IAEA-TECDOC-1442

Guidelines for prediction of irradiation embrittlement of operating WWER-440 reactor pressure vessels

Report prepared within the framework of the coordinated research project

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FOREWORD

This TECDOC has been developed under an International Atomic Energy Agency Coordinated Research Project (CRP) entitled Evaluation of Radiation Damage of WWER Reactor Pressure Vessels (RPV) using Database on RPV Materials to develop the guidelines for prediction of radiation damage to WWER-440 PRVs.

The WWER-440 RPV was designed by OKB Gidropress, Russian Federation, the general designer. Prediction of irradiation embrittlement of RPV materials is usually done in accordance with relevant codes and standards that are based on the large amounts of information from surveillance and research programmes. The existing Russian code (standard for strength calculations of components and piping in NPPs – PNAE G 7-002-86) for the WWER RPV irradiation embrittlement assessment was approved more than twenty years ago and based mostly on the experimental data obtained in research reactors with accelerated irradiation. Nevertheless, it is still in use and generally consistent with new data.

The present publication presents the analyses using all available data required for more precise prediction of radiation embrittlement of WWER-440 RPV materials. Based on the fact that it contains a large amount of data from surveillance programmes as well as research programmes, the IAEA International Database on RPV Materials (IDRPVM) is used for the detailed analysis of irradiation embrittlement of WWER RPV materials. Using IDRPVM, the guideline is developed for assessment of irradiation embrittlement of RPV ferritic materials as a result of degradation during operation. Two approaches, i.e. transition temperatures based on Charpy impact notch toughness, as well as based on static fracture toughness tests, are used in RPV integrity evaluation.

The objectives of the TECDOC are the analysis of irradiation embrittlement data for WWER-440 RPV materials using IDRPVM database, evaluation of predictive formulae depending on chemical composition of the material, neutron fluence, flux, and development of guidelines for prediction of radiation embrittlement of WWER-440 of operating reactor pressure vessels including methodology for evaluation of surveillance data of a specific operating unit.

Lead contributions were made by M. Valo (Finland), A. Kryukov (Russian Federation), F. Gillemot (Hungary) and L. Debarberis (EC/JRC). M. Brumovský (Czech Republic) was the Chief Scientific Investigator. The IAEA officer responsible for this report was Ki-Sig Kang of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Prediction of irradiation embrittlement of reactor pressure vessel (RPV) materials is performed usually in accordance with relevant codes and standards that are based on large amount of information from surveillance and research programmes. The existing Russian Code (Standard for Strength Calculations of Components and Piping in Nuclear Power Plants (NPPs) – PNAE G 7-002-86)[1] for the WWER RPV irradiation embrittlement assessment was approved more than 20 years ago and based mostly on the experimental data obtained in research reactors with accelerated irradiation.

Nevertheless, it is still in good use and generally consistent with new data. The validation of the above Code has been made without the surveillance specimen results that were produced in 1980-1990s. Thus, new analysis of all available data is required for more precise prediction of radiation embrittlement of RPV materials WWER-440/V-230 RPVs do not have surveillance programmes. This fact was a reason to take so-called “boat-samples” from the RPV inner surface of operated units. The results of boat samples testing could be used as verification for the RPV irradiation embrittlement assessment. However, boat sampling cannot be performed for the cladded RPVs.

It means that the irradiation embrittlement prediction for cladded WWER-440/V-230 RPVs could be made using surveillance specimens results from WWER-440/V-213 and materials irradiated in WWER-440/V-213 surveillance channels, as well as results from testing boat samples. Even for RPVs with surveillance programmes there is no standard procedure for using the surveillance results in the RPV integrity assessment.

Up to now a lot of surveillance data have been obtained from operating WWER type reactors. Also some years ago the IAEA International Database on RPV Materials (IDRPVM) has been created. A large amount of RPV irradiation embrittlement data have been obtained in the framework of national research programmes as well. Using these data with the surveillance results could considerably expand the WWER RPV material database and assist in the RPV integrity assessment.

The IDRPVM is a research database under the responsibility of the IAEA and has 21 files describing the materials production history, chemistry, mechanical properties in as received and in aged conditions, and the diagrams and visual information can be possible for drawing the pictures e.g. metallography etc. The database has the function to filter the data according to property or any combination of properties defined by the user, and export the selection in Excel Dbase, or text file format.

The IDRPVM mainly collects raw data, which can be analysed and reanalysed in the future according to the new developments in evaluation methods, standards and codes. The IDRPVM has two sections: one is the research data from research reactor, and the other is the utility data from operating Npps. The use of utility data is strictly restricted to keep the confidentiality and only the participants signed the agreement and provided utility data can access the IDRPVM.

The research section collected data from the IAEA Coordinated Research Project (CRP) programmes run during the past ten years, and there is an interest from the European Union (EU) 5th and 6th Framework Programmes projects, OECD projects and from other national research programmes to join. Presently more than 25 000 Charpy records, several thousands of fracture mechanical and tensile test data obtained on various materials are included. Analysis of WWER embrittlement data will help in understanding and developing general radiation embrittlement models for RPV steels.

1.2. OBJECTIVES OF THE TECDOC

Based on the fact that large amount of data from surveillance programme as well as some research programmes, IDRPVM could be used for the detailed analysis radiation embrittlement of WWER RPV materials. To develop the guidelines of prediction of radiation damage of WWER-440 PRVs, a CRP on "Evaluation of Radiation Damage of WWER Reactor Pressure Vessels using Database on RPV Materials" was started. Main tasks of this CRP were:

- Collection of complete WWER-440 surveillance and other similarly important data into the IDRPVM,
- Analysis of radiation embrittlement data of WWER-440 RPV materials using IDRPVM database,
- Evaluation of predictive formulae depending on material chemical composition, neutron fluence and neutron flux,
- Guidelines for prediction of radiation embrittlement of operating reactor pressure vessels of WWER-440 including methodology for evaluation of surveillance data of a specific operating unit.

The content of this TECDOC is based on presently applied methods, the results of direct analysis within the CRP, and the analysis and evaluation of other relevant information and experience.

1.3. SCOPE OF THE COORDINATED RESEARCH PROJECT

The CRP has been divided into four phases, which cover both collection and analysis of experimental data together with their evaluation for RPV integrity assessment.

1.1.1. Phase A

- Data collection and validation of all existing surveillance and research data from radiation embrittlement of WWER-440 RPVs from national surveillance and research programmes: Main interest are focused on collecting data of units not yet included in the IAEA database on RPV materials and irradiated in surveillance channels of operating plants. Data collection will cover primary irradiation embrittlement, using standard specimens. Collection of the extended information on neutron dosimetry for WWER data was included in the database.

1.1.2. Phase B

- Analysis of database raw experimental data using unified methods for determination of transition temperatures and other parameters of radiation embrittlement: Such analysis is necessary to obtain comparable data from different national programmes to be used for a generalisation of data results into material predictive formulae.

1.1.3. Phase C

- Analysis of data from the extended DRPVM database on WWER-440 RPV materials: The predictive trend formulae are elaborated for radiation embrittlement of WWER-440 RPV materials taking into account material chemical composition, type of metal (base and weld metal), neutron fluence/flux as well as possible effect of thermal ageing. Comparison of elaborated trend formulae with existing formulae from Russian "Standards for Strength Calculations of Components and Piping of NPPs".

1.1.4. Phase D

- Guidelines are developed for prediction of radiation embrittlement of operating reactor pressure vessels of WWER-440 type including evaluation and use of surveillance data from a specific unit.

The CRP group consisted of 7 Member countries as shown in Table 1.1.

Table 1.1. Participants in the CRP on evaluation of irradiation damage of WWER RPV using database

COUNTRY	ORGANIZATION	PARTICIPANT
Bulgaria	Bulgarian Academy of Sciences	Stefan Vodenicharov
Czech Republic	NRI Rez plc.	Milan Brumovsky
Finland	VTT Industrial Systems	Matti Valo
Hungary	KFKI	Ferenc Gillemot
Russian Federation	Kurchatov Institute	Aleksandr Kryukov
Slovakia	VÚJE	Ludovit Kupca
Ukraine	Ukrainian Academy of Sciences	Edvard Grynik / Ludmila Chyrko
Netherlands	JRC-IE, Petten	Luigi Debarberis

2. REACTOR PRESSURE VESSELS

The WWER-440 RPV was designed by OKB Gidropress, of the Russian Federation, the general designer for all NPPs in the former Soviet Union and the Community for Mutual Economical Assistance (CMEA) countries [2]. Some small modifications were made in the Czech Republic designs by Skoda company. There are two designs for the WWER-440/V-230 and the WWER-440/V-213. The Type V-230s were built first. The WWER-440 pressure vessels were manufactured at two plants, the Izhora Plant near Saint Petersburg in Russia and the Škoda Nuclear Machinery Plant in the Czech Republic

2.1. DESIGN FEATURES OF WWER 440 RPVS

The WWER-440 RPVs consist of the vessel itself, vessel head, support ring, thrust ring, closure flange, sealing joint, and surveillance specimens. In case of WWER/V-230, they have no surveillance specimens. The RPVs belong to the normal operation system, seismic Class I and are designed for:

- Safe and reliable operation for over 30–40 years;
- Allowing the Non-Destructive Testing (NDT) of the base and weld metal and decontamination of the internal surfaces;
- Surveillance programme to monitor materials properties degradation due to radiation and thermal ageing (not in the case of WWER/V-230 type of reactors); and
- Withstand all operational, thermal and seismic loadings.

The WWER RPVs have some significant features that are different from the western designs:

- The WWER-440 RPVs (as well as all other components) must be transportable by land, i.e. by train and/or by road. This requirement has some very important consequences on vessel design, such as a smaller pressure vessel diameter, which results in a smaller water gap thickness and thus a higher neutron flux on the reactor vessel wall surrounding the core and, therefore, requirements for materials with high resistance against radiation embrittlement.
- Transport by land also results in a smaller vessel mass and, therefore, thinner walls which require higher strength materials.
- The upper part of the vessel consists of two nozzle rings, the upper one for the outlet nozzles and the lower one for the inlet nozzles. An austenitic stainless steel ring is welded to the inside surface of the vessel to separate the coolant entering the vessel through the inlet nozzles from the coolant exiting the vessel through the outlet nozzles. This design results in a rather abrupt change in the axial temperature distribution in the vessel, but uniform temperatures around the circumference.
- Some WWER-440 RPVs/V-230 type have no or only partial austenitic cladding.
- The WWER-440 RPVs are made only from forgings, i.e. from cylindrical rings and from plates forged into domes. The spherical parts of the vessels, (the bottom and the head) are either stamped from one forged plate, or welded from two plates by electroslag welding, followed by stamping and a full heat treatment. There are no axial welds.
- The WWER-440 RPVs inlet and outlet nozzles are not welded to the nozzle ring but they are machined from a thicker forged ring.

A sketch of typical WWER pressure vessels is shown in Figure 2.1 and the main design parameters are listed in Tables 2.1 and 2.2.

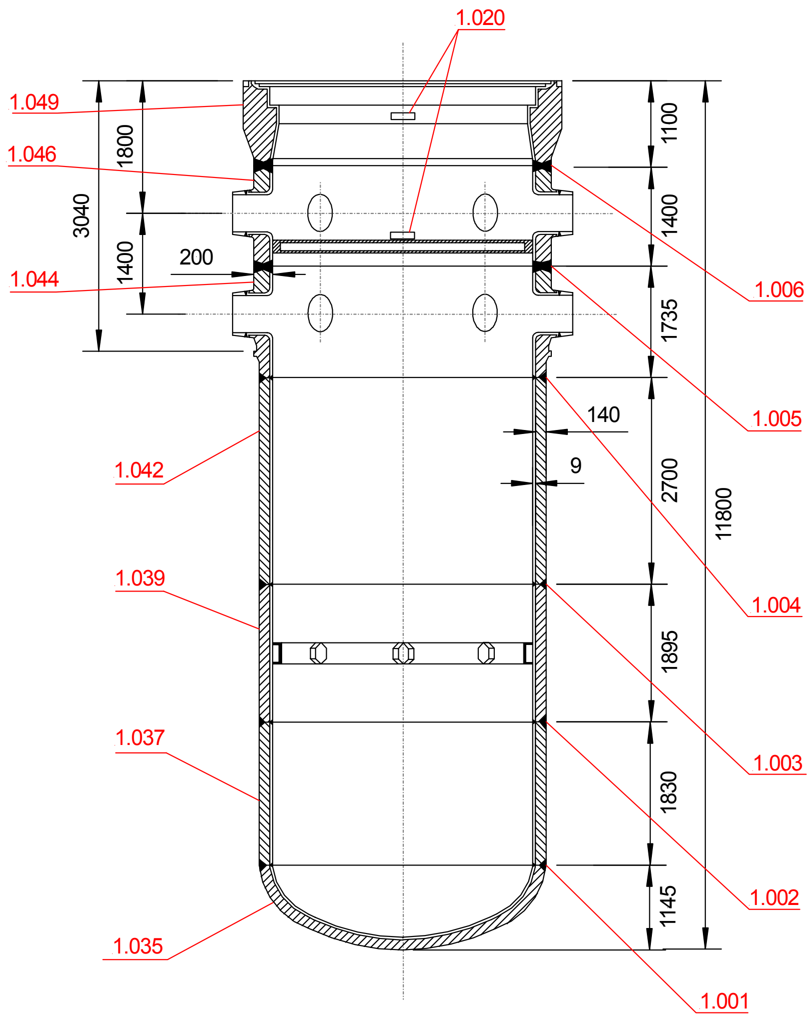


FIG. 2.1. Scheme of a typical WWER-440 reactor pressure vessel.

2.2. MATERIALS AND FABRICATION OF WWER-440 RPV

The chemical compositions of the WWER RPV materials are listed in Table 2.3. The allowable impurities in the beltline region are listed in Table 2.4., and the guaranteed mechanical properties are listed in Table 2.5. The WWER RPV materials are basically different than the Western RPV materials. The Type 15Kh2MFA(A) material used for the WWER-440 RPVs contains 0.25 to 0.35 mass percent vanadium and very little nickel (maximum of 0.40 mass percent).

Material with vanadium alloying was first used in the former Soviet Union naval RPVs because the vanadium carbides make the material relatively resistant to thermal ageing, fine grained (tempered bainite), and high strength properties. However, the Type 15Kh2MFA(A) material is more difficult to weld than nickel steels and requires very high preheating to avoid hot cracking. The influence of vanadium on the susceptibility of those materials to radiation embrittlement was shown to be negligible.

Not all the WWER-440 RPVs were covered by austenitic stainless steel cladding on their whole inner surface: only approximately half of the WWER-440/V-230 RPVs were cladded. However, all of the WWER-440/V-213 RPVs were covered by cladding on the whole inner surface. The cladding was made by automatic strip welding under flux with two layers — the first layer is made of a Type 25 chromium/13 nickel non-stabilized austenitic material (Sv 07Kh25N13), and the second layer is at least three passes made of Type 18 chromium/10 nickel niobium stabilized austenitic stainless steel (Sv 08Kh18N10G2B) to achieve a required total thickness of cladding equal to 8^{+2}_{-1} mm. Therefore, all the austenitic steels which are in contact with water coolant are stabilized .

The stabilized austenitic stainless steels for cladding contain an alloying element (niobium), which forms stable grain boundary carbides. This prevents chromium depletion along the grain boundaries and makes the material immune to stress corrosion cracking. Non-stabilized material was used for the first layer because the thermal expansion coefficient of that material is closer to the thermal expansion coefficient of the low-alloy pressure vessel material.

The WWER-440 RPVs head contains penetrations with nozzles. The nozzles are welded to the vessel head from inside (buttering) and are protected by stainless steel sleeving (0Kh18N10T). List of abbreviations used for nomenclature of WWER materials based on their chemical composition is given in Table 2.6.

Reactor pressure vessels of the WWER-440 V-230 were manufactured in 60–70s. The base composition of the reactor steel was chosen successful and there were no essential changes in it. However, at that time there was lack of information concerning materials behaviour under irradiation. It was not known, that rather small content of such impurities as phosphorous and copper could essentially reduce the radiation stability of the steel. The content of harmful impurities, for example, in weld metal, was not measured at manufacturing. The most WWER-440/230 RPVs were manufactured from the materials with high phosphorous and copper content in the welds. It caused the problem concerning further operation of a part of WWER-440/230 RPVs that had arisen much earlier than it was supposed.

The specific issues of WWER-440/V-230 RPVs are the following:

- Relatively high content of phosphorus in the weld located in beltline region;
- Limited understanding of the role of phosphorus and copper in radiation damage of the RPVs material;
- Absence of surveillance specimen programmes concerning steel properties behaviour;
- Limited information about initial transition temperature T_{K0} of the RPVs material.

The values of phosphorus and copper contents of WWER-440/V-230 RPV materials are given in Tables 2.7. and 2.8.

Table 2.1. WWER -440 RPV design parameters and materials

Reactor	WWER-440	
	V-230	V-213
mass [t]	215	
length [mm]	11,800	
outer diameter [mm]		
- in cylindrical part	3,840	
- in nozzle ring	3,980	
wall thickness (without cladding) [mm]		
- in cylindrical part	140	
- in nozzle ring	190	
number of nozzles	2 x 6[1]	2 x 6[1] + 2 x 3[2]
working pressure [MPa]	12.26	
design pressure [MPa]	13.7	
hydrotest pressure [MPa]	17.1	19.2[3]
operating wall temperature [°C]	265	
design wall temperature [°C]	325	
Vessel design lifetime [y]	30	40

1 Primary nozzle
2 Emergency Core Cooling System (ECCS) nozzle
3 Test pressure has been recently decreased to 17.2 MPa in Hungary, Czech Republic and Slovakia

Table 2.2. Effective full power years (EFPY) fluence for WWER-440 RPV

REACTOR TYPE	FLUX, $m^{-2}s^{-1}$ (E > 0.5MeV)	30 full-power effective years FLUENCE, m^{-2} (E > 0.5MeV)
WWER-440 core weld maximum	1.7×10^{15}	1.6×10^{24}
WWER-440 base metal maximum	2.5×10^{15}	2.4×10^{24}

Table 2.3. Nominal chemical composition of WWER- 440 forging and weld materials (mass%)

MATERIAL	C	Mn	Si	P	S	Cr	Ni	Mo	V
WWER-440 15Kh2MFA	0.13 0.18	0.30 0.60	0.17 0.37	max 0.025	max 0.025	2.50 3.00	max 0.40	0.60 0.80	0.25 0.35
Submerged arc weldSv-10KhMFT + AN-42	0.040 .12	0.60 1.30	0.20 0.60	max. 0.042	max 0.035	1.20 1.80	max 0.30	0.35 0.70	0.10 0.35
Submerged arc weldSv-10KhMFT + AN-42M	0.04 0.12	0.60 1.30	0.20 0.60	max 0.012	max 0.015	1.20 1.80	max 0.30	0.35 0.70	0.10 0.35
Electroslag weld Sv-13Kh2MFT + OF-6	0.11 0.16	0.40 0.70	0.17 0.35	max 0.030	max 0.030	1.40 2.50	-	0.40 0.80	0.17 0.37

Table 2.4. Requirements for aa rpv steel/weld quality (maximum allowable content, mass %)

Element	P	S	Cu	As	Sb	Sn	P+Sb+Sn	Co
AA-quality for beltline materials	0.012	0.015	0.08	0.010	0.005	0.005	0.015	0.020

Table 2.5. Guaranteed mechanical properties of WWER -440 RPV materials*

MATERIAL	20 °C				350 °C				$T_{k0}^{(1)}$ $RT_{NDT}^{(2)}$
	$R_{p0.2}$	R_m	A_5	Z	$R_{p0.2}$	R_m	A_5	Z	
	[Mpa]	[Mpa]	[%]	[%]	[Mpa]	[Mpa]	[%]	[%]	[^{O}C]
15Kh2MFA - base metal	431	519	14	50	392	490	14	50	$0^{(1)}$
A/S weld metal	392	539	14	50	373	490	12	45	$20^{(1)}$

*$R_{p0.2}$ is the 0.2 percent offset yield strength, R_m is the ultimate tensile strength, Z is the percent reduction in area at failure, and T_{k0} is the initial ductile-brittle transition temperature.

