Reconstitution of Time-limited Ageing Analyses for Justification of Long-Term Operation of Paks NPP

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ABSTRACT

The WWER-440/213 type units of Paks Nuclear Power Plant in Hungary were designed, manufactured in accordance with the Soviet codes and standards valid in late sixties and seventies. Design lifetime of the units was 30 years and the operational licence had been limited to that term. A very important aspect of justification of safe operation beyond this term is the review and validation of the time-limited ageing analyses. In the specific case of Paks NPP the designer made time limited ageing analyses are partly missing, partly inadequate because of lack of proper documentation or obsolescence. For the solution of this issue the necessary analyses (stress calculations, PTS, fatigue, etc.) have to be performed by state of the art methods, which involves also verification of strength calculations of most important structures and components. The goal of this paper is to describe the specific aspects of reconstitution of time limited ageing analyses and verification of strength calculations for Paks NPP WWER-440/213 units. Especially the scope time limited ageing analyses, their methodology and peculiarities of the adaptation of ASME BPVC will be considered in the paper.

INTRODUCTION

The WWER-440/213 units at Paks NPP were designed and constructed at the technical level of late seventies early eighties. Design, operation and maintenance practice of the units had been determined in great extent by the Soviet rules and regulations. Beginning from mid eighties safety deficiencies of WWER-440/213 had been identified. The subsequent comprehensive safety upgrading programmes resulted in essential achievements, e.g. the core damage frequency has been decreased more than order of magnitude, and the level of safety of these units is comparable today to the safety level of PWR's of same vintage around the world. In the same time the national regulation had been developed in accordance with international regulatory practice.

The design life-time of the NPP Paks WWER-440/213 units is 30 years and the operational licence had been limited to that term. The Paks Nuclear Power Plant strategy is to extend the operational lifetime of the plant and renew the operational license for 20 years over the designed and licensed lifetime. For the licence renewal the U.S. Nuclear Regulatory Commission approach and the 10 CFR Part 54 was adapted, which requires integrated plant assessment focusing on review of ageing management programmes and review and validation of time-limited ageing analyses. The adaptation of the U.S. licence renewal rules has been done taking into account the peculiarities of national regulation.

An overview of preparation of licence renewal is given in [1].

Review and validation of the time limited ageing analyses (TLAAs) is an essential element of the justification of licence renewal. However this task seems to be rather complex in case of Paks NPP, similar to any other WWER-440/213 plant. The issue is related to the availability of design base information and incompleteness of the delivered design documentation.

Recently Paks NPP performed a design base (DB) reconstitution project in the frame of the renewal of the Final Safety Analysis Report (FSAR). The project consisted of collection and review of original design base information and updating those taking into account all essential changes in the licensing requirements. The collection and review of supplied documentation in frame of DB reconstitution provided some understanding of the design. However part of design assumptions, inputs and the design conditions remained unknown.

Considering the available information, review and validation of TLAAs required for licence renewal turned to a complex issue. Often the final results of the analyses are known only, which are documented sometimes just in form of limitations in the Technical Specification. In some cases the analyses are presumably obsolete. It had to be recognised that the recovery and review of original TLAAs would be insufficient for justification of licence renewal because of essential changes in the design bases. The TLAAs have to be reviewed and verified for most important structures and components (SCs) by control calculations using state-of-the-art methods. In many cases the time limited ageing analyses have to be newly performed in accordance with the recent requirements and guidelines. Review, validation and reconstitution of TLAAs implicate also verification of existing strength calculations for selected most important SCs.

This is a comprehensive programme for justification of licence renewal of Paks NPP flow-chart of which is shown in figure 1.

