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ABSTRACT

The Power Machinery Plant of the Škoda Works in Plzeň carries out in-service inspections of the WWER-440 MW reactor pressure vessels by means of a remote controlled inspection equipment - the TRC reactortest system, and some other inspections means. The results of the in-service inspections have been evaluated by employing methods based on the fracture mechanics approach, the knowledge of stress and strain distribution, and the operating history of the pressure vessels. Examples of detected types of defects and their analysis are shown.

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Introduction

Recurring in-service inspections seem to be the most effective form of nuclear reactor diagnostics as they allow to determinate the actual state of the most exposed parts of the reactors. The main attention is concentrated on the pressure vessel because it is the most critical component of the whole nuclear power plant.

Admittance to the reactor pressure vessel after a short time of operation is practically impossible and for in-service inspection it is thus necessary to use special devices, especially remote controlled manipulators.

The Power Machinery Plant of the Škoda Works in Plzeň executed recurring in-service inspections of reactor parts since 1982. For this purpose, a special group of specialists has been formed within the Reactor Research and Development Base. System Reactortest TRC has been used for performing the recurring inspections of reactor pressure vessels from the inner surface. This system is seated down on the upper surface of vessel flange. This manipulator contains a television camera for visual inspection of inner surface, and also ultrasonic probes for volume examination of base material, welding joints, and austenitic cladding.

All WWER-440 MW reactors produced by Škoda Works under the marking of V-213.Č contain, as a part of the supply, also another system (USK 213) which is established for examining some chosen parts of reactor vessels from the outer surface. This system is not transportable, but it is also controlled by the same group of specialists from the Škoda Works. This second system has been used since 1987; both systems well complement each other during inspections.

Just before the start-up of reactor examination, the so-called "zero" in-service inspection is carried out to know the initial "zero" state of the pressure vessel material; these results are important for evaluating the changes during further reactor operation.

By June 1989, 15 reactors were inspected by the TRC system, together 22 inspections were realized, with 7 reactors being inspected twice. A survey of all inspections and plans till 1991 are given in table 1.

The total number of performed inspections allows to evaluate used devices, methods of inspections, auxiliary operations, and, which is very important, to get some information about the defectness of pressure vessel materials. At the same time it is possible to evaluate results obtained by the TRC system, and to compare them with those provided by two different systems and two apparatuses - from the inner and the outer surface.

As a result of the evaluation of used apparatus and methods of inspection it is possible to say that the initial suppositions have been approved. The mechanical parts of the TRC manipulator get worn as a result of operation as well as of recurring decontaminations. During the use of this system several details and parts had to be exchanged. This system is kept on the desirable operational level by proper maintenance.

For realizing an inspection of a reactor in a planned time - normally 12 days - it is necessary to ensure the presence of highly qualified specialists in electronics. Assembly of the system, the exchange of measuring heads, proper inspection, dismantling and operations associated with packing and transportation - all have been substantially improved during last years and thus the whole time of inspection has been shortened and a decrease of personal radiation dose achieved.

Inspection methods

Inspection methods and the observed defects are closely associated. While inspection methods have not changed, gradual removing of found imperfections and increasing experience and certainty with the use of the TRC system have led to a substantial improvement of received results. Thus, as a result of this procedure, during recurring inspection some defects have been found, even though they were not observed during the first inspection.

The evaluation of measured parameters is based on the determination of the "equivalent diameter" and measuring the relative length of the defect. It is shown that the first parameter - the equivalent diameter - is rather untrustworthy. The austenitic cladding gives only the scatter in amplitude of signal equal to ± 6 dB. It was supposed that this local value for a given defect and probe would be constant during the repeated examination, which would allow to estimate the change of defect size from the change of its amplitude. On the contrary, experience shows that this supposition is not correct. First, in most cases it is not possible to use the same probe, from many reasons. Second, even when using the same probe, scatter of ± 6 dB is again observed. This fact is associated with the mechanical movement and with practical impossibility to put the probe directly to the same place with a precision which is necessary to repeat the value of deviation.

The second defect parameter, the relative length, is more reliable. Even though it is also dependent on signal amplitude, its repeated measurements give substantially smaller relative error. This fact is approved by repeated measurements during one inspection, on one side, as well as during recurring inspection on the other side. It has therefore been decided to measure not only the length, but also the depth of the defect (i.e. defect dimension in the direction of wall thickness).