Table 2.6. List of abbreviations used in WWER materials

Chemical elements			
A	high quality	**AA**	very high quality/purity
U	improved		
B	niobium	**F**	Vanadium
G	manganese	**Kh**	Chromium
M	molybdenum	**N**	Nickel
Sv	welding wire	**T**	Titanium
Beginning of the designation:			
0	lower than 0.1 mass % C	**08**	mean value 0.08 % C
15	mean value 0.15 % C		
Centre of the designation:			
Kh2	mean value 2 % Cr	**M**	lower than 1 % Mo

Table 2.7. Phosphorus and copper contents in base metal of WWER-440/V-230 RPVs, mass %

	Cu	P
Bohunice-1	0.18	0.012
	0.091	0.01
Bohunice-2	0.08	0.01
	0.082	0.01
Greifswald-1	0.17	0.01
Greifswald-2	0.18	0.015
Greifswald-3		0.012
		0.012
Greifswald-4	0.12	0.016
Kola-1		0.012
Kola-2		0.012
Kozloduy-1	0.15	0.01
Kozloduy-2	0.17	0.017
Kozloduy-3	0.17	0.016
Kozloduy-4	0.1	0.012
Novovoronezh-3	0.16	0.012
Novovoronezh-4		0.011
Minimum value	0.017	0.010
Maximum value	0.18	0.017

Table 2.8. Phosphorus and copper contents in weld metal of WWER-440/V-230 RPVs, mass %

Reactor	Cu	P
Medsamor-1	0.16	0.03
Bohunice-1	0.15	0.043
	0.103	0.036
Bohunice-2	0.2	0.036
	0.11	0.026
Greifswald-1	0.10	0.043
	0.1	0.043
Greifswald-2	0.15	0.036
	0.15	0.032
		0.036
Greifswald-3	0.12	0.035
Greifswald-4	0.16	0.035
Kola-1	0.13	0.032
	0.146	0.033
Kola-2	0.154	0.036
		0.038
Kozloduy-1	0.12	0.051
		0.036
Kozloduy-2	0.18	0.036
		0.038
Kozloduy-3	0.2	0.036
Kozloduy-4	0.04	0.021
Novovoronezh-3	0.15	0.031
Novovoronezh-4	0.17	0.031
Minimum value	0.04	0.021
Maximum value	0.20	0.051

3. SURVEILLANCE PROGRAMMES

Surveillance programmes as a mandatory part of reactor design were implemented only for WWER-440/V-213 [2].

3.1. STANDARD SURVEILLANCE PROGRAMMES

The same standard surveillance programmes were incorporated to all WWER – 440/V-213 RPVs practically in NPPs in Russia, Ukraine, Finland, Czech Republic, Slovak Republic and Hungary.

3.1.1. Description of the Standard Surveillance Programme

Test materials

Test specimens were manufactured from the following materials used in pressure vessels:

- Base metal (steel 15Kh2MFA) from a ring used for beltline region;
- Weld metal (submerged arc weld -Sv-10KhMFTU) from a welding coupon represented the weld in the lower part of beltline region (weld 0.1.4 = 5/6);
- Heat affected zone from this welding joint; all materials are of Cr-Mo-V type.

Test specimens

Three type of specimens are part of the programme:

- Static tensile (dia 3 mm x 30 mm);
- Charpy V-notch for impact testing;
- Pre-cracked Charpy (COD) type for static fracture toughness testing.

In all cases specimens were cut from the central part of material thickness not closer than 1/4 of thickness to pressure vessel surface.

Neutron flux monitors

Fast neutron flux (and fluence) measurements are based on measurement of the following activation monitors which are used according to their lifetime: 54Fe (n,p), 63Cu (n, alpha), 93Nb(n,n´).There are 18 containers with neutron flux monitors in each set irradiated up to two years, and 6 containers for sets that are irradiated longer. Monitors are made from thin foils that are put into aluminum tubes and located in upper part of specimen container.

Temperature monitors

Natural diamond powder is used as a temperature monitor. Information from this monitor is not reliable as several parameters play a non-identified role.

Capsule assemblies

The test specimens are placed into thick wall pressure resistant capsules made from austenitic stainless steel that prevents corrosion effects on specimens. Scheme of such capsules is given in Fig. 3.1. Each capsule contains either 6 tensile specimens or 2 specimens of Charpy type specimens (Charpy V-notch or COD specimens). Some of them contain also neutron flux monitors. Capsules are assembled into chains — each contains 19 or 20 capsules in active core position. Two chains contain also another 13 capsules that are located in upper part of reactor pressure vessel, out of neutron field, and these specimens are determined for determination of thermal ageing affect.

Two chains of capsules represent one set of specimens, that is located on outer surface of active core basket, symmetrically to the corner of hexagonal active core structure. Thus, six sets of chains are placed into each pressure vessel. Capsules with the same type of specimens are located in chains in such a way to reach approximately the same neutron fluence — in some cases they are located in one chain in one row-compartment, in other way in two chains in the same location.

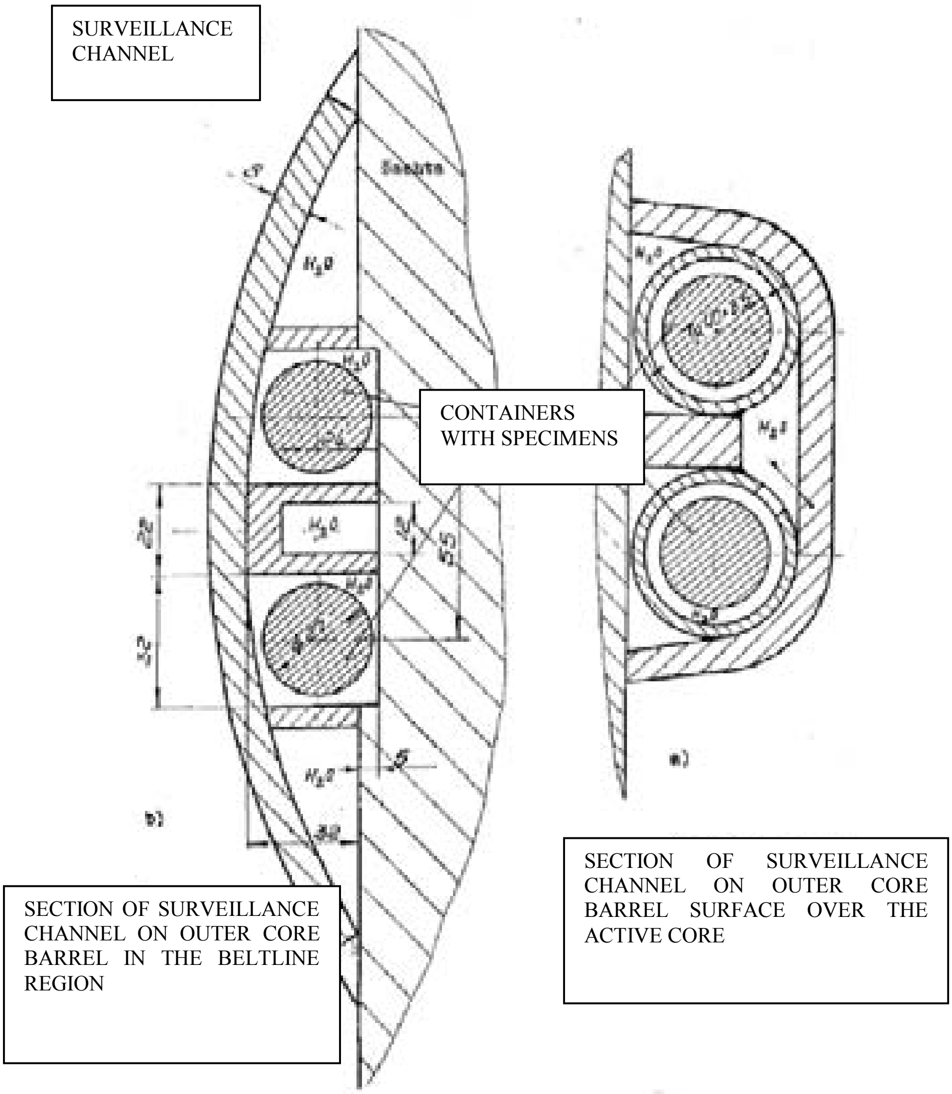


FIG. 3.1. Capsule assembly.

Removal schedule

Removal schedule is given by operation schedule of reactor, i.e. by fuel elements reloading. Interval between these changes is approximately one year, thus modified schedule of withdrawal with respect to lead factors are 1, 2, 3, 4 or 5 years of operation + 2 sets for 10 to

20 years. This schedule removal is slightly different in different NPPs, for example, in NPP Dukovany one set is withdrawn after 5 years of operation, annealed and then tested.

3.1.2. Technical issues of the Standard Surveillance Programme

During years of operation, wide experiences and testing for surveillance specimens have been collected and results were analyzed. Based on test results, the necessary steps for improvement of surveillance programmes were implemented. Here are given technical issues connected with standard surveillance programmes of this type of reactor:

Test materials

Base materials were taken from a technological addition to the real beltline region ring, In accordance with general worldwide practice, weld metals are taken from special test coupons, made by the same technology and from the same materials as a critical weld, where is located in lower part of active core. Specimens from heat-affected zone do not fully represent material from active core region as welding coupons were mostly made from special manufactured plates only.

Neutron flux monitors and neutron fluences

There is only a limited number of fast neutron flux monitors and in most programmes, thermal neutron monitors are not instralled. Location of these monitors in capsules does not allow the determination of exact neutron fluence on each specimen: its location in capsule does not allow determination of exact orientation of capsule with respect to the reactor core. Moreover, capsules are located in a very high radial gradient of neutron flux, which has an influence not only on relative fluences on both (or six) specimens, but also on a capsule as a whole.

Design of capsules in long chains (2 550 mm) results in an asymmetrical axial gradient of neutron flux, where differences between central and outer part of a chain represent more than 10 — see Figure 3.2. Only a limited number of containers in central part are receiving practically similar neutron fluence, in outer parts there is a steep gradient even for one set of specimens. This design also does not allow comparing results from different types of tests, for example of Charpy impact (located in central part) and static fracture toughness tests (located in upper or lower boundary).

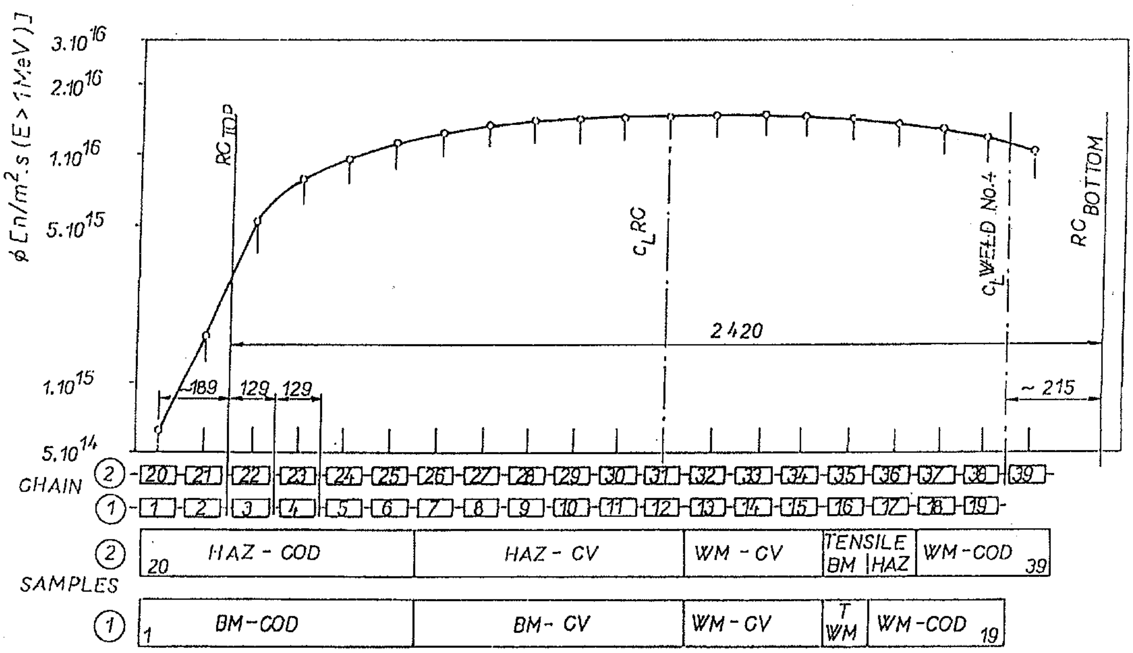


FIG.3.2. Typical distribution of neutron flux along irradiation chain with containers.

Where BM: Base Material, WM: Welding Material, HAZ: Heat Area Zone, COD: Crack Opening Displacement.

Due to the location of capsules on outer surface of active core basket a high lead factor with respect to pressure vessel wall is received: Between capsules and pressure vessel wall there is a water reflector, even though not as thick as for Western PWR type reactors. Thus, lead factor for base material is about 12 to 13 and for weld metal even up to about 18 for full core, but only about 3 for base metal and 4 for weld metal for reduced core. As a result, problem of flux intensity effect is very important and under very frequent discussions.

Temperature monitors

Temperature monitor of diamond powder type were shown to be unusable. Some experiments performed with irradiation of “opened” capsules (i.e. with specimens that were in contact with water) did not show any observable effect on material properties, similar results have been obtained also by direct measurement of irradiation temperature of specimens in special chains using thermocouples — in reactors of Loviisa, Jaslovske Bohunice and Kola NPPs.

Capsule assemblies

Capsules were designed only for two Charpy type specimens (see Figure 3.3) thus for one set of specimens for testing at least 6 capsules are necessary. These capsules have thick walls and hence good thermal contact should be guaranteed by tight tolerances. Design of these capsules does not allow determination of exact orientation of specimens with respect to the reactor active core and thus also an exact determination of fluences on each of specimens. Thus, gamma scanning of individual specimens and/or containers is used for such fluence determination. Capsules with COD specimens are located in positions with flux gradient positions in the surveillance channel.

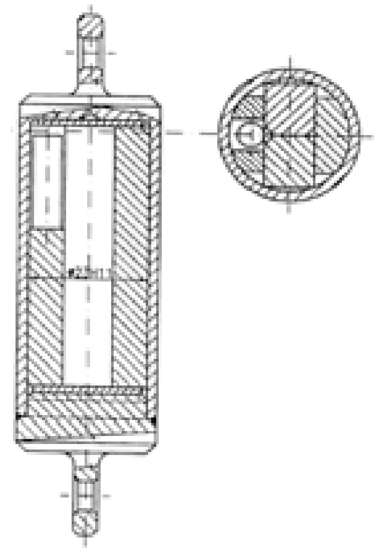


FIG. 3.3. Scheme of a container for standard surveillance programme.

Removal schedule

High lead factor needs very fast removal schedule — irradiation by about 3 years of operation represents practically full design lifetime of vessel material. Thus, maximum of 5 years was chosen to obtain some reliable trend lines, even though it does not respect neutron flux intensity effect.

Now, in most of reactors there is only one remaining set irradiated in each reactor — it has been decided to remove it after not less than 15 years of operation, as it has a meaning practically from thermal ageing point of view. There would not be any neutron measurement in surveillance as well as in inner pressure vessel wall. Only ex-vessel neutron measurements in cavity are performed in some reactors — now they serve as the only information on neutron fluxes on vessel wall.

Static fracture toughness testing

Reactor pressure vessel integrity assessment is fully based on fracture mechanics, i.e. on knowledge of fracture toughness of RPV materials and their changes due to irradiation damage. Nevertheless, the most importance in old surveillance programmes has been concentrated in testing of Charpy V-notch specimens for determination of impact notch toughness, KV or KCV, and its transition temperature.

In the standard surveillance programme, these RPVs specimens for static fracture toughness (COD) determination were planned, but their location is fully unfavourable for such determination: specimens were located in periphery of chains — see Fig.3.1. Thus, each set of two specimens (located in one container) received different neutron fluences, even in one chain and grouping of these specimens into one testing set is in quite impossible. To overcome this problem a modified procedure has been applied — specimens with similar neutron fluences from chains irradiated by different time are grouped together to obtain one transition curve. Another approach uses an advantage of specimen reconstitution — halves of broken Charpy specimens were reconstituted and tested for static fracture toughness.

3.2. PRINCIPLES OF THE SUPPLEMENTARY SURVEILLANCE PROGRAMMES

3.2.1. Dukovany Nuclear Power Plant

Using an experience from a standard surveillance programme, requirements for reliable assessment of reactor pressure vessel residual lifetime, as well as requirements, given by a trend of harmonization of WWER standards and procedures with PWR ones, the following requirements for the supplementary surveillance programmes have been established:

Scheme and main principles

Supplementary surveillance programme has been designed taking into account the experience from the standard surveillance and verification programme:

- Monitoring of RPV during the rest of lifetime, i.e. not only of material degradation but also of neutron field on RPV,
- Use of only archive materials, each container contains twelve inserts for further reconstitution either to Charpy V-notch toughness or static fracture toughness type specimens,
- Irradiation with a low lead factor to be able to fully apply results from surveillance to RPV embrittlement assessment,
- Irradiation of cladding materials as this material was not inserted into the standard surveillance programme and its behaviour is important for RPV integrity assessment during pressure thermal shock (PTS) regimes,
- Determination of an annealing efficiency and a re-embrittlement rate of real RPV materials after annealing as a useful and effective tool for potential NPP (RPV) life extension,
- Use of IAEA reference JRQ material for qualification of irradiation conditions — these specimens are placed in one container in the central part of each chain (i.e. in the same irradiation conditions). Charpy impact tests will be performed to compare irradiation conditions in individual chains — neutron fluences and irradiation temperatures. Thus, this material will serve as a reference solid-state neutron monitors, too.
- Monitoring individual containers has been modified with respect to the experience from the standard surveillance programme:

Specimens materials

Materials for specimens are only of archive type, i.e. the same as for standard surveillance programme. Even though some archive materials still exist in manufacturer store, halves of tested specimens in unirradiated state from standard surveillance programmes have been used. Specimens, put into programme, were manufactured from base and weld metals, only.

Use of reconstitution technique for these specimens after their irradiation allows irradiation of a whole set of specimens in one capsule, only. For this case, specimens of 10x10x14 mm are used, in three rows in one capsule = 12 specimens together in one capsule. Moreover, IAEA reference material (ASTM A 533-B type JRQ from the IAEA programme) is used for comparison of results in different reactors and with different materials.

Irradiation temperature

New design of capsules will contain small gaps between specimens and capsule walls and/or aluminium inserts. Capsules contain melting monitors according to ASTM and DIN standards. Two sets of temperature monitors are placed at least into three containers of each chain (upper, middle and lower part):

- First set is prepared according to ASTM standard (wires in quartz tubes) and contains (inlet water temperature is equal to 268°C): Bi (melting temperature = 271°C), Pb-12.5 In (280°C) and Pb-10 In (291°C),
- Second set is prepared according to DIN standard (hollow rings located in Al-fillings of containers): Bi (271°C), Pb-1.9 Ag-5 Sb (272°C), Pb-12.5 Sn (278°C), Pb-2 Ag-3 Sb (288°C), Pb-10 In (291°C).

Neutron fluence

Proper determination of neutron fluences as well as placing in locations with lower neutron fluxes is the main aim and reason of this supplementary programme. Capsules are located in such places where lead factor would be much lower than for a standard surveillance programme, at least in the range between 2 and 3. This can be reached by placing the capsules in the upper position of capsules, i.e. in place of nowadays nos., 3 and 4 as shown on Figure 3.4.

Two type of monitors are used:

- Type A: two spectrometric sets of monitors are placed in each container: Fe, Ni, Nb, Al-0.1%Co, Ti, Cu and Np - one set is inserted in Al-tube while the other in Gd-tube to be able to determine a correction factor for thermal neutrons;
- Type B: wire type monitors for determination of relative neutron field in each container and for each specimen — O wire (in both ends of container) to determine azimuthal distribution (and also the orientation of container with respect to the reactor active core), and I-wire (in specimens corners) to determine an axial distribution of neutron flux; wires are made either from Fe (for short irradiation) or Nb (for long irradiation),

Moreover, ex-vessel neutron measurements in cavity are realized during each reactor campaign to be able to compare results from surveillance as well as from ex-vessel position every year of operation and supplementary calculations.

