual stresses of the main weld seam and the cladding were included in the analyses. The magnitude of residual stresses in cladding was determined by simulating the last heat treatment and hydro test pressure load of the RPV with finite element methods.

The utility has decided to anneal the critical weld of the RPV in Loviisa-1 during the 1996 summer outage. Hence, definition of the effects of annealing on the RPV materials is an urgent task. Because the amount of irradiated material is limited, the research programs are carefully planned. Reconstitution of the broken surveillance specimens was developed and the method is in use in annealing and re-embrittlement studies since the mid 1980s.

Loviisa-1 RPV was annealed using technology from Czech Skoda Nuclear Machinery, Plzen Ltd with equipment and staff from Slovak Bohunice NPP during the 1996 summer outage.

Amir D. Amayev, Aleksandr M. Kryukov, Vladimir J. Levit, Mikhail A. Sokolov "Radiation Stability of WWER-440 Vessel Materials.", Published in: "Radiation Stability of WWER.440 Vessel Materials, " Volume J, ASTM STP 1170. Lendell E. Sttele, Ed., ASTM, Philadelphia. 1993. pp. 9-29

The main results of a complex investigation of radiation embrittlement of WWER-440 reactor vessel (RV) materials carried out in the former USSR are presented in this paper. The subject of the investigation was surveillance specimen (SS) evaluations of RV materials. It has been found that at an irradiation temperature of 270^0C neither the base metal (steel 15Kh2MFA) nor weld metal exhibits saturation of radiation embrittlement in irradiation of specimens up to neutron fluences of 7-x $l0^{20}$ n/cm^2 (E > 0.5 MeV).

Regularities in the influence of impurity' elements (copper and phosphorus) on radiation embrittlement of RV materials have been investigated too.

It is shown that radiation embrittlement of the weld metal is determined by at least four processes:

- Associated with the individual effects of copper
- Individual effects of phosphorus,
- Their joint effect
- The mechanism of direct buildup of radiation defects in the metal.

The effect of dependence or the radiation embrittlement characteristics of the WWER-440 RV materials on the neutron flux has been found. It is shown that within the range of fluences from 1 to 5 x 10^{18} n/cm^2 at a neutron flux of 4 x 10^{11} n/cm^2.s radiation embrittlement is stronger than in irradiation at neutron flux of 3 to 4 x 10^{12} n/cm^2.s. A tendency has been observed for the values of the radiation embrittlement coefficients A_F, characterizing sensitivity of the steel to radiation embrittlement, to reduce with an increase of neutron fluence in irradiation of specimens with a neutron flux of 4 x 10^{11} n/cm^2.s

The procedure for irradiation of the RV steels in NPP reactors has some advantages over that for irradiation in research reactors:

- It is possible to obtain experimental data on the fast neutron fluence effect on the mechanical properties of the RV materials exposed to irradiation for a long time under comparable conditions with respect to irradiation temperature, neutron flux parameters, coolant effect, and gamma rays.
- Identity of the irradiation parameters is possible for a great number (a few hundred) of specimens with various contents of alloying and impurity elements; the degree of the influence of these elements on radiation embrittlement of the steel can be determined.
- Activation neutron flux indicators set directly in the specimens or in the irradiation containers are used; a relatively high accuracy (±5%) of measurement of the fast neutron fluence affecting the specimens is assured.
- Irradiation of specimens within unsealed containers simulates operation conditions for the unclad RV, thus, simultaneous evaluation of the neutron f1uence and the

reactor coolant on the whole complex of the mechanical properties of the steel is possible,

The determination of the SS irradiation temperature is a complicated problem because no absolute temperatures or the current monitor exists. A method of using diamond powder as a temperature monitor, accepted earlier in the former USSR, has proven to be very sensitive to the neutron flux parameters. Therefore, to establish the temperature of SS irradiation in WWER-440, in the Armenian NPP-2 reactor, special calorimetric experiment was realized.

Bieth M., Rieg C., Ahlstrand R. "New TACIS regional projects on radiation embrittlement and integrity assessment of WWER reactor pressure vessels", Published in: International Journal of Pressure Vessels and Piping 81 (2004) 677–682

The WWER 1000 and 440/213 NPP's are at the centre of the electricity generating capacity in Russia and Ukraine, and will remain so for a significant period of time. Therefore, it is of major importance to address the safety issues concerning the RPV's and related ageing mechanisms, in order to provide pertinent figures for decision making about further safe operation of these plants.

Projects and Shared-Cost Actions in the area of the reactor pressure vessel (RPV) WWER 440 type of NPP core weld neutron embrittlement of the is the topics of this paper. As a basis for the development of the necessary material data to be used for upgraded RPV integrity assessment, two new TACIS projects are being launched, jointly with Russia and Ukraine. The paper presents the objective and the scope of these projects, which are concentrating on:

- Upgrading the surveillance databases by upgrading the neutron dose measurements,
- Acquisition of new impact test and toughness results on reconstituted surveillance specimens, including the evaluation of the 'Master Curve Approach',
- Further validation tests of the shape of the fracture toughness curve and the base and weld metal and characterization of the cladding,
- Preparation of some selected upgraded RPV integrity assessments, with insights on the latest approved methodology.

Clarification of the embrittlement mechanisms that take place in the RPV wall during operation and the development of more adequate and accurate procedures for the characterization of the material toughness, which are able to reduce the inherent margins of the codified indexation procedure, is the main goal of these projects too. Adequate surveillance program evaluation is also an important target, especially for demonstrating the possible safe long-term operation.

The final goal is to provide the Russian and Ukrainian operators the evidence of acceptable safety margins and the expected remaining lifetime of the plants. Recommendations for immediate or later implementation of mitigation measures will be given to these beneficiaries. WWER 1000 RPV's, with particular insights on those having a high Ni containing core weld, and some sensitive WWER 440/213 RPV's are of most concern.

The major concern is the evaluation of upgraded deterministic margins against catastrophic failure of the RPV in case of potential pressurized thermal shocks, which could occur during normal, upset or emergency conditions.

There are basically three main parameters to be looked at: the expected materials' toughness, the accurate loadings at the postulated crack tip and the justification of the size and shape of the postulated cracks. The program concentrates mostly on the prediction of the materials' toughness, mainly by reassessing the surveillance results and developing more accurate prediction procedures. It also aims to provide the Ukrainian reference laboratory with the necessary equipment and experience on reconstitution and testing of irradiated specimens. The program also provides updated generic PTS assessments, by use of advanced stress analyses and fracture mechanics methods. To optimize the benefit of the whole program, the European Commission has chosen to have the detailed scope approved by a specific Senior Advisory Group during the first phase of the project implementation.

Brumovský M., Novosad P., Falcnik M., Kytka M., Vacek M., Málek J. "REEVALUATION OF IRRADIATION EMBRITTLEMENT OF SURVEILLANCE SPECIMENS", Published in: Nucleon 3. 1998

In this paper are summarized the results from the surveillance specimen programs of WWER-440/V-213Č type reactors. Comparison of transition temperature shifts, induced by irradiation, and determined from impact Charpy V -notch toughness as well as from static fracture toughness tests on pre cracked Charpy size specimens (COD) is given. Two methods have been used for determination of static fracture toughness shifts - standard COD type specimens from a standard survei1lance programs and reconstituted COD type specimens from broken halves of Charpy V-notch specimens, thus identical irradiation conditions have been assured. The results are discussed and show that transition temperature shifts from static tests are, generally, larger than those from dynamic tests.

From the limited set of data are presented conclusions:

- The reconstitution technique was proved as a very effective and valuable method for reevaluation of specimens from the standard surveillance programs.
- The use of this method is very important mainly for determination of difference material parameters from the same specimens.
- Identical conditions for the different tests can be assured.
- Differences between static fracture toughness transition temperatures can be described as a linear dependence on neutron fluence or on transition temperature shift values.
- The higher is transition temperature shift, the higher is a difference between static and dynamic transition temperature shifts.
- These differences can be neglected up to the WWER-440 end-of-life design neutron fluence (approx. 2 x 1024 n.m2), because the differences can reach values up to about 50 oC for neutron fluences equal to about 5x1024 n.m2, i.e. for double RPV design lifetime.
- Practically no difference between transition shifts from different dynamic type tests was observed.
- Shifts from Charpy impact and dynamic fracture toughness test are close to each other.
- Charpy impact transition temperature shifts can be probably used for WWER RPV integrity assessment up to EOL neutron fluence.

- Reconstitution of specimens from standard surveillance program provides very important results and shows some dependence between static and dynamic type transition temperature shifts.
- Great difference still exists between static transition shifts from reconstituted specimens and from direct tests of pre-cracked Charpy specimens within the surveillance program. One possible explanation could be an existence of flux rate effect, which could increase these shifts comparing with shifts from specimens irradiated by much higher neutron flux.
- This question has been one of the reasons for a Supplementary Surveillance Program for RPV's in NPP Dukovany has been designed and finally implemented.

Brumovský M., "SURVEILLANCE SPECIMEN PROGRAMS FOR WWER TYPE REACTORS", Proceedings of ASME Pressure Vessels and Piping Division Conference, July 17-21, 2005, Denver, Colorado USA

Surveillance specimen programs are the best method for monitoring changes in mechanical properties of reactor pressure vessel materials if they are designed and operated in such a way that they are located in conditions close to those of the vessels. Reactor Codes and standards usually included requirements and conditions for such programs to assure proper vessel monitoring. WWER reactor pressure vessels are designed according to former Russian Codes and rules with somewhat different requirements using different materials comparing e.g. with ASME Code.

Two principal types of WWER reactors were designed manufactured and are operated in several European countries and also in China, Iran: WWER-440 and WWER-1000. Their surveillance programs were designed in quite different way, with some modifications due to the time, country of manufacturing and experience gained from their operation. The paper gives a critical comparison of these programs in both types of reactors with requirements of both Russian and ASME/ASTM Codes and Standards.

In this paper the information about Surveillance Programmes for reactor pressure vessels of both WWER-440 and WWER 1000 types located in the Czech Republic, also principles and content of the Integrated Surveillance Programme, which uses NPP Temelín reactors as a "host" reactors for several RPV's of operating reactors in Ukraine, Russia and Bulgaria are described.

Disadvantages of Standard Surveillance Programmes lead to their modification – Supplementary Surveillance Programme for NPP Dukovany with WWER-440 type and Modified Surveillance Programme for NPP Temelin with WWER-1000 type reactors.

