EMBRITTLEMENT TREND CURVE FOR SPANISH PWR PRESSURE VESSEL STEELS*

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INTRODUCTION

Neutron irradiation causes changes in the mechanical properties of pressure vessel steels and welds. The extent of these changes can be quantified by means of empirical models developed on the basis of tests data from material irradiated in experimental reactors, or from surveillance capsules irradiated in commercial reactors.

By using empirical and semi-empirical models, it is possible to determine the increase in reference temperature RT_{NDT} and the decrease in USE that the vessel material will under undergo for a given neutron fluence. Conclusions can be drawn from these predictions regarding the vessel state and the evolution of its mechanical properties. The results of these studies can contribute to extending the effective lifetime of power plants, either by ensuring compliance with vessel fracture toughness requirements and providing the information required to extend the plant's operating license beyond its design life, or else by providing the basis for a possible vessel annealing.

This article presents the authors' own correlation for the calculation of ΔRT_{NDT}, developed on the basis of surveillance data obtained from the analysis of capsules removed from the Spanish PWR reactors.

SURVEILLANCE PROGRAM OF SPANISH (PWR) NUCLEAR POWER PLANTS

In Spain, the regulations followed in surveillance programs are the ones from the country where the reactor was designed. Thus, German regulations are applied in the case of Trillo NPP, whereas for the rest of the Spanish PWR reactors (Westinghouse design), American regulations are applied to the surveillance program. However, all activities relating to surveillance programs, capsule analysis, impact of surveillance data on pressure vessel integrity and revision of operating limit curves are carried out by Spanish organisations [1,2].

Table I contains a summary of the principal characteristics and status of the vessel surveillance programs of Spanish PWR type reactors.

FORMULA PWR-ES/BC

An essential part of reactor pressure vessel surveillance is evaluation of its structural integrity at the end of its design life by means of empirical and semi-empirical models, in order to determine the additional effective life margin. In Spain, the main method used to extrapolate surveillance results to end of life, 32 effective full power years, is provided by Revision 2 of Regulatory Guide 1.99 [3].

In accordance with analyses performed on surveillance capsules removed from the Spanish PWR reactors, the changes in reference temperature RT_{NDT} are not as great as those resulting from surveillance data obtained from U.S. nuclear power plants, basically because the level of impurities in copper is not high in the Spanish PWR reactor vessel steels. For this reason, it was necessary for the authors to obtain their own correlation, which is more consistent with the ΔRT_{NDT} values observed in Spanish vessels.

Figure I shows the ΔRT_{NDT} values measured for base material with a copper content of 0,02% to 0,06% by weight, as is the case of most Spanish PWR reactors. This figure reveals that for low copper content and low fluence level, R.G. 1.99 Rev. 2 seems to be slightly conservative.

To estimate the embrittlement level, the proven fact that the magnitude of change caused by irradiation primarily depends on the material chemistry and the neutron fluence it receives is used:

$$\Delta RT_{NDT} = CF \cdot FF \qquad (1)$$

where FF is the fluence factor and CF is the chemistry factor.

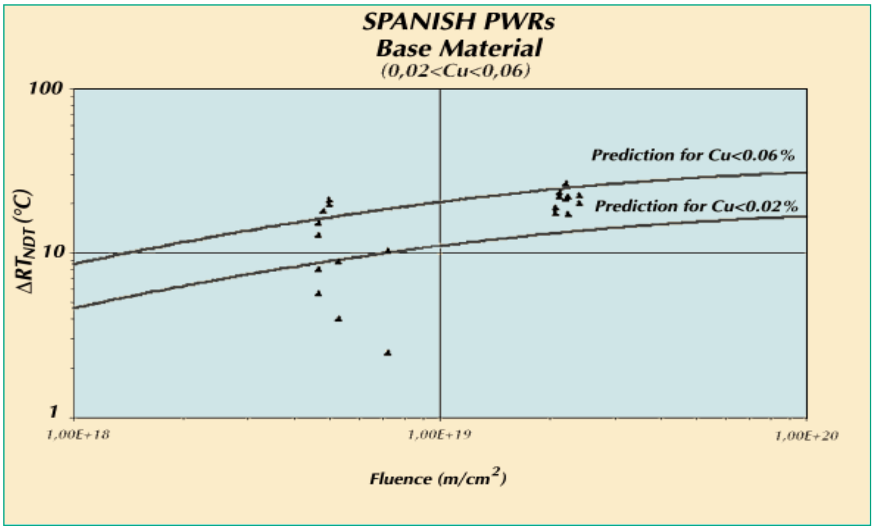
The equation for calculating ΔRT_{NDT} has been established on the basis of surveillance data of irradiated materials with low copper content (Cu < 0.1) from the Spanish PWR reactors. On the basis of the results of the Charpy impact tests performed, the aim is to correlate the ΔRT_{NDT}