Supplementary surveillance programme is divided into four parts:

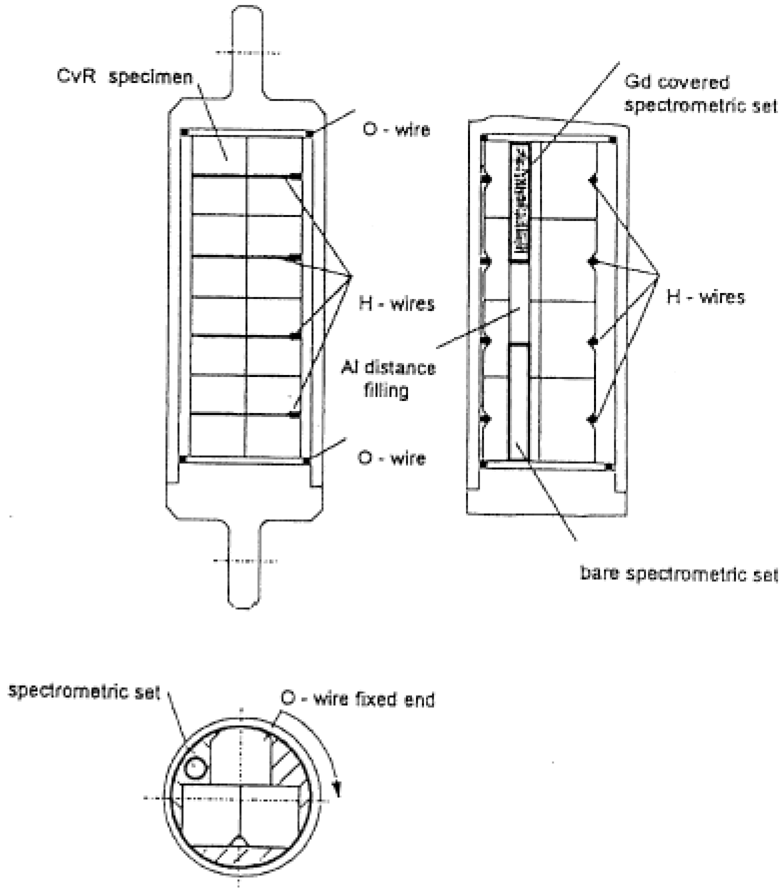


FIG. 3.4. Scheme of a container from the supplementary surveillance programme.

(1) Irradiation with low lead factor

Irradiation of archive materials — base and weld metals — are prepared in all four units with such a schedule to cover the whole planned residual lifetime of RPVs; individual chains are prepared for base metals and other for weld metals — position of containers with base metals are slightly different than those with weld metals in such a way that their lead factors will be the same for both materials (fluence on weld metal is equal to about 70% of base metal in beltline region), the following locations of containers have been chosen:

- Base metal: containers position
  - 2 - Charpy impact
  - 4 - static fracture toughness
  - 6 - tensile
- Weld metal: containers position
  - 1 - Charpy impact
  - 3 - static fracture toughness
  - 5 - tensile
- Reference steel JRQ: containers position
  - 12 - centre of active core
  - 19 - weld No.4

These locations are used also in all parts of the programme and principal time schedule for their withdrawal is 1, 2, 3, 5, 10 and some also 15 years of operation. Specimens will be tested for notch impact toughness as well as static fracture toughness determination (a limited number of tensile specimens is also included).

(2) Determination of re-embrittlement rate

Portion of specimens from base and weld metals irradiated within the standard surveillance programme has been annealed by a standard regime (475°C-100 h) in laboratory and then inserted into RPV in similar positions as non-irradiated specimens, i.e. in positions with low lead factor (between 2 and 3) to be fully applicable to RPV embrittlement; irradiated and annealed specimens will be withdrawn in one, three and five years intervals to assess re-embrittlement rate of RPV archive materials in real RPV irradiation conditions, both type of testing (notch impact as well as static fracture toughness) will be performed on both materials.

(3) Irradiation of cladding materials

Specimens for characterization of cladding properties are cut from outer and inner cladding layers as well as from heat-affected zone in base metal. This cladding material comes from an archive part of vessel-nozzle bottoms after their removal from the vessel. Cladding materials will be irradiated in two groups: one group in low lead factor position to obtain RPV end-of-life fluence (with similar schedule as base and weld materials, i.e. up to 15 years), the second group will be irradiated for 3 years in high lead factor, then it will be annealed and again inserted into the reactor. Further withdrawal is planned to cover an assessment of re-embrittlement rate after annealing for another 3 years of operation (represents more than 30 normal operating years), only static fracture toughness tests will be realized for these materials

(4) Neutron dosimetry chains

Even though some of chains within parts (1) to (3) will cover practically the whole remaining RPV lifetime, their long term irradiation will not allow to determine real neutron fluences in all operation years; thus, special chains with only a limited number of containers are inserted into reactors practically through the whole lifetime beginning five years from now; these chains for intermediate irradiation (for two years only, in principle) will contain only IAEA reference material (JRQ steel) and neutron monitors in position of maximum neutron fluence and weld metal locations — both reference material and neutron monitors will serve for determination/precisioning of neutron fluxes and irradiation temperatures.

JRQ specimens will be tested by notch impact method, only. Thus, all four parts of the supplementary surveillance programme will ensure necessary and reliable information about RPV materials behaviour, mainly about their irradiation embrittlement under conditions very close to real ones - similar irradiation temperature as well as close neutron flux values. At the same time, information for potential RPV life extension — without and/or with thermal annealing — will be gathered well in advance till the designed RPV lifetime.

This programme started in 1997 but not all chains are inserted immediately — there is a time schedule prepared in such a way to cover their whole remaining design lifetime of reactor pressure vessels.

3.2.2. Paks Nuclear Power Plant

The WWER-440 surveillance programme is unique due to the fact that the specimens are located near the core, which results in accelerated irradiation. Most of the specimens are withdrawn when the units are still relatively new. In case of beltline base metal, four years exposition of the specimens equals 48 years of reactor wall age. After four years of operation only one set remains in each unit, which is overexposed. This surveillance system gives early information about the vessel lifetime, but it doesn't follow the change of the operational modes. To eliminate this disadvantage and to give a more extensive database for PTS analysis and for exact lifetime calculation thea surveillance extension programme has been elaborated and implemented at Paks NPP.

The specimen sets elaborated for the extended surveillance consist of three different types of forged (base) material. The materials are:

- A special heat of the 15Kh2MFA material;
- The IAEA reference material JRQ; and
- The original archive material of every unit (reconstituted specimens from the remnants of the zero level testing).

Every specimen set consists of 12 to 20 Charpy and 6 tensile specimens of each of the abovementioned materials. All specimens are located against the middle of the core to be exposed with the same irradiation fluence, with the exception of 4 Charpy specimens in every set, which are located in low flux positions. These specimens will be collected from the four units, and they will be used to evaluate the flux rate effect, which may affect the surveillance results.

In the extended surveillance programme the foil holders centrally located are used. The foil holders are in the head of the selected capsules. There are 6 to 10 dosimetry foil sets in every extended surveillance set. The foils used in the extended surveillance programme are: Nb; 54Fe; Cu; Co. No melting or diamond powder temperature monitors are in the capsules. Since the high flux irradiation position and the high cross section of the monitor materials they heat themselves and could show false results. European Union (EU) Framework Programme (FP) 5 research programme COBRA proved, that the WWER-440 type specimen capsules are not sensitive for overheating.

For the extended surveillance programme every specimen set is to be irradiated for four years in the reactor. Table 3.1 shows the schedule for the extended surveillance sets at NPP Paks.

Table 3.1. Surveillance extension reloading itinerary at PAKS NPP

Unit	Capacity [MWe]	First operation	Extended specimen set reloading year
Paks1	440	1982	94, 98,2002, 06,10,14,18
Paks2	440	1984	89*,91*,91,92*,92*,95,99,2003,07,11,15,19
Paks3	470	1986	92,96,2000,04,08,12,16,20,24
Paks4	460	1987	93,97,2001,05,09,13,17,21,25

* Remark: preparation sets for research purpose

Evaluation method of the extended surveillance programme

After every four years period the results of every unit will be compared with the original surveillance results. If the new results differ greatly from the original ones, the end-of-life (EOL) calculated from the original surveillance programme results will be modified. The modification will be performed by the use of the following formulae.

$$\mathbf{E_m} = \mathbf{E_t} - \mathbf{C} - \sum_{1}^{n} 4 * \mathbf{K} * \frac{\Delta \mathbf{T_{k[n]}}}{\Delta \mathbf{T_{k[1]}}} \tag{3.1}$$

where E_m is the remaining lifetime, E_t is the full lifetime, calculated from the original surveillance programme, C is the number of operational years until the loading of the first new surveillance set, n is the serial number of the new sets in the unit, K is a correlation factor characterising the relationship between the original and the first extended surveillance period, finally $\Delta T_k(n)$ is the transition temperature shift (in K) measured on the n sets of new specimens. K is evaluated from the operational history, its value generally is equal to one. If in case of any extended set the value of: $\frac{\Delta \mathbf{T_{k[n]}}}{\Delta \mathbf{T_{k[1]}}} > \mathbf{1.5}$, the utility must documented why the embrittlement rate increased.

3.2.3. Jaslovske Bohunice and Mochovce Nuclear Power Plant

Four types of surveillance programmes were (are) realized in Slovak NPPs:

- In Jaslovské Bohunice V-2 NPP (Units 3 and 4), the original Standard Surveillance Specimen Programme (SSSP) was finished,
- Extended Surveillance Specimen Programme (ESSP), which is in progress now, was prepared with aim to validate the SSSP results (see Figure 3.5),
- For the Mochovce NPP Unit 1 and 2 new surveillance programme Modernized Surveillance Specimen Programme (MSSP) was completely prepared(see Figure 3.6).
- For the Bohunice V-1 NPP, New Surveillance Specimen Programme (NSSP)which is coordinated by IAEA, was prepared and is under realization.

Schedule of the SSSP mentioned above and the planned activities in the future are presented in this paper too. Surveillance specimens were prepared from:

- BM: steel 15Kh2MFA — additions to the beltline forgings
- WM and HAZ: so-called R-weld (reference).

The chemical composition has an important role, Cu and P contents must be limited as low as possible. For the neutron fluence monitoring the following procedures were used:

- Fe, Cu, Nb and Co foil detectors placed in the six capsules along the core.
- The neutron fluence was calculated for energy En > 0,5 MeV and En > 0,1 MeV.

The irradiation temperature of the SSSP samples was determined by using of diamond powder monitors. But the results of these measurements were not satisfactory. The irradiated chains withdrawal schedule is shown in Table 3.2:

Table 3.2. Schedule of standard surveillance specimen testing

Unit	1984	1985	1986	1987	1988	1989	1990
EBO 3	Start irradiation	1G1 1G2	6G1 6G2	3G1 3G2		5G1 5G2	
EBO 4		Start irradiation	2G1 2G2	6G1 6G2	3G1 3G2		5G1 5G2

(1) Extended surveillance specimen programme realization

ESSP was prepared in VUJE Institute for two units in Jaslovské Bohunice NPP V-2 (EBO-3 and EBO-4) equipped with WWER-440/213 RPV. This programme is successfully in operation now as it is seen in Table 3.3. The main problem for the ESSP project preparation was in the limited amount of original RPV material. After SSSP applications the rest from sets of un-irradiated specimens were available for ESSP only.

Table 3.3. Extended surveillance specimen programme realization

Unit	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006
EBO3	Start irrad	R31	R32	R33			R34		R35			R36
EBO4	Start irrad	R41	R42	R43		R44			R45			R46

(2) The modernized surveillance specimen programme (MSSP) for NPP Mochovce

The preparation of completely new surveillance specimen programme for the Mochovce unit-1 and 2, which are in operation now, was started in the year 1996. VÚJE prepared the MSSP with new philosophy for monitoring of irradiation embrittlement. This programme has one great advantage to compare with ESSP: complete sets of SSSP specimens. On the basis of a common decision of the utility (SE) and Slovak regulatory body (NRA SR), these problems were included in the range of the Mochovce NPP safety improvements IAEA Safety Issue — CI-01-Irradiation embrittlement monitoring.

3.2.4. Principles of additional surveillance programme in Loviisa NPP

With the experience gained in the irradiations and testing of original surveillance chains Loviisa surveillance positions have become a well-characterized irradiation position. The original surveillance programmes for Loviisa reactors have been completed except some thermal ageing capsules, which still remain in the reactors. The surveillance positions have been widely used in national and international research work during the conduction of original surveillance work and thereafter

Some technical improvements were made to the techniques used in the original surveillance programmes. The capsule diameters were slightly increased and the capsule wall thickness reduced in order to be able to include three standard specimens in one capsule instead of two (the outer diameter was increased from 27.00 mm to 27.50 mm, the inner diameter was increased from 23.0 mm to 24.75 mm and the wall thickness was reduced from 2.00 mm to

1.38 mm). It is also necessary to cut away small amount of material from two corners of standard specimens (1 mm away from the corner).

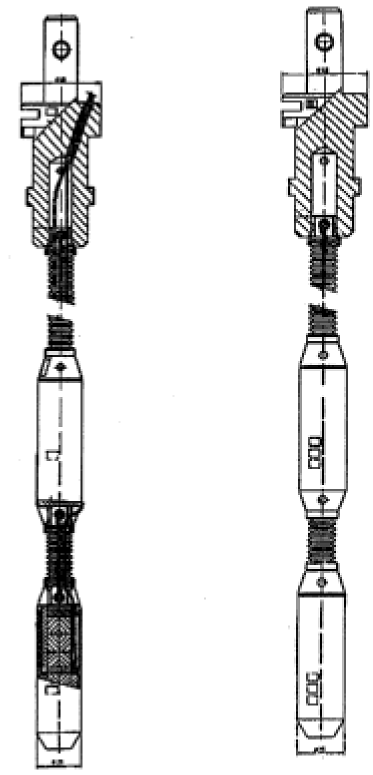


FIG. 3.5. Scheme of irradiation chains with containers connected by bellows in the MSSP and ESSP.

In standard irradiations only the portion of the surveillance channel is utilized, where neutron fluence is relatively constant. In special irradiations of small specimens also the part of the channel above the reactor core, where the fluence gradient is high, is utilized. In these irradiations several fluence levels can be reached during one irradiation period (variation by a factor of 10).

Relatively extensive neutron dosimetry was made for the original surveillance chains. During this period also fluence calculation code called PREVIEW was developed. This code is a kernel based code, which takes into account the fuel source term (fuel loading) and burn-up during one irradiation campaign. With this code irradiation fluence can be calculated in the surveillance channel as well as in the vessel wall. The code has been verified with vessel external dosimetry as well as with scraping samples taken from the inner wall of vessel cladding.

Currently Fe-Ni foils are located on the top and bottom locations inside irradiation capsules in order to be able to determine neutron fluences for each specimen. However, the use of specimen specific fluence values add little to the final analyses as compared to the use of capsule average fluence values.

Specimen irradiation temperature in the irradiation channel was directly measured during one irradiation period by a mock-up capsule, which was instrumented by thermocouples. The temperature remained constant during the whole period and it was only slightly higher (couple of degrees centigrade) than the incoming water temperature. The irradiation temperature has also been monitored at once with special temperature monitoring capsules, which contained several tens of melting alloys.

Conclusion from these temperature measurements is that gamma heating is no problem as concerns specimen irradiation temperature in the reduced core surveillance irradiations. However, cooling water of the surveillance channel flows through a 6 mm hole, which is located in the bottom part of the channel. The flow is driven by the pressure difference across the reactor core. Free water flow in the channel is a requirement of temperature stability in the channel. No constantly repeated supplementary surveillance programmes supporting the study of vessel original embrittlement are going on. Intensive research programmes for re-embrittlement behaviour are going on, which are described in section 4.4.

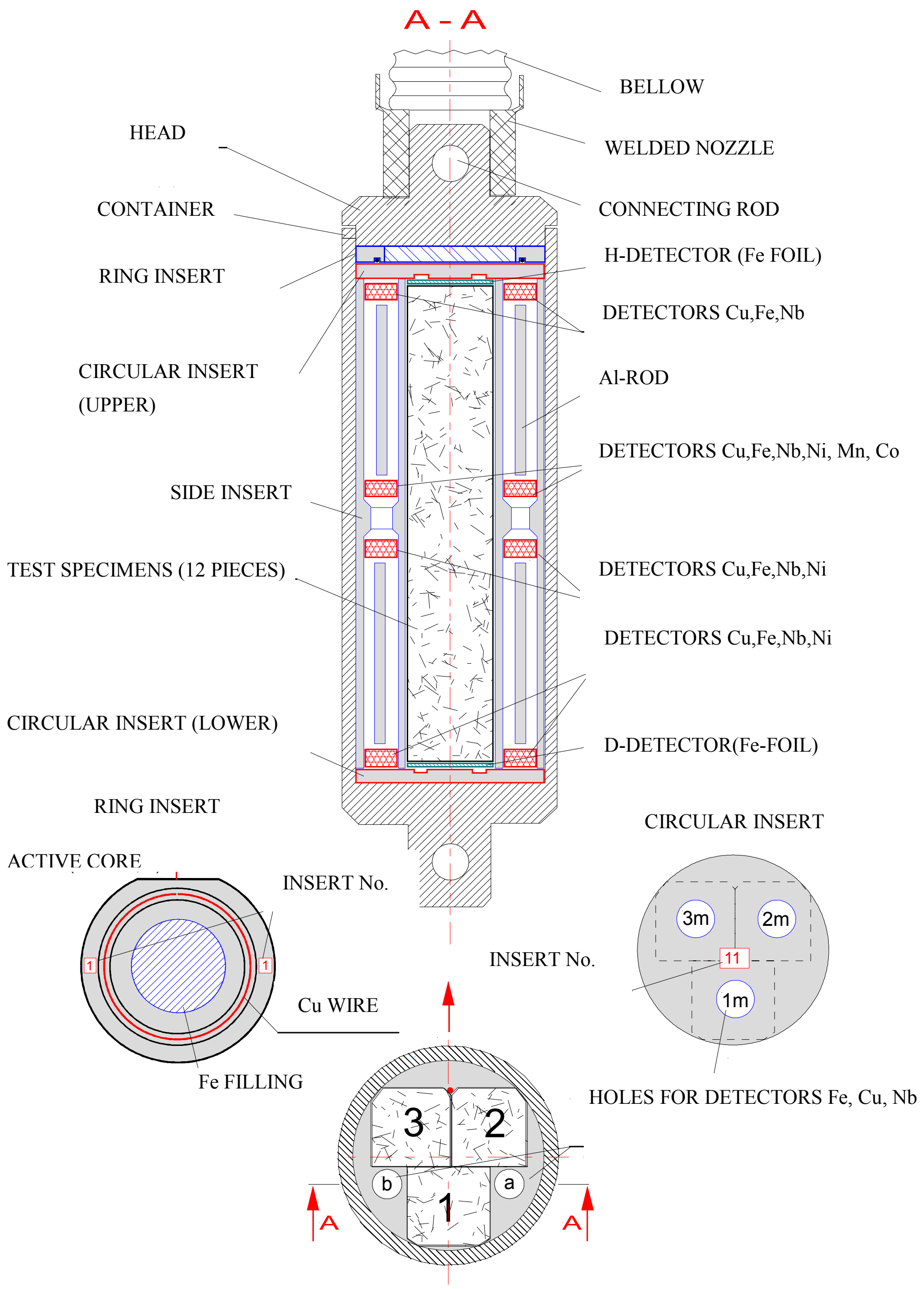


FIG. 3.6. Scheme of irradiationed container in the MSSP.

4. IRRADIATION PROGRAMMES IN SURVEILLANCE CHANNELS — NATIONAL/INTERNATIONAL PROGRAMMES

4.1. RUSSIAN FEDERATION

The activities aimed to more precise irradiation embrittlement assessment in the Russia Federation are described.

4.1.1. TACIS programmes

TACIS programmes were established since 1991 along with other EU programmes as support mechanisms through which nuclear safety aspects could be identified and addressed satisfactorily. The most important TACIS programme for WWER-440 is TACIS-91/1.1 — Reactor Vessel embrittlement. The objectives of this programme was focused on:

- Determination of the correct values of the parameters used to characterize the irradiation embrittlement of RPV material for the operating WWER-440/230 Units;
- Verification of the Russian standards for RPV irradiation embrittlement assessment;
- Validation of the annealing as a way for material toughness recovery;
- Validation of the boat sampling procedure for current RPV embrittlement assessment;
- Improvement of the knowledge on the re-embrittlement after annealing;
- Improvement of the understanding of the mechanisms of RPV material irradiation embrittlement.

