Methodology research on prediction for operating lifetime of PWR RPV

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1 ABSTRACT

The residual lifetime of the operating PWR NPP is always concerned by the owners and the RPV plays a critical role in the lifetime prediction of PWR NPP due to irradiation induced embrittlement of the active core beltline and its non-replaceablity. The PWR NPP in question have been operating over half of the initial design lifetime in China and overall survey and investigation for the RPV have been carried out focusing on the Time Limited Aging Analysis (TLAA). Based on the assessment of the pressure - temperature limits on maintaining the reactor coolant pressure boundary, integrity of RPV under the potential pressurized thermal shock (PTS) and the upper shelf energy (USE) of irradiated beltline, the residual lifetime of RPV is predicted in this paper.

2 INTRODUCTION

The residual lifetime of the operating PWR NPP is always concerned by the owners and the RPV plays a critical role in the lifetime prediction of PWR NPP due to irradiation induced embrittlement of the active core beltline and its non-replaceablity. The PWR NPP in question (it is named as a specific NPP in this paper later) have been operating over half of the initial design lifetime and overall survey and investigation focusing on the Time Limited Aging Analysis (TLAA) for the RPV have been carried out.

The reactor coolant transient monitoring and irradiation surveillance data have been collected to preliminarily estimate the cumulated fatigue damage for the high stress concentrated regions and irradiation embrittlement for the active core beltline of RPV. It has been revealed that the irradiation embrittlement is the most significant aging mechanism for the specific NPP RPV comparing with others, for example the fatigue damage etc.

Based on the above knowledge the efforts were moved on estimation of irradiated material, the transition temperature enhance and the upper shelf energy decline. Since more than two capsule specimens have been withdrawn the specific transition temperature RT_{PTS} in the end of lifetime (EOL) of NPP can be derived using the specimen tests data of the specific NPP and the results have been compared with the data from the NPP final safety analysis report (FSAR), which was predicted in design stage by the approach of R.G 1.99Rev.2. Furthermore, the maximum allowable transition temperature has been estimated under the typical pressurized thermal shock (PTS) for the beltline with postulated axial surface semi-elliptical flaw. According to the balanced fuel cycles verified by dose measurement from the withdrawn capsule the fluence for the end of design lifetime and the predicted extension lifetime of the specific NPP is extrapolated. The transition temperatures at different lifetimes are predicted and compared with the maximum allowable transition temperature [RT_{PTS}] obtained from PTS analysis. Finally, the residual lifetime of RPV has been predicted based on the above investigations. It is discovered from the diagram of RT_{NDT} versus NPP operating years (EFPY) that the transition temperature increases with the fluence but the slope become lower with the increasing of fluence.

In other aspect, pressure - temperature limits (P-T limits) for the primary reactor coolant system during heat up, cool down and normal operation of the NPP have been updated in accordance with the withdrawn specimen tests. Because the NPP was designed and constructed about 20 years ago, K_{IR}, the lower bound of arrest and static toughness, was used to determine the P-T limits. It is suggested that the K_{IR} can be replaced by the static toughness K_{IC} for P-T limits calculation to widen the NPP's operating window if the national nuclear authority approves those changes.

3 CODES, REGULATIONS AND WORKING SCHEME

Based on the fact that the specific PWR NPP was mainly designed, manufactured, constructed and inspected in accordance with the US codes or equivalent ones, the assessment of RPV integrity and prediction of its residual lifetime were performed following the relevant national nuclear safety laws (HAF103, 2004), US Code of Federal Regulations (10CFR54, 1995), (Appendix G to 10CFR50, 1997), (Appendix H to 10CFR50, 1997), (10CFR50.61, 1996), NRC Regulatory Guides (US NRC R.G 1.99, 1988), ASME B&PV Code (Appendix G to ASME B&PV Code XI, 2001) and IAEA guidelines (IAEA-EBP-WWER-08, 2005).

According to the requirements of the Codes and regulations the detailed working scheme is stipulated as shown in Figure 1. The implementary procedure of integrity assessment and lifetime prediction for RPV are divided into two main aspects: one is the Time-Limited Aging Analyses (TLAAs), which mainly focus on the RPV physical aging issues and the other is the Aging Management Program (AMP), which mainly focuses on installing or updating the AMP as per the latest or proper codes and regulations.