Table I. Spanish PWR Reactors and Status of Their Surveillance Programs

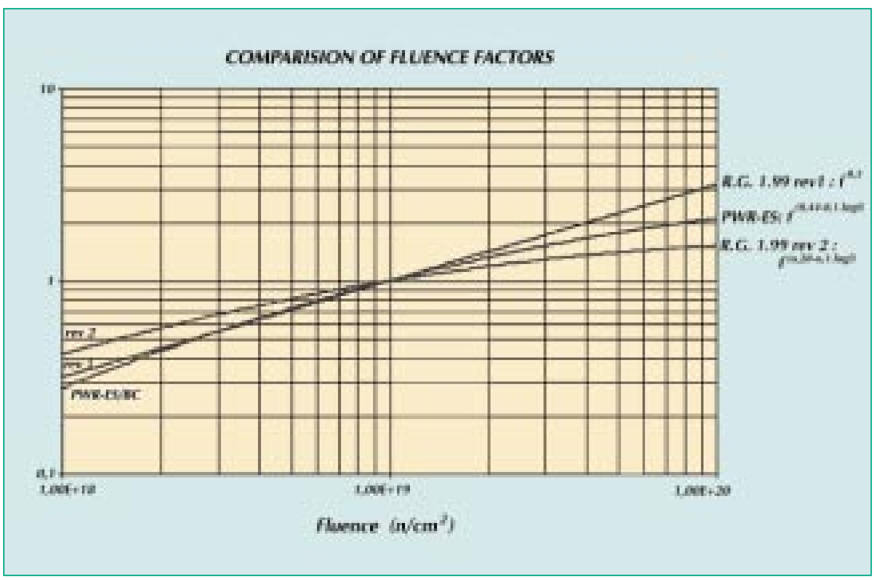
Nuclear Power Plant	Capacity MWe	Supplier	Began operation	Number of capsules	Capsules Removed at 31.10.98
José Cabrera	160	Westinghouse	1968	8 (a)	5
Almaraz I	930	Westinghouse	1981	6 (b)	3 (d)
Almaraz II	930	Westinghouse	1983	6 (b)	3 (d)
Ascó I	930	Westinghouse	1983	6 (b)	3 (d)
Ascó II	930	Westinghouse	1985	6 (b)	2
Vandellós II	992	Westinghouse	1988	6 (b)	2
Trillo	1041	KWU	1988	3 (c)	1

(a) Three of the eight capsules are in reserve, e.g. for life extension.
(b) Two of the six capsules are in reserve.
(c) Two of the three capsules are in reserve.
(d) Results were only available for the first two capsules at the time this work was prepared.