In the framework of programme round-robin and parallel tests were carried out and Quality Assurance implemented. The testing was performed in compliance with Russian standard GOST but the experimental data evaluation carried out according western and Russian Codes. The main achievements of the TACIS programme are summarized as follows:

- Novovoronezh NPP (Russian Federation) Units 2–4 RPV materials have been investigated;
- Actual properties of RPV have been evaluated by sub-size impact and tensile specimens fabricated out of samples taken from the vessel inner surface;
- Evaluation of T_{K0} and chemical composition of critical RPV welds were performed;
- The correlation between sub-size and standard Charpy specimens was confirmed;
- It was demonstrated that initial transition temperature T_{K0} values, calculated from chemical composition are not conservative;
- There is an agreement between predicted and measured transition temperature shift values caused by primary irradiation;
- Annealing procedure was validated by investigation of annealed and as-received material,
- Prediction by “lateral shift” model of transition temperature shift during re-irradiation seemed to be reliable for the material behaviour assessment;
- On the base of microstructure investigation the agreement between mechanical properties recovery during annealing and metal structure changes was demonstrated.

On the basis of information, which was gathered in the framework of such projects, the following important positive conclusions were reached:

- The predictive formulas used to assess the embrittlement of RPV (on the basis of the content in impurities and of the neutron fluence) are generally applicable;

Annealing is an efficient procedure to restore the material toughness.

The replacement and modernization of I&C systems and components are discussed in details in several IAEA publications as listed in the Reference section of this TECDOC.

4.1.2. EdF programme

The EdF initiated a research programme on model materials representing WWER-440 weld, base and heat affective zone materials as they were connected with several PHARE and TACIS project as well as with bilateral contracts with Bulgarian and Ukraine NPPs. The objective of the programme is to provide a basis for a comprehensive parametric study of the phosphorus and copper effect for this type of weld materials. The Russian part of work consisted of:

- Retrospective evaluation of T_{K0},
- Temper embrittlement risk during annealing,
- Re-embrittlement rate assessment

The results are important for the evaluation of the residual temperature shift for materials with high phosphorus content up to 0.05%. The initial irradiation and re-irradiation have been carried out in the surveillance channels of Kola NPP.

4.2. UKRAINE

Aiming to determine the effect of the coolant pressure on the characteristics of the crack growth resistance of the materials of the reactor vessels under irradiation, two container assemblies filled with 0.5T CT specimens of 15Kh2MFA and 15Kh2MFAA steels were irradiated. The CT specimens were in two states: unstressed and under the mechanical load imitating the pressure of the coolant. At the same time in the assemblies there were tensile and impact toughness specimens.

The assemblies were open and specimens contacted with the coolant. The irradiation temperature in this case was equal to the temperature of the primary circuit and in the place of the specimen location the temperature was about 290°C.

The fluences of the fast (E > 0,5 MeV) neutrons were $\sim(10.4\pm1.3)\cdot10^{23}$ m^{-2} and $\sim(3.4 \pm 0.4)\cdot10^{23}$ m^{-2} for 15Kh2MFA and 15Kh2MFAA steel specimens, respectively. Irradiation by high fluences of fast neutrons resulted in considerable change as of strength as well as of ductility properties.

Coefficient of radiation embrittlement for 15Kh2MFA steel was A_F = 12.5, and for 15Kh2MFAA steel which melted from more clear furnace charge was A_F = 5.6. The effect of neutron irradiation on radiation embrittlement of Ni-free vessel steel 15Kh2MFA is studied in two states: under mechanical load and without it. The effect of stressed state due to mechanical load, imitating the pressure of the coolant, on the Ductile-Brittle Transition

Temperature (DBTT) of the fatigue pre-cracked specimens of steel is discovered. The above effect is revealed in the reference temperature T_0 for stressed irradiated specimens being higher than for irradiated unstressed ones.

The effect of the stressed state caused by mechanical load, imitating the pressure of the coolant, on the ductile-brittle transition temperature is qualitatively comparable with the effect of neutron irradiation. From the physical point of view this phenomenon is conditioned by the higher rate of the formation of radiation defects due to the total effects of neutron irradiation and stress.

4.3. CZECH REPUBLIC

Surveillance channels of RPVs in NPP Dukovany are frequently used for research programmes dealing with determination of either detailed and precised parameters of surveillance programmes or special material properties of WWER-440 RPVs.

4.3.1. Surveillance Programme

Czech part of the programme was concentrated on the irradiation of materials from WWER-440/V-213 type RPVs. The following materials were chosen:

- Base metal
- Weld metal
- Heat affected zone

and the following tests were performed in unirradiated as well as in irradiated conditions (up to three years of irradiation which represents approximately design end-of-life fluence):

- Impact charpy tests,
- Static fracture toughness tests,
- Dynamic fracture toughness tests.

4.3.2. Measuring chains

The aim of this programme was concentrated in the following items:

- Precision of neutron dosimetry within the Standard Surveillance Programme, i.e. determination of the effect of self-shielding within the capsules and preparation of a standard procedure for evaluation of neutron fluences on each of specimens (correction factor with respect to container orientation to the reactor core, determination of a correct course of neutron fluence along the whole chain length),
- Characterization of newly designed capsules for Supplementary Surveillance Programme,
- Checking the use and application of two types of melting wire monitors for irradiation temperature determination,
- Determination of changes in mechanical properties (static fracture toughness) of cladding materials (first and second layer as well as heat-affect zone) due to irradiation.

4.4. FINLAND

Finland has taking part in many international programmes most of which have been organized by IAEA (CRP 3 to 5, Reactor Pressure Vessel on irradiation embrittlement, annealing and re-embrittlement of weld 502). Loviisa nuclear power plants (NPPs) surveillance position has been utilized for specimen irradiations. The relatively fast irradiation shift of Loviisa NPP1 NPP weld was realized during the study of the first surveillance chain. This information started long term research programmes on the annealing and re-irradiation behaviour of Loviisa welds. The broken specimen halves from original surveillance programmes were used in the studies. Specimen reconstitution was widely used in these studies. Altogether 15 new irradiation chains have been prepared and tested until now. Most of the chains have been pair of chains.

The core weld of Loviisa Npp 1 was annealed in 1996. During the same outage 15 new irradiation chains were installed into the Loviisa Npp 1 reactor. This research programme is called the surveillance programme for Loviisa Npp annealing. A tailored weld 501 prepared by Izora plants was chosen as the monitoring material. Both ISO CH-V and Charpy-size fracture toughness specimens are used for monitoring the irradiation response. All chains were first irradiated for three years in the reactor, the chains were thereafter annealed at the reactor site and re-inserted into the reactor for re-irradiation. In this programme the following materials condition will be measured both with Charpy-V and fracture toughness tests: In, IA, IAIn, IAIA, IAIAIn. The letter n gives the number of different fluence conditions, which will be created in the programme. The number n varies from 5 to 7. The programme is nearly finished now. One irradiation/re-irradiation chain was removed from the reactor each year.

4.5. IAEA CRPs

The IAEA Technical Group of Lifetime Management of Nuclear Power Plants (TWG LMNPP) together with the Division of Nuclear Power (NENP) organized several CRP dealing with study of irradiation embrittlement of RPV steels. In some of these programmes (CRP-4 and CRP-5) also materials from WWER RPVs were included and tested. Some of these materials were also irradiated in surveillance positions.

The Nuclear Safety Nuclear Installation (NSNI) organized in 1996–2003 a Round Robin Exercise on “WWER-440 Weld Metal Irradiation Embrittlement, l Annealing and Re-Embrittlement” where a characteristic weld 502 was irradiated, partially in surveillance positions, partially in research reactors.

The following countries took part in the programme: Russian Federation, Finland, Norway, France, Belgium, Slovak Republic, and Hungary. The programme was concentrated on irradiation; annealing and re-irradiation of specimens from weld metal — Charpy V-notch standard and subsize specimens for impact testing, pre-cracked Charpy size and subsize bending specimens for static fracture toughness as well as tensile data. Final report is under final approval.

5. EXAMINATION OF RPV SAMPLES FROM OPERATING NPP

There are some old plants that have no surveillance specimens in the reactor pressure vessel to monitor irradiation embrittlement. For these plants the taking of so-called "boat samples" or templates was proposed. Up to now the sampling is used only for uncladed RPV.

This procedure is used widely for the Russian design WWER-440/230 RPV. For these RPVs, besides the lack of surveillance programs, the archive material to perform supplementary evaluation is also not available. The evaluation of the vessel material status in terms of T_k was based on an empirical relationship using the assumed chemical composition. Later, the chemical composition was verified by the analysis of scraps taken from the vessel surface.

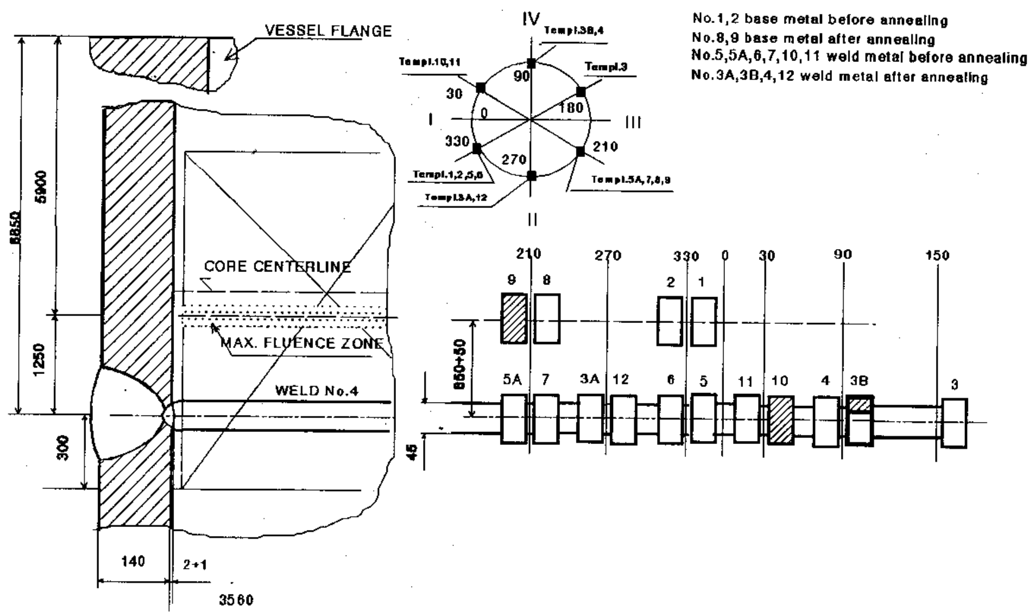
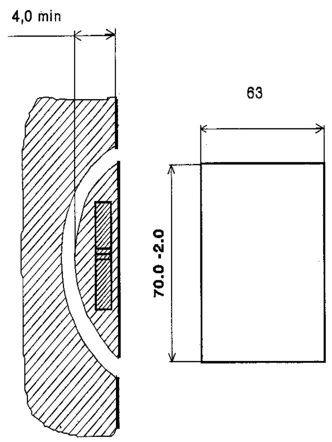


FIG. 5.1. The scheme of boat sampling for Kozloduy-2 reactor pressure vessel.

Since 1991 boat samples were taken from all uncladed WWER (several times for several of them) to verify the material status (the comparison of experimental data to the predicted values determined by the Russian standards), and the effectiveness of the annealing technology applied for the recovery of the RPV. The mechanical testing of sub-size Charpy specimens was used.

The typical scheme of the boat samples cutting is presented in Figure 5.1. [3]. Part of samples were used for the material embrittlement assessment before annealing, some for the annealing effectiveness evaluation. In some cases a few boat samples could be used for additional accelerated irradiation, to provide information about the re-embrittlement rate during reactor operation after annealing.

As a rule the 3x4x27 mm sub-size Charpy specimens are machined from the base metal and 5x5x27 mm from the RPV welds. The shape of sub-size specimens and their notch orientation are presented in Figures 5.2. and 5.3.



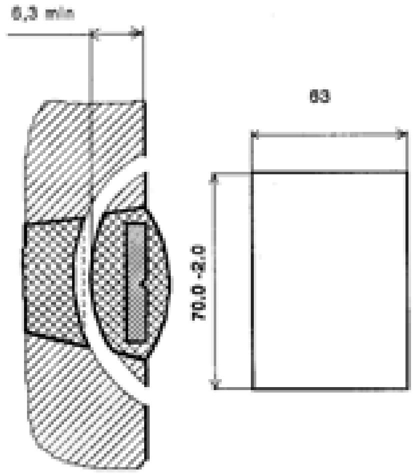


FIG. 5.2. Boat sampling of base metal.

FIG. 5.3. Boat sampling of weld metal.

Correlations between standard and sub-size Charpy specimens has been performed by Russian organizations in the beginning of the 90-ties. They performed experiments devoted to determine those dependencies and their relation to the irradiation. Correlation of transition temperature shifts between standard Charpy V-notch and subsize specimens is given in relation (5.1):

$$
\begin{aligned}
T_{k\,(10x10x55)} &= T_{k\,(3x4x27)} + 65\ \mathrm{K} \\
T_{k\,(10x10x55)} &= T_{k\,(5x5x27,5)} + 50\ \mathrm{K}
\end{aligned}
\qquad (5.1)
$$

However in later experiments an increase in the constant of the correlation (5.1) has been observed, and that results in an increase of T_k. The boat samples can be used also for the leading assessment of RPV material condition. Research is being undertaken in Russia with sub-size specimens be irradiated in a commercial operating WWER with channels for the surveillance specimens.

6. CURRENT NATIONAL PROCEDURES USED FOR IRRADIATION EMBRITLEMENT ASSESSMENT/PREDICTION

6.1. SURVEILLANCE DATA EVALUATION METHODOLOGY

The WWER surveillance programmes have been elaborated on the base of the PNAE-72 (Russian) code. But the PNAE-72 do not provide full methodology for the surveillance evaluation and for life management. Due to this the WWER users have developed their national regulations and practice, which generally is a mix of the PNAE-86 [1] and ASME methodology [4–5]. This practice generally lead to over conservative methods, but sometimes it can result in reduction of the safety, too. Only the PNAE provides material data and toughness trend curves for the WWER materials, so these have to be used or new fracture toughness curve has to be established for PTS and lifetime calculations.

For use of the toughness trend curves the critical transition temperature is obtained from the surveillance programs. PNAE uses the Charpy transition temperature, which is generally conservative as it is based on dynamic testing, which results in transition temperatures 30-60°C higher than for static fracture testing. To reduce this over- conservatism, the PNAE does not correct for the scatter of the test results.

The USNRC [6] methodology is based on the RT_{NDT} (Reference Temperature of Nil-Ductility Transition), which is the nil-ductility transition temperature measured by drop-weight testing. The Drop-weight test is semi-dynamic test, it provides only slightly over conservative transition temperature values. The latest development of the fracture mechanics is the elaboration of the Master Curve. Master Curve method allows determining a fracture toughness curve based on a transition temperature obtained by using small size static fracture toughness specimens.

For WWER life management recently some WWER operating countries (Czech Republic, Slovak Republic, Hungary, Finland, Bulgaria) elaborated a Unified Procedure document VERLIFE [8] within the EU Framework 5th Frame Work programme. VERLIFE – "Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs" also provides a correct evaluation methodology for use of the traditional Charpy testing. These countries consider to follow the VERLIFE in the future. The summary and the comparison of the three methodologies (PNAE, US NRC, VERLIFE) are given below.

6.1.1. Determination of Charpy transition temperature according to PNAE-G002-86

The critical brittle fracture temperature is obtained on the basis of the Charpy (V notched specimen) impact energy, KCV, the fibrous fracture ratio, or the lateral expansion measurements.

Within the Acceptance tests of RPV materials, for determination of critical temperature of brittleness in initial condition, T_{k0}, the temperature range of the tests has to include the following points: $(T_k \pm 10)$°C, $(T_k \pm 20)$°C, $(T_k \pm 30)$°C and $(T_k \pm 40)$°C. Tests are allowed to be performed in the temperature range of $(T_{k1} - 10)°C < T < (T_{k1} + 40)°C$ where T_{k1} is approximate value. At each temperature at least three specimens have to be tested. Impact energy, fibrous fracture ratio and lateral expansion should be determined for each specimen. The results received on three specimens are averaged and plotted as a function of temperature. These mean values are connected by straight lines. The resulting diagrams are used to determine the critical brittle fracture temperature.

The critical brittle fracture temperature, T_k, is the temperature, which meets the following conditions:

- At the temperature T_{k0} the average value of impact energy has not been lower than the values shown in the Table of requirements (Table 6.1);
- The average value of impact energy at a temperature of (Tk + 30)°C has not been lower than the values shown in the Table 6.1; and
- The minimal value of impact energy has not been lower than 70% of the values shown in Table 6.1; and fibrous fracture percentage not lower than 50% of the indicated values minimally.

Table 6.1. Pnae-minimal requirements for the average of impact energy at a temperature T_{k0},°C according to the PNAE-86

Yield point at a temperature of 20 °C, $R_{p0,2}$ [MPa]	Impact energy at T_{ko} (KV) [J]	Impact energy at T_{ko}+30°C (KV),[J]
up to 304	23	37
from 304up to 402	31	47
from 402up to 549	39	60
from 549up to 687	47	71

In case of surveillance specimen testing when usually maximum 12 specimens are available, the impact energy-temperature diagram can be constructed by fitting of a tangent hyperbolic curve:

$$KV = A + B * \tanh\left[\frac{T - T_0}{C}\right], \qquad (6.1)$$

where A = average value of impact energy (KV); B = (KV_{max} - KV_{min})/2; T = temperature, T_0 = temperature relating to the value of KV = B; C = empirical constant.

The values of A, B, C and T_0 have to be determined from the measured values by the least squares fit method. The values of impact energy measured on each tested specimen have to be indicated on the diagram. The PNAE does not require to use minimum values, but a reliable fitting is required to use the artificial lower constraint value for A, which is usually from 0 to5 Joules.

In the case of the initial state material and the irradiated one the required value of impact energy $(KV)_1$ is selected according to the yield point measured at 20°C from the Table 6.1. During this the average value of the measured yield point has to be used, if at least three measurements have been performed, and the maximal value if there are only two measurements. At T_k +30°C temperature the minimum acceptable value is 1.5 $(KV)_1$. Shift of the critical brittle fracture temperature caused by irradiation can be determined with the following formula:

$$\Delta T_F = T_{KF} - T_{ki} \qquad (6.2)$$

where $\Delta T_F \geq 0$; T_{kF} = the critical brittle fracture temperature of the material after irradiation; T_{ki} = the critical brittle fracture temperature of the material at the initial state (before irradiation) determined by tanh fitting. The radiation damage factor can be determined by the following relation:

$$A_F = \Delta T_F (F_n/F_0)^n \qquad (6.3)$$

where F_n = fluence above E>0,5 MeV; $F_0 = 10^{22}$ neutron/m^2; n = 1/3

6.1.2. Measurement of the transition temperature according to US NRC practice

All the current valid regulations determine the transition temperature needed to use the reference curves on the basis of the so-called Charpy impact test. Only the ASME Code is different, which uses the so-called RT_{NDT}. The value of RT_{NDT} used to be determined by drop-weight test. At the same time ASME Code allows the determination by Charpy test – during periodic (surveillance) tests only the later one is used – on the basis of the following formula:

$$RT_{NDT} = T_k - 33\ °C \qquad (6.4)$$

where T_k is the Charpy transition temperature determined according to the US regulations for 68 J criterion.

During the acceptance tests after manufacturing the specimens have to be tested around the expected critical transition temperature. Three specimens have to be examined at each temperature. Results of the specimens examined at the same temperature are averaged and the average values are conjugated with a curve drawn by the eye or a tangent hyperbolic curve. This diagram is used to prepare the factory quality assurance certificate. The transition temperature T_k is assigned by the energy level 68J or lateral expansion of 0.9 mm. Hereafter it is marked as T_{k0}.

The US practice, which is well followed by the practice of the European countries, is described partly in ASME Code, ASTM standards (6.2–6.3), and NRC regulatory guides (6.5–6.7). There is no uniform US regulation, which summarises the whole methodology. In the following text the US methodology is referred as ASME method. In order to determine the increase of the transition temperature due to irradiation, two methods can be used:

- If there is no set of specimens, the change has to be estimated from the chemical composition and the fluence value on the basis of the Regulatory Guide 1.99 (6.5).
- If there is a surveillance test program, at least 12 specimens have to be used.