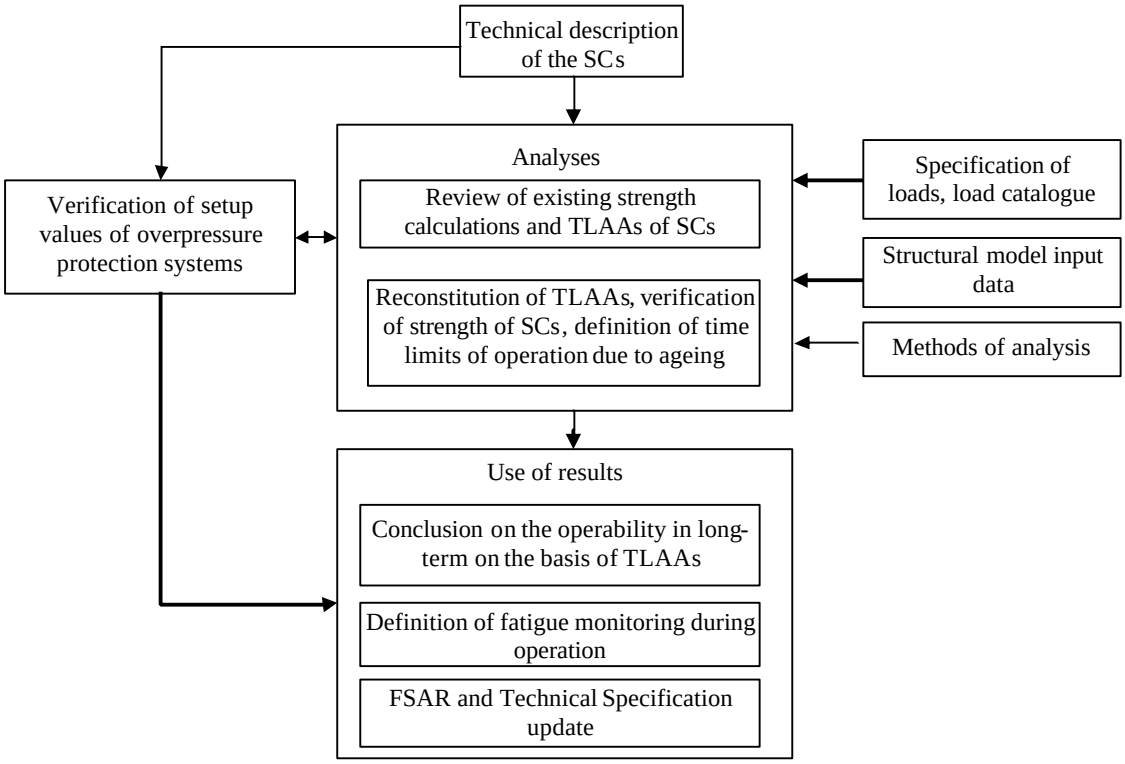

Fig. 1 Flow-chart of TLAA reconstitution

In this paper some aspects of the work outlined above will be discussed demonstrating the importance of use of state-of-the-art methodology and proper understanding of the design, manufacturing, construction and operational peculiarities of the WWER-440/213.

SCOPE OF REVIEW, VALIDATION AND RECONSTITUTION OF TLAAs

Scope of review, validation and reconstitution of TLAAs is defined by scope of SCs to be examined and scope of analyses to be performed.

Review, validation and reconstitution of TLAAs shall cover the SCs of Safety Class 1 and 2 in accordance with Hungarian regulations. Although the concept of definition of Safety Classes in Hungarian regulations not fully identical with those in U.S. NRC 10CFR50 and Regulatory Guide 1.26 (and it is far away from Regulatory Guide 1.201), the scope mentioned above is practically overlapping with Class 1 and 2 as per U.S. regulation.

The scope of SCs to be examined includes:

- reactor pressure vessel (RPV),
- steam generators (SG),
- pressurizer vessel,
- cases of the main circulating pumps and the main gate valves,
- other pipes, vessels, pumps, heat exchangers and valves in Safety Class 1 and 2.

Considering the required analyses, besides of the review and validation of existing TLAAs, verification of essential strength calculations has to be performed for the given above scope of SCs. The scope of fatigue analysis to be reviewed or newly performed shall cover the Safety Class 1 and 2 components. Analysis of the reactor pressure vessel includes PTS calculation, and developing new p-T curves. Where appropriate the thermal stratification has to be analysed. Justification of safe operation for 50 years includes also checking the safety margins for 60 years of operation. Obviously, the scope of analyses required by Hungarian regulation exceeds the scope of regular review of TLAAs performed for licence renewal. If it is necessary, modification of in-service-inspection, maintenance, testing programmes have to be identified and developed for management of aging during the period of extended operation.