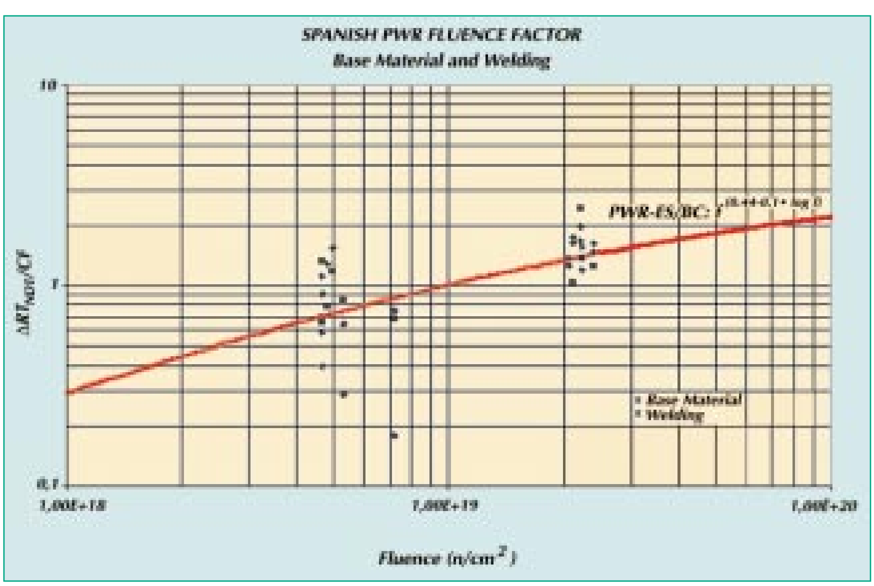
This article is based on Beatriz Acosta's dissertation, wich in 1997 obtained the SNE´s award for the best work submitted as thesis or university project in the nuclear field.



F I - Prediction of ΔRT_{NDT} as per R.G. 1.99 Rev. 2



F II - Fluence Factor as per R.G. 1.99 and PWR-ES/BC



F III - Values of ΔRT_{NDT}/CF versus fluence. CF as per PWR-ES/BC; CF = 12.3+34.52·%Cu

values observed and the fluences measured in the surveillance capsules, according to a formula as follows:

$$\Delta RT_{NDT} = CF \cdot f^{\,(a-b \cdot \log f)} \qquad (2)$$

where f is the neutron fluence. The exponent decreases with the fluence in order to consider the saturation effect that occurs at high fluences when, as the number of neutrons that penetrate the crystalline grid of the steel increases, the recombination of defects increases, thus decreasing the damage caused by the irradiation.

After imposing the conditions required for performing the regression analysis between the fluences and the ΔRT_{NDT} measured in the capsules, the following formula is obtained for the fluence factor:

$$FF = f^{\,(0.44-0.1 \cdot \log f)} \qquad (3)$$

where f is the fast neutron fluence (E>1 MeV) at which ΔRT_{NDT} is evaluated in units of 10^{19} n/cm^2.

Figure II shows the evolution of the above equation in comparison to the fluence factors proposed in Revisions 1 and 2 of Regulatory Guide 1.99. It can be seen that the fluence factor, when adjusted for Spanish reactors, is in an intermediate position between those of R.G. 1.99 Rev. 1 and R.G. 1.99 Rev. 2 for high fluences, and is lower than the other two at low fluences and more approximate to the data observed in Spanish vessels.

The following formula has been obtained for the chemistry factor:

$$CF = 12.3+34.52 \cdot \%Cu \qquad (4)$$

This chemistry factor is valid for vessel steels with lower copper content (Cu < 0.1), as in the case of most Spanish reactors. It would also be applicable to future reactors with vessel material SA-533 B Class 1, since the ASME code [4] establishes a copper limit in heat analysis of 0.1% by weight.

Therefore, the correlation for calculating ΔRT_{NDT}, called PWR-ES/BC, takes on the following form:

$$\Delta RT_{NDT} = (12.3+34.52 \cdot \%Cu) \cdot f^{\,(0.44-0.1 \cdot \log f)} \qquad (5)$$

where ΔRT_{NDT} is given in ºC and the fluence in units of 10^{19} n/cm^2.

Figure III shows the fluence factor of PWR-ES/BC, equation (3), together with the measured values of ΔRT_{NDT}, normalized by the chemistry factor obtained on using expression (4).

Figures IV and V show the ΔRT_{NDT} values, for base material and welding respectively, measured against the values calculated by using the formula PWR-ES/BC. The points on the diagram are very near the graph diagonal, which indicates a good consistency in predicting ΔRT_{NDT}.

The standard deviations of the residual values (observed values minus calculated values of ΔRT_{NDT}), obtained by using the PWR-ES/BC correlation, are 5ºC for base material and 6ºC for welding, which are lower than the values resulting from applying American standards.

