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PREDICTION OF IRRADIATION EMBRITTLEMENT IN WWER-440 REACTOR PRESSURE VESSEL MATERIALS

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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 programs. The existing Russian Code (Standard for Strength Calculations of Components and Piping in Nuclear Power Plants (NPPs) – PNAE G 7-002-86) 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. 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 was required for more precise prediction of radiation embrittlement of RPV materials. Based on the fact that large amount of data from surveillance program as well as some research programs, IAEA International Database on RPV Materials (IDRPVM) has been used for the detailed analysis radiation embrittlement of WWER RPV materials. Thus, the following activities have been performed within the IAEA Co-ordinated project: 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, Development of the 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.

INTRODUCTION

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 programs. The existing Russian Code [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.

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 big 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.

Reactor pressure vessels for WWER-440 type reactors were manufacture from the steel of 15H2MFA(A) grade which is of Cr-Mo-V type. Such steels are not used in any other RPVs in other type or reactors, thus there does not exist any other prediction formulae that could be applied for WWER-440 materials – their chemical composition is quite different when compared with usual LWR RPV materials – ASTM A 533-B or A 508 that are of Ni-Mo-Cr type without any amount of vanadium.

Thus, new analysis of all available data were required for more precise prediction of radiation embrittlement of RPV materials. This analysis was performed within the IAEA Cvo-ordinated Research Program (CRP) with these main tasks:

– 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;

– Development of the 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.

These IAEA Guidelines were finished in 2004 and are now in the printing procedure.

1. CURRENT PROCEDURE [1]

The similar procedure is used for the radiation embrittlement assessment in Russia, Bulgaria and Ukraine and is based on Russian Code [1] for the embrittlement assessment based on the brittle fracture temperature T_k. Its value should be evaluated experimentally by Charpy impact testing entirely. However, if it is not applicable, empirical relations given next have to be used:

Weld metals:

$$T_k=T_0 + \Delta T_F; \quad (1)$$

$$\Delta T_F=A_F^{270} (F\text{x}10^{-22})^{1/3} \text{ for } 10^{22}<F<3 \text{ x}10^{24} \quad (2)$$

A_F = 15°C for weld metal Sv-10KhMFT(U) ($T_{irradiation}$ = 270°C); (3)

= 800 (P + 0.07 Cu) for weld metal Sv-10KhMFT ($T_{irradiation}$ = 270 °C). (4)

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. (5)

Base metals: Formulae (1.1-1.2) 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. Coefficients A_F for base metals are given in **[1]** as:

A_F = 18 °C for steel 15Kh2MFA ($T_{irradiation}$ = 270 °C) (6)

= 12 °C for steel 15Kh2MFAA ($T_{irradiation}$ = 270 °C) (7)

Nor in [1] neither in other official document there is a valid procedure for evaluation of results from the surveillance specimen programs for WWER reactor pressure vessels. There is also no recommendation for the use of such data in reactor pressure vessel integrity and lifetime assessment.

2. 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.

2.1. PRINCIPLES OF THE TREND CURVE DERIVATION

Basic physics research has not resulted in justification of a unique physically based functional forms. The number of candidate functions, which fulfill 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)

The exponential function $\Delta T = a * (1 - e^{-n*\Phi})$; (9)

The tanh function

$$\Delta T = c1 * [1/2 + 1/2 * \tanh(\frac{\Phi - \Phi_0}{c2})], \quad (10)$$

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 characterization 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 nucleation 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.

The data are characterized 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 (1.4) given in the Russian Code PNAE [1].

The distribution of weld data in the copper-phosphorus plane is shown in Fig. 1.

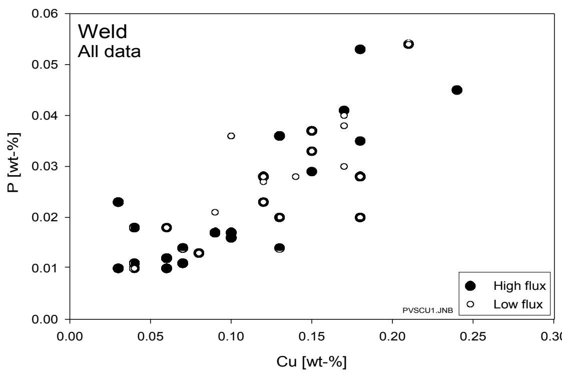

Fig. 1. Distribution of weld data in the Cu-P plane. Low flux marks are located on the top of high flux marks, i.e. open circles in bold include both flux values

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 Fig. 2.