Mato Cvitanovič V., Gordan Krpanec "KOZLODUY NPP UNIT 1 REACTOR PRESSURE VESSEL BOAT SAMPLlNG", Source unknown, probably proceedings of Conference in Prague, Czech Republic, 8-10 December, 1997

This paper is an abstract only. In the abstract the facts about the boat sampling of the Kozloduy Unit-1 reactor NPP in Bulgaria are described. The Bulgarian Kozloduy NPP, Unit 1, Reactor Pressure Vessel had no reactor vessel material surveillance program. Changes in the material fracture toughness resulting 'from fast neutron irradiation cannot 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 were determined.

Before to boat sampling from RPV wall, ultrasonic base metal wall thickness measurement was taken, and also weld centerline and the form of weld crown were found by replica method on all sample sites. On the basis of replica measurement and analysis as well as known EDM tool geometry mini-Charpy specimens cut from templates were modeled. Special magnets were placed on the RPV wall and they secure precise positioning on sample locations.

Electrical discharge machining performs boat sampling and after this on the depth of the divot 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. After all other activities divot depth measurement by replica and ultrasonic technique enable precise determination of all geometrical dimensions of the divot, which are needed for further structural analyses of the RPV.

Davies L.M., Gillemot F., Lyssakov V. “PTS AND THE IAEA DATABASE“, Source unknown, probably proceedings of IAEA Specialist Meeting in Vienna

This paper briefly describes Presurized Thermal Shock (PTS) and draw attention to the relevance of the International database in the area of irradiation effects.

Pressurized Thermal Shock (PTS), in a Pressurized Water Reactor Pressure Vessel, is a strain, which derives from an overcooling transient happening at the same time as/or followed by, re-pressurization. Rapid cooling of the vessels surface results in a tensile stress at the inside surface, the magnitude of this stress being dependant on the temperature gradient as a function of time. The often-quoted example is the incident, which triggers the safety injection system while coincidentally a significant pressure is still being maintained or reestablished in the system. Other transients can be initiated by instrument and control system malfunctions, which can include stuck open valves in primary or secondary systems and postulated accidents such as the 'small break loss of coolant accidents', 'main steam line breaks' and 'feed water line breaks'.

The risk to the vessel in terms of the impact of PTS on its integrity is assessed in two ways:

- Deterministic - combinations of defect sizes and transients and applying minimum safety factors.
- Probabilistic **-** distributions of flaw density, flaw size RT_{NDT} etc. and calculating the conditional probability of failure from a given transient.

If the probability and location of the occurrence of a critical defect is low and if the fracture toughness of the pressure vessel is sufficiently high then its integrity is not challenged by the coincidence of the transients.

Approach to PTS and irradiation effects internationally: Except for those countries which follow US practice, like Belgium and Spain, the approach is different in the methodology. It reflects the national Regulatory and engineering approaches ranging from those countries which have no prescriptive regulatory requirements to those where a more sophisticated approach is adopted. In the consideration of irradiation effects the Regulatory Guide 1.99 Rev 2 is used to predict the degradation of mechanical properties during neutron irradiation. This version relies on the copper and nickel concentration. For steels made and used in other countries different deleterious elements are identified in the trend equations. For example, in France and Japan phosphorus is included in the trend equation. In Russia, for example, phosphorus is a dominant embrittling agent. Prediction of irradiation therefore becomes unique to a particular national database and the derived trend curve. Differences with other trend curve databases may reflect:

- data scatter,
- differences in irradiation temperature,
- differences in composition and irradiation behavior,
- inadequacy of the national data base.

The IAEA database: The International database on Reactor Pressure Vessel Materials development is to do with irradiation effects and that database is being set-up at this time. The creation of this database as well as the analysis of the data collected will help substantially:

- generating utilities in a more sophisticated, supportive,
- analyses of surveillance data or in the provision of generic data-for RPV integrity, and operational lifetime assessment for plant operational life assurance,
- design authorities in the provision of data of materials behavior during operation and for early consideration design or mitigating features to extend the designed operational life time,
- safety authorities, in some countries, to prepare information in order to assess safety margins as well as an aid in licensing processes and safety report analyses,
- 'researchers' in assisting the understanding of damage mechanisms producing degradation in mechanical properties.

The IAEA IRPVM-DB database will support its participating organizations it can add to the treatment of national data with regard to operational plant life assurance and understanding of degradation mechanisms in PV steels.

Debarberis L., Acosta B. , Zeman A., Sevini F., Ballesteros A., Kryukov A., Gillemot F., Brumovsky M., "Analysis of WWER-440 and PWR RPV welds surveillance data to compare irradiation damage evolution", Journal of Nuclear Materials xxx (2006) xxx–xxx

Available surveillance results from WWER and PWR vessels are used in this article to compare irradiation damage evolution for the different reactor pressure vessel welds.

The analysis is done through the semi-mechanistic model for radiation embrittlement developed by JRC-IE. Consistency analysis with BWR vessel materials and model alloys has also been performed within this study. The two families of studied materials follow similar trends regarding the evolution of irradiation damage. Moreover in the high fluence range typical of operation of WWER the radiation stability of these vessels is greater than the foreseen one for PWR.

Both WWER and PWR/BWR pressure vessels are made of welded ferritic steel sections. The Russian designed WWER-440 employed Cr–Mo–V steels (i.e., 15Kh2MFa, 15Kh2NMFAA) while western PWR/BWR selected Mn–Mo–Ni type of steels (as 16MND5, A302 B, A508 cl. 2 and cl.3, A533 gr.B).

The WWER-440 RPV's are made only from welded forgings, i.e., from cylindrical rings and from plates forged into domes, so there are no axial welds. The cylindrical portion of a PWR RPV may have longitudinal (axial) welds in addition to circumferential welds if the vessel is made from plates. Both RPV types, and their respective welds in particular, are sensitive to radiation, which induces embrittlement of the material and is one of the life limiting factors of the component. The most important effect of the radiation damage is the increase in the ductile-to-brittle transition temperature (DBTT). RPV embrittlements is monitored by means of surveillance programmes, which have surveillance capsules containing specimens representative of the vessel beltline materials, and are placed at particular locations inside the vessel where they experience accelerated exposure. These capsules are withdrawn at regular intervals and the testing of their specimens is used for prediction of the vessel material properties degradation. WWER and PWR approaches differ somewhat. For example WWER's sampling location ranges from the ¼ thickness to the deep 3/4 thickness, whereas for PWR`s base and heat affected zone samples are taken near the 1/4 thickness depth plane and weld sampling is done through the thickness with the exception of location within 12.7 mm from the root (or surface) of the weld. Another difference can be found in the chosen orientation of the specimens that for weld metal it is taken transverse to the weld (axial to vessel) for WWER`s and following ASTM E-185 (both axial and transverse to vessel) for PWR`s.

It is rather complex to properly compare the stability of WWER-440 and PWR RPV welds to radiation embrittlement. The first difficulty arises from the fact that WWER-440 reactors operate at a temperature of 260–265 ^{o}C while PWR operate in general at 290–295 ^{o}C. In this study the comparison is made using available data, mostly assembled in the frame of the AMES (Ageing Materials European Strategy) European Network. Using the semi-mechanistic model under development at the JRC Institute for Energy performs the determination of the different evolution of the irradiation damage.

In spite of the encountered difficulties to compare both PWR and WWER weld types some important conclusions can be drawn:

- Matrix damage rates are basically equal for both weld types, the observed differences can be attributed to the different operation temperatures of WWER and PWR.
- For modeling purposes unified temperature dependent matrix damage parameter can be used for the two families of materials.
- Precipitation and segregation coefficients are quite similar for the different welds, the apparent higher observed damage for the WWER-440 welds is, for the same phosphorus level, mainly due to the higher radiation level at which the materials are exposed and the slightly lower temperature at which they are operated.
- Fluences at which segregation and precipitation effects start to saturate are higher for WWER- 440 than for PWR welds due to the lower operational temperature, the Cr–Mo–V structure of the material and the different content of other alloying elements.
- Taking into account the different fluence indexation for WWER-440 the two families of materials follow similar trends for what regards the evolution of the irradiation damage.
- The temperature effect the radiation stability of WWER-440 is in the high fluence range greater than the one foreseen for PWR.
- The influence of other radiation field parameters needs to be analyzed in more details.

Gérard R. (Tractebel Energy Engineering),Karzov G. (CRISM Prometey),Margolin B. (CRISM Prometey),Shtrombakh J. (RRC Kurchatov Institute),Zaritsky S. (RRC Kurchatov Institute),Vasiliev V. (Rosenergoatom), Langer R. (Framatome ANP), Van

Walle E. (SCK.CEN), "MAIN ACHIEVEMENTS OF THE TACIS R2.06/96 PROJECT: SURVEILLANCE PROGRAM FOR WWER 1000 REACTOR PRESSURE VESSELS", Source is unknown

The international activities in the frame of the TACIS R2.06-96 project according to the validity of the surveillance programs for WWER 1000 reactor pressure vessels are described in this paper.

The main objectives of the project were:

- Define the adequate and reliable evaluation methods for the determination of the real irradiation conditions of the surveillance specimens in WWER 1000 reactors.
- Apply advanced evaluation techniques (Specimen reconstitution, Master Curve, Local approach)
- Reassess the current situation of the surveillance programs.
- Define corrective actions and/or alternative surveillance programs.

In order to evaluate precisely the situation and to define remedial measures, a number of actions were undertaken:

- A special irradiation set instrumented with a large number of dosimeters and temperature monitors was irradiated during one cycle in Balakovo unit 1 in order to determine the real irradiation conditions of the surveillance specimens. It was shown that the irradiation temperature is below 300°C, which is close enough to the vessel wall temperature to be considered as representative.
- The use of the reconstitution technique of Charpy specimens was validated having received a fluence varying by no more than 15%, in order to determine reliably the transition temperature.
- Fracture toughness measurements on pre-cracked Charpy specimens were interpreted using the Master Curve and the local approach models for brittle fracture and compared to large specimens results.
- A set of specific recommendations was established for the improvement of WWER 1000 surveillance programs.

