WWER STEAM GENERATOR TUBE STRUCTURAL AND LEAKAGE INTEGRITY

K. Šplíchal, Vl. Krhounek, J. Otruba, M. Ruščák

ABSTRACT

The integrity of heat exchange tubes may influence the life-time of WWER steam generators and appears to be an important criterion for the evaluation of their safety and operational reliability. The basic requirements are to assure very low probability of radioactive water leakage, preventing unstable crack growth and sudden tube rupture. These requirements led to development of permissible limits for primary to secondary leak evaluation and heat exchange tubes plugging. The stress corrosion cracking and pitting are the main corrosion damages of WWER heat exchange tubes and are initiated from the outer surface. Both the initiation and crack growth cause thinning of the tube wall and lead to part thickness cracks and through wall cracks, oriented preferentially in the axial direction. The paper presents the leakage and plugging limits for WWER steam generators, which have been determined from leak tests and burst tests. The tubes with axial part-through and through-wall defects have been used.

The permissible value of primary to secondary leak rate was evaluated with respect to permissible axial through-wall defect size of WWER 440 and 1000 steam generator tubes. Blocking of the tube cracks by corrosion product particles and other compounds reduces the primary to secondary leak rate. The plugging limits involve the following factors: permissible tube wall thickness which determine further operation of the tubes with defects and assures their integrity under operating conditions and permissible size of a through-wall crack which is sufficiently stable under normal and accident conditions in relation to the critical crack length. For the evaluation of burst test of heat exchange tubes with longitudinal through-wall defects the instability criterion has been used and the dependence of the normalised burst pressure on the normalised length of an axial through-wall defect has been determined. The validity of the criterion of instability for WWER tubes with through-wall cracks have been experimentally confirmed. An expert system of WWER steam generators has been suggested with the main goal to predict the corrosion damage rate, the residual life-time and the life-time extension. The assessment is based on the evaluation of local environment parameters by MULTEQ Code combined with SCC experiments.

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INTRODUCTION

The integrity of heat exchange tubes may influence the life-time of WWER steam generators and appears to be an important criterion for the evaluation of their safety and operational reliability. The basic requirement is to assure a very low probability of radioactive water leakage, preventing unstable crack growth and sudden tube rupture. These requirements led to development of permissible limits for primary to secondary leak evaluation and heat exchange tubes plugging based on eddy current test (ECT) inspection and burst test experiments.

CORROSION DEFECTS

WWER 440 and WWER 1000 steam generators are horizontal bodies with 5536 and 11000 heat exchange tubes respectively. According to the WWER steam generator tube experience the titanium stabilized austenitic steel similar to the 321 type is susceptible in the secondary environment to SCC and pitting. Corrosion damage of the outer tube surface occurs especially in the form of crevice corrosion at the tube support plates in the higher thermally-loaded hot leg and only to a limited extent in the cold leg tube section. In tube bends it is insignificant. The corrosion damage of tubes in different steam generators and NPP units varies in different extent (Fig.1). The total number of plugged tubes is given for four NPP units in Tab.1 [1].

Table 1. WWER 440 steam generator plugged tubes

NPP unit	Number of SG	Operating time [years]	Number of plugged tubes	Plugged tubes [%]
V 1	6	17	0	0
V 2	6	15	0	0
V 3	6	11	118	0.35
V 4	6	10	0	0
EDU 1	6	10	38	0.11
EDU 2	6	9	49	0.15
EDU 3	6	9	43	0.13
EDU 4	6	8	39	0.12