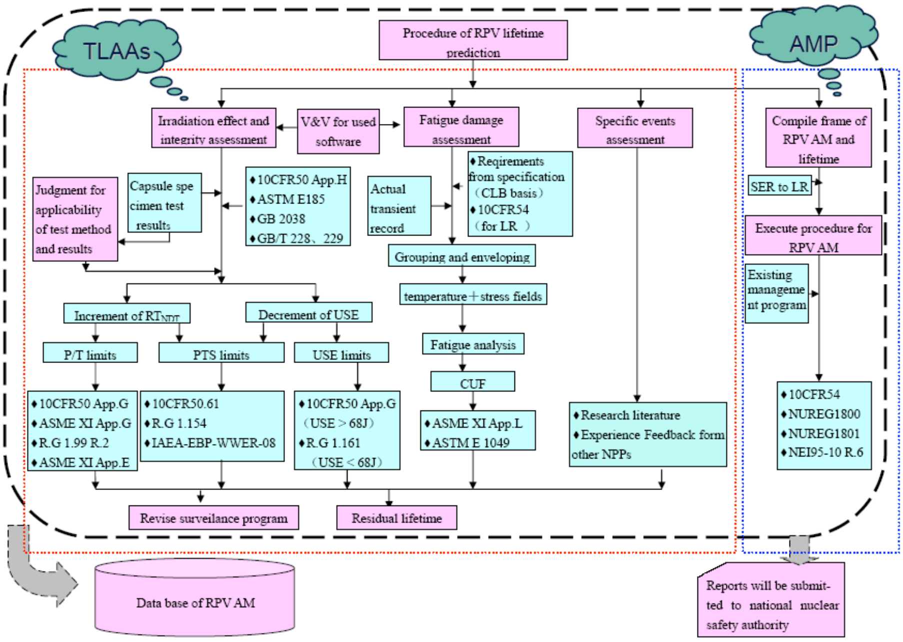

Figure 1. Implementary procedure of integrity assessment and lifetime prediction for RPV

4 DETERMINATION OF INCREMENT OF REFERENCE TEMPERATURE

The increment of reference temperature (RT_{NDT}) is a part of appearances of embrittled RPV material due to neutron irradiation. There are two approaches to calculate R T_{NDT}. For the NPP which has not been put into operation or the number of withdrawn capsule specimens is less than two, the approach recommended in R.G 1.99 Rev.2 should be used and the chemistry factor can be obtained from the tables listed in R.G 1.99. And for the specific plants there are more than two withdrawn capsule specimens and their test results have been obtained, so the specific approach is available for getting the RT_{NDT} related to the fast neutron (E > 1MeV) fluence and Charpy tests of the capsule specimens.

4.1 Approach using R.G1.99Rev.2 table

According to the requirements of R.G 1.99Rev.2 the adjusted reference temperature of irradiated RPV material is expressed as follows:

$$ART=RT_{NDT(U)}+ \quad RT_{NDT}+M \tag{1}$$

$$\Delta RT_{NDT} = (CF)\, f^{(0.28-0.10\log f)} \quad (2)$$

$$M = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2} \quad (3)$$

f is fluence at deepest point along the crack front of the flaw assumed in the analysis, 10^{19} n/cm^2. The fluence at inner surface was applied conservatively instead of that at deepest point of flaw.

CF is chemistry factor related to copper and nickel content and given in R.G 1.99. If there is no information available, 0.35% copper and 1.0% nickel should be assumed, °C;

M is margin to conservatively uncertainties, °C;

$_U$ and are the standard deviation for the initial RT_{NDT} and the RT_{NDT} respectively. If there is no information available, the recommended values are given in R.G 1.99.

For the specific NPP, 32°C of ART was obtained in end of lifetime (EOL) as per R.G 1.99 and it is the same as the RT_{PTS} defined in 10CFR50.61.

4.2 Approach using surveillance results

For the case that there are more than two withdrawn capsules, the above *CF* and *M* can be determined by the test results and the specific RT_{PTS} for the specific RPV can be obtained also.

As more than two capsules had been withdrawn for the specific NPP RPV, the following calculation formula can be employed based on the test results as per R.G 1.99 or 10CFR50.61.

$$CF = \frac{\sum_{i=1}^{n}\left[A_i \times f_i^{(0.28-0.10\log f_i)}\right]}{\sum_{i=1}^{n}\left[f_i^{(0.56-0.20\log f_i)}\right]} \quad (4)$$

A_i is the T_{41J} for capsule *i*;

f_i is the fluence for capsule *i*;

M is determined based on test error (uncertainties).

The ART in EOL (or RT_{PTS}) is 24°C using the parameters from tests to substitute the corresponding ones in the formulae mentioned in Section 4.1. It is observed that the ART using recommended by R.G 1.99 is higher than that from specific plant tests and more conservative.

