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Technical Note

Comparison of applicability of current transition temperature shift models to SA533B-1 reactor pressure vessel steel of Korean nuclear reactors

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A R T I C L E I N F O

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A B S T R A C T

The precise prediction of radiation embrittlement of aged reactor pressure vessels (RPVs) is a prerequisite for the long-term operation of nuclear power plants beyond their original design life. The expiration of the operation licenses for Korean reactors the RPVs of which are made from SA533B-1 plates and welds is imminent. Korean regulatory rules have adopted the US Nuclear Regulatory Commission's transition temperature shift (TTS) models to the prediction of the embrittlement of Korean reactor pressure vessels. The applicability of the TTS model to predict the embrittlement of Korean RPVs made of SA533B-1 plates and welds was investigated in this study. It was concluded that the TTS model of 10 CFR 50.61a matched the trends of the radiation embrittlement in the SA533B-1 plates and welds better than did that of Regulatory Guide (RG) 1.99 Rev. 2. This is attributed to the fact that the prediction performance of 10 CFR 50.61a was enhanced by considering the difference in radiation embrittlement sensitivity among the different types of RPV materials.

1. Introduction

The long-term operation of nuclear power plants (NPPs) beyond the original design life is essential to satisfy the increasing global demand for nuclear power as a clean and sustainable energy source. A specific design-basis life such as 40 years was originally not based on technical studies of material degradation. The current target for most plants in many countries in Europe, Japan and USA is long-term operation beyond 60 years [1].

The practical operating life of a reactor is determined based on the safety margin of the reactor pressure vessel (RPV) as it is impossible or economically unviable to replace the RPV if its mechanical properties degrade significantly. RPVs are thick steel containers that hold nuclear fuel while the reactors operate. The vessels provide one of several barriers that keep radioactive fuel contained and out of the environment. Reactor operation generates subatomic particles called neutrons. Some of these neutrons hit atoms in the steel as they leave the core. The exposure to high-energy neutrons can result in embrittlement of radiation-sensitive RPV steels. The neutron radiation embrittlement is considered to be the primary aging degradation phenomenon that occurs in the materials for RPVs.

Pressurized-water reactors (PWRs) take embrittlement into account because of a phenomenon called pressurized thermal shock. This is an accident scenario in which cold water enters a reactor while the vessel is pressurized. This rapidly cools the vessel and places large thermal stresses on the steel. Under these conditions an embrittled vessel can crack and even fail. This would seriously challenge the plant's ability to keep the public safe [2].

The nuclear regulatory rules require reactor surveillance programs including plans for installation of surveillance capsules containing specimens, the removal of surveillance capsules at specific intervals and testing of encapsulated specimens exposed to neutron irradiation to monitor changes in the fracture toughness and tensile properties of the beltline materials of the RPV.

Irradiation embrittlement of RPV beltline materials has been evaluated according to the US Nuclear Regulatory Commission (NRC) Regulatory Guide 1.99, Radiation embrittlement of reactor vessel material, Revision 2 (RG 1.99 Rev. 2), which presents methods (based on data correlations) for estimating a Charpy transition temperature shift (TTS) at 41 J (30 ft-lb) [3]. The irradiation hardening and embrittlement of RPV steels depend on a combination of many metallurgical and irradiation variables.

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Therefore, it is difficult to expect a specific TTS model to be able to universally estimate the embrittlement trends of RPVs of various product types, fabrication practices, and irradiation conditions [4]. The last few decades have seen remarkable progress in the development of a mechanistic understanding of irradiation embrittlement for an RPV. This understanding has been exploited in formulating robust, physically based, and statistically calibrated models of Charpy V-notch-indexed TTSs [5]. A mechanistic and sophisticated TTS model is given in Title 10, Section 50.61a, Alternate fracture toughness requirements for protection against pressurized thermal shock events of the US Code of Federal Regulations (10 CFR 50.61a) and draft RG 1.99, Rev. 3 [6,7].