The results from the recurring inspections of seven reactors with the use of the TRC system suggest an increase of the total sum of defect indications. This fact must be considered particularly in austenitic cladding and in circumferential weld joints (in all other inspected places there are no defects, or their number does not increase). In the cladding, increase of defect indications of the type of tearing off of the cladding (i.e. defects of laminar type, parallel to the surface) is observed. An increase of this type of defects can be supposed also for reactors inspected for the first time only after some time of operation. The mechanism of this growth is not known as it has not been possible to remove the defects, even partially. It follows from the principle of ultrasonic reflection thatthe ability of reflecting the defect can be increased even without any growth of the defect - this is considered to be the main reason of observed increased number of these indications. In the case of defect type of "underclad flaws" which are perpendicular to the line of fusion, no increase of their dimensions has been observed.

In circumferential welding joints and surrounding base metal, some increase of defect number has been found during recurring inspections (in comparison with the first ones), but all new indications could be explained by the improvement of examination methods or by having been omitting during the first inspection. From the point of view of their amplitudes, all new defects have amplitudes within the scatter of ± 6 dB, even though deviation in positive direction (i.e. toward defect increase) are somewhat more frequent than in the opposite one. In case of normal length measurement all deviations are observed only within the precision levels of measurements. Generally, till now no substantial growth of these types of defects between the recurring inspections has been observed.

Comparison of the results from ultrasonic inspections from the inner and outer surfaces is made more difficult because of the substantially lower sensitivity of the USK 213 system in comparison with the TRC system. From this reason, only several defects have been found using the USK system but not all have been observed also by the TRC system from the inner surface. This fact is valid for the defects observed from the outer surface using tandem probe. All other defects have been found also by the TRC system and these data allow to determinate, with a high reliability, also their character. For this reason, even though there were more defects found from the inner surface, both methods are necessary. We suppose that in future the USK system will be supplemented by a moderner ultrasonic apparatus in order to receive better results.

Lifetime and defect allowability

Evaluation of reactor pressure vessel lifetime, and allowability of defects found during recurring inspections, are two closely associated and dependent processes.

Design lifetime of reactor pressure vessel is based on the fictive existence of the so-called "calculated (fictive) defect" (surface semielliptical crack with depth equal to 25% of wall thickness), the design values of radiation embrittlement coefficients (received on the basis of research and development experimental programs of pressure vessel materials), and initial guaranteed material properties which are also incorporated into design. The reactor pressure vessel must have, even at the end of its lifetime, an appropriate resistance against brittle fracture in such a way that the "calculated defect" placed into the area of maximum irradiation embrittlement is not yet critical one. It is clear that allowable defect sizes must be substantially smaller with respect to the "calculated defect" as it is necessary to ensure a high reliability of reactor operation (probability of its fracture should be lower than 10^{-6} per year). Moreover, all the parameters used in the calculations are given only with some confidence (this is especially valid for the results obtained from recurring inspection At this time the standard for evaluating the defects found before reactor startup, the "Rules for examinations of welding joints and claddings of reactors" (PK 1514-72), is used. For evaluation of defects found during in-service inspections, a new CMEA standard is being prepared. The main parameters put into calculation of reactor lifetime, are the dimensions and shape of the defect, as determined by non-destructive examination (in most cases by ultrasound methods) and transition temperature shift (characterizing the measure of irradiation embrittlement) as determined by surveillance specimens of reactor pressure vessel materials.

Evaluation of the allowability of defects is done by such a way as to ensure the designed reactor lifetime taking into account the real irradiation embrittlement of materials (in cases when it is larger than the design one).

The following steps are characteristical for this evaluation:

1) analysis of the design of non-destructive examination - determination of the most probable defect (inclusion, non-fusion, slag, crack, etc.) for a given indication;

2) choice of a proper "calculated-crack-like-defect" for a given indication - determination of real dimensions of this defect (semiaxis lengths of inner elliptical or surface semi-elliptical defect), its configuration from the point of view of working stresses (projecting into the plane perpendicular to the direction of largest tensile stresses);

3) calculation of the number of operation cycles necessary for the transformation of real type of defect into "fatigue-crack-type-defect" which is put into calculations based on fracture mechanics;

4) determination of the subcritical growth of "fatigue-crack-type-defect" during the whole design reactor lifetime (design or real number and level of operation cycles);