From the TACIS project realization was achieved:

- the improved evaluation of the fluence received by each individual specimen, determined by activity measurements and precise calculation based on the local power distribution and its history in the core periphery.
- application of Charpy reconstitution techniques to provide uniform sets of 12 specimens having received a uniform fluence (in a range +/- 15%).
- In support to the regulatory evaluation, use of fracture toughness measurement on pre-cracked Charpy specimens, interpreted by the Master Curve and the local approach models for brittle fracture, can provide an increase confidence in the conservatism of the regulatory results.

These measures can be implemented at a reasonable cost, and contribute significantly to the improvement of the follow-up of the WWER 1000 irradiation embrittlement, and thereby to the safety of operation of these units.

Ferenc Gillemot, "EXTENSION OF THE SURVEILLANCE PROGRAM AT NPP PAKS", Source is unknown

At Paks NPP a new surveillance program extension is started to eliminate of this disadvantage of the original program. The reasons for extension of original surveillance program in Hungarian NPP Paks are described in this paper. The new specimen sets will consist of three forging materials.

The materials are:

- a special heat of the 15Kh2MFA material,
- the IAEA reference material JRQ, and
- the original archive material of every unit.

Every specimen set consists of 15 Charpy and 6 tensile specimens of each of the above-mentioned materials. For forty years operation 150 uniform Charpy and 60 uniform tensile specimens are required plus specimens for zero level testing. Reconstituted type specimens are widely used, due to the limited availability of archive materials. In case of the reference materials is the requirement to use specimens cut from the same specific layer of the material also necessitates the use of reconstituting techniques. The dosimetry foils are in special foil holders. The shielding of the holders is equal to the self-shielding of a Charpy specimen, so the foils are showing real results. In WWER-440-s the surveillance specimens are located in accelerated irradiation positions. The lead factor is 11.2 for base material and even higher, 18 for the most critical circumferential weld. Because of this high fluence the surveillance specimens were withdrawn after four years of service represent the lifetime irradiation, or even more. This accelerated surveillance system has the advantage that the utility is informed in time, if the radiation embrittlement rate is higher than the designed value. At the same time it has the disadvantage that the operational changes (like the use of low leakage core, or change of fuel type, etc.) are not monitored. To eliminate this disadvantage new sets of specimens are to be loaded in every unit. The surveillance testing of WWER-s is different from that prescribed in ASME IX Code. The remaining lifetime is calculated after every four-year testing period.

Gillemot F., Fekete T., Tatár L., Horváth M. "R&D Background for Life Management in Hungary", Source is unknown labelled as IAEA-CN-92/P6, paper is available in abstract form only.

The purpose of this paper is to show some of the realistic solutions for WVWER-440 V-213 RPV PLIM and to propose the direction of further studies. Although this paper specifically considers the life management of WWER-440 V-213 type vessels, other type RPV-s need similar R&D activity.

In the first part of the paper are summarized the already existing research results obtained in the frame of PLIM related research in Hungary.

The second part of the paper introduces the ongoing research providing a strong background for the PLIM of NPP Paks.

Several actions are available to extend the operational time of the WWER-440 V213 RPV's.

The lifetime of structural materials influences the safe service time of equipment to a great extent. Longer lifetime is generally more advantageous both from the economic and environmental aspect, as waste volume decreases if decommissioning occurs less frequently. Enhancement of lifetime calculation is an important economic aspect in case of valuable equipment. WWVER-440 V-213 units provide about 40% of the electricity of

Hungary, and play a similarly important role in several countries in the area. In lifetime management of nuclear power plants reactor pressure vessels play a crucial role in safety.

Their safe lifetime can be prolonged by mitigation actions, thus plant lifetime is not limited technically by the RPV. Generally the life limiting consideration of the WWER-440 vessels are the PTS calculations results. To extend the safe operational lifetime of the WVWER-440 V2 13 vessels among other options using hardware solutions the following actions or at least one of them are suggested:

- Consideration of the clad in elastic-plastic PTS analysis,
- Application of less over-conservative trend curves, especially the Master Curve,
- Extension of the analysis with crack arrest considerations,
- Increasing the understanding of vessel annealing, and elaborate the optimal annealing strategy,
- Development non-destructive measurement methods of material degradation.

These actions require further study of the material ageing mechanism.

Gillemot F., "Recent development in Life Management of the Pressurised Components", Proceedings from TCM of the International Working Group on Life Management of Nuclear Power Plants Vienna 6-8 October 1997.

The main activities in the field of Life Management and Safety implemented in Paks NPP are the topics of this paper.

The present developments in the field of Life Management and Safety can be divided into four groups:

- Governmental actions
- Regulatory actions
- Utility actions
- R&D

Governmental actions are the following:

- The new law on peaceful use of nuclear energy was accepted by the parliament (it requires to present a new periodic safety report for every nuclear units in each 10 years period of operation. On evaluation of these reports the authority can extend the operational licenses for the next 10 years period)
- The Nuclear Regulatory Body and the National Committee of Atomic Energy had been reorganized

Nuclear Regulatory Committee actions:

- New national Code of Safety had been issued (is based on the ASME and on the present issue of the Russian Code PNAE. Finnish experience is used during elaborating of it.)
- The first periodic safety reports were evaluated and the licenses of the Paks unit 1 and 2 were issued for the next 10 years.
- The Regulatory Committee initiated and sponsoring the elaboration of a series of studies and guides dealing with ageing evaluation and management.

Utility actions:

- NPP Paks elaborated and presented the first periodic safety reports to the authority
- Life management program was started
- Several safety enhance actions started (Fire insulation's of cables, updating the computer systems, inserting safety valves for automatic protection against cold over-pressurizing etc.)
- Enhanced condensers were built into secondary circuits to increase the efficiency
- Revision of the operation instructions according to the enhanced safety requirements of the new national Code
- Enhancements of the ISI equipment's
- Surveillance program extensions
- Built up a new training centre.

Research actions:

- study of thermal ageing of 15H2MFA steel and of the cladding properties,
- development of the PTS methodology,
- participation in the IAEA coordinated research programmes on use of Master curve,
- study of the properties of the welding of WWER-440 reactors,
- Round-Robin on the use of the new IAEA PTS guide,
- elaboration of FM method for crack arrest measurement on small size specimens,
- elaboration of national database on RPV materials and management of the IAEA RPV database.

Gillemot F., "PLANT LIFE MANAGEMENT IN HUNGARY", source is unknown

The general idea of the technical life management used in Hungary on NPP Paks is a topic of this paper. The plant age is an important factor in the elaboration of the life management strategy. During the previous ten years the safety enhancement of the plant was the main issue. NPP Paks and the scientific institutes started to elaborate a common life management programme about 3 years ago. Several important factors affect the life management in Hungary:

- The new Hungarian Safety Code - based on the ASME code - has been elaborated. Earlier the translation of the Russian Code PNAE and the Interatomenergo Normative documents were used.
- These changes cause some practical problems, as not all of the ASME regulations can be applied to a WWER reactor, which was designed on the base of the PNAE code. E.g. the material characteristics of a WWER plant (Kic curve etc.) can be found only in the PNAE code. Since the PNAE code uses a different transition temperature than the ASME code the use of the original Russian Kic curve is difficult.

In 1992 the so-called periodic safety review system has been introduced on the base of recommendation of the IAEA. First the review period was 12 years, but 4 years later the Nuclear Safety Act of Hungary shortened it to 10 years. The introduction of this system means not only safety enhancement but also change of the previously established fixed (designer given) lifetime. Using the periodic review system, NPP has to prove that the level of safety satisfies the requirements for the following ten years, and the Regulatory Committee extends the license for this period if the results of the review allow it. This provides the possibility to extend the lifetime more than the designed life, but in the case of non-satisfying results the plant can be shut down before reaching the planned life.

The codes, rules and the responsibility of the operator clarify the acceptable safety level. Generally the development of the codes, rules, and safety enhancement practice requires increasing the safety margin. The calculation of the safety margin - due to the uncertainties - has a big scatter. The lower bound line of this scatter band has to be over the required safety margin. Mitigation actions (maintenance etc.) can increase the safety margin.

The technical lifetime of the following components are considered to be limiting the lifetime of the plant:

- RPV-s
- Steam generators
- Box system
- Spent fuel storage

It is presumed that the basic equipment determining the plant lifetime is the pressure vessel. The WWER-440 RPV is a compact design. The reactor pressure vessels are made from a 15Kh2MFA type Cr-Mo-V alloyed steel and have a 9 mm thick stainless steel cladding inside produced by submerged strip welding. The wall thickness is 140 mm. The small diameter has a unique consequence: the water gap between the core and the reactor wall is small.

At calculation of the technical lifetime of the RPV-s at NPP Paks the following ageing mechanisms are considered:

- Radiation embrittlement
- Low cycle fatigue
- Head cracking
- The failure of the threads and thread housing
- The state of the concrete supports
- The ageing of the inside structures.

The inside structures can be changed (repaired) at acceptable costs, and the nondestructive testing results prove that the thread holes and the head penetrations have not been damaged until now. Since the failures of these elements are not typical at PWR-s it can be expected that these elements will not cause the limit of the lifetime.

The state of the concrete supports is still satisfactory, but until now no reliable calculation of their lifetime has been performed. The designer gives the allowed number of fatigue cycles in a list. The NPP hasn't used too many of them, and if it will operate the units at a constant load - like until now - the cyclic lifetime can be as much as 60 years, or even more.

Due to the high neutron flux the radiation embrittlement is the life-limiting load of the WWER-s. Several steps have been taken in the framework of the life management program in order to reduce the embrittlement rate:

- All the surveillance results have been collected, written into a database and evaluated.
- Tangent hyperbolic curves were fit for all Charpy data by the same fitting procedure, and transient temperature shifts were evaluated for all the surveillance specimen sets of the four units.

- Trend curves of the embrittlement rate were also elaborated considering different operating modes. Material data obtained from these trend curves were used to predict the material degradation in the function of the lifetime.

During the previous years many transient cases (PTS pressurized thermal shock events) were analyzed and evaluated by Atomic Energy Research Institute. The stress and strain analysis of these PTS calculations was repeated using the new material degradation trend curves.

The results show that the lifetime of every unit of NPP Paks is longer than 24-operation year without any change in the operation mode.

With proper life management all four RPV-s can reach a lifetime of 48 years or even more, without annealing.

The suggested mitigation methods are as follows:

- reduction of the scatter of the material testing data base,
- reduction of the scatter of the analysis,
- reduction of the severity of the transients by revising the operation manuals,
- application of the low leakage core,
- heating up of the emergency core cooling water.