A total of 20 PWRs are operating in Korea. Five of the six RPVs of Westinghouse type PWRs, manufactured by combustion engineering (CE) in the 1980s and on the verge of reaching their original designed lifetime, are made of SA533 low alloy steel, Grade B, Class 1 (SA533B-1) rolled plates and their welds. The RPV of the oldest Westinghouse reactor (Kori Unit 1) was made of SA508 Gr.2 forging. The others constructed after the 1980s are made of SA508 Gr.3 forgings. Kori Unit 1 will be shut down in June 2017 after 10 years of extended operation. Thus, the focus is on the continued operation of the five PWRs whose original license expirations are imminent.

The purpose of this study is to verify the applicability of current TTS models for SA533B-1 RPV materials, to predict more accurately the embrittlement trend of the aged Korean RPVs using accumulated surveillance data.

2. RPV surveillance program in Korea

The RPV surveillance program has been in operation for all NPPs in Korea since 1979. Korean nuclear regulatory rules for RPV surveillance are based on 10 CFR 50. The surveillance capsules containing prefabricated specimens are installed in capsule holders attached to inner structures of the reactor vessels. The lead factor for the Westinghouse type reactors in Korea, which is the ratio between the neutron flux at the capsule and the maximum flux at the vessel's inner wall, ranges from 2.0 to 3.8. A series of surveillance tests have been conducted by the Korea Atomic Energy Research Institute. Information such as surveillance data, material information, and neutron irradiation conditions were obtained from the surveillance tests reports. Five surveillance campaigns for each of the five RPVs made of SA533B-1 steel plates and welds have been conducted, and thus the total number of TTS datasets that were obtained from the campaigns was 25 for the plates and welds equally. The chemistries of the surveillance specimens are listed in Table 1.

The prediction of embrittlement shift in transition temperature is generally uses the correlations of the measured surveillance Charpy TTSs with the specific chemistry variables and fluences for the materials of interest [8]. Currently, the surveillance Charpy test data are being evaluated using the procedure in the US NRC RG 1.99-Rev. 2 for determining the embrittlement trend curve at the reference temperature (RT_{NDT}). These values are compared with the screening criteria in the requirements for fracture toughness in 10 CFR 50 Appendix G.

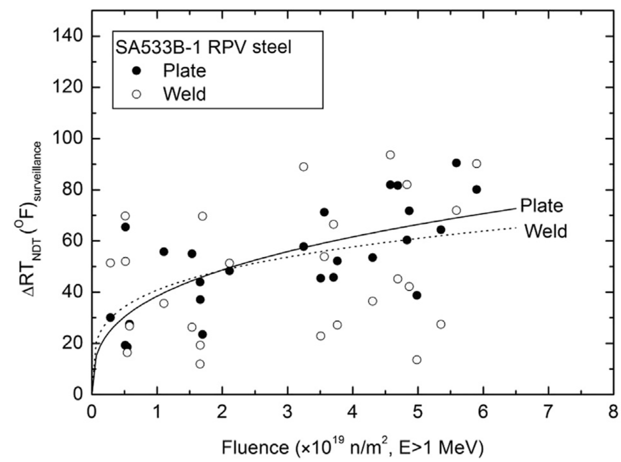

Fig. 1. Relationship between transition temperature shift and neutron fluence for SA533B-1 reactor pressure vessel (RPV) steel plates and welds.

3. Comparison of radiation embrittlement prediction models with Korean RPV surveillance data

The TTS data obtained from the surveillance campaigns are plotted with respect to the neutron fluence in Fig. 1. The embrittlement trend curves for the SA533B-1 plates and welds were arbitrarily constructed through a simple power-law fitting of the data. The fitting curves intersected at a fluence of approximately 2×10^{19} n/cm^2 ($E > 1$ MeV). Beyond this intersection point, the embrittlement values of the plates are larger than those of their welds. However, it is notable that the upper bound of the TTS values for the welds is higher than that for the plates, and the data for the welds are very scattered. The scattering of data plotted in TTS versus the neutron fluence graph is attributed to the variability of the chemical composition and microstructure of the materials, the differences in the reactor operating conditions, uncertainties in the surveillance and other factors [4].

