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PTS ASSESSMENT -THE BASIS OF LIFE TIME EVALUATION AT NPP PAKS.

by

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Abstract

Plant specific PTS analysis at NPP Paks was performed in the frame of the AGNES (Advanced General New Evaluation of Safety) project.

NPP Paks belongs to the second generation of the WWER-440/213 NPP-s. To verify the safety during transient events and predict the lifetime of the RPV-s several transient cases have been analysed.

The paper summarizes:

- *- the general scheme elaborated for the assessment,*
- *- the safety philosophy used,*
- *- the applied and available codes and methods,*
- *- the ongoing and planned developments*

Keywords: *PTS, life time evaluation, fracture mechanics*

1. INTRODUCTION

The very high tensile stress caused by pressurised thermal shock [PTS] at the vessel inside wall could cause the initiation of the propagation of a pre-existing flaw of small dimensions. It would stop when it reached a zone of higher temperature, where the material toughness is high enough to arrest the running crack. This crack could restart propagating later on during the transient if the pressure is increased, or due to the progressive cooling of the vessel wall. This could ultimately lead to vessel failure after a sequence of arrest re-initiation events.

The WWER-440 V-213 type design was developed in the seventies, without PTS assessments. Since then several calculations were performed by GIDROPRESS (USSR) - main designer of the WWER units - and other institutes, but they did not take into account the plant specific conditions.

In the frame of the AGNES project PTS study was performed on unit 3 of NPP Paks. The specific design and material characteristics of the WWER-440-s (and NPP Paks) made it impossible to refer directly to the results of the generic US studies in case of the PTS assessment of the pressure vessel integrity, but the methodology of the PTS evaluation, the safety factors and the acceptance limit met the requirements of the ASME code [1], and the 10 CFR 50 code [2].

During the elaboration of the methodology for updated evaluation of PTS integrity of the PAKS vessels the experience of the PTS assessment at Loviisa (Finland) [3], RPV Stade (Germany) [4], RPV-s DOEL (Belgium) [5]; US practice [6], and IAEA documentation were used [7].

This paper is giving an overview of the full analysis, including the already completed and planned development of the methodology.

2. OVERVIEW OF PTS METHODOLOGY USED AT NPP PAKS

The PTS assessment includes several separated actions as shown in FIG. 1.