GENERAL METHODOLOGY

The Hungarian Nuclear Safety Regulations have to be followed while performing the analyses. Several guidelines had been published containing applicable methodological information how to comply with requirements. This is the generic framework for performance of review, verification and reconstitution of TLAAs. On the other hand the Hungarian regulations require the use of state-of-the-art methods, codes and standards while performing design and any analyses. ASME Boiler & Pressure Vessel Code, Section III: Rules for Construction of Nuclear Facility Components, edition 2001 (hereafter, ASME BPVC III) had been selected for the reconstitution of TLAAs and associated strength verification. Use of ASME code is a generic decision of the Paks NPP. For example, currently the ISI programs at Paks NPP are under comprehensive review in order to modify them to meet the requirements of the ASME BPVC Section XI. The code selection required proper understanding of the both the Russian (Soviet) design standards and the ASME code. Different studies had been performed for ensuring the adequacy of ASME implementation for WWER-440/213 design (see e.g. [2]).

In the framework of preparation of licence renewal and in overall context with ASME adaptation Paks NPP developed a detailed specification for review, validation and reconstitution of TLAAs. A comprehensive and very detailed Methodology and Criteria Document has been developed on the basis of the utility specification [3]. This describes the routine application of ASME BPVC III as it is shown in figure 2 and also provides additional considerations with regard to specific aspects of strength verification and validation/reconstitution of TLAAs.

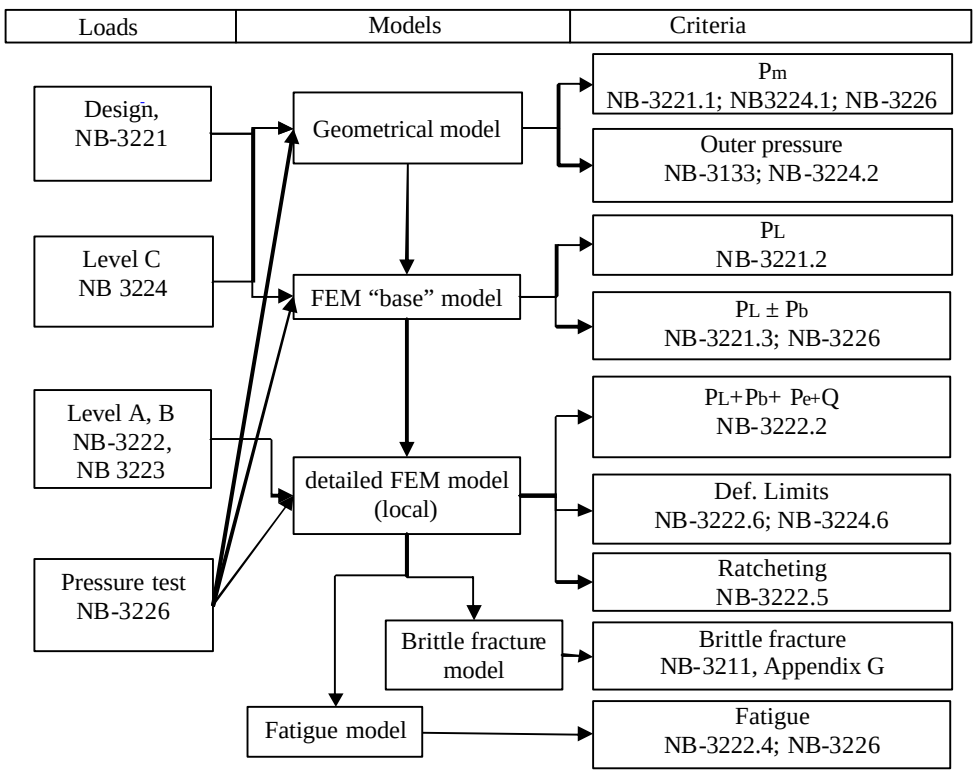

Fig. 2 Flow-chart of use of ASME BPVC III

It is obvious that unavoidable deviations shall between a routine ASME BPVC III analysis and recent methodology of reconstitution of TLAAs for Paks NPP and associated calculations. These deviations are caused either by specific regulatory environment or by technical peculiarities of the WWER-440/213 units. In the frame of this paper some of these specific aspects features of TLAA validation and reconstitution will be discussed.