The irradiation embrittlement of the reactor vessel materials is evaluated using the procedure in US NRC RG 1.99-Rev.2. Changes in the TTS or ΔRT_{NDT} due to neutron irradiation are calculated as follows [3]:

$$\Delta RT_{NDT} = (CF)\, f^{(0.28-0.10 \log f)} \quad (1)$$

[where CF (°F) is the chemistry factor and f is the neutron fluence at any depth in the vessel ($\times 10^{19}$ n/cm^2, $E > 1$ MeV)]

Table 1
Chemical composition of SA533B-1 RPV materials (in wt%).

		C	Mn	P	S	Si	Ni	Mo	Cr	Cu	Al
A-2	plate	0.23	1.38	0.004	0.008	0.22	0.63	0.52	0.10	0.05	0.020
	weld	0.11	1.70	0.01	0.01	0.41	0.07	0.50	0.16	0.03	0.009
A-3	plate	0.20	1.36	0.008	0.01	0.26	0.65	0.58	0.05	0.06	0.040
	weld	0.13	1.53	0.012	0.007	0.51	0.18	0.46	0.15	0.02	0.016
A-4	plate	0.23	1.31	0.023	0.014	0.25	0.66	0.58	0.058	0.043	0.040
	weld	0.12	1.54	0.019	0.014	0.50	0.12	0.53	0.066	0.023	0.025
B-1	plate	0.23	1.45	0.012	0.018	0.23	0.52	0.51	0.18	0.054	0.016
	weld	0.13	1.38	0.016	0.011	0.47	0.11	0.50	0.07	0.031	0.015
B-2	plate	0.20	1.50	0.015	0.006	0.20	0.54	0.49	0.16	0.051	0.020
	weld	0.11	1.44	0.018	0.012	0.49	0.11	0.53	0.18	0.029	0.009

The chemistry factors are given as a function of the copper and nickel content for the base metal and welds respectively, in RG 1.99-Rev.2. The forgings and plates in the RG 1.99-Rev.2 model are not distinctive.

An alternative TTS model, given in 10 CFR 50.61a, is a more mechanistic and sophisticated model. It differentiates among plates, welds and forgings. The model is described as follows:

$$\Delta T_{30}(\Delta RT_{NDT}) = MD + CRP \quad (2)$$

$$MD = A \times (1 - 0.001718 \times T_C) \times \left(1 + 6.13 \times P \times Mn^{2.471}\right) \times \varphi t_e^{0.5}$$

[where: A = 1.140 × 10^{-7} for forgings, 1.561 × 10^{-7} for plates, and 1.417 × 10^{-7} for welds; T_C is the coolant temperature, ϕt_e is the effective neutron fluence, $\phi t_e = \phi t$ for $\phi \geq 4.39 \times 10^{10}$ n/cm^2/s and $\phi t_e = \phi t \times (4.39 \times 10^{10}/\phi)^{0.2595}$ for $\phi < 4.39 \times 10^{10}$ n/cm^2/s]

$$CRP = B \times (1 + 3.77 \times Ni^{1.191}) \times f(Cue, P) \times g(Cu_e, Ni, \phi t_e)$$

[where: B = 102.3 for forgings, 102.5 for plates in non-combustion engineering manufactured vessels, 135.2 for plates in combustion engineering vessels, and 155.0 for welds.]

(See 10 CFR 50.61a for more detail) [6].

The TTS values for SA533B-1 plates and welds predicted using RG 1.99-Rev.2 and 10 CFR 50.61a models are plotted in Figs. 2A and 2B with respect to the TTS values obtained from the surveillance campaigns.