Figure VI, obtained by using typical Cu and Ni values of Spanish PWR pressure vessel steels, shows the joint behavior of [$\Delta RT_{NDT} + 2 \cdot \sigma\Delta$] in terms of the neutron fluence received by the vessel. It has been indicated previously that $\sigma\Delta$ is 5ºC for base material and 6ºC for welding when PWR-ES/BC is applied, where as R.G. 1.99 Rev. 2 gives a standard deviation value of 9.5ºC for base material and 15.5ºC for welding. These values were obtained in the regression analysis performed by Guthrie [7].

It can be seen from the figure that at 32 effective full power years, i.e. for a typical fluence of $4.5 \cdot 1019$ n/cm^2, the value of [$\Delta RT_{NDT} + 2 \cdot \sigma\Delta$] is approximately 8ºC less for the prediction made on the basis of PWR-ES/BC than for the estimation provided by R.G. 1.99 Rev. 2. This means a lower RT_{NDT} value for the end of the vessel's lifetime, i.e., a value farther from the limit established by the regulation, thereby ensuring greater safety in plant operation and the possibility of extending the operating window (pressure-temperature limits) and lengthening the vessel's effective lifetime.

CONCLUSIONS

The use of an embrittlement trend curve or formula specific to Spanish PWR reactor pressure vessel materials, such as the one presented herein, would help to optimize plant operation, increase safety margins and extend the plant's effective lifetime beyond the predictions established by use of standing American regulations.

REFERENCES

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[2] Bros, J. And Ballesteros, A., "Overview of the Activities in Spain on Irradiation Embrittlement of RPV Steel", Reactor Dosimetry ASTM STP 1228, Philadelphia, 1994.

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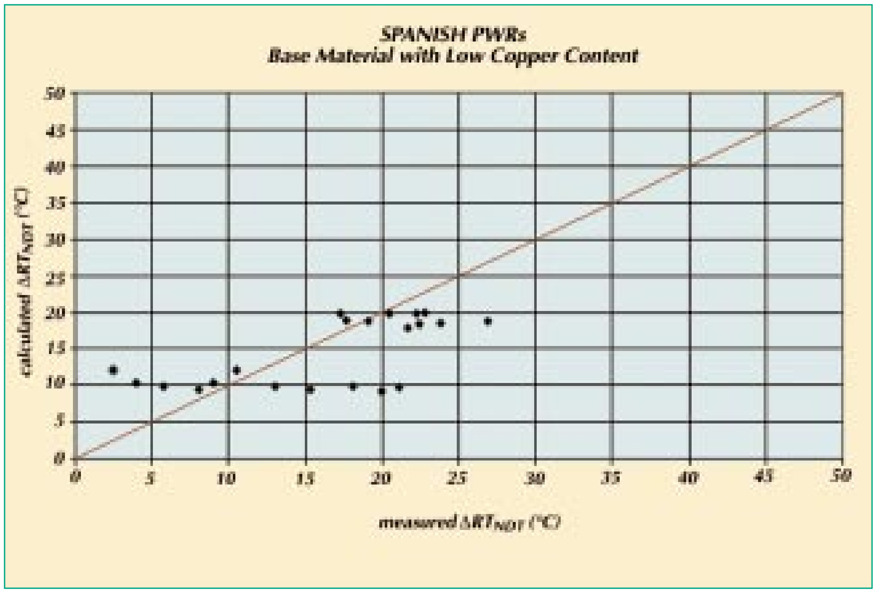
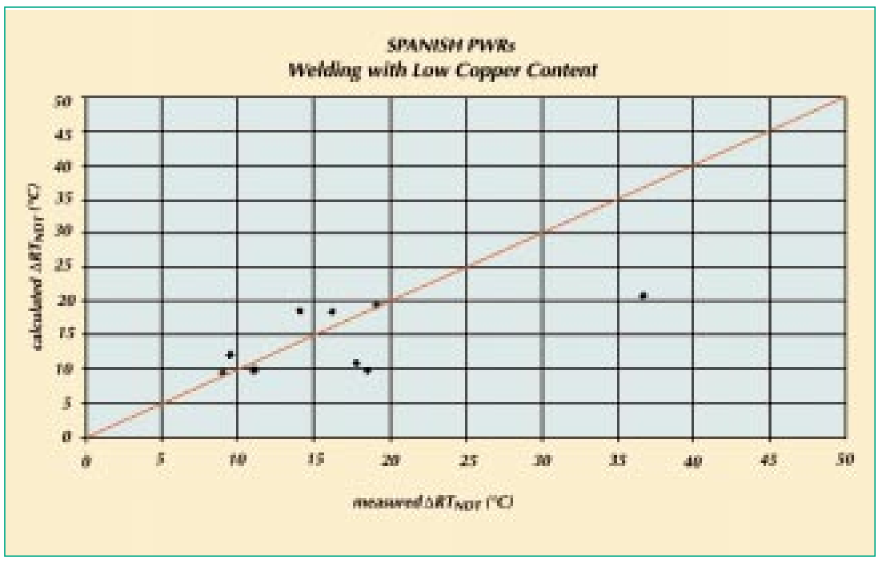
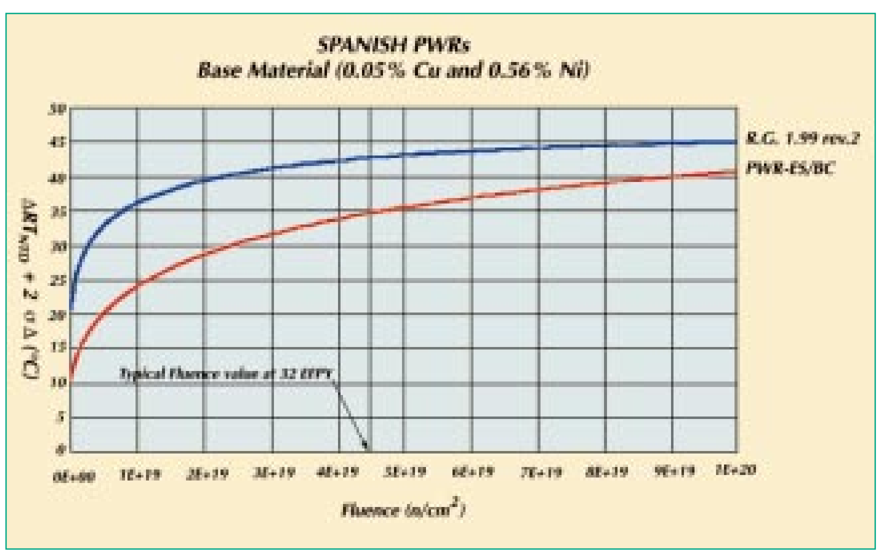