5 PRESSURE – TEMPERATURE LIMITS

The 10CFR50.60 put forward the acceptance criteria for fracture prevention measures for NPPs during normal operation. Appendix G to 10CFR50 provides the general basis for pressure – temperature (P-T) limits and requires that P-T limits must be at least as conservative as the value obtained according to Appendix G to Section XI of the ASME Code.

The specific RPV was designed about 20 years ago in accordance with codes and regulations of the version of that time. P-T limits were compiled following Appendix G to Section III of the ASME Code and K_{IR} was used for the fracture estimation. The P-T limits for the specific RPV have been recalculated using K_{IC} as per appendix G to Section XI of the ASME Code and the comparison has been made. It is noticed that the new curve is on the left side of the old one around 15°C as shown in Figure 2. The new P-T limits are quite beneficial for the NPP operation during the approaching EOL or even the extended lifetime (beyond design lifetime). The new operating window can be derived combining the P-T limits of RPV, saturation curve and operation limits for RCP etc. and the judgement should be made whether the new operating window has enough space for continued operation.

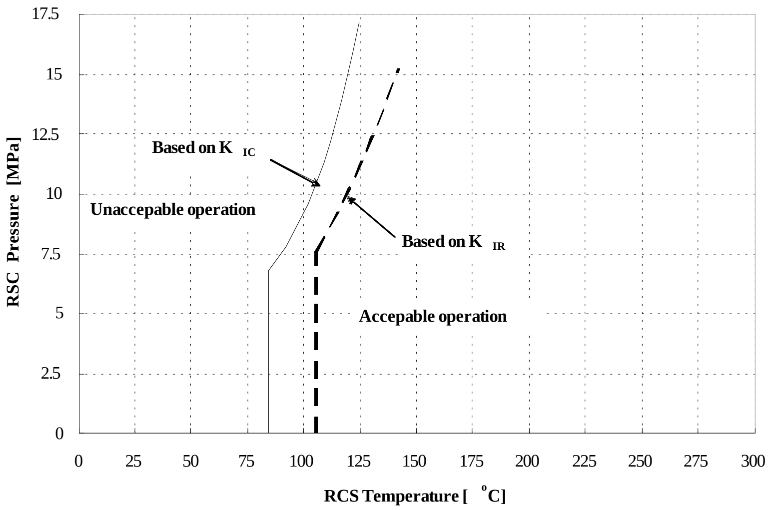

Figure 2. Comparison between old and new P-T curves

6 PRESSURIZED THERMAL SHOCK

The projected values of RT_{PTS} have been determined for the beltline of the specific NPP RPV in Section 4 and are less than the screening criterion of 132°C specified by 10CFR50.61. That means the risk due to PTS events is acceptable for US NRC if the expected penetration frequency (probability) of through-wall crack is less than 5×10^{-6} per reactor year. But the problem is that the regulation for PTS has not been established in China until now. After the detailed investigation and research it will be determined whether the USA screening criteria are fully available for the RPV of NPP in China. Facing this status in China, the PTS analysis procedures recommended by IAEA (IAEA-EBP-WWER-08, 2005) are employed to the specific RPV for PTS analysis also.

6.1 Models and methods

Since there are no welds in the active core beltline of the shell of RPV and the axisymmetrical loading conditions are assumed without considering the plume effect, the breaking cladding axial surface semi-elliptical flaw was postulated near the inner surface of beltline wall. The flaw depth is taken as one tenth of wall thickness and the aspect ratio is one third. The illustration sketch is shown in Figure 3.

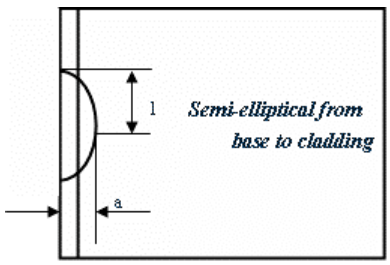

Figure 3. Postulated flaw for PTS analysis

6.2 Material properties and PTS events

Since the lead facors of capsules are not great than two and the fluences of all the withdrawn specimens have not reached the level equivalent to the fluence in EOL yet till now, the postulated irradiated materials are applied to the PTS structural analysis assuming the yield strength of beltline will enhance by 20% in EOL based on the tested material properties of unirradiated specimens except for cladding. The fracture toughness of beltline in EOL is obtained by shifting a T from unirradiated fracture toughness curve.