The slopes of the secant lines in Figs. 2A and 2B are the ratios of the predicted values to the measured values for the plates and welds. A slope of <1 means the TTS model underpredicts the embrittlement of the RPV materials. The poor correlation between surveillance and model-estimated data in Figs. 2A and 2B is mainly attributed to the scattering of surveillance data. The scatterings in surveillance datasets are attributed to the variability of the chemical composition and microstructure of the materials, the differences in the reactor operating conditions, uncertainties in the surveillance, and other factors.

As shown in Figs. 2A and 2B, the TTS model of RG 1.99-Rev.2 largely underpredicts the embrittlement of both SA533B-1 steel plates and welds. By contrast, the TTS model of 10 CFR 50.61a matches the trend of radiation embrittlement in the SA533B-1 plates and welds relatively well. Even though a slight underprediction still exists in the TTS model of 10 CFR 50.61a, it is within the acceptable range. Recently, Yoon [9] have been trying to develop a Korean RPV-specific TTS model, which will eliminate the slight underprediction by optimizing the coefficients of the TTS model of 10 CFR 50.61a. In conclusion, the 10 CFR 50.61a model predicted the TTS of SA533B-1 plates and welds much better than did the RG 1.99-Rev.2 model.

The TTS model of 10 CFR 50.61a given in Eq. (2), combines two terms: the first term represents the matrix damage (MD), which depends on the irradiation temperature (T_c), material chemical composition and the effective fluence (ϕt_e), and the second term is a Cu-rich phase precipitation hardening (CRP) term that depends only on the latter two factors. Because the Cu content is below 0.072 wt% for all SA533B-1 surveillance materials, it was considered that there was no contribution of precipitation hardening to the transition temperature shifts. Therefore, coefficient A in the MD term determines the performance of the model at predicting the embrittlement of materials containing low level of Cu. The TTS model in 10 CFR 50.61a enhanced the performance in predicting the radiation embrittlement of the plates and welds by making coefficient A 1.37- and 1.24-times larger, respectively, than that value for the forgings. As a result, it was possible to better predict the embrittlement trend of Korean RPVs made of SA533B-1 plates and welds using the TTS model of 10 CFR 50. 61a than that of RG 1.99 Rev.2.