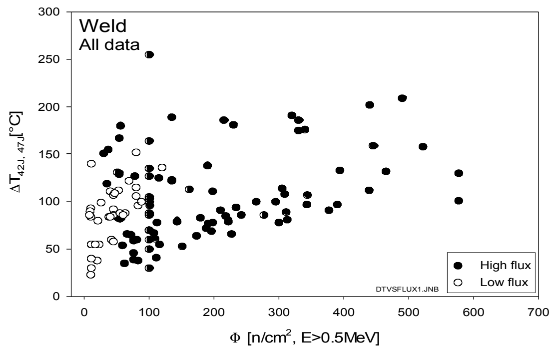

Fig. 2. The measured transition temperature shifts as a function of neutron fluence

The measured weld data compared to the prediction by formula (1.4) are shown in Fig. 3. The formula (1.4) 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).

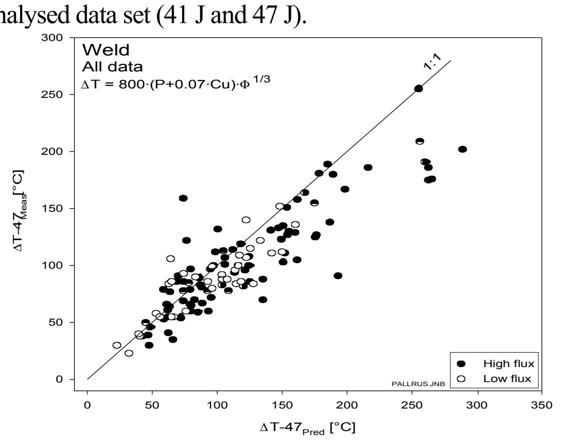

Fig. 3. The measured shift versus the shift calculated from the formula (1.4), weld data

The distribution of base metal data in the copper-phosphorus plane is shown in Fig. 4, where low and high flux data points are indicated.

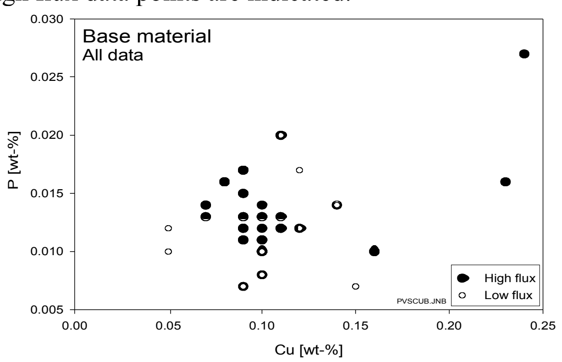

Fig. 4. Distribution of base metal data in the Cu-P plane. Low flux marks are located on the top of high flux marks, i.e. open circles in bold include both flux values

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 Fig. 5 as a function of neutron fluence. The shifts correlate well with neutron fluence.

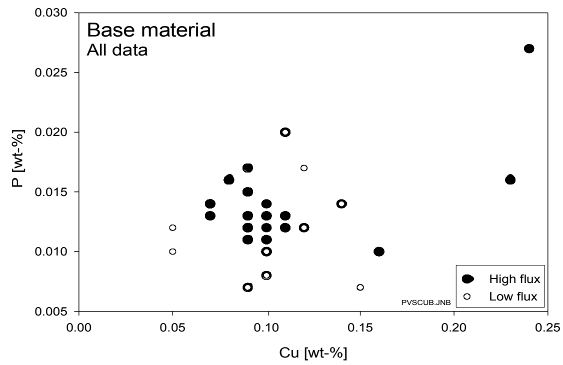

Fig. 5. The measured transition temperature shifts of base metal as a function of neutron fluence

The measured shifts of base metal are compared to the prediction (1.4) used for weld metal in Fig. 6. For most of the points the weld prediction formula (1.4) represents an upper boundary for base metal but there are points in the range of ΔT_{Pred} = 80 to 140 ºC, which exceed the prediction for weld.