The application of any of these methods or their combination should increase the lifetime. The final lifetime of every unit has to be evaluated on the base of the electricity cost minimization, and not on the basis of the final technical limit of the lifetime. To reach a lifetime of over 60 years of operation annealing can be included in the list of mitigation measures.

Håkon K. JENSSEN/IFE Norway, BLANC J.Y./CEA France, DUBUISSON P./CEA France, MANZEL R./Framatome-ANP Germany, EGOROV A.A./Minatom Russia, GOLOVANOV V./RIAR Russia, SOUSLOVv/RIAR D. Russia, "IAEA Post Irradiation Examination Facilities Database", Proceedings from "HOTLAB" Plenary Meeting 2004, September 6th - 8th, Halden, Norway

The situation in the field of possibility to perform the post irradiation examination (PIE) in the future is the topic of this paper. The number of hot cells in the world in which PIE can be performed has diminished during the last few decades. This creates problems for countries that have nuclear power plants and require PIE for surveillance, safety and fuel development. With this in mind, the IAEA initiated the issue of a catalogue within the framework of a coordinated research program (CRP), started in 1992 and completed in 1995, under the title of "Examination and Documentation Methodology for Water Reactor Fuel (ED-WARF-II)". Within this program, a group of technical consultants prepared a questionnaire to be completed by relevant laboratories. From these questionnaires a catalogue was assembled. The catalogue lists the laboratories and PIE possibilities worldwide in order to make it more convenient to arrange and perform contractual PIE within hot cells on water reactor fuels and core components, e.g. structural and absorber materials. This catalogue was published as working material in the Agency and was converted to a database and updated through questionnaires to the laboratories in the Member States of the Agency. The database consists of five main areas about PIE facilities:

- acceptance criteria for irradiated components;
- cell characteristics;

- PIE techniques;
- refabrication/instrumentation capabilities;
- and storage and conditioning capabilities.

The content of the database represents the status of the listed laboratories as of 2003, the IAEA Member States will be able to use this catalogue to select laboratories most relevant to their particular, needs'. The success of the new database depends mainly on the quality of data that IAEA have received from the hot laboratories, e.g. the technical description of the PIE methods and laboratory equipment. The data transfer between the IAEA and the laboratories was arranged by using Microsoft Access software. All relevant laboratories have received an Access template with user instructions included for filling out the PIE data and sending it back to the IAEA. The updating of the IAEA PIE facility database with this new input mode will be performed during 2005. A similar PIE facilities database is under construction at the LHMA-SCK-CEN hot laboratory in Belgium. However, this database is only implemented with the PIE facilities of European countries. But, the database will be open to everybody and it will be in use during 2005.The two databases should develop in a similar manner to ensure maintenance compatibility. Also, some cross references/hyper couplings should be implemented in both databases to make it easy to access information for customer users.

Bukanov V.N., Vasiljeva E.G., Gavriljuk V.I., Grinik E.U., Demechin V.L., Nedelin O.V., Department of nuclear energy problems the Ukraine Academy of Sciences,Vorobej V.V., Poralo G.I.,Titov A.S., Šadrincev S.V. - Khmelnitski NPP "Determination of fast neutron fluence in Surveillance specimen of unit 1 NPP Khmelnitski", Source is unknown

In this article (written in Russian) is described the Surveillance Specimen Program (SSP) used in WWER-1000 units in Khmelnitski NPP. Except brief information about the structure and SSP specimen placement, there is in more detail described the system of fast neutron dosimetry, which consists from threshold detectors Fe, Mn, Cu, Co type. The detectors activity is measured by Ge-detector; the neutron flux is determined by effective cross section calculations.

From the results of calculations follows:

- the fluence in individual SSP sets are greater than 3 what is not consistent with the goals of SSP.
- Except these problems, there is very difficult to recalculate the neutron spectra on the irradiated surveillance specimen because of its placement above the reactor core especially for the upper set of surveillance specimen.
- For the solution of this problem will be placed special set of detectors in the operated unit, which enable to set the precision of the fluence measurement wit energies above 0,5 MeV.

Kee-Ok Chang, Se-Hwan Chi, Kwon-Jae Choi, Byeong-Chul Kim, Sam-Lai Lee, Korea Atomic Energy Research Institute, P.O. Box 105, Yusong, Taejon 305-600, South Korea, „Changes in magnetic parameters of neutron irradiated RPV Linde80 high copper weld surveillance specimens", International Journal of Pressure Vessels and Piping 79 (2002) 753–757

This paper is dealing with irradiation-induced changes in the magnetic parameters and mechanical properties of a reactor pressure vessel (RPV) Linde80 high copper weld. The specimens employed in the present study were obtained from a RPV surveillance programme.

The specimens were tested in following conditions:

- in the unirradiated,
- and two different post-irradiation conditions:
    - Irradiation I: 1,23 x 1019 n/cm2,
    - Irradiation II: 3.94 x 1019 n/cm2,
    - Neutron energies E > 1.0 MeV,
    - Irradiation temperature: 288 0C.

By the examination procedure were measured following magnetic parameters:

- Saturation magnetization (Ms),
- remanence (Mr),
- coercivity (Hc),
- Barkhausen noise amplitude (BNA).

By mechanical property parameters evaluation were implemented following methods:

- Vickers microhardness,
- tensile tests,
- Charpy impact tests.

The principal results after irradiation showed:

- hysteresis loops appeared to turn clockwise, resulting in an increase in Hc,
- BNA appeared to decrease after irradiation,
- all mechanical property changes followed the same trend as previously observed for low copper weld.

Main conclusion from experiments is, that although limited, Hc and BNA were confirmed to be viable magnetic parameters that can be used in monitoring the mechanical parameter changes due to neutron irradiation.

Kryukov A.,Erac D. RRC-KI Moscow, Russia, Debarberis L., Sevini F., Acosta B. JRC Petten, Vodenicharov S.Institute of Metal Science Bulgaria, "EXTENDED ANALYSIS of WWER-1000 SURVEILLANCE DATA", Source unknown, probably Proceedings from Meeting in Belgrade, Yugoslavia, July 2003

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

Both of these papers are practically identical. The extended analysis of the results from 17 surveillance sets testing is the topic of these papers.

Up to now 20 surveillance specimen sets of 14 WWER-1000 Reactor Pressure Vessels (RPV) have been evaluated in Russia, Ukraine and Bulgaria by different testing organizations:

- Kurchatov Institute, Russia - (10 RPV's)

- Institute for Nuclear Research, Ukraine - (2 RPV's)
- Institute of Metal Science, Bulgaria - (2 RPV's).

The materials involved in analyses contain very low and homogeneous levels of P and Cu and a significant variation of Ni and other elements like Mn etc.

- The observed temperature transition shifts are showing consistent behavior:
- The steels with low Ni content are embrittled at a much lower rate than the one predicted by the Guide (chemistry factor AF = 20)
- The Guide looks to be conservative also for the higher Ni steels if the content of Mn is lower than 0.8 wt%.
- The steels embrittling at much higher rates are those with high Ni and high Mn contents at the same time.

The threshold for Ni is evaluated to be at Ni > 1.5 wt% and for Mn at Mn > 0.8wt%.

Mn together with Ni seems to play a key role in low Cu and P steels embrittlement.

Correlation analysis that considers Ni and Mn and fluence dependence to 1/3 power is showing predictive capabilities within 20°C scatter band in most cases. Other elements, like for example C and S could also explain the residual scatter in the data. In the papers are given the recommendations for the future activities. A more precise analysis in order to reduce the uncertainties can be done in cooperation with the WWER-1 000 surveillance-testing laboratories. Such a precise analysis for WWER-1000 surveillance data must be performed as soon as possible in order to allow for improved:

- RPV integrity assessment,
- Plant life prediction,
- Mitigation methods/plant life management.

The following steps need to be undertaken:

- Harmonization of the method for transition temperature determination for involved testing laboratories (common database, training, Round-robin and/or benchmarking exercise)
- More precise assessment of Ukrainian & Bulgarian testing results e.g. Individual fluence value for each specimen; verification of actual fluence rates, etc.
- Decrease of data scatter by using re-constitution of broken Charpy specimens (based on the experience acquired in TACIS 96 projects and AMES projects like RESQUE , etc.)
- Direct measurement of the irradiation temperature of surveillance specimens
- Possibly update of the Russian Guide.

Kupca L., Beno P., „Progress of irradiation embrittlement monitoring in the Slovak Republic NPP`s",Department of Structural Analysis VUJE Inc., Trnava, Slovak Republic, Proceedings of the1-st AMES Biennial Conference on: "Through life toughness prediction in reactor steels" Hevíz Hungary 6-th to 8-thFebruary 2006

Kupca L. "IRRADIATION EMBRITTLEMENT MONITORING PROGRAMS OF RPV`S IN THE SLOVAK REPUBLIC NPP`S", Department of Structural Analysis VUJE Inc., Trnava, Slovak Republic, Proceedings of ICONE14 International Conference on Nuclear Engineering, July 17-20 2007, Miami, Florida, USA

These papers are dealing with state of the art information about five surveillance programs which were (are) realized in Slovak NPP`s:

- „Standard Surveillance Specimen Program“(SSSP) was finished in Jaslovské Bohunice V-2 Nuclear Power Plant (NPP) Units 3 and 4,
- „Extended Surveillance Specimen Program“ (ESSP), was prepared for Jaslovské Bohunice NPP V-2 with aim to validate the SSSP results,
- For the Mochovce NPP Unit 1 and 2 was prepared completely new surveillance program “Modern Surveillance Specimen Program” (MSSP), based on the philosophy that the results of MSSP must be available during all NPP service life (planned till 2011),
- For the Bohunice V-1 NPP was finished (2005)“New Surveillance Specimen Program” (NSSP) coordinated by IAEA,
- New Advanced Surveillance Specimen Program (ASSP) for Bohunice V-2 NPP (units 3 and 4) and Mochovce NPP (units 1, 2) is under realization in the Bohunice V-2 plant now. ASSP is dealing with the irradiation embrittlement of heat affected zone (HAZ) and RPV`s austenitic cladding, which were not evaluated till this time in surveillance programs.