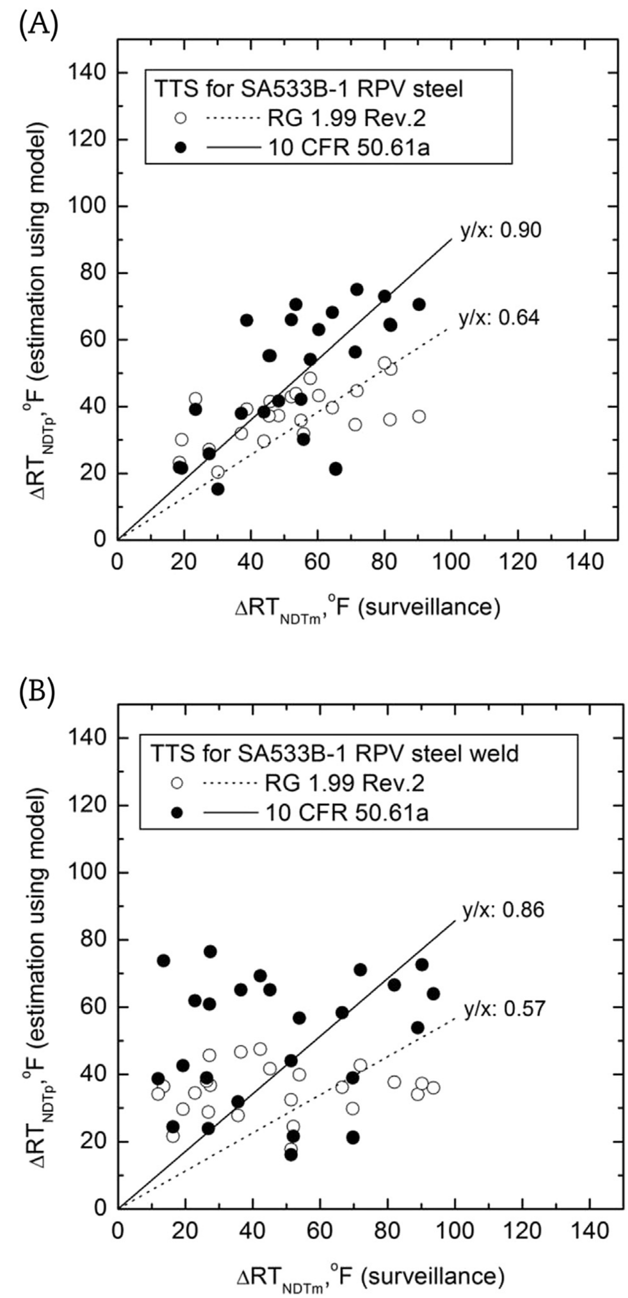

Fig. 2. Comparison of transition temperature shift (TTS) values predicted by models and measured TTS values for (A) SA533B-1 plates and (B) welds.

4. Summary

The applicability of the TTS model in predicting the embrittlement of Korean RPVs made of SA533B-1 plates and welds was investigated through a comparison of the predicted values and measured TTS values obtained from surveillance tests. The RG 1.99-Rev.2 model largely underpredicted the embrittlement of both SA533B-1 steel plates and welds. By contrast, the TTS model of 10 CFR 50.61a matched the trend of radiation embrittlement in the

SA533B-1 plates and welds relatively well. The prediction performance of 10 CFR 50.61a was enhanced by considering differences in radiation embrittlement sensitivity among the different types of RPV materials.

Conflicts of interest

All authors have no conflicts of interest to declare.

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References

[1] IAEA, Integrity of Reactor Pressure Vessels in Nuclear Power Plants: Assessment of Irradiation Embrittlement Effects in Reactor Pressure Vessel Steels, IAEA Nuclear Energy Series No. NP-T-3.11, International Atomic Energy Agency, 2009.

[2] C.E. Carpenter, B.J. Elliot, A.L. Hiser Jr., M.J. Khanna, A.D. Lee, J. Medoff, M.A. Mitchell, S.C.F. Sheng, K.R. Wichman, J.K. Strosnider, W.H. Bateman, Reactor Pressure Vessel Status Report, NUREG-1511, Supp.2, US NRC, 2000.

[3] US NRC, Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, 1988.

[4] J.A. Wang, N.S. Rao, New methodologies for developing radiation embrittlement models and trend developing radiation embrittlement models and trend curves of the Charpy impact test data, in: M.L. Grossbeck (Ed.), Effects of Radiation on Materials, ASTM STP 1447, ASTM International, West Conshohocken, PA, 2003, pp. 1–17.

[5] E.D. Eason, G.R. Odette, R.K. Nanstad, T. Yamamoto, A physically-based correlation of irradiation-induced transition temperature shifts for RPV Steels, J. Nucl. Mater. 433 (2013) 240–254.

[6] U.S. Government Publishing Office [Internet]. Available from: https://www.gpo.gov/fdsys/pkg/CFR-2012-title10-vol1/pdf/CFR-2012-title10-vol1-sec50-61a.pdf.

[7] M. Kirk, Cayetano S. Santos, E. Eason, J. Write, G.R. Odette, Updated Embrittlement Trend Curve for Reactor Pressure Vessel Steels (Paper # G01-5), Transactions of SMiRT 17, Prague, Czech Republic, August 17-22, 2003.

[8] W.L. Server, R.K. Nanstad, Integrity and embrittlement management of reactor pressure vessels in light-water reactors, in: N. Soneda (Ed.), Irradiation Embrittlement of Reactor Pressure Vessels in Nuclear Power Plants, Woodhead Publishing, Cambridge, UK, 2015, pp. 132–155.

[9] J.H. Yoon, Development of Advanced Technology for Enhancing Long Term Operation Safety of Nuclear Materials, KAERI/RR-3929/2014, KAERI, 2015.