The specimens have to be distributed equally around the transition temperature. At least three specimens are needed to determine the upper shelf energy. A tangent hyperbolic curve (equation 6.1) has to be fit to the results similarly as it described at the PNAE (the eyeball method is allowed, but it is not used any more) and change in the transition temperature has to be evaluated at the energy level of 41 Joule (ΔTk41J). This value has to be added to the transition temperature value determined on the unirradiated material by the criterion of 68 Joule (T_{ko}) to get the transition temperature of the irradiated material (T_{kirrad}).

6.1.3. Measurement of the critical Charpy transition temperature according to the VERLIFE

VERLIFE uses a method similar to the PNAE for evaluation of the Charpy tests with the exception that it uses the fixed 41J criteria for the evaluation of the transition temperature shift, and considers the scatter of the material properties.

The transition temperature is determined from minimum 12 specimens. Difference of fluence in specimens of one set should be smaller than ±15% of their mean value and difference in irradiation temperatures of individual specimens should be within a ± 10°C and their mean value should not be higher than + 10°C the temperature of the reactor pressure vessel inner wall.

Obtained experimental values of KV are evaluated using the tangent hyperbolic equation according to (6.1) evaluation. Shift of the transition temperature is determined for the criterion KV = 41 J. It is also required that the upper shelf energy after irradiation must not fall below 68 J.

6.2. VALIDITY OF THE CHARPY TRANSITION SHIFT MEASUREMENT AND UPPER SHELF ENERGY

ASME and VERLIFE requires the satisfaction of the following criteria: The procedure can result in valid values of ΔT_F only in cases when upper shelf energy, derived from the formula (6.1), i.e. sum of (A+B) is reaching values larger than 68 J. PNAE-86 does require that at temperature T_k +30°C the minimum mean acceptable value of KV is 1.5 $(KV)_1$. That is even more conservative.

6.2.1. Evaluation of the brittle fracture temperature T_k of RPV materials during operation

Temperature T_k is given by the following relationship:

$$T_k = T_{ko} + \Delta T_F + \Delta T_T + \Delta T_N, \qquad (6.5)$$

where T_{ko} : brittle fracture temperature based on results from Acceptance Tests, if such value is not known, then guaranteed value from technical requirements for a given material is used, [°C],

ΔT_F - Shift of the brittle fracture temperature due to irradiation, [°C],

ΔT_T - Shift of the brittle fracture temperature due to thermal ageing, [°C],

ΔT_N - Shift of the brittle fracture temperature due to cyclic damage, [°C].

T_{ko} determination however uses different criterion in the different codes. In ASME the 68J Criteria is used, in PNAE the criteria have to be selected according to the yield strength like it was mentioned above (see Table 6.1).

VERLIFE is concentrated to the evaluation of operating plants, consequently it accepts the Acceptance Test results. US NRC RG and VERLIFE consider the scatter of the mechanical properties of the forgings and welds to be equal to standard deviation, δT_M. PNAE uses zero scatter value, because T_{ko} is obtained from a dynamic test that are always conservative compared with the static crack initiation. (ASME uses RT_{NDT} that is a less conservative dynamic fracture toughness value and can be obtained as Tk_{68J}-33°C. The Russian code

PNAE compensates this conservatism by neglecting the scatter of testing and the scatter from material in-homogeneity.)

The δT_M (in VERLIFE and US NRC RG) is the standard deviation of T_{ko} determined for the given forgings and welds in the frame of Qualification Tests or in the frame of a set of identical semi-products established during production of the component by the identical technology.

If this value is not available, the following values are suggested for the application.

$$\delta T_M = 10°C \text{ for the base material,} \qquad (6.6)$$
$$\delta T_M = 16°C \text{ for weld metals.}$$

6.2.2. Calculation of the actual transition temperature of component

VERLIFE [8] uses the results from surveillance specimens tests if they were irradiated by at least three different neutron fluences (the difference between the fluences has not be smaller than the value of the lowest fluence) using the relationship:

$$\Delta T_F^{mean} = A_F^{exp\ mean} \text{ x } (F\text{x}10^{-22})^n \qquad (6.7)$$

where F is the fluence of fast neutrons with the energy higher than 0.5 MeV, $A_F^{exp\ mean}$ and n are empirical constants obtained by statistical evaluation .

ASME uses the formulae given in Reg. Guide 1.99, Rev.1, 2 (6.5) and Rev. 3 is under preparation. For RPV integrity evaluation, ASME code uses the corrected reference temperature. It can be calculated with the following formula:

$$ART_{NDT} = \text{initial } RT_{NDT} + \Delta RT_{NDT} + \text{margin} \qquad (6.8)$$

The margin can be generated by the following formula:

$$M = 2\sqrt{\sigma_1^2 + \sigma_\Delta^2} \qquad (6.9)$$

where σ_1 is normal scatter of the transition temperature of the factory state material and σ_Δ is the scatter of the shift. The σ_Δ value is used as 28°F (16°C) for the weld and 17°F (10°C) for the forging. The value of σ_1 has to be determined from scatter of the measured values or from accuracy of the measuring method. If data of two or more surveillance tests are available, the value of σ_Δ can be reduced to its half. So the margin is calculated with following conditions;

$M = 2\sqrt{8^2+8^2} = 22.6$°C for the weld and
$M = 2\sqrt{5^2+5^2} = 14$ °C for the forging.

ASME and US NRC Reg. Guide 1.99 evaluate the ΔT_F by using the Reg. Guide 1.99 formula and consider the scatter margin. PNAE uses the ΔT_F^{mean} curve according to formula (6.7) to evaluate the ageing effects. VERLIFE shifted up vertically the mean trend curve obtained from formula (6.7) with the value of δT_M. If any experimental point is over of this adjusted trend curve, the curve should be shifted further up until the highest experimental value. This upper boundary of the shifts is to be used in assessment of RPV resistance against fast fracture.

PNAE-86 does not provide any requirements to determine the scatter in mechanical properties of the forgings and weld metals, in the scatter of the Charpy shift measurement as well as in the scatter of the fluence evaluation. It considers that the resulted scatter is less than the difference between the Charpy transition temperature and the real transition temperature obtained by static fracture toughness testing.

VERLIFE is more conservative, as it considers the scatter of the mechanical properties of the semi products δT_M. VERLIFE does not allow to extrapolate experimentally determined shifts of the transient temperatures for the fluences higher than 2-multiple of the maximum fluence of the tested surveillance specimens. Figure 6.1 shows the evaluation of surveillance data using following three steps.

Step 1: Draw the surveillance transition temperature mean points.
Step 2: Fit the $\Delta T_F^{mean} = A_F^{exp\ mean} \times (F \times 10^{-22})^n$ *curve.*
Step 3: Shift up the men curve by the margin and check whether all surveillance results are below the curve.

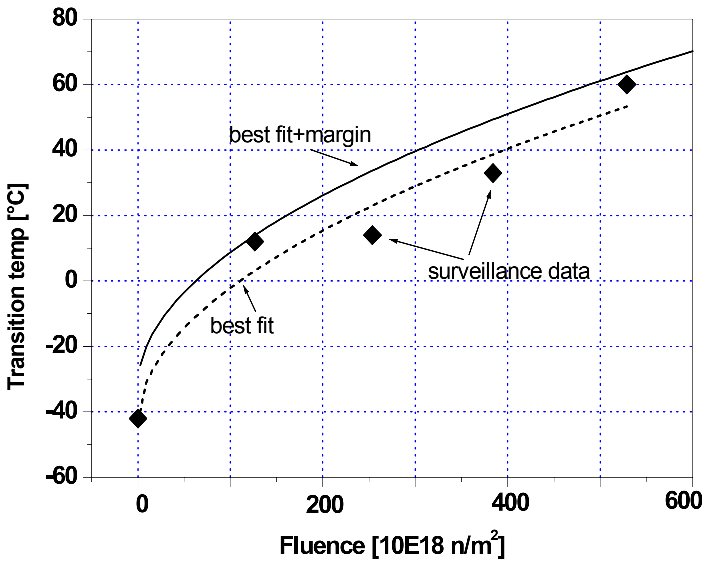


FIG. 6.1. Evaluation of surveillance data.

6.2.3. Evaluation of the aged material properties using direct fracture toughness measurements

VERLIFE and different national regulatory bodies allows the use of direct fracture toughness measurement to determine the properties of the aged materials. Master Curve method or the Prometey method is used in all this countries. The application of Master curve method is summarized as it is follows:

Reference temperature T_0 is determined from static fracture toughness tests using Master Curve approach in accordance with the standard ASTM E 1921-02 with single- or multiple-temperature Then, a chosen lower tolerance bound (usually 5%) should be applied for a determination of fracture toughness temperature dependence for the calculation of the integrity /lifetime. In principle, transition temperature T_0 is usually determined for a fluence required for the RPV integrity assessment, i.e. for end-of-life fluence or for extended life fluence. In these cases, one set of specimens is sufficient for the determination of value T_0 for a required neutron fluence.

In addition, in special cases during RPV design or for prediction of RPV behaviour during future operation, these temperatures could be also evaluated using similar procedure as for brittle fracture temperature. Reference temperature, T_0, as determined in accordance with the standard ASTM E 1921-02 is suggested to adjust a margin to cover the uncertainty in T_0 in case of a few specimens. The standard deviation σ of the estimate of T_0 is given by:

$$\sigma_1 = \beta / N^{0.5}, {}^{o}C \tag{6.10}$$

where N = total number of specimens used to establish the value of T_0, $\beta = + 18^{o}C$.

To consider the scatter within the forgings and weld metals, margin δT_M also should be applied. If this value is not available from experimental data, the application of the following values is suggested

$$\delta T_{M1} = 10°C \text{ for the base material,} \tag{6.11}$$
$$\delta T_{M2} = 16°C \text{ for weld metals.}$$

The sum of the margins is:

$$\sigma=(\sigma_1^2+\delta T_M^2)^{1/2} \tag{6.12}$$

Thus, reference temperature when used in integrity evaluation, RT_0, is defined as:

$$RT_0 = T_0 + \sigma \tag{6.13}$$

6.2.4. PNAE procedure for brittle fracture temperature determination

The similar procedure is used for the radiation embrittlement assessment in Russia, Bulgaria and Ukraine and is based on Russian Code for the embrittlement assessment the brittle fracture temperature T_k. Its value should be evaluated experimentally by Charpy impact testing entirely, as summarized in detail in TECDOC-659 [9]. However, if it is not applicable, empirical relations given next have to be used:

Weld metal:

$$T_k=T_0 + \Delta T_{F'} \tag{6.14}$$

$$\Delta T_F=A_F^{270} (Fx10^{-22})^{1/3} \text{ for } 10^{22}<F<3 x10^{24} \tag{6.15}$$

$$A_F^{270} =800(P+0.07Cu) \tag{6.16}$$

Here T_{k0} is the initial value of brittle fracture temperature from Acceptance Tests and ΔT_F is irradiation induced shift in T_k, A_F^{270} is the irradiation embrittlement factor for irradiation at temperature 270°C, F is the neutron fluence in m^{-2}, E>0.5 MeV and P, Cu, is the concentration in mass %. The irradiation embrittlement factor A_F is irradiation temperature dependent and its value for irradiation at 250°C is equal to:

$$A_F^{250} = 800(P+0.07Cu) + 8 \qquad (6.17)$$

Base metals:

Formulae (6.14-6.16) are used in general to evaluate the brittle fracture temperature of the WWER-440/230 reactor pressure vessels. In some cases these results are being verified by testing of the material samples, templates, cut directly from the vessel wall. Subsized Charpy specimens are used for this purpose. The special procedure has been developed to correlate results of subsized and standard Charpy testing, and obtained data supporting the procedure.

This approach has been applied to date to some of the uncladded vessels where templates were taken from inside. For cladded vessels only scraps for chemical analysis were taken from outer surface; no templates were taken from outside or through the cladding.

When specimen data are to be used in vessel integrity assessment, the orientation of the specimen notch (crack) should be similar to the postulated defect. Otherwise the conservativeness of the specimen orientation should be demonstrated. Results from laboratory tests and from vessel templates indicate that the formula to calculate A_F is an upper limit formula; it is basically conservative. It is however, based on a database which is small as compared to data available today and might not cover all the cases.

There is some uncertainty concerning the temperature profile in the pressure vessel wall during normal operation; the profile is plan specific and depends on the cooling of the reactor cavity. The lower temperature could affect the degree of embrittlement near the outer surface.

7. DESCRIPTION OF THE IAEA DATABASE ON WWER-440 RPV MATERIAL DATA

It has been proposed by the IAEA Technical Working Group on Life Management of NPPs that the IAEA should expand that activity and put forward a proposal for the development of an “International Database on NPP Life Management”. The RPV material database is the first part of it [10–12].

7.1. THE IAEA INTERNATIONAL DATABASE ON RPV MATERIALS

A short description of the main features of the IAEA International Database on Reactor Pressure Vessel Materials is given in this report. The database is being used for the following purpose:

- Licensing authorities in the preparation of improved design curves, reducing or decreasing conservatism but ensuring the maintenance of adequate and reliable safety margins in the field of licensing and safety report analyses.
- Utilities in providing supportive data complementing those gained from surveillance experiments and providing data for remaining life assessment and mitigation activities
- Authorised research organisations to better understand underlying mechanisms and to provide advice in the areas outside the data base

The Database organization includes: IAEA, Database Custodian, Steering Committee and the Database Members. The IAEA organizes the Database according to requests from Member States.

The **Custodian** acts as the agent for the IAEA in operating and maintaining the database and providing an effective interface for Member States/Participating Organizations. The Custodian will also arrange data acquisition from Member States and Database members, and will assist with data evaluation and distribution as appropriate.

- The Steering Committee supervises the data flow, and use.
- The Database Members are the persons or organizations from Member States that provide and are entitled to use the database as well as to receive database information. Each Database Member is responsible for data acquisition, as well as validation and verification and for the strict observation of the database rules.

Present status of the database

Fourteen countries supplied large quantities of surveillance data, enlarging the surveillance section. They include several WWER user countries e.g. the Russian Federation, Ukraine, Finland, Slovakia, Bulgaria and Hungary. Five large IAEA research programmes also joined the database research section:

- IAEA coordinated research programme on “Optimizing of Reactor Pressure Vessel Surveillance Programmes and their Analysis”
- IAEA coordinated research programme on “Assuring Structural Integrity of Reactor Pressure Vessels”
- IAEA Round-Robin Experiment on “WWER-440 weld metal radiation embrittlement, annealing and re-embrittlement”.

- IAEA CRP on “Mechanism of Nickel Effects on Radiation Embrittlement of RPV materials.

The data quantity collected and processed until now is shown in Fig. 7.1. This data collection is one of the largest in the world, and contains the widest range of data covering all type of existing RPV materials, and a large amount of laboratory heats providing wide variations of the chemical compositions.

Participation and data access

Membership of the Database is restricted to States, which are Members of the Agency, and organizations in Member States, recognized by those states, which provide data relevant to the International Database. This is an important statement and a specific feature of this database is that only those organizations, which contributed to the Database, can join it and get all the benefits of being a Database Member.

An intended member of the Surveillance Database shall provide the following information from the NPPs to be included in the database together with the application for membership to the Agency:

- Number of units
- Types of reactors
- Number of irradiated capsules/sets
- Number of irradiated specimens tested
- Type of specimens tested
- Future trend of data provided to the Agency
- Year – number of capsules/sets – number of specimens.

The first set of data shall be provided for inclusion into the database within 6 months after the membership is granted. Every Member State of the Agency, which is a member of the International Database, shall appoint a Liaison Officer. The Liaison Officer is expected to serve as a key person in the country for interaction with the Agency regarding the Database. The Member of the Database is expected to:

- Collect, categorize and prepare reactor pressure vessel material data on a "best efforts basis" as well as data validation at its own expense;
- Contribute advice and recommendations on matters relating to the maintenance, improvement and development of the Database;
- Provide information services to a monitoring contact with, to the extent practicable, users of the database within its territory and representing user's news at meetings of the Database;
- Obtain clearance from the Agency Database members before providing information derived from the International Database to non-members of the Database.

Database Members who have fulfilled all the Database requirements and have supplied data shall have full access to all non-confidential information from the International Database. The Agency may release information from the Database to non-members only after obtaining clearance from the data supplier.

Surveillance and some other data in the Database will be treated as confidential. In the developed database format the Agency suggests to include such data in specially marked (italic printed and shaded) areas in the Database Handbook. The Agency prepared a special coding system to protect this data. A Database member may, at its option, prepares its own coding system for its confidential information. Access to confidential data shall be only through the Agency. Such data can only be released with the express approval of the Member who supplied the data. The Agency shall have the same rights of access to the International Database as a Member. The Agency may use only non-confidential data in the International Database for elaborating guidance and recommendations for developing countries and without releasing raw data.

Important features

- Database can include not only data (raw data are collected) but also store visual information (diagrams and metallographic pictures);
- Software was elaborated and provided to the participants (in MS Access) to make the database user friendly. Data are stored in Access format, but data export in Dbase or several other formats is also available.

The database structure

The International Database on Reactor Pressure Vessel Materials is a research database as it derives from the purpose and tasks of the IAEA. The database consists of 21 files (see Table 7.1). Not all is used for every programme. In the case of a research database there is no need for an on-line data access, as generally there is time enough to distribute the data by post. These conditions allow the use of diskettes (CD ROM), which means that every participant could use the database without special equipment. At the same time the use of CD and diskettes significantly reduces the possibility of unauthorised persons accessing the data.

Visual section

Existing databases on RPV ageing include data mostly in number format only. One of the main purposes of the database is to enhance the understanding of ageing mechanisms and to help the development of effective methods for the timely detection and mitigation of ageing effects. Mathematical analysis of the mechanical testing data on new and service-aged materials may serve new mechanistic models, and may enhance the design curves. This is a very important benefit of the database, but is in itself not enough to understand the ageing mechanism. A new metallography section and previously collected testing curves are included as an extension to the IAEA International Database on Ageing Management and Life Extension of Reactor Pressure Vessel Materials.

Metallography section

There are two possible levels for the metallography section.

- Level one is a simple reference file of the available photomicrographs.
- The enhanced level also includes the collection of the digitised pictures. Several hundred high-resolution photomicrographs can be collected on an easy to use and CDs. Some typical photomicrographs are shown as examples in Fig. 7.1.

Table 7.1. The file structure of the Database

Material identification information

Type of information	File name
Material code, type	RPV_MAT
Manufacturer; utility data	RPV_GEN RPV_UTIL
Technology; welding	RPV-TEC RPV-WEL
Ageing history information	
Irradiation history	RPV_IRR
Thermal ageing	RPV_THR
Mechanical testing	
Tension tests	RPV_TEN
Charpy testing	RPV_CV
Static fracture testing	RPV_SFR
Dynamic fracture testing	RPV_DFR
Hardness testing	RPV_HRD
Evaluated data	
Constants of exponential curve fit on fracture toughness data	RPV-EXP
Charpy transition temperatures and constants of the fit tanh curves	RPV-TT.DBF
References	
References	RPV-REF.DBF
Related documentation	RPV_REL.DBF
Visual data	
Metallography, fractography in TIF format	RPV_MET TIF picture files
Spectra, flux distribution, instrumented impact, tensile, static fracture, J curves, etc. in scanned pictures	RPV_PIC TIF picture files
Spectra, flux distribution, instrumented impact, tensile, static fracture, J curves, etc. in digitalised form	RPV_DIA TIF picture files

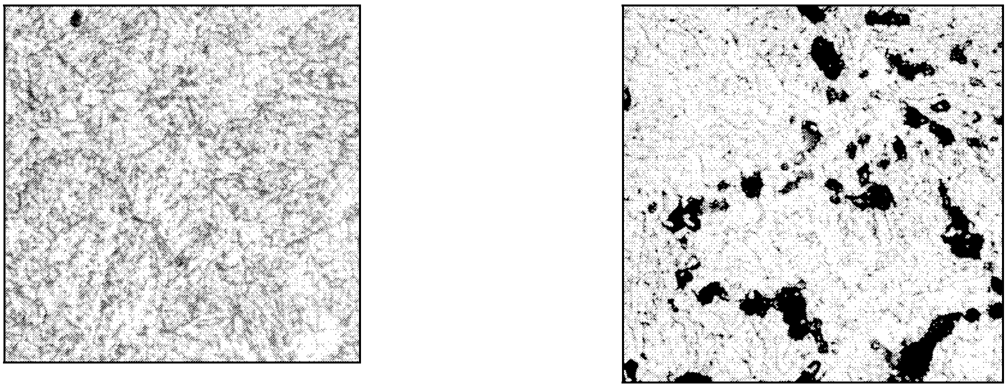
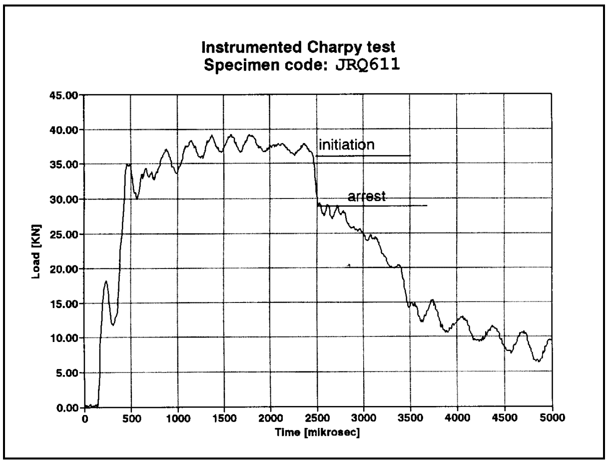


FIG. 7.1. Digitised metallography pictures reprinted. Cr-Mo-V materials.