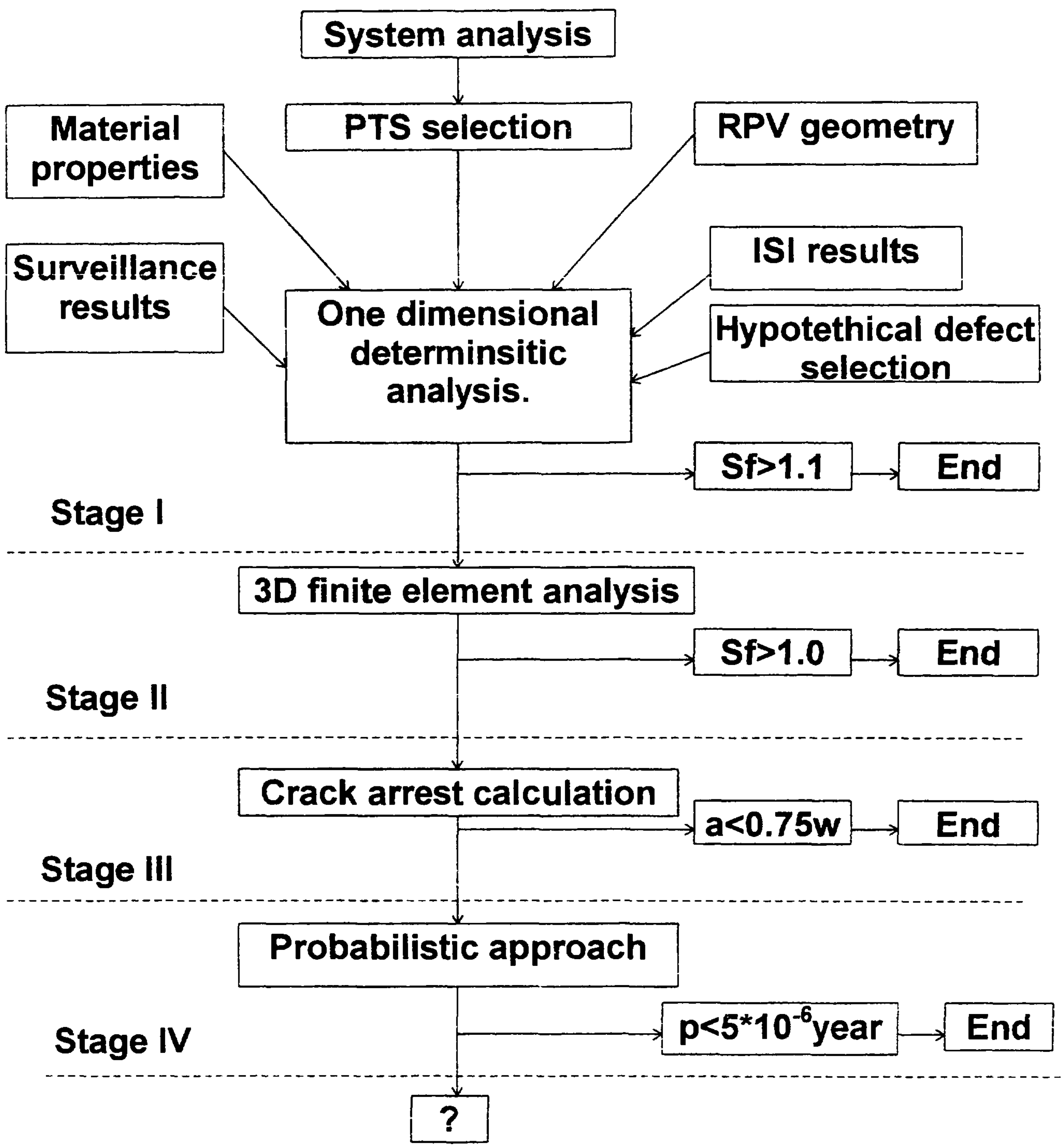

FIG. 1. The flowchart of the PTS assessment in the frame of the AGNES project.

These actions were performed by different groups of engineers and scientists, and were coordinated by the manager of the AGNES project. The main actions are further discussed below:

2.1. Transients selection

The selection of the transients to be analysed is the most difficult part of the analysis. The selection needs good knowledge of the reactor system, and analysis of the plant specific behavior of it. Many transients are plant specific, that is they can occur at one plant, but are not typical for another similar plant operated in a different way, or located at a site, where the weather conditions are different. Generally the so called similar plants only mean plant specific equipment, and during operation the operational mode and maintenance history are very often different even in case of the units of the same plant. In the frame of the AGNES project the PTS transient pressure, temperature data are always based on thermohydraulic transient simulation, which incorporates the unit specific characteristics. The calculations contain not only the effect of the overcooling-depressurization event, repressurisation-reheating situations are also considered. Altogether 7 transients with 18 cases were analyzed. Case means here different operator actions after the same transient event started or variations of the transient according to random behavior of the damaged equipment. (E.g. after the inadvertent opening of the safety valve it automatically closes again after a certain time.)

The selected transients are given in table 1.

TABLE 1. The tested tranzients.

Tranzient name	Cases
Inadvertent Opening of the Pressuriser Safety Valve	4
Opening of Steam Generator Cover.	3
Line break ∅ 233	1
Line break ∅ 73	3
Cold leg large break LOCA	3
Inadvertent Operation of the Emergency Core Cooling System	2
Steam line break	2
Total: 7 transients	18

2.2 RPV geometry

Paks units 1-4 are WWER-440 V-213 pressure vessels. The main characteristics of the V-213 vessels are:

- the vessel is welded from forged rings, and has an inside cladding made by submergerged arc strip welding. Inside diameter: 3542 mm
- the base material is 140 mm thick against the core zone, and 190 mm at the nozzle zone. The material is 15H2MFA. The cladding is 9 mm thick welded 18/8 type stainless steel.: 08H18N12B.