SSSP program was the part of WWER-440/213 type RPV’s delivery and it was used in NPP V-2 for irradiation embrittlement monitoring. SSSP started in 1979 and was finished in 1990.Conclusions from SSSP:

- After 10 years of irradiation, was observed the significant growth of strengths and decrease of ductility.
- TT shifts of WM calculated from the Charpy-V and fracture toughness tests results were in the range of prediction after 10years of irradiation.
- Higher AFWM values of unit 3 are possible to explain due to the lower precision of fluence measurements for this unit.

ESSP program started in 1995 and finished in 2007, was prepared with aim of:

- increasing of neutron fluence measurement accuracy,
- substantial improvement the irradiation temperature measurement,
- fixed orientation of samples to the centre of the reactor core,
- minimum differences of neutron dose for all the Charpy-V notch and COD specimens,
- the dose rate effect evaluation.

ESSP FINAL RESULTS

Till December 2007 were finished all planned stages of ESSP according to the schedule. Results of the testing of mechanical properties changes were positive. Subsequent analyses of all relevant data analysis gave good arguments for planned operational lifetime prolongation of the units 3 and 4. All ESSP results and relevant data were archived into the computer database.

MSSP started in the year 1996 as the new surveillance specimen program for the Mochovce RPV’s unit-1 and 2, based on the fundamental postulate – <u>to provide the irradiation embrittlement monitoring till the end of units operation.</u>

MSSP PRELIMINARY RESULTS

Till the December 2005 were finished following stages from MSSP:

- analysis of the fourth chain from the unit 1,
- analysis of the third chain from the unit 2,
- measurement of activation foils and special samples for:
  - Mössbauer spectroscopy and electron-positron analysis,
  - neutron dosimetry measurement,
  - transmission electron microscopy.
- melting monitors evaluation was finished and documented,
- the computer database preparation for processing and archiving all MSSP results and relevant data.

NSSP the "New Surveillance Specimen Program" prepared during the year 1999 for the Bohunice V-1 NPP was successfully finished in the year 2004. Main goal of this program was to evaluate the weld material properties degradation due to the irradiation and recovery efficiency by annealing too.

NSSP FINAL RESULTS

- All experiments of NSSP were finished by the schedule
- mechanical properties evaluation in as received conditions,
- realization of all cycles (I, IA, IAI, IAIA, IAIAI) for all chains,
- complete analyses of samples from chains N-1 tillN-5,The results showed that the V-1 RPV`s can be operated without any limitation as minimum for the next 20 years.

ASSP Advanced Surveillance Specimen Program for Bohunice V-2 NPP (units 3 and 4) and Mochovce NPP (units 1 and 2) was prepared as the part of research project dealing with the WWER-440 units ageing management and is under realization now in NPP V-2.

FINAL CONCLUSIONS from the SSP in Slovak Republic are following:

- Projects SSSP ESSP and NSSP were finished by planned schedules.
- All experimental and operational activities involved in the MSSP surveillance projects are in progress by planned schedules.
- The Project ASSP is under realization in NPP V-2 and under preparation for the Mochovce units 1 and 2.
- The results of all projects are very satisfactory and enable to prepare for the operators and Slovak Nuclear Regulatory Authority the good arguments for Slovak NPP's EOL prolongation.
- The broad international cooperation during the stage of Slovak irradiation monitoring programmes preparation organized by IAEA or WANO was good example how to apply the international „know-how" for complex projects.

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

This paper described in detail the substructure analysis of the surveillance specimen from the Russian steel used for production of RPV's WWER 440 by an atom probe application.

The analyses have 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 295^0C and after neutron irradiation at 270^0C for 10 years to a fluence of 1.0 x 10^{25} m^2 (E > 0.5 MeV).

The ductile-to-brittle transition temperatures (DBTT) of the base metal were -49, -70 and +141^0C, 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 18 h at 690^0C plus a simulated stress relief treatment of 43.5 h at 680^0C, after thermal aging for 5 years at 295^0C, and after neutron irradiation at 275^0C for 5 years to a fluence of 5.2 x 10^{24} m^2 (E > 0.5 MeV). The DBTT`s were 7, 11 and 123^0C, respectively, for these three conditions.

A high number density of ultra fine 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.

From the atom probe analyses follows:

- This investigation has determined that there were significant increases in the yield and ultimate tensile strengths and reductions in the elongation and reduction in area in both base and weld materials after neutron irradiation.
- In addition, the ductile-to-brittle transition temperatures increased significantly after neutron irradiation.
- The changes in mechanical properties correlate with the presence of spherical and cylindrical manganese, silicon, copper, phosphorus and carbon-enriched features in the matrix of the neutron irradiated base and weld materials.

Miller M.K., Russell K.F., Kocik J., Keilova E., "Embrittlement of low copper WWER 440 surveillance samples neutron-irradiated to high fluences" ,Journal of Nuclear Materials 282 (2000) 83-88

The topic of this publication is application of atom probe tomography on special low copper surveillance samples.

In this study, an atomic level microstructural characterization has been performed on low copper surveillance specimens (0.06 at. % Cu) that were exposed to high fluences in a WWER 440 reactor in order to evaluate the susceptibility of low copper reactor pressure vessel steels to embrittlement.

In order to extend the lifetime of the reactor, it is essential to be able to predict the changes that will occur in the microstructure of the steel and their influences on the mechanical properties.

Previous atom probe experiments on both Western and Russian type steels have revealed that the matrix copper content was reduced to ~ 0.05 at. % Cu after neutron irradiation to fluences in the low 10^{23} m^{-2} and that this reduction was accompanied by the formation of copper-enriched precipitates.

This matrix copper level was found to be reduced slightly by annealing the steels for 168 hours at 454°C. Therefore, it has been proposed that the embrittlement may be reduced or eliminated through the use of low copper steels.

Examination of the embrittlement database (EDB) of mechanical properties from low copper (< 0.06 %) neutron-irradiated pressure vessel steels reveals that the maximum shift in the ductile-to-brittle transition temperature is of the order of 30°C; however, the range of fluences examined is limited to less than 6 x 10^{23} m^{-2} .E > 1 MeV.

From the presented study follows:

- The results indicate that there is an additional mechanism of embrittlement during neutron irradiation that manifests itself at high fluences.
- Significant increases were found in the DBTT, yield and ultimate tensile strengths and reductions were found in the elongation and reduction in area in both base and weld materials after neutron irradiation.
- The analyses revealed manganese and silicon segregation to dislocations and other ultra fine features in neutron-irradiated base and weld materials (fluences 1 x 1025 m-2 and 5 x 1024 m-2, E > 0,5 MeV, respectively).
- These changes in mechanical properties correlate with the presence of manganese, silicon, copper, phosphorus and carbon-decorated dislocations and other features in the matrix of the neutron-irradiated base and weld materials.

Nikolaev Yu. A..,"Radiation Embrittlement of Cr–Ni–Mo and Cr–Mo RPV Steels", Published in: Journal of ASTM International, Vol. 4, No. 8 Paper ID JAI100695

Nikolaev Yu. A.,"RADIATION EMBRITTLEMENT OF WWER-1000 RPV STEELS", Published in: 18th International Conference on Structural Mechanics in Reactor Technology (SMiRT 18)Beijing, China, August 7-12, 2005 SMiRT18-D08-2

Both of these publications are similar. Radiation embrittlement of WWER-440 and WWER-1000 RPV steels exposed to irradiation for 17–22 years are analyzed in these publications. Extremely high levels of thermal aging of WWER-1000 RPV steels at 320°C are pointed out. The neutron flux effect problem and the problem of accelerated radiation embrittlement of welds with high nickel contents are emphasized. The effect of P, Cu, Si, Mn, and Ni contents on radiation embrittlement of WWER-440 and WWER-1000 RPV steels is studied using the Russian surveillance database and representative results of research programs. The trend curves for WWER-440 and WWER-1000 RPV steels are proposed. The mechanisms of radiation embrittlement of Cr–Ni–Mo and Cr–Mo RPV steels are supposed. Irradiation can affect the safety of an RPV operation and cause changes of working parameters during operation and under transient conditions. Therefore, surveillance programs should be used for monitoring radiation embrittlement of RPV steels.

The objective of surveillance programs consists of periodic estimation of RPV material properties after neutron irradiation and thermal aging using test specimens of different types.

The specimens are located inside an RPV both near the core region and in the zone with vanishing neutron flux (to study the thermal aging). Embrittlement evaluated using specimens exposed to irradiation or thermal aging is compared to the one calculated using standard reference dependences specified for the prediction of mechanical properties variation during operation in the Guide. Only conservative dependences are specified for prediction of radiation embrittlement and thermal aging in the Russian Guide.

From the presented results follows:

- All the standard reference dependences for evaluation of radiation embrittlement for WWER-440 and WWER-1000 RPV base metal and weld seams are not conservative and should be revised.
- Under development of the functional form of the new model of WWER-440 RPV weld seam radiation embrittlement, in which following results were taken into account:
    - the effect of the threshold levels of phosphorous and copper contents on radiation embrittlement was found to be significant;
    - the copper effect on radiation embrittlement was found to have relatively rapid saturation (before 5x1023 m−2);
    - 80 % of the analyzed database had the fast neutron fluence (above 5x1023 m−2);
    - copper in copper-rich precipitates was always observed combined with phosphorus;
    - aggregates of phosphorus atoms were observed practically at all micro structural features of steel and even form their own atmospheres;
    - the fluence exponent was found to be different from 1/3 and different fluence exponents can not be equal.
    - several models for characterization of WWER-440 RPV weld seam radiation embrittlement are proposed in this publication.
    - On average, the flux effect on radiation embrittlement for the WWER-440 surveillance welds irradiated with fast neutron fluence above 5x1023 m−2 and for all studied WWER-1000 surveillance materials was found to be vanishing.
    - The extremely high level of thermal aging at 320°C was observed for the WWER-1000 RPV steels.
    - Practically in all cases the transition temperature of WWER-1000 RPV steels was observed to be increasing with increasing damage doze, and for materials with high nickel contents the character of radiation embrittlement dependence on neutron fluence was found to have close to linear, not fading character.
    - The results of the WWER-1000 surveillance programs were shown to be the most representative for radiation embrittlement evaluation of WWER-1000 RPV materials.
    - The best choice for the exponential coefficient in dependence of transition temperature shift on the fast neutron fluence for the WWER-1000 surveillance steels was found to be of 1/3.
    - Relatively low variation of nickel contents in WWER-1000 surveillance base metals did not reveal any significant effect of the chemical composition on radiation embrittlement.
    - The model ΔTTF=12.7F1/3 was found to be the best for characterization of WWER-1000 RPV base metal radiation embrittlement.
    - The model ΔTTF =12.7F1/3+35°C can be used for determination of the end-of-life fluence for WWER- 1000 RPV base metal under plant life time extension.