(a) as produced. *(b) aged at 680C 2000 h.*

Section related to testing diagrams

During mechanical testing of structural materials testing diagrams are obtained. In case of traditional evaluation of the results only some specific points of the curves are used. The measured diagrams, however, include much more information. For example application of the "Local Approach" method needs correct flow diagram, not only tensile data. All of these features justify the storage and acquisition of the original diagrams from mechanical testing. Similarly the neutron spectra cannot be characterised with a single number, but can easily be described by a diagram. All of these examples justify the use of digitised diagrams within a material ageing database. Figure 7.2 shows a part of a digitised data file, and the printed curve.



Time	Load
[µsec]	.[kN]
...	...
550	22,15
552	22,43
554	22,65
556	22,91
...	...

FIG. 7.2. Reprinted instrumented impact testing curve, and a part of the ASCII file.

The coding system

The coding system ensures the anonymity of the data sources as well as the connection among the data files, and facilitates the selection and use of the data. The identification codes include from 1 to 3 parts. Every group of data (data characterizing a certain heat of a material and collected from the same data source e.g. one utility, or laboratory) has a different Mcode. During elaboration a separate Mcode will identify the data groups belonging together.

IAEA_MCode:(9 characters), i.e. the material identification code is given randomly by the IAEA as shown on Table 7.2. This code will identify all of the processed data to be supplied to the participants. This code is built up from the following characters (digits):

Table 7.2. Identification of material (Mcode)

FOR	Forging
WEL	Weld metal
HAZ	Heat affected zone
PLA	Plate
LAB	Laboratory heat
LBW	Laboratory welding
CLA	Cladding
CAS	Casting
REF	Reference material
OTH	Other

Characters 1 to 6 = reference identification number given randomly by IAEA
Characters 7 to 9 = material type

The following files include general information connected with material production and the data sources: RPV_MAT; RPV-GEN; RPV_TEC; RPV_WEL. Multiple information may belong to the same heat of materials. In this case a second indentifier the IAEA_NUM is used for the correct characterization. IAEA_NUM is a three digit random or artificial number, uniquely identifying the data or specimens belonging to the same group. Files in the RPV-CHEM; RPV_REF and RPV_REL fields are identified by MCode and IAEA_NUM.

IAEA_ACODE (3 digits)

In case of aged material the **IAEA-ACode** characterises the type of the ageing. See Table 7.3.

Table 7.3. Identification of aged of material (Acode)

IAEA_Acode	type of ageing
NUL	zero level testing
IRR	irradiation in research reactor
IRA	irradiation in research reactor and annealing
IPR	irradiation in power reactor (surveillance)
APR	thermal aged in power reactor
IPA	irradiation in power reactor and annealing
IAR	irradiation and ageing in research reactor
IAP	irradiation and ageing in power reactor
TMP	template cut from power reactor wall
CAV	irradiation in power reactor cavity
THA	thermal aged in laboratory
HFA	high cycle fatigue
LFA	Low cycle fatigue

Remark: further ageing codes can be included. Information characterising a group of similarly aged material identified by MCode and ACode: RPV_IRR; RPV_THR; RPV_EXP; RPV_TT; RPV_MET; RPV_DIA; RPV_PIC. Finally the specimen data are characterised by MCode+ACode+NUM. By this way, every individual data can be separately stored and identified. The following data files are using all three identifiers: RPV_CV; RPV_SFR; RPV_DFR; RPV_TEN; RPV_HRD.

Data supply requirements

The data supplied by the participants are required to be consistent with the database requirements.

Supplied data and data source must be correctly identified.

- The data should preferably be in SI units, but they are also accepted in national units.
- Missing data can be supplied later as modification.
- Correct description of specimen orientation and location is essential.
- Terminology should be used according to the ASTM Standard terminology and EPRI ageing terminology.

The data can be supplied in computer files -Dbase; Quattro, MS Excel, MS Access, Paradox, Lotus, and ASCII text or WinWord doc files for standard IBM PC. Data can be provided in electronic format. Data provided in typed datasheets or via Internet are also welcome. Preferably supply of a copy of the original report or summary of the reports, which will be used to check the data and stored at IAEA, is recommended.

8. IRRADIATION EMBRITTLEMENT MODELING

Relatively large Charpy-V surveillance data set for WWER-440 pressure vessel materials was collected in the CRP but only limited number of fracture toughness data. Hence trend curve fitting in this report is based on ISO Charpy-V data only. The weld data consists of 34 low flux and 87 high flux data points, i.e. of 121 weld data points altogether, and the base metal data of 24 low flux and 76 high flux data points, i.e. of 100 base metal data points altogether.

8.1. PRINCIPLES OF THE TREND CURVE DERIVATION

8.1.1. Functional description of embrittlement

Basic physics research has not resulted in justification of a unique physically based functional forms. The number of candidate functions, which fulfil the requirements given above, is naturally large. In the statistical fitting several combinations of commonly issued functions are applied in order to have a selection for choice.

The following three basic functions are used in fitting:

The power law function $\Delta T = a * F^{n}$ (8.1)

The exponential function $\Delta T = a * (1 - e^{-n*\Phi})$ (8.2)

The tanh function

$$\Delta T = c1 * [1/2 + 1/2 * \tanh(\frac{\Phi - \Phi_0}{c2})] \quad (8.3)$$

where ΔT is the transition temperature shift, F is the neutron fluence and a, n, c_1, c_2 and F_0 are coefficients. The acceptable values of *n* in the power law function (1) are 0 < n < 1.

The power law function never saturates fully and its derivative at zero fluence is infinite for the realistic values of n (0< n < 1). The derivate of the exponential function at zero fluence is "a∗n" and it saturates fully to the value of "a" at high fluences. Hence the exponential function suits well for the characterisation of early development of embrittlement.

The embrittlement description may include also threshold values for fluence and impurity contents. These threshold values can be connected to real physical phenomena. Threshold fluence can describe the hnucleation fluence, which is needed before a physical response can be observed. Threshold on chemistry may be linked to the amount of the chemical element permanently bound to some structures or to solubility of the element in the matrix at the operation temperature of the material.

A realistic model can include only positive threshold values, i.e. at zero value of the element its response to embrittlement shall be zero. The same holds for fluence. Negative threshold values may be real parameters in special situations, if for instance there is a systematic bias in chemistry analyses or fluence estimation. However, there is no indication on biases and hence negative threshold values will not be accepted.

Threshold values have been added to the trend curve functions by using a unit step function g(x), which is described as follows

$$g(x) = 0, \text{ when } x < 0 \tag{8.4}$$

g(x) = 1, when x > 0

Threshold fluence and phosphorus descriptions for the power law function can be formulated as follows:

$$\Delta T = a * (F - \Phi_0)^n * g(\Phi - \Phi_0) \tag{8.5}$$

$$\Delta T = a * (P - P_0) * g(P - P_0) * \Phi^n \tag{8.6}$$

where F_0 is the threshold fluence value and P_0 the threshold phosphorus value. The same formulation can be made for the copper content and for the exponential functions.

8.1.2. Determination of the fitting parameters

The fitting parameters are determined by multi parameter regression or by other means. The fittings are made to full or sub-data sets as described in the reference. The number of candidate functions is large and hence it is possible in principle that the absolute minimum of standard deviation has not been found. On the other hand it is important to keep a physical insight on the problem and not to try to find the last but insignificant improvements. The choice of the representative trend curve function is not always straightforward, as several candidate functions will work equally well. In addition the minimum may be shallow, which does not give a firm bases for setting the parameter values. In the choice of the function also practical considerations shall be taken into account (simple form and easy to use).

An acceptable fit defines the fitting parameters with reasonable accuracy. However, those parameters, which are only poorly supported by the data, remain uncertain. For instance phenomena, which occur at low neutron fluence values, can not be identified, if the data includes only few low fluence data points. The same situation holds for responses to chemistry values, which do not occur in the data.

Two input parameters may also be strongly correlated. Any of the two correlated parameters can explain the behaviour of the material and their relative contributions may be difficult to separate from each other. Standard deviation is the measure of the goodness of the fit and it shall be minimised. However, it is important to explain the main behaviour, not the last details. For instance, threshold values in chemistry or neutron fluence obtained from the applied fitting shall not necessarily reflect real physical phenomena.

Overview of the data:

The data are characterised by the phosphorus and copper contents, neutron fluence and neutron flux. The distribution of these parameters in the database may limit the information, which can be derived from the data. The measured data are also compared to the prediction function given in the Russian Code PNAE [1]

$$\Delta T = 800 * (P + 0.07 * Cu) * \Phi^{1/3} \tag{8.7}$$

where neutron fluence is given in units of [10^{22} n/m^2, E > 0.5 MeV] as it will be used in the whole document.

The distribution of weld data in the copper-phosphorus plane is shown in Figure 8.1. Low and high flux data points are identified in the figure. The measured transition temperature shifts of weld data are shown as a function of neutron fluence in Figure 8.2.

The measured weld data compared to the prediction by formula (8.7) are shown in Figure 8.3. The formula (8.7) gives a relatively good description of the data but at high shifts the formula predicts on average too high shift values, which can be related to different approach for transition temperature shifts as it was described in PNAE (47 J and 71 J) and in the analysed data set (41 J and 47 J).

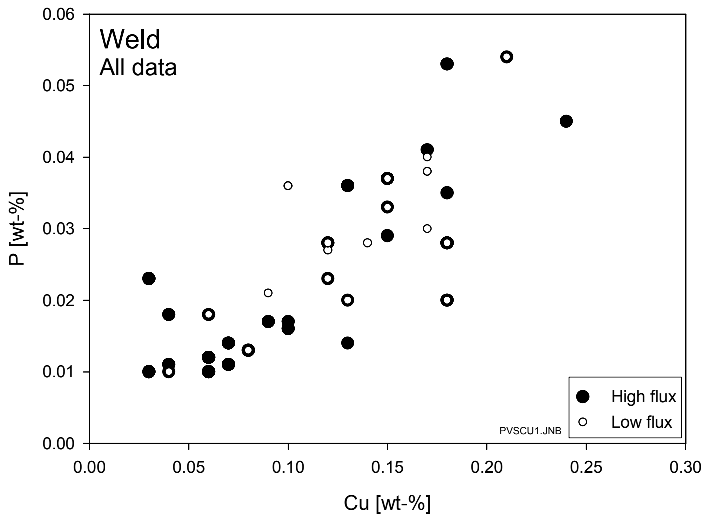


FIG. 8.1. Distribution of weld data in the copper-phosphorus plane. Low flux marks are located on the top of high flux marks, i.e. open circles in bold include both flux value.

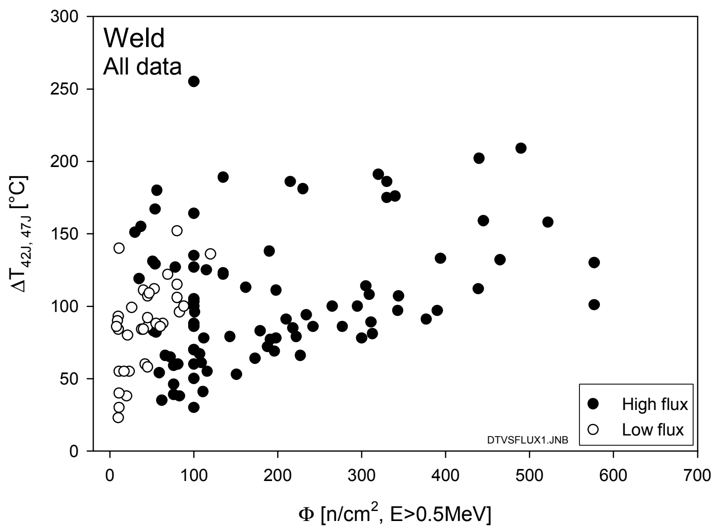


FIG. 8.2. The measured transition temperature shifts as a function of neutron fluence.

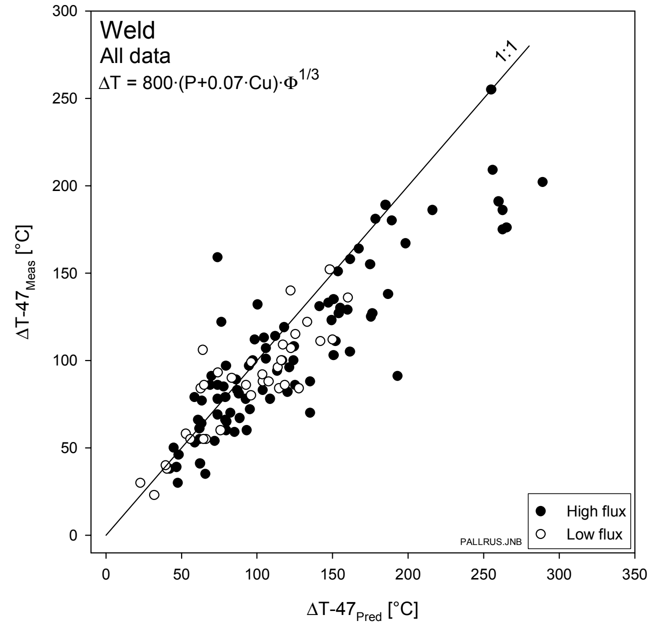


FIG. 8.3. The measured shift versus the shift calculated from the formula (8.7), weld data.

The distribution of base metal data in the copper-phosphorus plane is shown in Figure 8.4, where low and high flux data points are indicated. The copper and phosphorus contents are concentrated approximately around P= 0.013 mass% and Cu = 0.10 mass%. The measured transition temperature shifts of base metal are shown in Figure 8.5 as a function of neutron fluence. The shifts correlate well with neutron fluence.

The measured shifts of base metal are compared to the prediction (8.7) used for weld metal in Figure 8.6. For most of the points the weld prediction formula (8.7) represents an upper boundary for base metal but there are points in the range of ΔT_{Pred} = 80°C to 140°C, which exceed the prediction for weld.

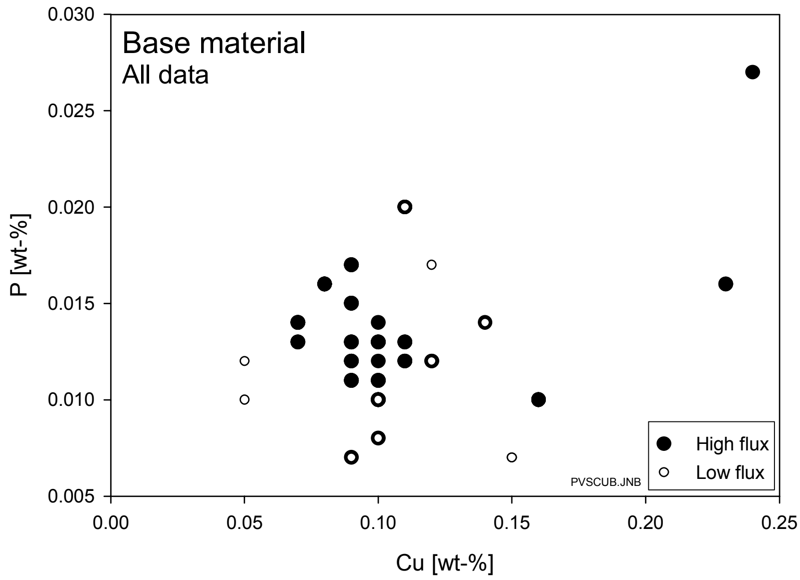


FIG. 8.4. Distribution of base metal data in the copper-phosphorus plane. Low flux marks are located on the top of high flux marks, i.e. open circles in bold include both flux values.

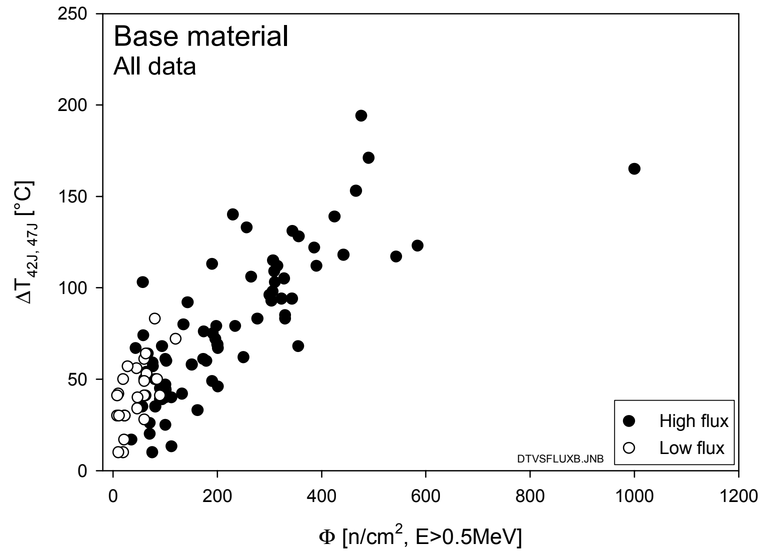


FIG. 8.5. The measured transition temperature shifts of base metal as a function of neutron fluence.

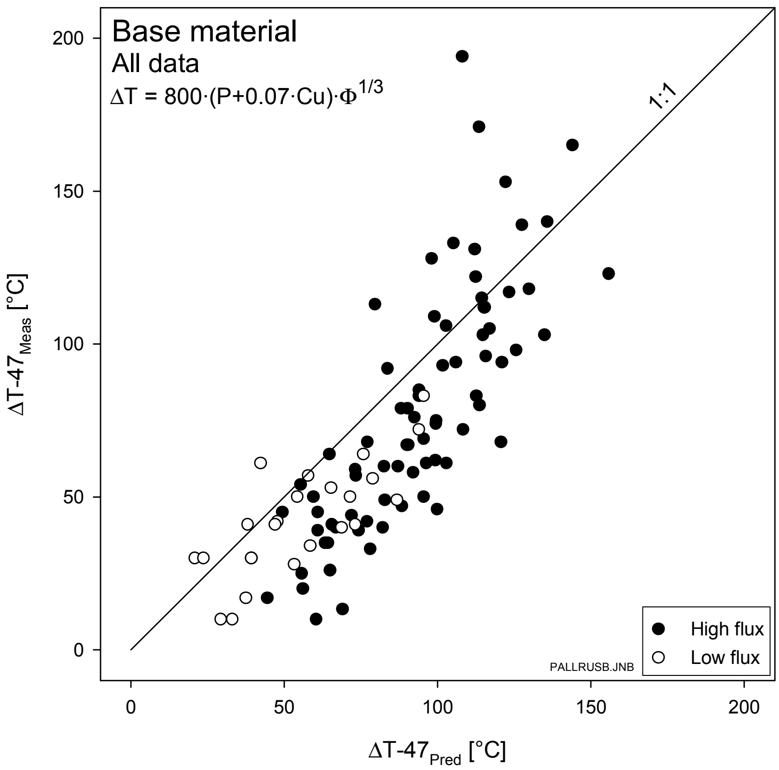


FIG. 8.6. The measured base metal shifts versus the shifts calculated from the formula (8.7) for weld metal.