SPECIFIC ASPECTS OF THE METHODOLOGY

Material Properties

An important issue of the ASME BPVC III application is related to definition of material properties. With this respect the manufacturer's national and industrial standards, designers or manufacturer's technical specifications and former Hungarian regulatory rules related to materials, construction (welding) and quality assurance should be considered as

relevant. Basic source for the identification of material used for manufacturing is the equipment delivery documentation (called passports) supplied to Paks NPP. This documentation contains the drawing numbers, tables for material properties, also the as it is values measured during installation. Material identified in this documentation shall be considered in the analysis. In case of missing of passport of the given equipment, the designer technical specification for material has to be used. If the specification is missing, the data has to be sought in the former Soviet industry standards (GOST). If the relevant GOST could not be recovered, the Russian code PNAE G-7-002-86 [4] shall be applied.

A Methodology and Criteria Document provides the applicable material data collected, reviewed and systematised on the basis of material specifications and standards used during manufacturing and construction [3].

In some cases the certification of material properties could be an issue. For example the contemporary Soviet regulations for pipelines required hand over of quality certificates to the buyer only in case of an operating pressure above 100 bar. Below the value of 100 bar it was sufficient to keep the original quality certificate of the product at the supplier. The quality control organization of the manufacturer confirmed existence of quality certificates and their conformity to standards in the operating manual of the product. This could be considered as non-compliance with NCA-1221.1. However this kind of non-compliance is rather formal, it does not fundamentally affect the validity of results of the analysis of Class 2 components.

Dimensions and tolerances

With respect to the dimension and tolerance ranges overlapping exists between ASME BPVC III Chapters (see for example NB-4221.1) and the contemporary Soviet and Hungarian standards. In critical cases, compliance with the ASME criterion of difference between the highest and the minimum diameters has to be inspected. Manufacturing tolerances concerning each product can be obtained from the drawing of the product or from Chapter 4.1 of PNAE G-7-002-86 [4].

Consideration of Aging Effect on the Wall Thickness

In order to specify the necessary wall thickness, the design allowances have to be determined in accordance with Chapter 4.1 of Russian code PNAE [4]. It has to be pointed out that the as designed corrosion allowance corresponds to 30 years of operation. It means that for 60 years the allowance should be double of the existing one.

On the basis of strength calculations cross-sections have to be identified where the difference between calculated and selected wall thicknesses does not exceed the sum of double of the corrosion allowances and manufacturing tolerances according to PNAE. If it is the case the rate of wall thinning due to the general corrosion has to be investigated. For this purpose different information sources exists for original wall thickness and the wall thinning, e.g. construction documentations, records of NDT programmes.

The rate of wall thinning really measured has to be compared with the expected value for operating time elapsed in accordance with PNAE. The value of double corrosion allowance used as criterion for operating time of 60 years should be corrected in proportion with the ration of expected and measured wall thinning. Difference between the real wall thicknesses and the calculated ones should not exceed the sum of corrected corrosion allowance and fabrication allowances. If it is the case, the ageing effect has to be qualified as non acceptable for the operating time of 60 years. In this case the remaining operating time has to be identified.

Load Catalogue

The design input loads and conditions have been reviewed and newly defined for the most important SCs because of amendments of the regulations modifying the DB and extension of initiating events, transient and accident scenarios. The new load catalogue has been completed on the basis of the existing design information, results of analyses performed for the renewed Final Safety Analysis Report, and operational history [5]. Review, verification and reconstitution of TLAAs have to be performed using the load catalogue and forecast for the period of extended operation.

Use of other standards

Beside of the selected ASME BPVC III other standards or technical guidelines might be used for time limited ageing analyses, if it is necessary because of technical features of the WWER-440/213. According to the studies performed one of them is the already mentioned Russian Code PNAE G-7-002-86 [3]. As it will be demonstrated on an example below, in specific cases, the methodology and guidelines developed in frame of European Commission VERLIFE Project "Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs" might be also applied [6].

SPECIFIC ASPECTS OF FATIGUE ANALYSIS

Fatigue Curves

Materials of the equipment of WWER-440/213 within the scope of fatigue analyses are carbon steels, low-alloy steels (ST20, 22K, 15H2MFA, 18H2MFA) and stainless-steels (08H18N10T, 08H18N12T). From the point of view of reconstitution of TLAAs the fatigue curves have very important role.