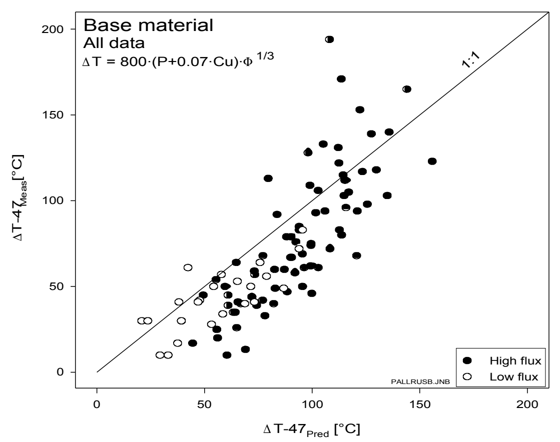

Fig. 6. The measured base metal shifts versus the shifts calculated from the formula (1.4) for weld metal

2.2. TREND CURVE FITTING

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

Fit-1: $\Delta T = (a1 * P + a2 * Cu) * \Phi^{n}$; (11)

Fit-2: $\Delta T = (a1 * P + a2 * Cu) * \Phi^{n} + a3 * \Phi^{n3}$; (12)

Fit-3: $\Delta T = a1 * P * \Phi^{n1} + a2 * Cu * \Phi^{n2} + a3 * \Phi^{n3}$; (13)

Fit-4:

$$\Delta T = a1 * P * (1 - e^{-n1 * \Phi}) + a2 * Cu * (1 - e^{-n2 * \Phi}) + a3 * \Phi^{n3}; \quad (14)$$

Fit-5:

$$\Delta T = a * F^{n} + b1 * \left[1 - e^{\left(-F/F_{sat}\right)}\right] + c1 * [1/2 + 1/2 * \tanh(\frac{F - F_{start}}{c2})]. \quad (15)$$

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.

2.3. 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 ºC 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 ºC. 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 analyzed 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 ºC.
- 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 ºC. Fits made to low flux data sub sets are typically 12 ºC. Because the number of low flux data points is relatively low, the derived functions are not so well defined as the all data fits.

3. 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 initial properties are required;
- 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

3.1. FRACTURE MECHANICS TEMPERATURES

3.1.1. 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 [2, 3]. 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, 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:

$$\sigma=(\sigma_1^2+\delta T_M^2)^{1/2}, \tag{16}$$

where margin σ_1 is defined as

$$\sigma_1 = \beta / N^{0.5},\ ^{\circ}C, \tag{17}$$

where N = total number of specimens used to establish the value of T_0,

$\beta = +18$ °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:

δT_{M1} = 10°C for the base material,

δT_{M2} = 16°C for weld metals.

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

$$RT_0 = T_0 + \sigma. \quad (18)$$

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.

3.1.2. BASE CURVE APPROACH

"Base Curve" approach is approved as a national procedure [4] in Russian Federation. It contains the followingparts:

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

3.2. 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:

$$T_k = T_{ko} + \Delta T_F, \quad (3.4)$$

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} was 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 the 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.

Value of 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 (3.4).

3.3. DETERMINATION OF THE EFFECT OF IRRADIATION EMBRITTLEMENT

3.3.1. THE SHIFT OF THE BRITTLE FRACTURE TEMPERATURE DUE TO IRRADIATION

First method: Results of tests of the surveillance program 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:

$$\Delta T_F = T_{kF} - T_{ki}, \quad (19)$$

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

$$KV = A + B \tanh [(T-T0)/C], \quad (20)$$

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

$$KV = 41 \text{ J}. \quad (21)$$

This procedure results in valid values of ΔT_F only when the upper shelf energy, derived from the formula (3.6) - 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:

$$\Delta T_F = A_F^{exp}. (F.10^{-22})^n, \quad (22)$$

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 (3.8) from surveillance tests is compared with the prediction for a given chemical composition of the tested material. If the real shifts are larger by more than 30 °C (approx. 1.5 SD given in (3.9+3.10)) for end-of-life fluence than predicted value, analysis of the difference should be performed and evaluated.

In addition, the mean line from (3.8) should be vertically shifted upward by the value of δT_M calculated according to 3.2. 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 a case, the following prediction formula can be used for prediction of 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 ºC	(23)
Base metal*	$\Delta T = 8.37 * F^{0.43}$	SD=21.7 °C	(24)

*Formulae are valid for neutron fluences in the range $10^{22} < F < 4 \times 10^{24}$ m^{-2} SD = standard deviation

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

3.3.2. 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.

$$T_0^{operation} = T_0^{initial} + 1.1\ \Delta T_F, \qquad (25)$$

where ΔT_F is determined by the same process as is shown in 3.3.1, i.e. using Charpy impact specimen testing and/or prediction using formula (3.8).

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 $T_0^{initial}$.