8.1.3. Trend curve fitting

The following five types of trial functions have been used in fitting:

$$\text{Fit-1: } \Delta T = (a1 * P + a2 * Cu) * \Phi^{n} \quad (8.8)$$
$$\text{Fit-2: } \Delta T = (a1 * P + a2 * Cu) * \Phi^{n} + a3 * \Phi^{n3} \quad (8.9)$$
$$\text{Fit-3: } \Delta T = a1 * P * \Phi^{n1} + a2 * Cu * \Phi^{n2} + a3 * \Phi^{n3} \quad (8.10)$$
$$\text{Fit-4: } \Delta T = a1 * P * (1 - e^{-n1*\Phi}) + a2 * Cu * (1 - e^{-n2*\Phi}) + a3 * \Phi^{n3} \quad (8.11)$$
$$\text{Fit-5: } \Delta T = a * F^{n} + b1 * \left[1 - e(^{-F/F_{sat}})\right] + c1 * [1/2 + 1/2 * \tanh(\frac{F - F_{start}}{c2})] \quad (8.12)$$

- Fit-1 has the same form as the Russian Code formula for weld metal.
- Fit-2 is a function, where a matrix damage term has been added to Fit-1.
- Fit-3 has separate power law exponents for the phosphorus and the copper terms in addition to the matrix damage term.
- Fit-4 describes the phosphorus and the copper terms with exponential function for both terms having own exponent values. In addition, it contains the matrix damage term.
- Fit-5 describes the phosphorus term with tanh function.

The schematic description of individual terms is given in Figure 8.7.

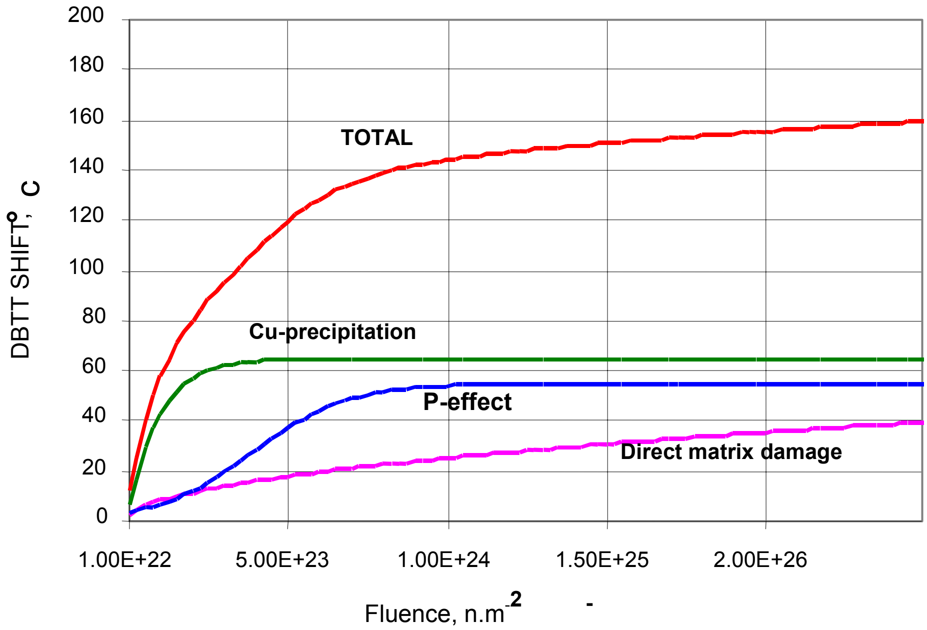


FIG. 8.7. Schematic description of individual terms in the fitting formulae.

The fits have been made to all data, low flux data and high flux data separately. The fits made to all data are presented in the report. In principle the matrix damage term can be estimated from low impurity content data. In practice the matrix damage description was dependent on the detailed definition of the low impurity sub data sets and was determined in many cases with multiparameter regression together with other terms.

8.1.4. The derived trend curves for Weld metals

The exact functional forms of the fitting functions are given below together with the best fit functions made to all data.

Fit-1

$$\Delta T = [a1*(P-P_0)*g(P-P_0)+a2*(Cu-Cu_0)*g(Cu-Cu_0)]*(\Phi-\Phi_0)^n * g(\Phi-\Phi_0) \tag{8.13}$$

***best fit to all weld data*:**

$$\Delta T = [884*P+51.3*Cu]*\Phi^{0.29} = 800*(1.11*P+0.064*Cu)*\Phi^{0.29}, \tag{8.14}$$

SD=22.6 ^oC, where SD is a standard deviation

All parameters are well defined.

Fit-2

$$\Delta T = \begin{bmatrix} a1*(P-P_0)*g(P-P_0)*(\Phi-\Phi_1)^n * g(\Phi-\Phi_1)+a2*(Cu-Cu_0)*g(Cu-Cu_0)*\Phi^n + \\ a3*\Phi^{n3} \end{bmatrix} \tag{8.15}$$

***Best fit to all weld data*:**

$$\Delta T = [1949*(P-0.015)*g(P-0.015)+132*Cu]*\Phi^{0.09} + 4.32*\Phi^{0.51}, \tag{8.16}$$

SD=17.9 ^oC

All parameters are well defined.

Fit-3

$$\Delta T = \begin{bmatrix} a1*(P-P_0)*g(P-P_0)*(\Phi-\Phi_1)^{n1} * g(\Phi-\Phi_1)+a2*(Cu-Cu_0)*g(Cu-Cu_0)*(\Phi-\Phi_0)^{n2} + \\ a3*(\Phi-\Phi_0)^{n3} * g(\Phi-\Phi_0) \end{bmatrix} \tag{8.17}$$

***Best fit to all weld data*:**

$$\Delta T = 1641*(P-0.015)*g(P-0.015)*\Phi^{0.13} + 220*(Cu-0.05)*g(Cu-0.05)+6.80*\Phi^{0.45}$$

SD=17.9 ^oC (8.18)

No well defined value could be found for parameter n^2. The candidate parameter values varied around zero and hence n^2 was fixed to zero. Other parameters could be defined well.

Fit-4

$$\Delta T = a1*(P-P_0)*g(P-P_0)*(1-e^{-n1*(\Phi-\Phi_0)})*g(\Phi-\Phi_0)+a2*Cu*(1-e^{-n2*\Phi})+a3*\Phi^{n3} \tag{8.19}$$

***Best fit to all weld data*:**

$$\Delta T = 3054*(P-0.015)*g(P-0.015)*(1-e^{-(\Phi-0.015)})*g(\Phi-0.015)+226*Cu*(1-e^{-0.03*\Phi})+4.03*\Phi^{0.53}$$

SD=18.1 ^oC (8.20)

In fit-4 the parameters a1, a2, a3 and n3 are well defined but the parameters n^1 and n^2 cannot be defined unambiguously from the data. Fit-4 can described the initial slope of the trend curve accurately and hence the slope was studied in more detail with this function. The fluence value corresponding to half of the total response of the exponential function is

$$\Phi_{1/2} = \frac{\ln 2}{n} \qquad (8.21)$$

The value of the exponent n1 of the phosphorus term can vary from 10 000 to 1 without any change in SD or in the other parameters. SD starts to grow slightly only below the value n1=0.1. These limits correspond to $\Phi_{1/2}$ values from $7*10^{-7}$ to 7. This means that a weak upper limit for the phosphorus response is $\Phi_{1/2} \leq 7$ n/cm^2, E> 0.5 MeV but no lower limit can be determined from the data. The lowest fluence value in the data is 8 and hence it is clear the embrittlement behaviour at lower fluence values cannot be described by the data. The value of exponent n1 is low but its real value cannot be determined from current data.

The value of the exponent n^2 of the copper term can vary from 0.02–10 000 without any change in SD or in the other parameters. These limits correspond to $\Phi_{1/2}$ values from 0.0007 to 35. There is a very shallow minimum around n2 ~ 0.02-0.04, which corresponds to $\Phi_{1/2}$ - values of 20–30. There are few data points in this fluence range but the embrittling response of copper is so low that practically no response is seen in the data analyses. The conclusion is that the fluence exponent of copper cannot be determined with the current data.

As concerns the initial slope of phosphorus and copper responses the differences between fit-4 and fit-2 and fit-3 are not essential, because the initial slope of the power law function is infinite, which corresponds to n1 and n2 values of zero.

Fit-5

$$\Delta T = a * F^n + b1 * \left[1 - e^{-F/F_{sat}}\right] + c1 * [1/2 + 1/2 * \tanh(\frac{F - F_{start}}{c2})] \qquad (8.22)$$

Best fit for all weld data:

$\Delta T_k = 3.7 \cdot F^{0.56} + (273.5\ Cu - 5) \cdot g(Cu - 0.04).\ (1 - e^{-0.12F}) + (2732\ P - 41.9) \cdot g\ (P - 0.015).$
$(0.5 + 0.5 \cdot \tanh(F - 15/10))\ g(F - 15)$ (8.23)
SD = 18.7 °C

8.1.5. The derived trend curves for Base metals

Fit-2

$$\Delta T = \begin{bmatrix} a1 * (P - P_0) * g(P - P_0) * (\Phi - \Phi_1)^{n1} * g(\Phi - \Phi_1) + a2 * (Cu - Cu_0) * g(Cu - Cu_0) * \Phi^{n1} + \\ a3 * \Phi^{n3} \end{bmatrix}$$

(8.24)

***Best fit to all base metal data*:**

$$\Delta T = [1851 * (P - 0.015) * g(P - 0.015) + 105 * Cu] * \Phi^{0.15} + 0.93 * \Phi^{0.76}, \qquad (8.25)$$

SD=18.8°C

All the parameters are relatively well defined except the parameter n^1, which refers to a shallow minimum. The parameters are sensitive to small changes in the values of P_0 and Cu_0, which suggest a valley like geometry for the minimum of the standard deviation (standard deviation is practically constant on some loci of parameters).

Fit-5

Fit for all base metal data:

$\Delta T = 8.37 \cdot F0.43$ (8.26)
SD = 21.7°C

Copper and phosphorus contribution in the formula were set to zero due to low contents in base metals.

8.1.6. Comparison of measured and predicted data:

Russian Code formula (8.7) was modified using the whole data set collected within this CRP and represented as formula (8.8) in Fit-1. For more physical application, formulae (8.9) to (8.12) were derived and the discussion were done in section 8.1.4 for Fit-1 to Fit-5. The comparison of the measured and predicted data for Fit-1 for all weld data is shown in Figure 8.8.

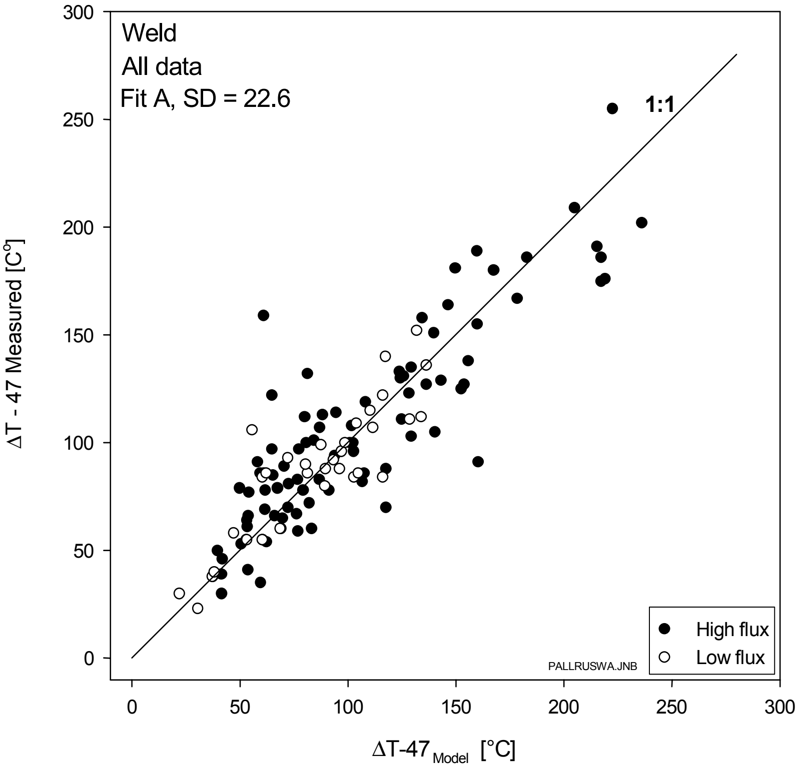


FIG. 8.8. The measured versus predicted shifts made to all weld data, Fit-1.

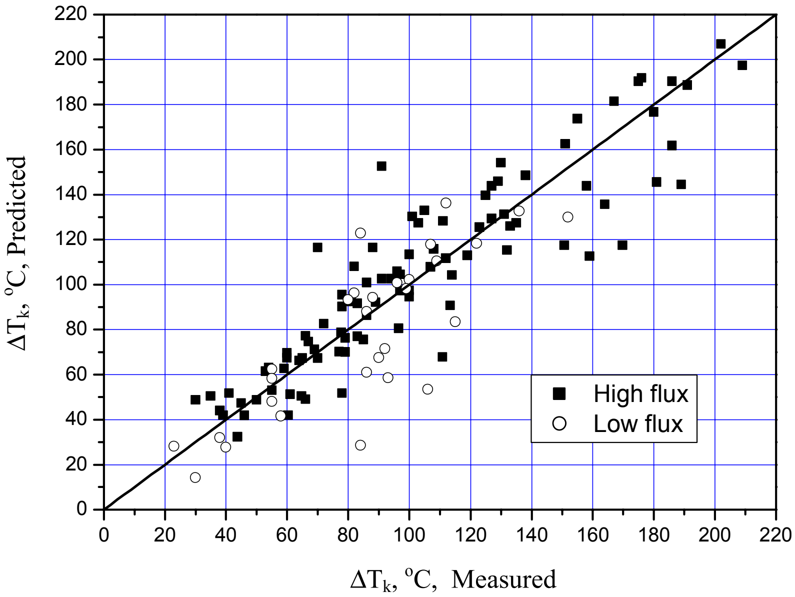


FIG. 8.9. The comparison of all experimental data with the prediction by Fit-5.

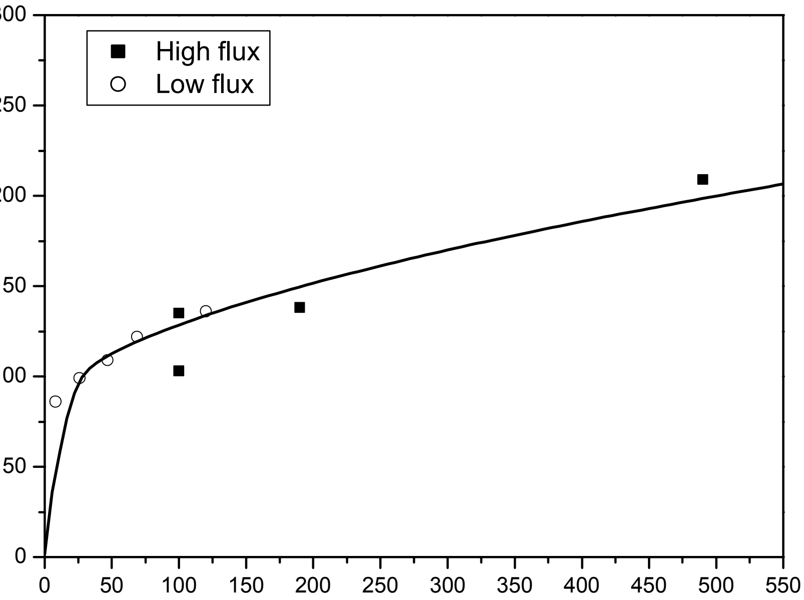


FIG. 8.10. An example of fitting by Fit-5.

8.2. SUMMARY OF THE FITS

The following conclusions can be made as results of fitting the candidate functions:

- In weld metal matrix damage term is needed. The addition of the matrix term reduces the standard deviation of the fit by approximately 5°C as compared to Fit-1, which has the functional form given in the Russian Code.
- The use of phosphorus threshold level reduces the scatter of all fits approximately by 0.5°C. The value of the derived threshold is P_0 = 0.015 mass-% for weld and base metals and it is clearly the property of the analysed data set.
- The data can be described equally well with different types of functions. Standard deviation of all best fits made to the whole data set is about 18°C.
- The early development of copper and phosphorus response as a function of neutron fluence is fast but it cannot be determined quantitatively from the data.

Fits were made also to low flux and high flux sub data sets separately and the analyses is described in the reference documents. Standard deviation of all data fits is typically 18°C. Fits made to low flux data sub sets are typically 12°C. Because the number of low flux data points is relatively low, the derived functions are not so well defined as the all data fits. The relative contributions of the phosphorus, copper and matrix damage terms in the collected data set are visualised in Figure 8.11 for weld data and in Figure 8.12. for base metal data.

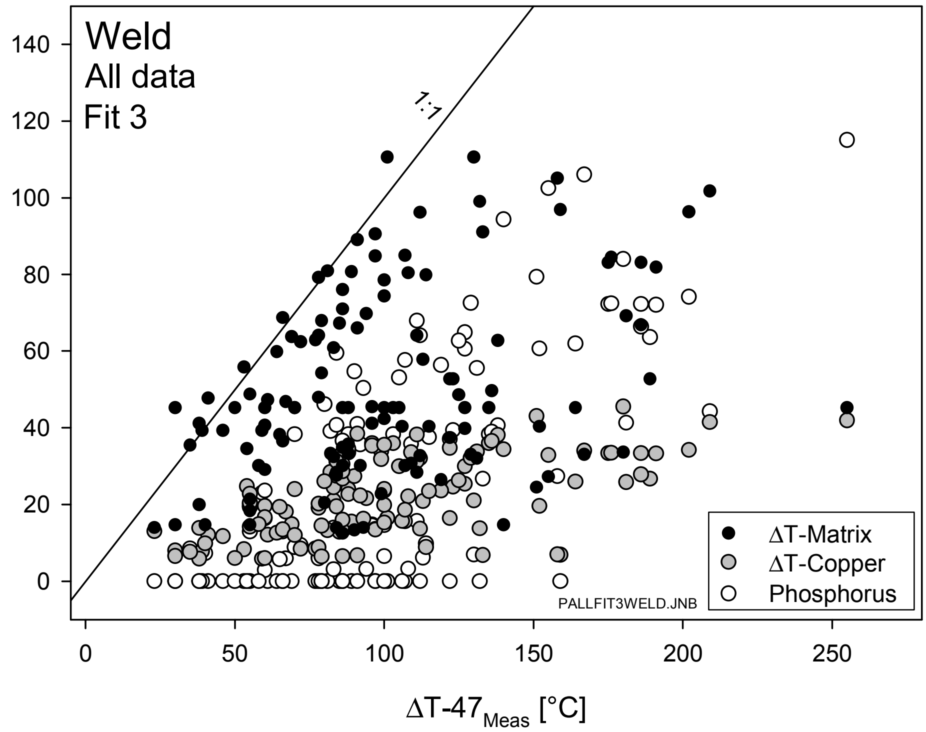


FIG. 8.11. The relative damage contributions from different embrittlement mechanisms in the whole weld data set.

The mechanisms are related to phosphorus and copper impurity levels and to pure fluence, i.e. matrix damage. The 1:1 shows the measured shift. The measured total shift is given on the x-axes and shifts due to different mechanisms are given on the y-axes. The relative shifts are data set specific.

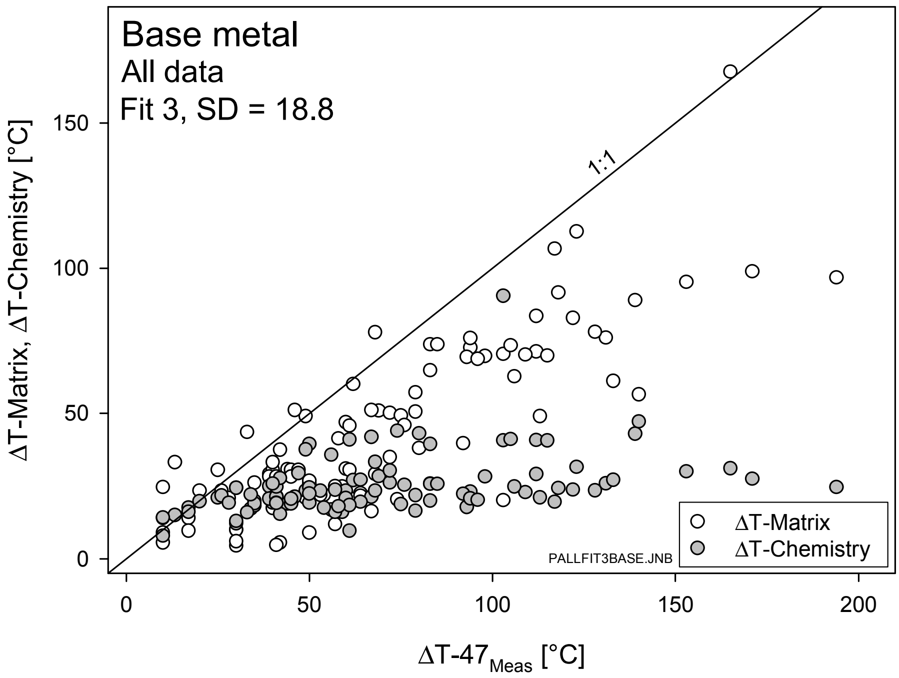


FIG. 8.12. The relative damage contributions from different embrittlement mechanisms in the whole base metal data set.