- Increase in nickel and manganese contents in WWER-1000 surveillance welds was found to considerably increase radiation embrittlement. The nickel and manganese effects on radiation embrittlement were observed to be synergetic rather than independent.
- Silicon was found to decrease radiation embrittlement of WWER-1000 surveillance welds.
- Since radiation embrittlement of WWER-1000 weld seams is much higher than for base metal, it can be supposed that the lifetime of WWER-1000 RPV is limited by radiation embrittlement of the former.
- Results of research programs were found to be unrepresentative for assessment of WWER-1000 RPV steel radiation embrittlement since 20 times the increase in neutron flux on the average resulted in a 20–30°C decrease in the transition temperature shift.

BRUMOVSKY M., ERBEN O., NOVOSAD P., ZEROLA L., HOGEL J., MASSOUD J.P., TROLLAT C., "NEUTRON DOSIMETRY IN EDF EXPERIMENTAL SURVEILLANCE PROGRAMME FOR WER-440 NUCLEAR POWER PLANTS", Unknown source

Cooperation of NRI and EdF in the field of experimental surveillance program for WWER-440 NPP's is the topic of this publication. Fourteen chains containing experimental surveillance material specimens of the WWER 440/213 nuclear power reactor pressure vessels were irradiated in the surveillance channels of the Nuclear Power Plant Dukovany in the Czech Republic. The irradiation periods were one, two or three cycles. The chains contained different number and types of containers, the omitted ones were replaced by chain elements. All of the containers were instrumented with wire neutron fluence detectors; some of the containers in the chain had spectrometric sets of neutron fluence monitors. For the absolute fluence values evaluation it was taken into account time history of the reactor power and local changes of the neutron flux along the reactor core height, correction factors due to the orientation of monitors with respect to the reactor core centre. Unfolding programs SAND-II or BASA-CF were used. The relative axial fluence distribution was obtained from the O-wire measurements. Neutron fluence values above 0.5 MeV energy and above 1.0 MeV energy in the container axis on the axial positions of the sample centers and fluence values in the geometric centre of the samples was calculated making use the exponential attenuation model of the incident neutron beam. Received fast neutron fluence values can be used as reference values to all WER-440 type 213 nuclear power plant reactors.

From the experimental results of this international program follows:

- The neutron fluence values above 0.5 MeV energy (F>0,5) and above 1.0 MeV energy (F>1,0) in the container axis on the axial positions of the sample centers and fluence values in the geometric centre of the samples were calculated making use the exponential attenuation model of the incident neutron beam and by evaluation of the neutron spectra in the position of spectrometric sets.
- The relative axial distribution of the neutron fluence was determined by the polynomial regression of the calculated values.
- The relative neutron fluence distribution obtained from the exponential attenuation model inside the container was confirmed by the H-wire values.
- Received fast neutron flux values can be used as reference values for all WER-440 type 213 nuclear power plant reactors.

Novosad P., Falcník M., Kytka M., Brumovský M., "SUPPLEMENTARY SURVEILLANCE PROGRAM", Published in: Nucleon 3, 1998

The description of program for the re-evaluation of the standard surveillance program for NPP Dukovany in Czech Republic is the topic of this paper. Except these additional irradiations in NPP reactor and in research reactor in NRI and a very detai1ed analyses of all results obtained served for a design of a Supplementary Surveillance Program. Experience from the standard surveillance program compared with present day requirements for reactor pressure vessel residual lifetime assessment indicated the necessity for re-evaluation and also implementation of a new supplementary program. The realization of Supplementary Surveillance Program (SSP) started in 1997 by loading first chains with specimens. For this Program archive materials were used from the Standard surveillance program and samples evaluation is based completely on the reconstitution technique using the electron beam welding technology.

The structure of SSP consists from four parts:

1. Irradiation of archive materials with a lead factor lower than 3,
2. Irradiation of cladding materials,
3. Re-irradiation of irradiated and annealed specimens from the Standard program,
4. Neutron monitoring chains containing the IAEA reference materials as well as neutron monitors to be loaded after the other chains are withdrawn.

This program will provide not new results under a realistically low lead factor only, but data after annealing and re-irradiation and data on re-irradiation rate effects on vessel materials and also new data on cladding properties under irradiation. At the same time the fourth part of the Program will provide for a permanent neutron monitoring until the actual vessel end-of-life of the vessel. Within the whole program, the IAEA reference material JRQ will be also used.

Platonov P., Shtrombakh Ya., Kryukov A., Gurovich B., Korolev Yu., Shmidt J., "Results on research of templates from Kozloduy-1 reactor pressure vessel"; Reactor Technologies and Materials Institute, Russian Research Centre, Kurchatov Square, 123182 Moscow, Russian Federation; SIEMENS AG KWU, W- 8520 Erlangen, Germany, Nuclear Engineering and Design 191 (1999) 313–325

This paper is dealing with following issues:

- The studies on the specimens manufactured from the templates cut out from the weld 4 of Kozloduy NPP Unit 1 reactor vessel.
- The data on chemical composition of the samples from weld metal No4.
- Hardness measurements and indentation diagram to evaluate the yield stress of the metal.
- The macro- and micro structural, fractographic and electron-microscopic research with aim to describe the embrittlement mechanism.
- Determination of neutron fluence, mechanical properties, ductile to brittle transition temperature (DBTT) using mini Charpy samples.
- DBTT determination after irradiation (Tk) with aim to evaluate the vessel metal state at the present moment,

- DBTT determination after heat treatment at the temperature of 475°C to simulate the vessel metal state after thermal annealing (Tan),
- and after heat treatment at 560°C to simulate the metal state in the initial state (Tk0) with the aim of Tk0 recovery in accordance with the methods accepted in Russia..

From the performed testing follows:

- A number of studies on the templates of weld 4 of Kozloduy-1 reactor vessel have been performed. These studies includes: determination of the chemical composition, fast neutron
- fluence, research into macro- and microstructure, electron microscopy, determination of mechanical properties and critical brittleness temperature.
- The data on chemical composition and fast neutron fluence well agree with the data obtained at Bulgarian research institutes.
- The structural particles, which may be responsible for radiation hardening and embrittlement of the vessel steel under study, have been detected with the electron microscope.
- The data on mechanical properties and critical brittleness temperature demonstrate that the state of the weld 4 metal is considerably better than may be expected on the basis of the conservative calculation method accepted as standard and even 'lateral shift' method.
- On the base of analysis was stated that re-annealing of the vessel is unnecessary.

Shtrombakh Ya I. and Nikolaev Yu A., "Monitoring of Radiation Embrittlement of the First and Second Generation of WWER RPV Steels", Journal of ASTM International, Vol. 4, No. 5Paper ID JAI100696 Available online at www.astm.org

This paper is very important from the precise analysis of the current condition on the prediction of radiation embrittlement for WWER RPV materials of the first and second-generation point of view. Following important question are discussed and analyzed:

- The radiation lifetime of pressure vessel.
- Re-irradiation embrittlement kinetics for the first generation of WWER RPV steels (WWER-440/179 and WWER-440/230 grades).
- The technique for the first generation RPV materials monitoring, based on sampling and studies of metal from the pressure vessel's inner surface.
- The current program of cutting out templates, for the inner surface of a RPV wall, and their re-irradiation in surveillance channels of WWER-440 units with full core and dummy assembles.
- The new approaches to re-irradiation embrittlement models.
- The basic problems of WWER-440/213 and WWER-1000 metal monitoring.
- The representativity of WWER-440/213 and WWER-1000 RPV surveillance programs.
- The necessity of updating of the current standard reference dependences for the prediction of radiation embrittlement of RPV materials.
- The most urgent questions of reactor dosimetry for WWER pressure vessel materials.

- The lifetime extension for the first and second generation of WWER units.

From this extensive analysis follows:

- The correct evaluation of radiating embrittlement rates for WWER-440/213 and WWER-1000 RPV materials requires improvement of the reactor dosimetry reliability.
- Both design and experimental parts of the dosimetry should be improved.
- Not only the core power history should be taken into account under neutron field calculations, but also local power distributions and the power history of fuel elements close to the surveillance specimens located on the periphery of the core.
- A method of fluence evaluation based on 93mNb activity measurements should be applied in parallel with the standard technique of 54Mn activity measurements.
- Development of a new, physically proven model for re-irradiation embrittlement of WWER-440/230 RPV materials will allow the validation of NPP lifetime extension for about 15–20 years beyond the design operation period for power units of the first generation on the condition that there will be metal condition monitoring using periodically sampling from the inner surface of reactor pressure vessels.
- Using the reconstitution technique can essentially increase representativeness of surveillance programs for WWER-440/213 and WWER-1000 reactor pressure vessels.
- Practically unique reliable data on radiation embrittlement of WWER-1000 RPV materials are the results obtained in the WWER-1000 surveillance programs.
- The standard reference dependences from the Russian Guide used at present for the prediction of radiation embrittlement of WWER-440/213 and WWER-1000 RPV materials should be updated because they are not conservative and do not represent real physical processes of radiation damage.
- The problem of the fast neutron flux effect on radiation damage rates should be solved at the development stage of new models for radiation embrittlement.

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Williams, T., Ellis, D., and O`Connell, W., “Dose rate effects in high and low nickel welds,” Proc. of the Conf. on Dose Rate Effects in RPV Materials, Olympic Valley, CA, 2001

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12 Cladding

Regarding the claddings the following can be mentioned for the WWERs:

- WWER-440/230 2-layer cladding, where seven vessels out of 16 are cladded
- WWER-440/213 2-layer cladding (8-10 mm in thickness), where all vessels are cladded
- WWER-1000 2-layer cladding, where all vessels are cladded
- the inner layer is mainly from 07Cr23Ni13 steel and the outside layer of 04Cr19Ni10Mn1Nb steel

12.1 Consolidated Conclusions

Very little literature is available on the properties and ageing of RPV cladding. The main conclusions from the papers:

- The WWER RPV cladding remains ductile after high dose irradiation
- No saturation effect on the cladding hardening until the EOL of the WWER-440 units
- The annealing of cladding results only partial recovery
- Re-irradiation embrittlement rate is similar to the first irradiation

12.2 Open Issues

Several open issues still exists in the field of RPV cladding:

- Collect further information and prepare new study on the presently available information
- Data base elaboration –a new chapter in the MAT DB
- Elaboration of missing data list, and initiate research to obtain them
- Application of the data for structural evaluation

12. 3 Reviewed papers and summaries

Karzov-001.pdf, Karzov G.P.,Mikhaleva E.I., Morazovskaia I.A.,"Creation of welding materials for anticorrosive cladding of new family WWER reactor pressure vessels", Radiation materials science and structural strength of reactor materials. Jubilee volume to 100- anniversary of Kurchatov and Aleksandrov's birthdays. p.26-36, St.Petersburg, 2002, CNII Prpometey.