Consideration has been made for the proper selection of fatigue curves and methodological guidelines have been also developed during the preparatory work [7]. Objective of performed analysis was to justify the applicability of material-specific fatigue curves specified by Russian code PNAE, which could be considered as relevant for the existing materials. For this reason the empirical, theoretical background of fatigue curves has been analysed. The fatigue curves based on minimum standard values in PNAEA have been compared with fatigue curves defined in ASME BPVC III Appendix I. Figure 3 shows that the fatigue curves based on the minimum standard values specified for the WWER-440/213 materials are more conservative than ones defined per ASME. The Interim Fatigue Curves as per NUREG-CR-5999 is also shown in figure 3.

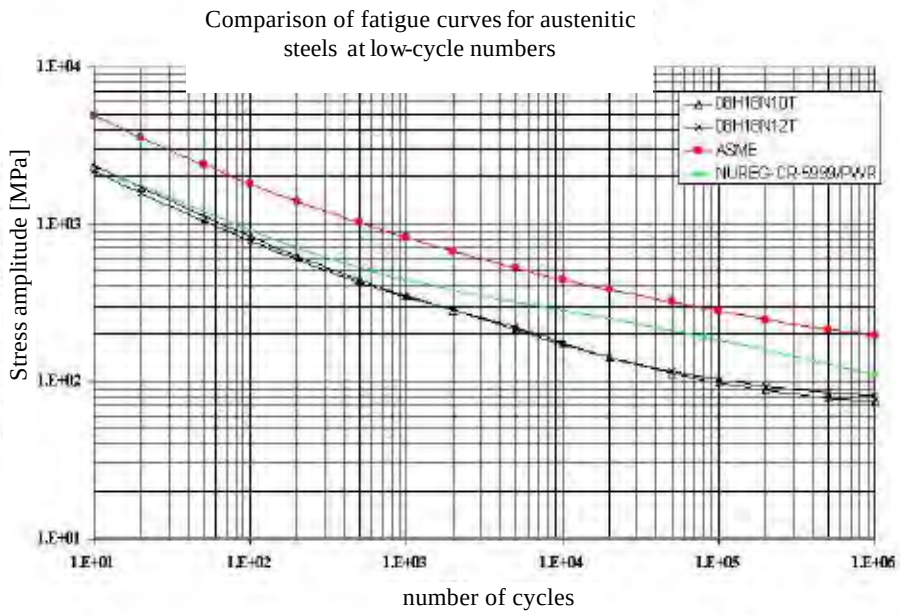

Fig. 3 Comparison of fatigue curves

Based on these studies the fatigue curves to be applied in the analyses are the material-specific curves in PNAEA. The impact of the environment on the fatigue will be accounted by adaptation of current U.S. NRC guidance.

Specification of Fatigue Strength Reduction Factor

In accordance with the relevant Hungarian draft guidelines, the Fatigue Strength Reduction Factor (FSRF) has to be specified as follows:

- In case of component without weld, FSRF might be set equal to the theoretical stress concentration factor.
- If the numerical method and consequently the analysis fully consider the local structural discontinuity, FSRF=1.
- In case of fatigue test of welded joints, FSRF might be taken as a function of category of the welded joint, according to NB-3350.
- In case of welded joints of vessels, FSRF=2 should be taken for welding categories of A, B, C, D. In case of D weld incompletely welded, the value of FSRF=4 is accepted.
- During assessment of welded joints of pipelines, stress indexes according to NB-3683 shall be used.
- If configuration of the weld can not be classified into one of categories mentioned above, then FSRF=5 has to be applied. Usage of value larger than 5 is not allowed. In case of plane-like defects perpendicular to the principal stress, rate of fatigue crack propagation should be specified instead of analysis of fatigue resistance.
- In case of threaded pocket, the value of FSRF=4 has to be used.

Relationships and diagrams included in Enclosure 2 of PNAE serve for specification of stress concentration factors.

Application of ASME XI Appendix L in Case CUF>1

Practically the nuclear power plants operate without repair of cracks detected by means advanced testing processes. However, the analysis of lifetime limit due to fatigue, the analysis and criteria of acceptance used so far, is performed on the basis of the principle of historical inviolability of the limit of CUF less than or equals to 1. This practically eliminate the

chance of occurrence of fatigue cracks in the environments analysed, while occurrence of cracks due to degradation processes not included in standard TLAAs made by designer (for example, local corrosion phenomena), as well as the practice of operation with cracks.

