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 and overall survey and investigation for the RPV have been carried out focused 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.

1. 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 (we call it as 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 transient monitoring and surveillance data have been collected to preliminarily estimate the cumulated fatigue damage for the high stress concentrated regions and irradiation embrittlement for the beltline of RPV. It has been revealed that the irradiation embrittlement is the most significant aging mechanism comparing with others, for example the fatigue damage etc.

Based on the above knowledge the efforts are moved on estimation of irradiated material, the enhance of transition temperature and upper shelf energy decline. Since more than two capsule specimens have been withdrawn the specific transition temperature $R_{\text{PTS}}$ in the end of lifetime (EOL) of NPP can be derived using the specimen tests data of the specific NPP and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 1.99. Furthermore, the maximum allowable transition temperature has been estimated under the typical pressurized thermal shock (PTS) for the beltline with surface axial semi-elliptical flaw and the flaw depth is 10 percent of the thickness according to the code requirements and the measuring error of NDT for such flaws. According to the balanced fuel cycles verified by dose measurement from the withdrawn capsule the fluence at the EOL and the predicted extension lifetime of the NPP in question is extrapolated. The transition temperatures at different lifetimes are predicted and compared with the maximum allowable transition temperature 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 $\Delta R_{\text{NDT}}$ versus fluence 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.

2. Codes, Regulations and Working Scheme

Based on the fact that the PWR NPP in question was mainly designed, manufactured, constructed and inspected in accordance with the US codes or equivalent ones, the assessment of RPV integrity and predicion of its residual lifetime were performed following the relevant national nuclear safety laws[1], US Code of Federal Regulations[2, 3, 4, 5], NRC Regulatory Guides[6], ASME B&PV Code[7] and IAEA guidelines[8].

According to the requirements of the Codes and regulations the detailed working scheme is stipulated as shown in Fig.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.

3. Determination of increment of reference temperature

The increment of reference temperature (\( \Delta R_{NT} \)) is a part of appearances of embrittled RPV material due to neutron irradiation. There are two approaches to calculate \( \Delta R_{NT} \). 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 \( \Delta R_{NT} \) related to the fast neutron (E > 1MeV) fluence and Charpy tests of the capsule specimens.

3.1. Approach using RG table

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

\[
\Delta R_T = R_{NTD}(U) + \Delta R_{NDT} + M
\]

\[
\Delta R_{NDT} = (CF) f^{0.28-0.10\log f}
\]

\[
M = 2\sqrt{\sigma_f^2 + \sigma_\Delta^2}
\]
$f$ is fluence at deepest point on the crack front of the flaw assumed in the analysis, $10^{19}$ n/cm$^2$;

$CF$ 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;

$\sigma_u$ and $\sigma_j$ are the standard deviation for the initial $RT_{NDT}$ and the $\Delta RT_{NDT}$ 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.
FIG. 1. Implementary procedure of integrity assessment and lifetime prediction for RPV
3.2 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.

\[
CF = \frac{\sum_{i=1}^{n} [A_i \times f_i^{0.28-0.10 \log f_i}]}{\sum_{i=1}^{n} [f_i^{0.56-0.20 \log f_i}]}
\]

\( A_i \) is the \( \Delta T_{FP} \) 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 24.12 °C using the parameters from tests to substitute the corresponding ones in the formulæ mentioned in Section 3.1. It is observed that the ART using recommended by R.G 1.99 is higher than that from specific plant tests and more conservative.

4. 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 Fig.2. The new P-T limits are quite beneficial for the NPP operation in 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.
5. Pressurized thermal shock

The projected values of $RT_{PTS}$ have been determined for the beltline of the specific NPP RPV in Section 3 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 $5 \times 10^{-6}$ 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[8] are employed to the specific RPV for PTS analysis.

5.1. 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 through cladding axial surface semi-elliptical was postulated in the 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 Fig.3.
5.2. Material properties and PTS events

Since the lead factors of capsules are not greater 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 initial unirradiated specimens. The fracture toughness of beltline in EOL are obtained by shifting the $RT_{PTS}$ 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.

5.3. Fracture analysis and integrity assessment

The virtual crack extension (VCE) method are 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 the following formula:

$$K_J = \sqrt{\frac{\pi E J}{1 - \nu^2}}$$

The integrity assessment for RPV are 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 Fig.4, 5 and 6.
FIG. 4. $[RT_{PIS}]$ for SB-LOCA

FIG. 5. $[RT_{PIS}]$ for LB-LOCA
According to the PTS analysis the critical \( [RT_{PTS}] \) is 44.8 °C. As calculated in Section 3 the ART in EOL (or \( RT_{PTS} \)) is 32 °C or 24.12 °C are less than 44.8 °C. So the structural integrity of RPV within its design lifetime can be maintained relied on the current test data and calculation.

6. 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 amount (about 16J) and the USE are far higher than the limit of 68J.

7. 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 fluence is shown in Fig.7.
Considering the average 80% power factor, the relationship between fluence and operation years also can be expected as shown in Fig.8.

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

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[4] USA, Appendix H to 10CFR50 “Reactor Vessel Material Surveillance Program Requirements”
[5] USA, 10CFR50.61 “Fracture Toughness Requirements for Pressurized Thermal Shock Events”