F IV – Calculated ΔRT_{NDT} versus measured ΔRT_{NDT}, calculation as per PWR-ES/BC

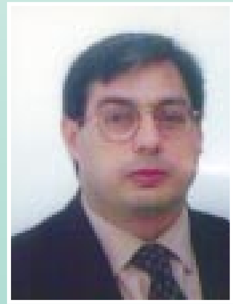


F V – Calculated ΔRT_{NDT} versus measured ΔRT_{NDT}, calculation as per PWR-ES/BC



F VI – Evolution of Reference Temperature

Beatriz ACOSTA is an industrial engineer and majored in energy techniques in the E.T.S.I.I. of Madrid. After completing her dissertation in Tecnatom's Division of Inspection Engineering, in 1997 she joined the Institute for Advanced Materials of the European Commission's Joint Research Centre in Petten (Holland), as part of the Material Ageing Evaluation and Studies group.



Antonio BALLESTEROS received a degree in theoretical physics from the Universidad Autónoma de Madrid. In 1987, he joined Tecnatom's Component Integrity Group to take charge of reactor pressure vessel surveillance programs. He has conducted the analysis of all capsules removed since that time, and has also done work in relation to pressure vessel integrity both in Spain and as coordinator of several international projects undertaken by the European Network AMES group.

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[4] ASME Code Section III, Rules for Construction of Nuclear Power Plants Components, Subsection NB 2331, "Materials for Vessels".

[5] "Survey of National Regulatory Requirements", AMES Report No. 4, June 1995.

[6] "A Review of Formulas for Predicting Irradiation Embrittlement of Reactor Vessel Materials", AMES Report No. 6, July 1996.

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