- The most critical parts of the pressure vessels (because of the highest neutron flux) are the welds 5/6, 3/5 and the forged ring against the core. Optionally the nozzle zone may be considered for analysis.

2.3 Thermohydraulic calculations

Most of the thermohydraulic analyses were performed by RELAP 5 code. The code has been modified to model the 6 loop WWER-440 design. The water mixing in the downcommer has also been considered in several cases using REMIX code.

2.4 Material database

All four units of NPP Paks have complete manufacturer documentation including material properties, production technology and quality assurance. These data were validated by the NPP owner during installation. Surveillance testing of the vessel material including surveillance of radiation embrittlement and thermal ageing were performed parallel with ISI testing. The operational parameters are also monitored. These databases and the knowledge of 15H2MFA steel properties gained from international cooperation and from the research performed in Hungary make a reliable integrity assessment during any possible PTS.

During the different stages of the fracture mechanical integrity analysis of the RPV different material characteristics are used. For the analysis of crack stability (no initiation) the K_{1c} reference curve is used, for calculation of arrest of a propagating crack the crack arrest reference (K_{1a} or K_{IR}) curve is needed, for probabilistic analysis a mean K_{1c} curve together with the scatter of the data.

K_{1c} reference curve

In case of the Paks units both the forged material and the welding satisfies the updated requirements of 15H2MFA steel, and this verifies the use of the new reference curves for PTS events given in the Russian Normative Documents [8]:

K_{1c}= 35+ 45*exp(.02*(T-T_K) [Mpam$^{0.5}$] 15H2MFA forging

and

K_{1c}= 35+ 53*exp(.0217*(T-T_K) [MPam$^{0.5}$] 15H2MFA weld.

where T_K is the transition temperature measured by Charpy impact testing belonging to unirradiated (T_{k0}) or irradiated (T_{KI}) values. The T_{K0} values are given by the producer, and the T_{KI} values are calculated from the unit surveillance results. The reference curves were compared with the results obtained on Charpy size TBP specimens of the surveillance program as it shown on FIG.2.

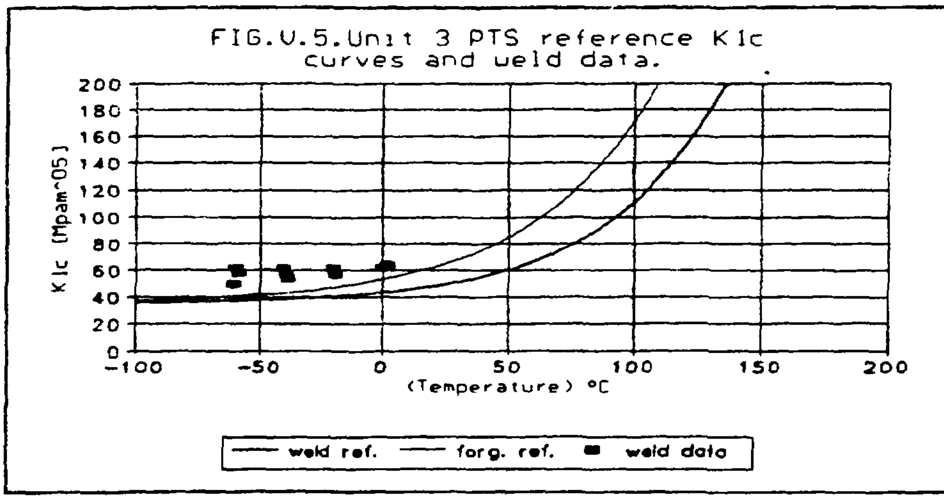

FIG. 2. Comparison of the K1c reference curves and surveillance results obtained on weldment samples.