The mechanisms are related to chemistry (phosphorus and copper impurity levels) and to pure fluence, i.e. matrix damage. The 1:1 shows the measured shift. The measured total shift is given on the x-axes and the shifts due to different mechanisms on the y-axes. Matrix damage gives the main contribution to large shifts (high fluence, low impurity data). Small shifts are largely explained by chemistry. The relative shifts are data set specific.

8.2.1. Choice of the trend curve functions

All the analysed candidate functions, which include the matrix damage term, describe the data equally well. However, all the best-fit functions contain the phosphorus threshold term P_o= 0.015 mass-%. This value is considered rather high and it has not been validated by independent methods. Hence for practical purposes a Russian Code type function in revised form is proposed to be used. The range of variation of the input parameters in the collected data set is large. Hence if local subsets of data with limited variation of the some input parameters are available, they may give a better description for the local data. The proposed formula is:

8.3. THE PROPOSED FORMULAE FOR ENGINEERING USE

As results of comparison and fitting, the following formulae are proposed for engineering use.

Table 8.1 Proposed formulate for engineering use.

Metal	Formula	SD	Equation No.
Weld metal*	$\Delta T = [884 * P + 51.3 * Cu] * \Phi^{0.29}$ $= 800 * (1.11 * P + 0.064 * Cu) * \Phi^{0.29}$	SD=22.6 °C	(8.14)
Base metal*	ΔT = 8.37* F0.43	SD = 21.7 °C	(8.25)

* valid for neutron fluences in the range, $10^{22} < F < 4 \times 10^{24}$ m^{-2}

8.4. DIRECT ESTIMATION OF FRACTURE TOUGHNESS USING “MASTER CURVE” APPROACH

The analysed dataset included 8 WWER units equipped with surveillance COD type specimens and test results within the standard surveillance programme, and most of the extended surveillance programmes are based on fracture mechanical specimens. These specimens were not used to evaluate the residual lifetime until the most recent time. The future is to use the so-called “Master Curve”. (ASTM E 1921-02) describes the method. The basis of the “Master Curve” is that instead of different K_{JC} reference curves a uniform curve is established for all ferritic steels (steels of RPVs). The evaluation of the transition temperature is based on fracture mechanics measurements. The Master Curve generally gives higher transition temperature shift than those measured by Charpy impact specimens, but the initial transition temperature T_o is generally lower than the transition temperature value T_{k0} measured by Charpy impact specimens.

Several programmes validated the use of the Master Curve method. In 1996–1999 IAEA organised a Coordinated Research Project on “Assuring Structural Integrity of Reactor Pressure Vessels” to verify the measuring technology. The second phase of the project on” Optimising Reactor Pressure Vessel Surveillance Programmes and Their Analyses” was between 2000–2002, and about 20 laboratories participated in it. The data show that the measurement could be performed adequately by all laboratories, the scatter of the results does not exceed the scatter of the Charpy transition temperature. It means that the measuring method can be used in practice. Further application is going on under the organisation of US NRC, and in the EU 5^{th} and 6^{th} FP .

The current version of Master Curve (ASTM E 1921-02) includes multi-temperature analysis of the data is expected to be useful for WWER-440 type reactors, where fracture mechanics specimens are available in the surveillance programme.

Requirements of the standard ASTM 1921-02 need to be fulfilled for the Master Curve measurement. Most of the surveillance specimens were manufactured about 20 years before establishment of the standard. Due to the non-standard manufacturing processes of the specimens, and testing before the Master Curve standard elaboration the deviation range is slightly larger; the results can still be used after a proper analysis and evaluation. Some COD specimens in the standard surveillance program are in the capsules located in a steep flux gradient position. The flux and fluence values of each specimen have to be carefully calculated and considered in the analysis. Reconstitution of Charpy impact specimen halves is a useful and effective method for determination of valid fracture toughness data for sets fulfilling requirements for homogenous neutron fluence.

There are two accepted methods to determine the Master Curve on the basis of calculations made

- from measurements performed at the same temperatures (single temperature method)
- from measurements performed at different temperatures (multi-temperature method).

To demonstrate the use of the Master Curve method, the following analyses have been performed on surveillance data collected from 4 similar units in as received and EOL irradiated conditions. The results are shown on Fig 8.13 and on Fig 8.14. The calculated shift is a slightly larger than the shift obtained by Charpy impact testing on the same surveillance sets, but the irradiated T_0 value is acceptable and realistic and it can be used for RPV lifetime assessment.

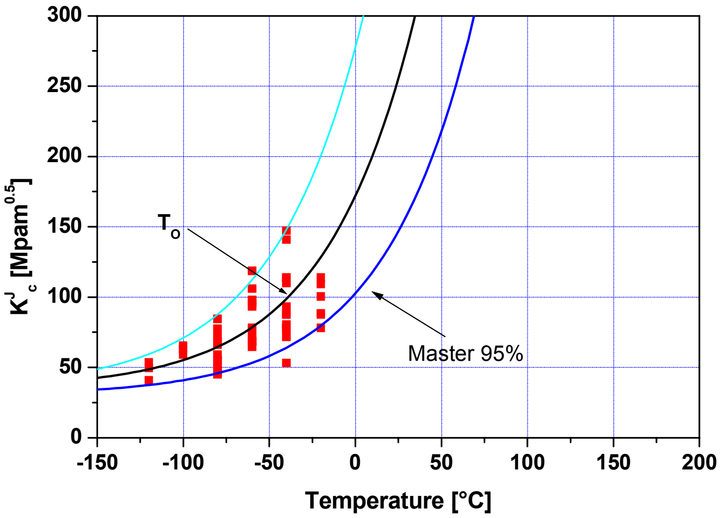


FIG. 8.13. Master Curve T_0 evaluation on as received weldments of four WWER units.

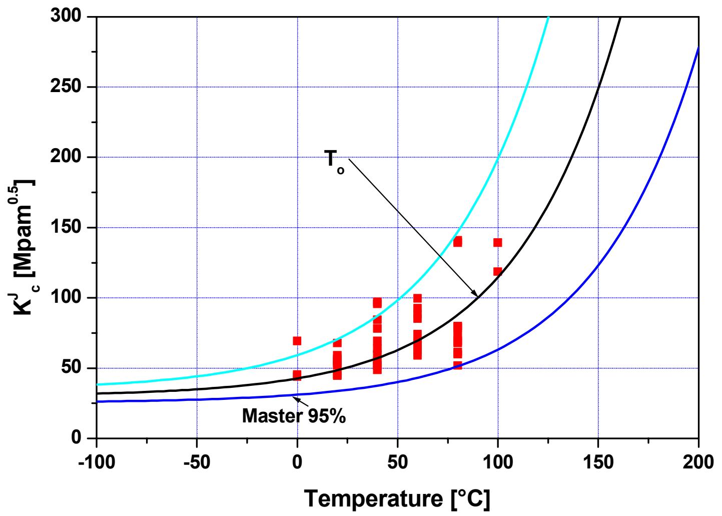


FIG. 8.14. Master Curve evaluation of the surveillance results of four WWER units weldments after irradiation corresponding to 48 years of operation.

9. GUIDELINES FOR PREDICTION OF RADIATION EMBRITTLEMENT OF OPERATING WWER-440 RPVS

This Guideline should be used for assessment of irradiation embrittlement of RPV ferritic materials as a result of degradation during operation. Both approaches, i.e. transition temperatures based on Charpy impact notch toughness as well as based on static fracture toughness tests are to be used in RPV integrity evaluation.

Integrity and lifetime assessment of reactor components are based, in principle, on application of fracture mechanics approach, thus determination of fracture toughness changes is of the main concern. Thus, two ways can be applied, depending on the number and type of specimens either in surveillance specimens or in the material qualification programme and other experimental/material validation programmes:

- direct determination of fracture toughness of RPV materials is required at certain periods during RPV operation, i.e. with given level of degradation — in this case, fracture toughness testing is performed and its temperature dependence is determined directly using either single- or multi-temperature testing approaches. Fracture toughness of the degraded materials shall be determined and no initial properties are required; and
- indirect determination of the fracture toughness of RPV material using Charpy V-notch impact test specimens and correlation formulae between brittle fracture temperature and temperature dependence of fracture toughness including their shifts. In this case, critical temperature of brittleness from Acceptance Tests, T_{k0}, must be well known, as surveillance Charpy specimens could serve only for determination of a shift of the transition temperature.

9.1. FRACTURE MECHANICS TEMPERATURES

9.1.1. Master curve approach

Reference temperature T_0 is determined from static fracture toughness tests using a single- or multiple-temperature "Master curve" approach in accordance with the standard ASTM E 1921–02 and its application is given in [13]. Then, a chosen lower tolerance bound (usually 5%) should be applied for determination of fracture toughness temperature dependence to be used in integrity/lifetime calculations.

In principle, transition temperature T_0 is usually determined for the required fluence for the RPV integrity assessment, i.e. for end-of-life fluence or for extended life fluence In addition, in special cases during RPV design or for prediction of RPV behaviour during future operation, these temperatures could be also evaluated using similar procedure as for critical temperature of brittleness, as it is explained in Chapter 6, together with shifts of brittle fracture temperature instead of shifts ΔT_{JC} or separately only for trend curves of shifts ΔT_0.

Similarly, this temperature could be determined also for a given time of ageing, i.e. for characterisation of thermal ageing of materials. The reference temperature, T_0, as determined in accordance with the standard ASTM E 1921-02 shall be increased by a margin, equal to a standard deviation σ (defined below) only for the tested condition, i.e. either initial or for a given degradation state. Reference temperature T_0 is defined from experimentally determined values of static fracture toughness, K_{JC}, adjusted to the thickness of 25 mm.

Margin σ_1 is added to cover the uncertainty in T_0 associated with using of only a few specimens to establish T_0 while margin δT_M characterizes the scatter of the properties of forgings and welds. The total standard deviation σ of the estimate of T_0 is given by:

$$\sigma=(\sigma_1^2+\delta T_M^2)^{1/2} \qquad (9.1)$$

where margin σ_1 is defined as

$$\sigma_1 = \beta / N^{0.5}, \ {}^{o}C \qquad (9.2)$$

where N = total number of specimens used to establish the value of T_0, β = + 18°C.

If the value of δT_M is not available from Qualification tests of given material, the use of the following fixed values is suggested:

δ TM1 = 10°C for the base material,

δ TM2 = 16°C for weld metals.

Thus, reference temperature that will be used in RPV integrity evaluation, RT_0, is defined as:

$$RT_0 = T_0 + \sigma \qquad (9.3)$$

In the case when surveillance specimens for determination of temperature T_0 were irradiated at a different neutron fluence as required (i.e. end-of-life or extended life), an interpolation between results from two adjacent fluences is permitted. In this case, interpolation can be performed using power law formula.

9.1.2. “Base curve” approach

“Base Curve” approach is approved as a national procedure in the Russian Federation. It contains the following parts:

- A new procedure of RPV brittle fracture resistance calculation; and
- A prediction procedure of fracture toughness temperature dependence on the base of testing small specimens (Prometey Probabilistic Model).

9.2. BRITTLE FRACTURE TEMPERATURE

To determine both the initial and actual temperature dependences of fracture toughness K_{JC}, respectively, the brittle fracture temperature T_k has to be used. The brittle fracture temperature T_k is usually determined for the fluence corresponding to the design or extended end of life. The brittle fracture temperature T_k as a result of irradiation embrittlement is given by the following relationship:

$$T_k = T_{ko} + \Delta T_F \ , \qquad (9.4)$$

where T_{ko} critical temperature of brittleness, [°C],

ΔT_F shift of the brittle fracture temperature due to irradiation, [°C].

The values of the brittle fracture temperature T_{k0} is to be obtained from the Acceptance Tests. If such value is unknown, then the so-called "guaranteed" value from the corresponding Technical Requirements or from Code for a given material is used. When using "guaranteed" values of T_{k0} in RPV integrity evaluation, then no temperature margin should be added as this value is considered to be enough conservative.

If the experimentally determined values of the initial brittle fracture temperature T_{k0} from component Acceptance Tests are known (based on component Passport) these should be increased by the temperature margin δT_M; the margin has to take into account the scatter of the values of mechanical properties in forgings and welds; δT_M.

δT_M is the standard deviation of T_{k0} determined for the given forgings or weld metal in the frame of Qualification Tests or in the frame of a set of identical materials established during production of the component by the identical technology. If this value is not available the application of the following values is suggested

δT_{M1} = 10°C for the base material,

δT_{M2} = 16°C for weld metals.

This margin should be applied to the temperature T_k determined in accordance with equation (9.4).

9.3. DETERMINATION OF THE EFFECT OF IRRADIATION EMBRITTLEMENT

9.3.1. The shift of the brittle fracture temperature due to irradiation

First method

Results of tests of the surveillance programme for specimens of the material of the vessel are available: respectively, also results for other vessels containing identical materials - for example, identical heat of the welding wire and flux:

Shift of the brittle fracture temperature is determined from the formula:

$$\Delta T_F = T_{kF} - T_{ki} \tag{9.5}$$

where T_{kF} is a value of transition temperature for a fluence F, T_{ki} is a value of transition temperature for initial conditions (unirradiated).

In both cases, these temperatures are determined from similar sets of specimens (minimum 12) using similar test equipment and procedure. The difference in fluence between specimens of one set should be smaller than ±15% of the mean value, and the difference in irradiation temperatures of individual specimens should be within 10°C. Finally, the mean value of irradiation temperature should be not higher than + 10°C above the inner wall temperature of the reactor pressure vessel.

Obtained experimental values of KV (impact notch energy) are evaluated using the following equation

$$KV = A + B \tanh [(T-T0)/C] \tag{9.6}$$

where A, B, C and T0 are constants derived by statistical evaluation.

It is strongly recommended to set lower shelf energy at values between 0 and 5 J to avoid incorrect fitting when a small number of specimens are tested in the lower shelf energy temperature region. Upper shelf energy should be fixed to the mean value of ductile fractured specimens. Shift of the transition temperature is determined for the criterion or consistent with national procedures.

$$KV = 41 \text{ J} \qquad (9.7)$$

This procedure results in valid values of ΔT_F only when the upper shelf energy, derived from the formula (9.6) — i.e., sum of (A+B), — is greater than 68 J. The results of determinations of the shift in the brittle fracture temperature obtained at least for three different neutron fluences are to be evaluated by the least squares method using the relationship:

$$\Delta T_F = A_F^{exp} \text{ x } (F\text{x}10^{-22})^n \qquad (9.8)$$

where F is the fluence of fast neutrons with the energy higher than 0.5 MeV, A_F^{exp} and n are empirical constants obtained by statistical evaluation of surveillance data.

Determination of shifts ΔT_F is to be based on unirradiated and irradiated test data obtained from the same type of testing equipment and using the identical procedures for statistically processed curves. Mean experimentally determined fluence dependence of ΔT_k in accordance with the equation (9.8) from surveillance tests is compared with the prediction for a given chemical composition of the tested material as shown in (9.9 + 9.11). If the real shifts are larger by more than 30°C (approx. 1.5 SD given in (9.10 + 9.12)) for end-of-life fluence than predicted value, analysis of the difference should be performed and evaluated.

In addition, the mean line from (9.8) should be vertically shifted upward by the value of δT_M calculated according to 9.2. If any experimental point exceeds this adjusted trend curve, the curve should be shifted further until it bounds all data. This upper boundary of the shifts is to be used in assessment of RPV resistance against fast fracture. It is not allowed to extrapolate shifts of the transient temperatures for the fluences higher than double of the maximum fluence used for the experiment.

Second method

If there are insufficient number or no surveillance test results: In such case, the following prediction formula can be used for the shift in brittle fracture temperature:

Table 8.2. Proposed prediction formulate for the shift in brittle fracture temperature

Metal	Formula	SD	Number
Weld metal*	$\Delta T = [884 * P + 51.3 * Cu] * \Phi^{0.29}$ $= 800 * (1.11 * P + 0.064 * Cu) * \Phi^{0.29}$	SD=22.6 °C	(8.14)
Base metal*	ΔT = 8.37* F0.43	SD = 21.7 °C	(8.25)

* Formulae are valid for neutron fluences in the range $10^{22} < F < 4 \text{ x } 10^{24} \text{ m}^{-2}$

Both formulae represent the mean trend line; this mean value with the margin should be used for RPV integrity assessment.

9.3.2. Determination of the Reference temperature T_0 for required time of operation

If this cannot be determined directly by fracture toughness testing, then the following mixed way (i.e. combination of static fracture toughness and Charpy V-notch impact test results) may be conservatively used for determination of temperature T_0 during operation, i.e.

$$T_0^{operation} = T_0^{initial} + 1.1\ \Delta T_F \tag{9.13}$$

where ΔT_F is determined by the same process as is shown in 9.3.1, i.e. using Charpy impact specimen testing and/or prediction using formula (8.14).

In this case, the same margin than for the scatter of the material and the margin equal to standard deviation, $\Delta\sigma$, in accordance with the standard E 1921-02 should be applied for determination of $T_0^{initial}$.

REFERENCES

[1] PNAE-G0002-86, Standard for Strength Calculations of Components and Piping in NPPs, PN AE G-7 002-86m Energoatomizdat, Moscow (1989).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Components Important to Safety: PWR Pressure V essels, IAEA-TECDOC-1120, Vienna (1999).

[3] KRYUKOV, A., et al., "Investigation of samples taken from Kozloduy unit 2 Reactor Pressure Vessel", Int. Journal of Nuclear Engineering and Design 59–76 (1996).

[4] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Vessels, ASTM-E 185, ASTM, West Conshohocken, PA. USA (2002).

[5] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Practice for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, ASTM E-636, ASTM, West Conshohocken, PA. USA (2002).

[6] NUCLEAR REGULATORY COMMISSION, Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," Office of Nuclear Regulatory Research, U.S. NRC, Washington, DC, May (1998).

[7] KIRK, M., et al., Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10CFR50.61), draft NUREG, U.S. Nuclear Regulatory Commission, Washington, D.C. (2003).

[8] VERLIFE – Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs, EU 5th FP, Rez, (2003).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Reactor Pressure Vessel Embrittlement (Report of the IAEA Extrabudgetary Programme on the Safety of WWER-440 Model 230 Nuclear Power Plants), IAEA-TECDOC-659, Vienna (1992)

[10] GILLEMOT, F., IANKO, L., DAVIES, M., "The IAEA data base ageing of reactor pressure vessel steels and welds", paper presented at the IAEA Specialist Meeting on Irradiation Embrittlement and Mitigation, Espoo, Finland (1995).

[11] STEELE, L.E.; BRUMOVSKY, M.; GILLEMOT, F.; KRYUKOV, A.; WALLIN, K., Phase III of the IAEA Coordinated Research Program on Optimizing of Reactor Pressure Vessel Surveillance Programs and Analysis, 18th Symposium on Effects of Radiation on Materials. June (1996).

[12] GILLEMOT, F., DAVIES, L.M., Examples of the Use of the Database, IAEA Specialist Meeting on Methodology for Pressurized Thermal Shock Evaluation, IWG-LMNPP-97/1 pp.281–287. Esztergom, Hungary May (1997).

[13] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Test Method for Determination of Reference Temperature, T_0, for Ferritic Steels in the Transition Range, ASTM E 1921, West Conshohocken, PA, USA.

SYMBOLS AND UNITS

T_{ko}	-	initial critical temperature of brittleness, [°C]
T_k	-	critical temperature of brittleness, [°C]
T_0	-	reference temperature from "Master Curve" approach, [°C]
ΔT_F	-	shift of the critical temperature of brittleness due to irradiation, [°C]
ΔT_T	-	shift of the critical temperature of brittleness due to thermal ageing, [°C]
ΔT_N	-	shift of the critical temperature of brittleness due to cyclic damage, [°C]
A_F	-	irradiation embrittlement coefficient, [°C]
D	-	fatigue damage factor
COD	-	crack opening displacement – pre-cracked Charpy size specimen

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