New welding materials with higher purity have been developed for anticorrosive cladding of reactor pressure vessels for NPPs of a new generation. These materials assure higher resistance of the cladding metal against embrittlement as a result of technological tempering and irradiation.

Badanin-001.pdf Badanin V.I.,Ignatov V.A., Nikolaev V.A., Rybin V.V., Timofeev B.T. "The resistance of austenite-pearlite steel cladded on 15Kh2MFA steel against brittle fracture", in Russian, Avtomaticheskaia Svarka, (1989), No.3, 4-7

The present paper gives an overview of studies on resistance of stainless austenite steel cladded on ferrite-perlite steel. The plate of 140 mm thickness from 15H2MFA was cladded by two layers of Sv07Cr25Ni13 (inner layer) and Sv-08Cr19Ni10G2B (outer layer). After welding it was heat treated at 665±10^{o}C, 30 hours hold, air cooling. There were carried out tensile tests, impact tests and static three-point bend tests have been performed to study the results. The heat treatment reduced the ductility.

Golovanov005.pdf, Tsykanov V.A., Golovanov V.N., Krasnoselov V.A.,.Kolesova T.N., Kozlov D.V., Prokhorov V.I., Karzov G.P., Filimonov G.N.,"Estimation of radiation embrittlement of steel 15X2MOAA containing 0,75 % of nickel and corrosion-resistant cladding after an exposure in facility "KORPUS"", International Atomic Energy Agency, Vienna (Austria)., International Working Group on „Life Management of Nuclear Power Plants Irradiation Effects and Mitigation", Workshop Proceedings of the IAEA Specialists Meeting, Working Material 1997 398 p.p. 250-262

RPV cladding properties have been investigated after it had been exposed by high level neutron fluence. The results obtained on radiation embrittlement rate of this material are given in the paper. The irradiation was carried out in the KORPUS" facility in the PBT-6 reactor.

RPV cladding 02X18H10B-BH after irradiation by fluence ~10^{20}n/cm$^{'2}$ (E>0,5 MeV) does not have a transition tempereture in the range of test temperature from -100 °C to +200 °C, and the value of impact ductility in this temperature range is always higher than 80 J. The data are slightly hesitating, since the specimens cut from a stainless steel welding and not from real cladding.

Novosad 002.pdf, Brumovsky M., Novosad P., Vacek M., "CLADDING PROPERTIES CHANGES DURING OPERATION", IAEA specialists' meeting on irradiation effects and mitigation, Vladimir, Russia 15-19 September, 1997

Austenitic cladding was originally designed as a protection of ferritic/bainitic base materials of reactor pressure vessels against corrosion. Nevertheless, its existence must be taken into account into reactor pressure vessel integrity evaluation from several reasons:

- cladding has different thermal properties with respect to base metal which affect temperature fields in a vessel,
- cladding has different mechanical and thermal-mechanical properties
- comparing with base metal which affect stress field in a vessel, austenitic cladding has different fracture mechanics properties than base metal, but they are also changing during operation due to radiation damage.
- instrumented hardness method is a very useful method for direct determination of changes in outer cladding tensile properties in real RPVs, as no surveillance programme contains specimens from cladding,
- The cladding toughness properties of the surface layer are relatively low at initial state, and irradiation reduces it further. No information from the first layer.

The surveillance extension programme of Dukovany includes new cladding specimens.

Lauerova-001.Pdf, Lauerova D., Brumovsky M., Simpanen P., Kohopaa J., "Problems of underclad Type Defects in resctor pressure vessel integrity evaluations, Transaction of

17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17) Prague, Czech Republic, August 17-22, 2003

NRI Rez tested 70*100 blocks cut from WWER-440/V-213 type RPV. The blocks included artificial underclad defects in the base and weld material below the cladding. In NRI Rez 16 experiments were performed. Materials were tested in as-received as well as in aged condition simulating of RPV end-of-life state. Thermal ageing used to simulate the irradiation embrittlement. Crack propagation in the base material and crack arrest in the cladding studied. In general the cracks initiated in the base or weld material and arrested in the first layer of the cladding

Gillemot-022.Pdf,Gillemot F.,Horvath M., Uri G.,Viehrig H-W.,Debarberis L.,Fekete T.,Houndeffo E.," Behaviour of irradiated RPV Cladding",AMES workshop 6--8 Febrary 2006, Hévíz

The paper presents the results obtained on trepans cut from Greifswald unit 8, which is a WWER-440-213 unit, and never operated.

First, the defects in cladding have been studied using UT testing. Only very small indications were found, the diameter of the largest was beyond 2 mm-s. To verify the NDT results some of these defects have been opened by surface milling in 0.1 mm steps. Most of the defects were small gas bubbles, or slag inclusions between two passes of the welding, or between the base material and welded cladding.

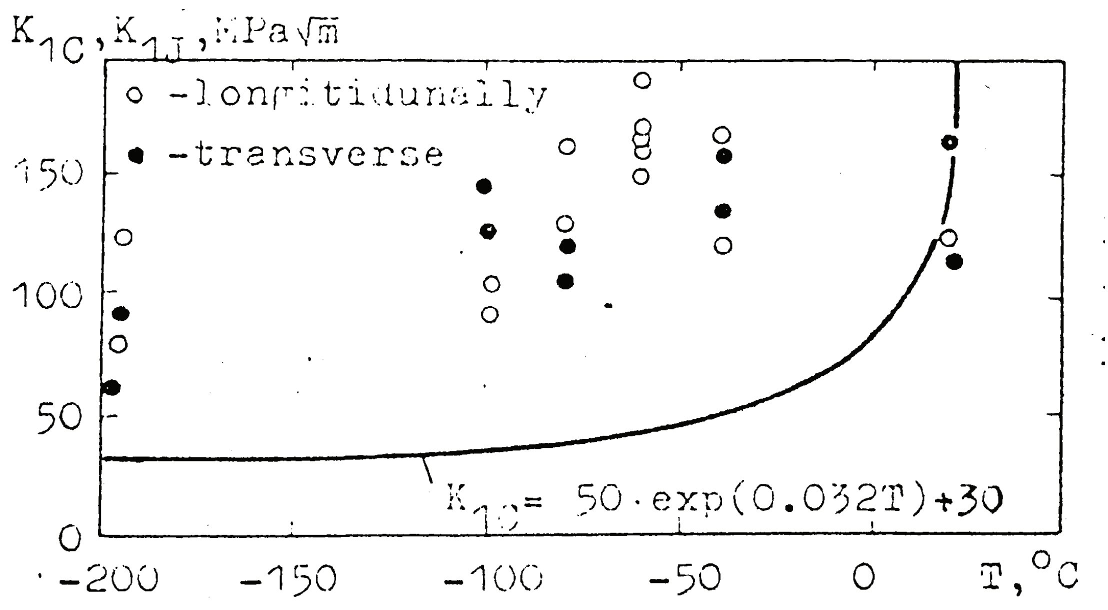
Mini Charpy and tensile specimens used to evaluate the strengths properties. The strengths properties determined in as received and in irradiated conditions.

Badanin-002.pdf, Badanin V.I.,Ignatov V.A., Nikolaev V.A., Rybin V.V., Timofeev B.T., "Brittle fracture resistance of anti-corrosive cladding on presure vessel", 10th Int. Conf. On Structural Mechanis in Reactor Technology, Annaheim, August 14-18, 1989 (1989) 221-225

The paper shows the properties of the austenitic cladding of RPV. The following main conclusions are obtained:

- Brittle fracture may occur in cladding in the areas where the δ ferrite is high.
- The fracture toughness of cladding below 20 °C is always higher than the fracture toughness of the ferrit-perlitic base steel.
- The as-produced cladding has a transition temperature -40°C using 40J Charpy energy criteria.
- The irradiation shifts up the trabsition temperature of the cladding, but the shift is considerable less than in the case of low alloyed steel ($1*10^{24}$ n/cm^2 E>0.5 Mev irradiation increases the shift by 50°C only).

The paper provides the K_{1c} curve of the cladding.



NikolaevV-002.pdf (in Russian), Nikolaev V.A., Kursevich I.P., Nesterova E.V., Prokoshev O.Yu., Rybin V.V.," Tendency to brittleness of RPV cladding metal for water-cooled reactors", Radiation materials science and structural strength of reactor materials. Jubilee volume to 100th anniversary of Kurchatov and Aleksandrov's birthdays, p.232-251, St.Petersburg, 2002, CNII PROMETEY.

The paper gives an overview of the properties and embrittlement behaviour of the RPV cladding. It discusses the effect of polluting elements (S and P) on the toughness of the cladding. It also provides data on thermal embrittlement. Austenitization and rapid cooling results good toughness, annealing at 670°C decreases the toughness especially in the case of „high" phosphorus and sulphur content.

005Kostylev.pdf., Kostylev V.I.,Margolin B.Z.," Determination of residual stress and strain Fields caused by cladding and tempering of reactor pressure vessels", International Journal of Pressure Vessels and Piping 77 (2000) p. 723

Constitutive equations for the calculation of stress and strain during cladding and post-weld tempering are presented. Theoretical and experimental investigations of residual stress and strain caused by cladding and tempering of reactor pressure vessels were performed.

In particular, stress and strain in reactor pressure vessels of WWER-440 and WWER-1000 types, thick plates and model specimens were determined. An engineering procedure for the determination of residual stress and strain in reactor pressure vessels for various temperatures and durations of post-weld tempering is proposed.