CONCLUSION

New Guidelines for prediction irradiation embrittlement in reactor pressure vessel materials of WWER-440 type reactors were prepared within the IAEA Co-ordinated Research project. These Guidelines are based on analysis of experimental data from irradiation of materials of these RPVs collected in the IAEA International Database on RPV Materials (IDRPVM).

These Guidelines contain formulae for prediction irradiation embrittlement for base and weld metals of this type of reactors, either based on brittle transition temperature, T_k, or reference temperature of Master Curve approach, T_0.

Recommendation for the use of real experimental data from testing surveillance specimens from these RPVs was also elaborated and recommended.

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ПРЕДСКАЗАНИЕ РАДИАЦИОННОГО ОХРУПЧИВАНИЯ ВНУТРИКОРПУСНЫХ МАТЕРИАЛОВ РЕАКТОРА ВВЕР-440

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Предсказание радиационного охрупчивания материалов внутрикорпустных устройств (ВКУ) обычно выполняется в соотвествии с кодами и стандартами, основанными на обширной информации, накопленной в ходе модельных и исследовательских программ. Существующий Российский Код (Стандарт для вычислений прочности и компонентов и трубопроводов в атомных электростанциях (АЭС) – PNAE G 7-002-86) для оценки радиационного охрупчивания ВКУ реакторов ВВЭР хорошо зарекомендовал себя на протяжении более чем 20 лет; он основан на экспериментальных данных, полученных в исследовательских реакторах с ускоренным облучением. Оценка упомянутого выше Кода была выполнена без результатов с образцов-свидетелей, которые были получены в 1980-1990 годах. Таким образом, необходим новый анализ всех имеющихся данных для более точного прогнозирования радиационного охрупчивания материалов ВКУ. На основании того факта, что было использовано большое количество данных с макетных и исследовательских программ, Международная База данных МАГАТЭ по материалам ВКУ была использована для подробного анализа радиационного охрупчивания ВКУ материалов для реакторов ВВЭР. Таким образом, в рамках Координационного проекта МАГАТЭ были выполнено следующее: сбор полных данных с образцов-свидетелей ВВЕР-440 и других подобных важных данных в Международную базу данных; анализ данных по радиационному охрупчиванию ВКУ материалов ВВЭР-440 с использованием международной базы данных; оценка формулы прогнозирования в зависимости от химического состава материала, флюенса нейтронов и нейтронного потока, разработка основных положений для предсказания радиационного охрупчивания эксплуатируемых внутрикорпусных устройств ВВЭР-440, включая методологию для оценки контрольных данных конкретной действующей установки.

ПРОГНОЗУВАННЯ РАДІАЦІЙНОГО ОКРИХЧЕННЯ ВНУТРІШНЬОКОРПУСНИХ МАТЕРІАЛІВ РЕАКТОРА ВВЕР-440

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Прогнозування радіаційного окрихчення матеріалів внутрішньо корпусних пристроїв (ВКП) зазвичай виконується у відповідності з кодами та стандартами, заснованими на численній інформації, що була накопичена на базі модельних та дослідницьких програм. Існуючий Російський Код (Стандарт для обчислення міцності компонентів і трубопроводів в атомних електростанціях (АЕС)-PNAE G 7 –002-86) для оцінки радіаційного окрихчення ВКП реакторів ВВЕР зарекомендував себе на протязі більш, ніж 20 років; він заснован на експриментальних даних, отриманих в дослідницьких реакторах з прискореним опроміненням. Оцінка вище згаданого Кода була виконана без результатів із зразків-свідків, які були отримані у 1980-1990 роках. Таким чином, необхідно провести новий аналіз усіх наявних даних для більш точного прогнозування радіаційного окрихчення матеріалів ВКП. На підставі того факту, що було використано велику кількість даних з макетних та дослідницьких програм, Міжнародна база даних МАГАТЕ по матеріалам ВКП біла використана для докладного аналізу радіаційного окрихчення ВКП матеріалів для реакторів ВВЕР. Таким чином, в межах Координаційного проекту МАГАТЕ було виконано наступне: Збирання повних даних із зразків-свідків ВВЕР-440 та інших подібних важливих даних в Міжнародну базу даних, аналіз даних по радіаційному окрихченню ВКП матеріалів ВВЕР-440 з використанням міжнародної бази даних, оцінка формули прогнозування в залежності від хімічного складу матеріалу, флюенса нейтронів та нейтронного потоку, розробка основних положень для прогнозування радіаційного окрихчення експлуатуємих внутрішньо корпусних пристроїв ВВЕР-440, включно з методологією для оцінки контрольних даних конкретної діючої установки.