The PTS events are mainly characterized by a rapid cooldown in RCS with a high primary system pressure or re-pressurization. Such events depend strongly on the specific plant and system configurations, systems operation, and operator actions. The best way to determine the PTS events for RPV integrity assessment is to perform the event sequences analysis based on PRA technology. For the specific NPP, such a PRA for PTS purpose has not been fulfilled. Alternatively, the PTS events are selected referring to the international research achievements and the experience feedback from the other NPPs. The SB-LOCA and LB-LOCA are chosen from the NPP design transients as the typical PTS events and additionally, a typical PTS with re-pressurization is supplemented.

6.3 Fracture analysis and integrity assessment

The virtual crack extension (VCE) method is employed to obtain the J-integral at the deepest and 2mm below interface points. The effects of thermal loads and pressure loads on the faces of the crack are considered. The stress intensity factor is derived from J-integral.

The integrity assessment for RPV is performed using the maximum allowable reference temperature [RT_{PTS}] which is obtained by tangent approach. The different [RT_{PTS}] can be determined for the SB-LOCA, LB-LOCA and typical PTS event with re-pressurization and shown in Figures 4, 5 and 6.

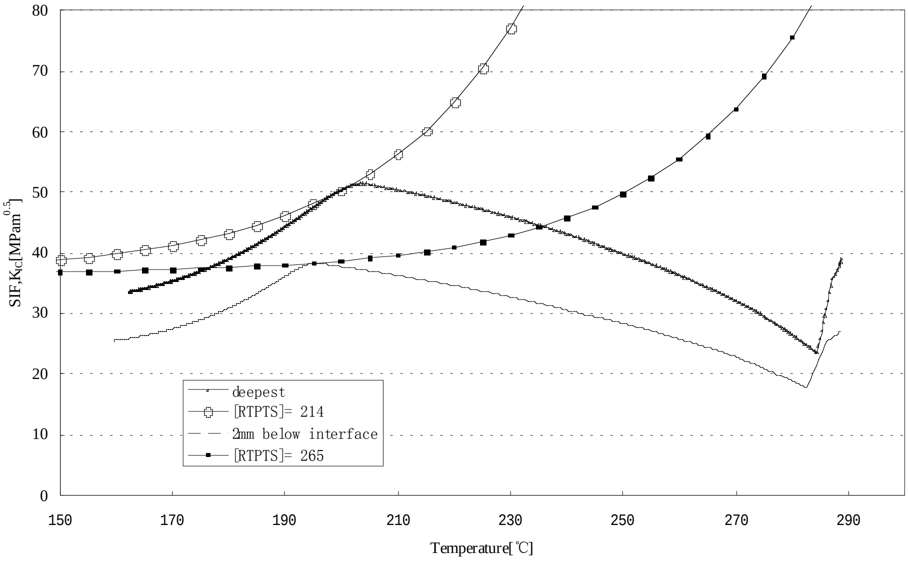