5) evaluation of the resistance against brittle fracture at the end of reactor lifetime (i.e. determination of the transition dependence of fracture toughness of materials including all degrading factors) - for this case only a conservative approach is permitted: the design curve of fracture toughness is given by standards; design values of transition temperature are given in Technical requirements imposed on materials (i.e. guaranteed values), as well as design values of irradiation embrittlement (if real values are not larger);

6) calculation of stress intensity coefficients K_I of "calculated crack-type-defect" operational regimes including calculated subcritical crack growth (according to 4) for all operational regimes (normal, upset, and emergency one);

7) comparison of calculated values of stress intensity coefficients (according to 6) with allowable values of fracture toughness (design curves) for a given operational regime (according to 5) and determination of real safety coefficients with respect to fracture toughness;

8) comparison of real safety coefficients with the required ones according to the standard; determination of allowable defect sizes.

Types of defects in reactor pressure vessels

Defects which are found by the ultrasonic method can be divided into the following groups according to their character, configuration, and the way of their calculational evaluation:

(a) defects in cladding

(a.1) surface defects: their evaluation is not carried out, but they must be removed;

(a.2) inner defects: their evaluation is not carried out from the viewpoint of brittle fracture, but only from the viewpoint of their subcritical crack growth till the moment of their changing into a defect through the whole thickness of cladding; in such a case there would occur a direct contact of cooling medium with the base material.

(b) defects in the interface between base material and cladding

(b.1) defects of slag type between individual weld beads in austenitic cladding (these defects are evaluated according to defects of (a.2) type);

(b.2) defects of non-fusion type between cladding and base material - tearing off the cladding. These defects cannot influence the initiation of brittle fracture because the main stresses are oriented parallel to these defects plane and can initiate only shear stresses;

(b.3) defects of the type of "underclad flaws". In conformity with calculation and experimental verification, the maximum depth cannot be larger than 8 mm; their ascertainability depends strongly on their real dimensions, the reliability of their ascertainment is around 75%; calculations suggest that these defects are not dangerous for reactor pressure vessels of V-213.Č type.

(c) inner defects

(c.1) defects in base material - according to the manufacturing technology of rings for pressure vessels these defects are in most cases of nonmetallic inclusion type which lie in a plane parallel to the pressure vessel surface; even if the maximum dimension is close to 8 mm ($D_{equivalent}$), their projection into the plane perpendicular to the surface is small for the transformation of the defects into fatigue cracks; the maximum calculated fatigue growth of these defects is smaller than 0.1 mm for the whole service life;

(c.2) defects of slag type in welding metal - their orientation can be different - in all cases the choice of "calculated dimensions of the defect" is made as a projection into the plane perpendicular to vessel surface; for these cases the number of cycles necessary for their transformation into fatigue cracks are also very high in comparison with the operational number of cycles (similar as in (c.1), even for defects of "sharp-slag" type); their total calculated subcritical crack growth is again smaller than 0.1 mm;

(c.3) defects in the boundary between weld metal and base metal (i.e. on the fusion line) - in most cases they are of non-fusion type, and lie in the area of compression stresses outside the irradiation embrittled zone. Their dimensions are not large.

Conclusions

During recurring in-service inspections realized from both inner and outer surface of reactor pressure vessel, different types of defects may be found. Results of these examinations suggest an increase in the number of indications as a result of reactor operation, but a detailed analysis has shown that this effect is caused by other effects than by the increase of their dimensions.

Calculations of the allowability of these defects for all inspected reactor units of the WWER-440 nuclear power plants have shown that all detected defects are allowable and the designed lifetime of the reactors is ensured.

Table 1. Realized and planned recurring inspections of WWER-440 MW units

system	realized	planned
REACTORTEST TRC	*	+
USK 213	●	•

		1982	1983	1984	1985	1986	1987	1988	1989	1990	1991
BOHUNICE	1				*			*			
	2	*			*				+		
	3			*			* ●				+ •
	4				*			* ●			
DUKOVANY	1				*		* ●				+ •
	2					*		* ●			
	3					*		* ●			
	4						* ●				
GREIFSWALD	1				*						
	2					*					
	3					*					
	4				*						
PAKS	1						*				+
	2							*			
	3								+		
	4									+	
KOLA	1								*		
	2								*		
MOCHOVCE	1								+ •		+ •
	2									+ •	
	3										+ •