<u>Crack arrest (K_{1a}) reference curve</u>

For the 15H2MFA steel and weldment no K_{1a} reference curves are available in the Rules or in the Russian Normative Documents. Instead of it the ASME K_{IR} (Reference Fracture Toughness) curve was used . The K_{IR} curve represents the lower bound critical stress intensity factors determined from static, dynamic and crack arrest curves. According to the ASME Code the K_{IR} is a function of temperature and RT_{NDT} (Nil-ductility temperature obtained by drop-weight test). The RT_{NDT} values can also be calculated from the surveillance impact testing results. The RT_{NDT0} value (belonging to as received material) is T_{K0}-33 °C where T_{K0} is determined using 68 J criteria. The RT_{NDTi} (belonging to irradiated material) value is $RT_{NDT0}+\Delta TTKV_{41}$, where $\Delta TTKV_{41}$ is the irradiation caused temperature shift measured with 41J criteria.

$$K_{IR}=29.4 + 13.44*\exp\{0.0261[T-(RT_{NDTi} - 88.89)]\} \quad [MPam^{0.5}]$$

To verify the use of the ASME K_{IR} curve the instrumented impact diagrams measured during the surveillance testing of NPP Paks on 15H2MFA steel were analyzed and compared with the results obtained. K_{1a} values on 15H2MFA forging were measured in the frame of the OMFB (National Committee on R&D) financed research project "Radiation damage of 15H2MFA steel" (91-97-42-0339).

<u>Mean K_{1c} curve for probabilistic analysis</u>

Mean K_{1c} curves are necessary for probabilistic analysis. Such curves do not exist in the Russian Code, therefore the US data used in VISA-II were accepted. These curves have been compared to the real measured data on 15H2MFA and weldment partly taken from the literature, partly obtained during the Hungarian OKKFT A-11/4 research program and in the frame of the Paks surveillance program. The result of the comparison on weld metal data is shown in FIG.2.

The VISA-II curves are: (please note that the following formulae are in US units) for crack initiation:

K_{1c}= 36.2+49.4*exp(0.0104(T-RT_{NDT}) T-RT_{NDT} < 50 °F and

K_{1c}=55.1+28*exp(0.0214(T-RT_{NDT}) T-RT_{NDT} .> 50 °F

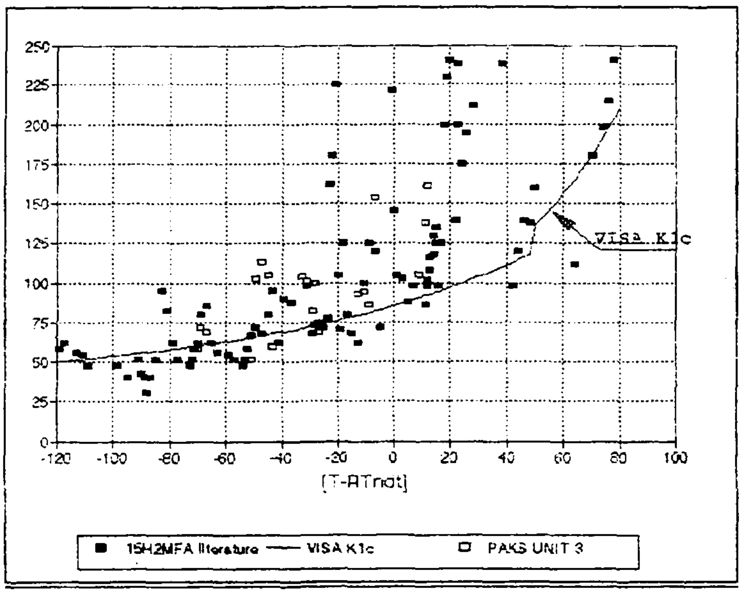

FIG. 3.

Verification of the VISA-II reference curve for 15H2MFA material

Flaw distribution.

The application of probabilistic codes requires a verified flaw distribution. The verification should be based on the ISI (In-service Inspection) results. The ISI testing of the RPV-s is very strict at NPP Paks. The core zone welds and beltline ring are partially checked yearly by automatic UT testing operating outside. During a four year period the outside UT testing covers the whole core zone. Moreover NPP Paks performs an inside UT testing in every four years (based on the good results this period is going to be changed for 8 years or even longer.) The results of the UT tests were compared with the most frequently applied flaw distribution functions. In case of Paks the so-called Marshall distribution gave the best fit, so this has been used.

2.5 Selection of hypotethycal defects

According to the Russian Code which was valid at the beginning of the AGNES projekt, and international studies [4,5] the following three models were selected for study.