The procedure in ASME Appendix L, applicable in case of a CUF higher than 1, provides an opportunity to consider the above mentioned circumstances of operating nuclear power plants. It makes possible to harmonize the practice with respect of both fatigue crack initiating phenomena considered in design standards and other crack initiating phenomena not managed by design standards.

Condition for the application of this procedure is that no-cracked condition should be demonstrated in the critical cross sections by means of non-destructive material tests.

If it is proved that the criterion of CUF=1 does not complied with for a critical cross section, then the crack propagation test according to L-3000 has to be performed for it.

SPAECIFIC ASPECTS OF BRITTLE FRACTURE ANALYSIS

Generally the brittle fracture analysis has to be performed for all critical cross-sections of the equipment, according to Appendix G.

It has to be examined whether the critical brittle fracture transition temperature of carbon steel material of the component does not exceed, even at the end of the extended lifetime, the value of assumed lowest end temperature of pressure test of the equipment, as well as of transients occurring during normal operation or accidents. If it is the case, it is not necessary to execute specific brittle fracture analysis.

In case of already operating pumps and valves, impact test results for materials are not available; therefore it is impossible to inspect acceptability of aged materials according to ASME.

In this case analysis has to be performed according to the method and procedure developed in frame of VERLIFE Project [6]. Change of transition temperature due to ageing might be calculated (according to both PNAE and VERLIFE) as follows:

Transition temperature of embrittlement T_k can be calculated by the following formula:

$$T_k = K_{ko} + DT_F + DT_T + DT_N$$

where K_{ko} - initial transition temperature, [°C],
DT_F - shift of transition temperature due to irradiation, [°C],
DT_T - shift occurred due to impact of thermal ageing, [°C],
DT_N - shift occurred due to impact of cyclical load, [°C].

The impact of irradiation has to be examined for RPV.

The thermal embrittlement has to be considered only in case of operating temperatures above 150 °C.

Impact of the cyclical load can be considered by a simple explicit formula.

THERMAL STRATIFICATION ANALYSIS

Effects caused by possible thermal stratification have to be analysed in the frame of TLAA reconstitution. This is a specific example of analysis which did not performed by the designer and required today by the regulation.

First step of analysis is a screening of pipelines with respect of possibility of stratification on the basis of own and international WWER experience.

For the critical pipelines the highest and the lowest temperature have to be defined on the basis of design data. Then, using a finite element model, and the formula shown below (NB-3600), an abrupt temperature field will be applied at critical section corresponding to extreme temperature values and the thermal stresses should be calculated. Using the reserves up to CUF=1, calculated for load history, and using the fatigue curve, the allowable cycle number could to be determined. If it exceeds the highest cycle number at the fatigue curve, no further actions needed. However, if it is lower than the highest cycle number at the relevant fatigue curve, then a special temperature monitoring will be recommended, in order to provide an opportunity to precisely specify the highest and lowest temperature values which affect at given part of pipeline. Based on measured temperatures, the real cycle number can be determined. According to the experience obtained up to now in similar analyses the real cycle number will be less than the allowable cycle number.

While applying the temperature jump onto the finite element model, a linear or parabolic temperature distribution for the element could be specified. Consequently, the size of element may significantly affect the results. This aspect and the speed of estimated real temperature-change have to be considered during modelling. In case of an analysis of pipes (NB-3600), the effect of thermal stratification is included in the formula for calculating the peak stress (NB-3653.2).

ANALYSIS OF RPV

Considering the time limit of operation of the RPV, neutron irradiation damage, thermal ageing and low-cycle fatigue decreasing the fracture toughness of the RPV materials should be analysed. According to extensive studies performed by designer and Hungarian experts the pressurised thermal shock (PTS) is the most critical lifetime limiting event for RPV

Effect caused by neutron irradiation is dominating near the core. This part of RPV is most sensitive from the point of view of brittle crack initiation and propagation. At other parts of RPV, e.g. in the vicinity of the nozzles, the stresses might be much higher and the neuron fluence much lower than in the area of beltline region. Such locations are the nozzles and their vicinity, the vicinity of the supporting flange and weld 3/5. According to the studies performed the vicinity of the weld 8/9 is not a critical location for PTS, nevertheless it is included into the analysis. The reactor main flange and its vicinity not exposed to high neutron radiation, consequently it is not critical from point of view of PTS. The same is valid for RPV head and the nozzles on it. These parts of RPV are not exposed to high neutron radiation and the stresses from internal pressure are considerably lower than in the nozzle region.