Figure 4. [RT_{PTS}] for SB-LOCA

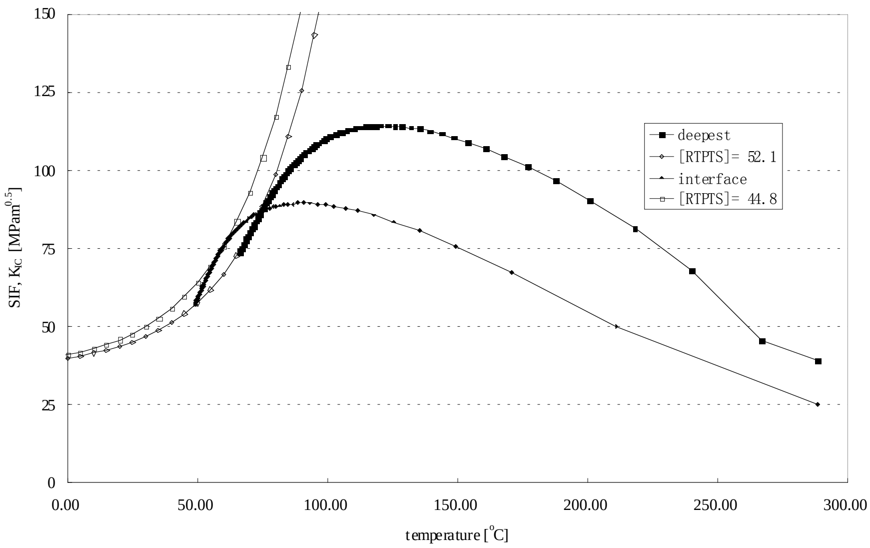

Figure 5. [RT_{PTS}] for LB-LOCA

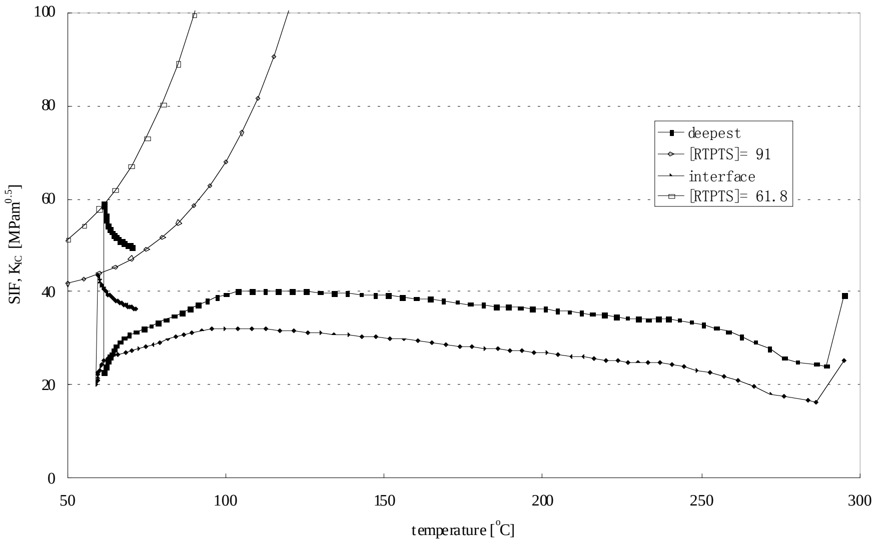

Figure 6. [RT_{PTS}] for Typical PTS event with re-pressurization

According to the PTS analysis the critical [RT_{PTS}] is 44.8 ^{o}C. As calculated in Section 4 the RT_{PTS} is 32^{o}C or 24^{o}C are less than 44.8 ^{o}C. So the structural integrity of RPV within its design lifetime can be maintained relied on the current test data and calculation.

7 CHARPY UPPER SHELF ENERGY

According to the results from Charpy impact tests of the several withdrawn capsule specimens the decrement of upper shelf energy is a small amout (about 16J) and the USE are far higher than the limit of 68J.

8 RESIDUAL LIFETIME ASSESSMENT

Based on the measurement and calculation of fast neutron flux the modified extrapolation factor can be derived and applied to calculate the peak fast neutron flux on inner surface of base metal for the balance fuel recycle. The relationship between ART and NPP operating years (EFPY) is shown in Figure 7.

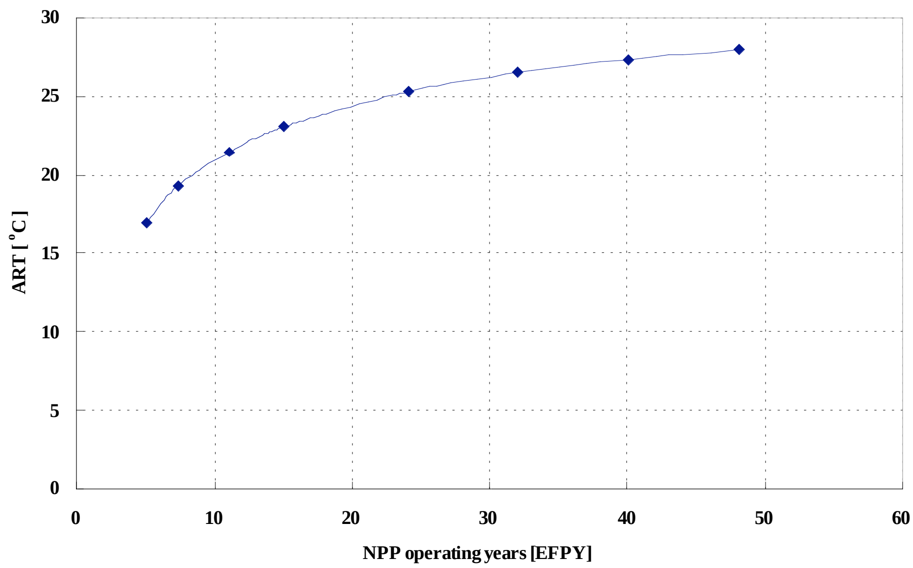

Figure 7. Relationship between ART and EFPY

9 CONCLUSION

The current test and calculation results show that the structure integrity of the specific NPP RPV will be maintained throughout the design lifetime and even for the potential extended lifetime.

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