The paper supplies data on the thermal, tensile, and toughness properties of the RPV cladding in the function of the temperature

12.4 Further References

none

13 Workshop Recommendations and Actions

- 3rd Consolidation Workshop on WWER RPV Embrittlement in 2nd Quarter 2009 in preparation for training workshop in 2010 (draft of lectures will be discussed and special Issue IJNKM) *(UvE, BH)*
- Joint Training Workshop JRC/IAEA in 2010 on WWER RPV Embrittlement in the 10 subfields in Petten or Vienna *(UvE, MB, KSK)*
- Special Issue in the IJNKM on the NKP&C Workshop with a general paper and10 specific papers by the reviewers of the 10 sub-fields for 2010 *(VS, UvE)*
- Strengthen co-operation with NULIFE *(MB, RA, UvE)*
- Supply all new papers per sub-field to Bianca Hirte *(reviewers, BH)*
- Comparison of the consolidated conclusions and open issues of 1st and 2nd WWER NKP&C Workshop (2007 & 2008) *(UvE)*
- JRC/IAEA to inquire the possibility to organize again the successful series of Specialist Meetings on Irradiation Embrittlement (IAEA SM) in 2010/11 *(KSK, UvE)*
- The final goal is to cover the State-of-the-Art of WWER RPV embrittlement issues in a book (MB)

14 Annexes

Annex I

Agenda
2nd Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop
WWER – WS2
Amsterdam, 26-28 November 2008

Date: 26/11/2008

12h00	Arrival and Lunch
14h00	Welcome and Planning (UvE)
14h10	Organisational Matters (BH)
14h20	IE NKPC&D – Strategy (UvE)
14h35	CAPTURE – WP 2009 (UvE)
14h45	IAEA – Nuclear Knowledge Preservation Activities (KSK)
15h00	NULIFE – Plans and Status (MB)
15h15	ENEF and SNE-TP (UvE)
15h30	General Info and Background 1st Workshop (UvE)
15h45	Coffee Break
16h15	Subject: 1-SOL – Presentation (MS)
16h30	Subject: 1-SOL – Discussion (all, MS)
17h00	Subject: 2-Irradiation Shift – Presentation (TW)
17h15	Subject: 2-Irradiation Shift - Discussion (all, TW)
17h45	Closure Day 1 (MB, UvE)

Date: 27/11/2008

09h30	Subject: 3-Property-Property Correlation – Presentation (AB)
09h45	Subject: 3-Property-Property Correlation – Discussion (all, AB)
10h15	Subject: 4-Annealing and Re-irradiation – Presentation (AC)
10h30	Subject: 4-Annealing and Re-irradiation – Discussion (all, AC)
11h00	Coffee Break
11h30	Subject: 5-Material Factors – Presentation (FG)
11h45	Subject: 5-Material Factors - Discussion (all, FG)
12h15	Subject: 6-Environmental Factors – Presentation (KI)
12h30	Subject: 6-Environmental Factors – Discussion (all, KI)
13h00	Lunch Break
14h30	Subject: 7-Mechanisms and Microstr. Evolution – Presentation (VS)
14h45	Subject: 7-Mechanisms and Microstr. Evolution – Discussion (all,VS)
15h15	Subject: 8-PLEX Issues – Presentation (RA)
15h30	Subject: 8-PLEX Issues – Discussion (all, RA)
16h00	Coffee & Closure Day 2 (MB, UvE)

Date: 28/11/2008

09h30	Subject: 9-Surveillance – Presentation (LK)
09h45	Subject: 9-Surveillance - Discussion (all, LK)
10h15	Subject: 10-Cladding – Presentation (FG)
10h30	Subject: 10-Cladding – Discussion (all, FG)
11h00	Coffee Break
11h30	Next Steps (Publication/ Dissemination/Database) (MB/UvE)
12h00	Conclusions and Closure Workshop (MB, LD, UvE)

Experts
1st Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop
WWER - WS2
Amsterdam, 26 - 28 November 2008

Participants	*Interested*	
	V Petrosyan	Armenia
	T Kamenova	Bulgaria
K Ilieva		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
L Kupca		Slovak Republic
V Slugen		Slovak Republic
	A Hanzel	Slovak Republic
A Ballesteros		Spain
M Serrano		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
J Degmova		EC – DG JRC
R Ahlstrand		EC – DG JRC
L Debarberis		EC – DG JRC
U von Estorff		EC – DG JRC

List of WWER units in the world

Table 1 - WWERs in operation

IsoCode Of country	Station	Type	NetElecCapacity	Model
AM	ARMENIA-2	PWR	376	WWER V-270
BG	KOZLODUY-5	PWR	953	WWER V-320
BG	KOZLODUY-6	PWR	953	WWER V-320
CN	TIANWAN 1	PWR	1000	WWER V-428
CZ	DUKOVANY-1	PWR	412	WWER V-213
CZ	DUKOVANY-2	PWR	412	WWER V-213
CZ	DUKOVANY-3	PWR	427	WWER V-213
CZ	DUKOVANY-4	PWR	412	WWER V-213
CZ	TEMELIN-1	PWR	930	WWER V-320
CZ	TEMELIN-2	PWR	930	WWER V-320
FI	LOVIISA-1	PWR	488	WWER V-213
FI	LOVIISA-2	PWR	488	WWER V-213
HU	PAKS-1	PWR	437	WWER V-213
HU	PAKS-2	PWR	441	WWER V-213
HU	PAKS-3	PWR	433	WWER V-213
HU	PAKS-4	PWR	444	WWER V-213
RU	BALAKOVO-1	PWR	950	WWER V-320
RU	BALAKOVO-2	PWR	950	WWER V-320
RU	BALAKOVO-3	PWR	950	WWER V-320
RU	BALAKOVO-4	PWR	950	WWER V-320
RU	KALININ-1	PWR	950	WWER V-338
RU	KALININ-2	PWR	950	WWER V-338
RU	KALININ-3	PWR	950	WWER V-338
RU	KOLA-1	PWR	411	WWER V-230
RU	KOLA-2	PWR	411	WWER V-230
RU	KOLA-3	PWR	411	WWER V-213
RU	KOLA-4	PWR	411	WWER V-213
RU	NOVOVORONEZH-3	PWR	385	WWER V-179
RU	NOVOVORONEZH-4	PWR	385	WWER V-179
RU	NOVOVORONEZH-5	PWR	950	WWER V-187
RU	VOLGODONSK-1	PWR	950	WWER V-320I
SK	BOHUNICE-2	PWR	408	WWER V-230
SK	BOHUNICE-3	PWR	408	WWER V-213
SK	BOHUNICE-4	PWR	408	WWER V-213
SK	MOCHOVCE-1	PWR	405	WWER V-213
SK	MOCHOVCE-2	PWR	405	WWER V-213
UA	KHMELNITSKI-1	PWR	950	WWER V-320
UA	KHMELNITSKI-2	PWR	950	WWER V-320
UA	ROVNO-1	PWR	381	WWER V-213
UA	ROVNO-2	PWR	376	WWER V-213
UA	ROVNO-3	PWR	950	WWER V-320
UA	ROVNO-4	PWR	950	WWER V-320
UA	SOUTH UKRAINE-1	PWR	950	WWER V-338
UA	SOUTH UKRAINE-2	PWR	950	WWER V-338
UA	SOUTH UKRAINE-3	PWR	950	WWER V-320
UA	ZAPOROZHE-1	PWR	950	WWER V-320

UA	ZAPOROZHE-2	PWR	950	WWER V-320
UA	ZAPOROZHE-3	PWR	950	WWER V-320
UA	ZAPOROZHE-4	PWR	950	WWER V-320
UA	ZAPOROZHE-5	PWR	950	WWER V-320
UA	ZAPOROZHE-6	PWR	950	WWER V-320
			35641	

Table 2 – WWERs already shut-down

IsoCode of Country	Station	Type	NetElecCapacity	Model
AM	ARMENIA-1	PWR	376	WWER V-270
BG	KOZLODUY-1	PWR	408	WWER V-230
BG	KOZLODUY-2	PWR	408	WWER V-230
BG	KOZLODUY-3	PWR	408	WWER V-230
BG	KOZLODUY-4	PWR	408	WWER V-230
DE	GREIFSWALD-1(KGR 1)	PWR	408	WWER V-230
DE	GREIFSWALD-2 (KGR 2)	PWR	408	WWER V-230
DE	GREIFSWALD-3 (KGR 3)	PWR	408	WWER V-230
DE	GREIFSWALD-4 (KGR 4)	PWR	408	WWER V-230
DE	GREIFSWALD-5 (KGR 5)	PWR	408	WWER V-213
DE	RHEINSBERG (KKR)	PWR	62	WWER-70
RU	NOVOVORONEZH-1	PWR	197	WWER V-120
RU	NOVOVORONEZH-2	PWR	336	WWER V-120
SK	BOHUNICE-1	PWR	408	WWER V-230
			5051	

Table 3 – WWER actually in built (2007)

IsoCode of country	Station	Type	NetElecCapacity	Model
BG	BELENE-1	PWR	953	WWER V-466
BG	BELENE-2	PWR	953	WWER V-466
CN	TIANWAN 2	PWR	1000	WWER V-428
IN	KUDANKULAM-1	PWR	917	WWER V-412
IN	KUDANKULAM-2	PWR	917	WWER V-412
IR	BUSHEHR-1	PWR	915	WWER V-446
RU	BALAKOVO-5	PWR	950	WWER V-320
RU	KALININ-4	PWR	950	WWER V-320
RU	VOLGODONSK-2	PWR	950	WWER V-320I
UA	KHMELNITSKI-3	PWR	950	WWER V-320
UA	KHMELNITSKI-4	PWR	950	WWER V-320
			10405	

European Commission

EUR 23719 EN– Joint Research Centre – Institute for Energy
Title: 2nd Nuclear Knowledge Preservation and Consolidation (NKP&C) Workshop – WWER – WS2 – 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
Luxembourg: Office for Official Publications of the European Communities
2009 –184 pp. – 21 x 29.8 cm
EUR – Scientific and Technical Research series – ISSN 1018-5593

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 second time for the present pilot study for consolidating and preserving WWER RPV safety related literature, which is part of a wider Nuclear Knowledge Preservation and Consolidation activity in the Nuclear Design Safety unit of the Institute for Energy. The WWER type reactors were widely built and used, mainly throughout Russia and the Eastern European countries. The WWER RPV was chosen due to the urgency regarding loss of knowledge and due to the proactive attitude of the involved experts and scientists.

Several reviewers received between 10-100 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.

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