Methodology of PTS analysis is described in dedicated Hungarian regulatory guidelines. Based on the utility specification and the regulation a Methodology and Criteria document has been developed for the analysis of RPV [8].

The PTS analysis required is completely deterministic. It includes the well-known tasks:

a) Selection of thermo -hydraulic transients;
b) Performance of thermo-hydraulic transient analyses;
c) Performance of reactor-physical calculations to determine the 3D neutron flux distribution in the RPV wall;
d) Evaluation of results of reactor dosimetry;
e) Evaluation of ageing of structural materials;
f) Calculation of temperature, deformation, stress field and fracture mechanics;
g) Evaluation of integrity criteria;

Task f) consists of three main steps:

- thermal field calculation, i.e. calculation of temperature distribution in the vessel wall during the transient as a function of coolant temperature, heat transfer coefficients between coolant and wall, assuming heat conductivity in wall and convective heat transfer between wall and coolant;
- stress calculation, i.e. calculation the deformation and stress fields occurring as a result of temperature transient and pressure in vessel by solving a system of elastic (and plastic) equations;
- fracture mechanics analysis, analysis of stability conditions of cracks (detected or postulated) in the vessel wall under transient conditions.

The analysis has been performed in two phases. In the first phase a simplified and conservative calculation method is used for large number of scenarios in order to screen out the most demanding cases which require more sophisticated and less conservative analysis.

With respect of RPV safety the programme of long term operation includes management of RPV properties of aged structural materials, examination of further surveillance specimens, evaluation of expected effects of the measures affecting integrity (heating of coolant in emergency core cooling system, etc.).

Ageing due to fatigue affects the components of RPV in different extent. The main designer (Gidropress) and the manufacturer (Škoda) performed the basic fatigue analyses for the RPV critical components. In the frame of preparation of licence renewal these analyses will be renewed taking into account the new and expected operating conditions. However, in order to determine time limit of operation of RPV and justification of long-term operation (50 years), it is necessary to account all effects of ageing, i.e. elements critical for PTS have to be checked also for fatigue and the elements critical for fatigue must be reviewed also for PTS.

In the frame of the TLAA reconstitution the critical brittle fracture temperature (T_k) characteristic of the aged condition of the reactor pressure vessel materials has been re-evaluated. The value of neutron fluence in the characteristic points of the capsule assemblies participating in the surveillance programme as well as the values of the relevant neutron fluence distributions in the critical cross sections of the reactor pressure vessels regarding the base material and the weld material of the 5/6 weld, for the 60th and 50th years of reactor operation, presuming 108% and 100% power rates.

CONCLUSIONS

Review, validation and reconstitution of TLAAs are essential part of the justification of safety of long-term operation and licence renewal of Paks NPP. The results of analyses have to be submitted for regulatory control in the frame of Programme of Long-term Operation by the end of 2008.

Review, validation and reconstitution of TLAAs of Paks NPP WWER-440/213 units is a complex task requiring adaptation of state-of-the-arts methods and codes as well as understanding the technical features of the design, manufacturing, construction and operational history.

For the reconstitution of TLAAs and complementary validation of stress calculations ASME BPVC III has been selected. Justification of the applicability of the ASME BPVC III required extensive studies as well as understanding the limits of the consequent code application.

Careful studies have been performed for those aspects of the analyses where the ASME BPVC III applicability might be questionable or even incorrect. Justified solutions are found for the major deviations from the routine code application, e.g. for definition of the material properties including fatigue curves.

Methodology developed for the reconstitution of TLAAs and complementary calculations is a unique trial for application of ASME BPVC III for an operating WWER-440/213 plant since verification of strength calculations made in design phase for the Loviisa NPP in Finland.

Authors believe that the appropriate use of the former Soviet designer codes and standards, the recent Russian code and the VERLIFE methodology and the ASME BPVC III in a procedure of time limited ageing analysis outlined in the paper is worth of discussion within the international community of experts and will result adequate and auditable justification of long-term operation of Paks NPP.

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