Model 1.:Axial semielliptical surface crack a/c=2/3, depth is 1/4T = 35 mm. Location: the forging against the middle of the core.

Model 2.: Underclad axial crack in the ferritic welds, 4 mm deep and 50 mm long touching the interface of the clad, which is in complete contact with the vessel wall, and free of defects. Location: weld 5/6; weld 3/5.

Model 3.: Elliptical circumferential surface crack with a depth of 4 mm in the 15H2MFA weldment, and the clad is postulated broken (i.e. Σ depth=13 mm), a/c=2/3. Location: weld 5/6; weld 3/5.

3. Organization of the FM (fracture mechanics) analysis

The selected 18 cases multiplied with three crack models result in 54 FM analyses to be performed. Even if the temperature and stress distribution is common in some cases the number of the FM analyses is large. To reduce the calculation time a working process has been elaborated as shown in FIG.1.

In phase 1 of the calculation a one dimensional fast analytical code was used to evaluate the effect of the transient on the vessel integrity. If the results had shown that the safety factor during the tranzient event can go below 1.1 a more detailed analysis would have followed the first guess. If it was necessary crack arrest calculation or probabilistic analysis would have been used in the continuation.

Phase I. is a deterministic analysis evaluation of whether crack initiation can occur or not during PTS events. The stress and temperature distribution in the function of time is calculated by a fast analytical code ACIB-RPV (Advanced Calculation of Integrity of the Beltline of RPV). The K_{1c} value belonging to the time scale and crack tip temperature divided by the actual K_1 value gives the safety factor. ($S_f = K_{1c}/K_1$). The safety criterion is that this factor must be higher than 1 plus 0.1 as safety margin for the ACIB-RPV program during the whole transient.

Phase II. If the calculated value in Phase I. had been less than 1.1 the whole calculation would have been repeated by a 3 dimensional finite element code. If the resulting safety factor is higher than 1.0, the assessment is finished, otherwise crack initiation may occur (the calculation is based on very conservative assumptions, and a calculated safety factor below 1.0 means only a low probability of crack initiation, not a vessel failure) and the calculation is continued according to Phase III with an analysis based on crack arrest assumption.

Phase III. If a crack is initiated during a PTS it generally runs into a hotter location where the material toughness is higher and it is arrested. If the crack is arrested before reaching 70% of the wall thickness the vessel is still considered safe. The calculation method is the same as in Phase I. but K_{1a} (crack arrest toughness) is used instead of K_{1c}. If the crack is arrested the evaluation continues according to Phase I, because during the remaining time of the PTS event crack initiation may occur again. If the crack becomes stable before reaching 70% of the wall thickness the vessel integrity is not affected by the tested PTS case.

Phase IV. The deterministic evaluation is based on very conservative assumptions. If the results of the assessment in Phase II do not prove the vessel to be safe, probabilistic analyses can be used. The acceptance criterion for probabilistic calculation is that the overall probability of through wall crack penetration (not the brittle fracture of the vessel) must be $<5*10^{-6}$ event per reactor year (the probabilistic approach is presently not accepted by the existing codes and rules).

The Hungarian version of US NRC donated VISA-II code was used for this analysis.

4. SUMMARY OF THE RESULTS:

The rather conservative PTS calculations performed in the frame of the AGNES project have shown that NPP Paks units 1-4 can be safely operated at least until the 24th operational year, or more.

To evaluate the real lifetime and to run a life management program further study and research are necessary . Some of them have already been started, some others are still planned.

A short list of the life management actions performed, planned or under consideration at Paks NPP.

1. Use of low leakage core
2. Extension of the surveillance program
3. Heating up the ECCS water to 50 °C
4. Revising the operational regulations
5. Measuring the real K_{1c} and K_{1a} values of 15H2MFA material and its weldments
6. Study of the thermal ageing effect.
7. Development of the calculation by considering the effects of:
    - The material properties distribution in the RPV wall
    - Cladding effect
    - Low-leakage core
    - Use of the extended surveillance results
    - Following the operational changes

According to preliminary calculations - after the suggested life management actions are done- the recalculated lifetime will reach 40-60 years of safety operation life for all NPP Paks units.

ACKNOWLEDGEMENTS

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