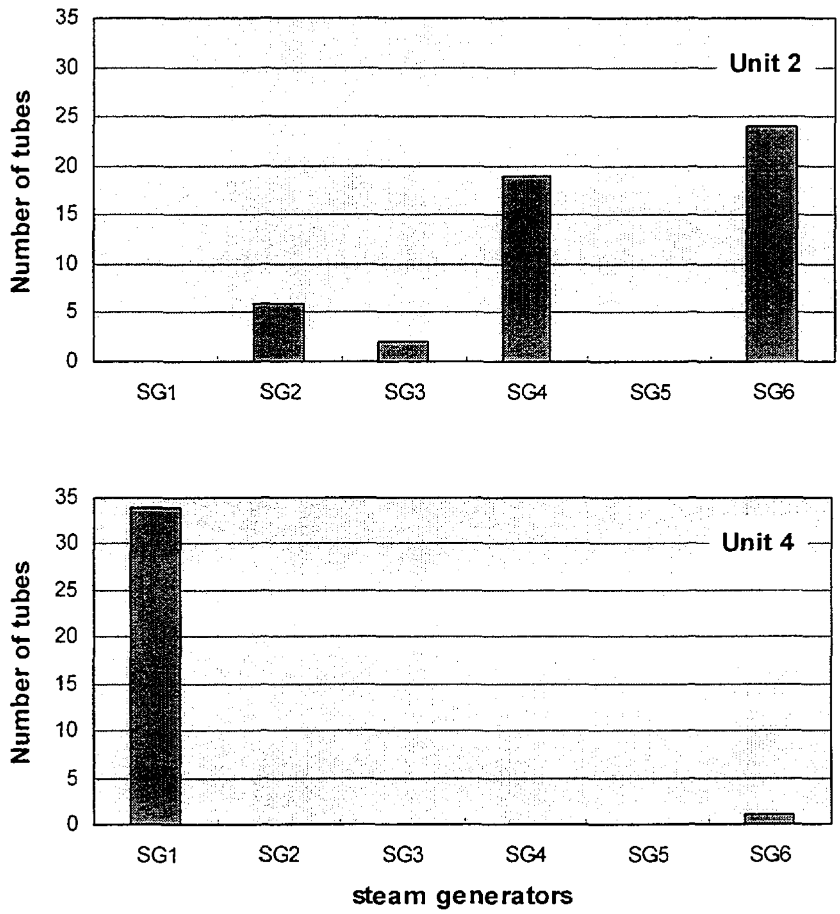

Fig.1. Number of plugged tubes of WWER 440 steam generators for two NPP units

Up to now any relationship between operating modes and defect initiation and propagation has not been established. Stress corrosion cracking results in the formation of non through-wall as well as through-wall cracks, oriented, primarily in the axial direction, and to a lesser extent also circumferentially. In the first period the defects propagate through the tube wall in the form of narrow cracks and their widening takes place probably by intergranular attact in the course of further exposure to the secondary environment. The observed maximum crack length at the inner surface of tubes pulled from operated steam generators was less than 12 mm.

LEAK RATE LIMITS

The basic approach to the determination of primary-to-secondary leak permissible limits is based on measurement on tubes with through-wall cracks [2]. The evaluation should comprise the location of defects in the tube free spans, the region of the tube support plate, and the tube expansion zones. In our work the limits have been derived only for the tube free spans. The tests were carried out on equipment which enabled simulation of the primary and secondary circuit temperatures and pressures of WWER 440 and 1000 steam generators. Leakage of medium through an axial crack was measured in time dependence of exposure to the pressure inside the tube pressurized by the primary water from the experimental reactor water loop. Resulting relationships between the leak rate and the inner crack length or the inner crack cross-section were determined and are shown in Fig.2,3 as an example for WWER 440 steam generator tubes [2]. The smaller scatter of measured values of the leak rate in dependence on the crack cross-section is due to the measured cross-section parameter involving the crack width, which is decisive for clogging of cracks with particles present in the primary water.

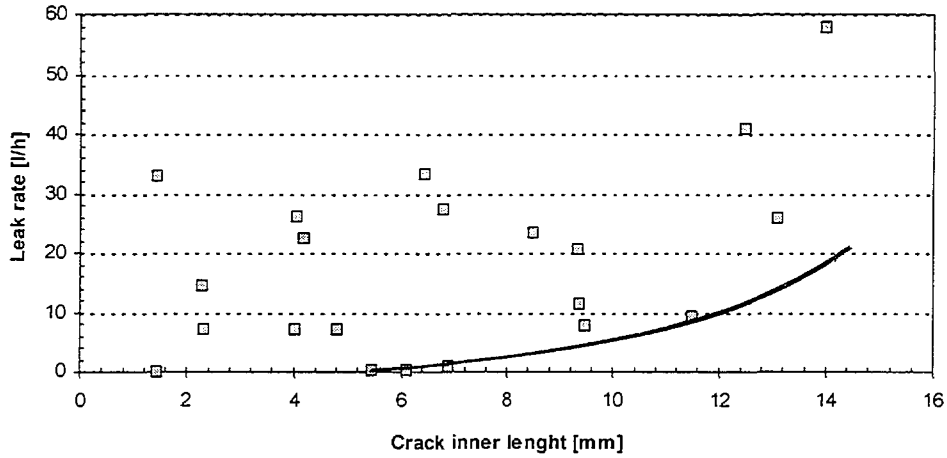

Fig.2. Relationship between leak rate and crack length on the tube inner surface for WWER 440 steam generator tubes

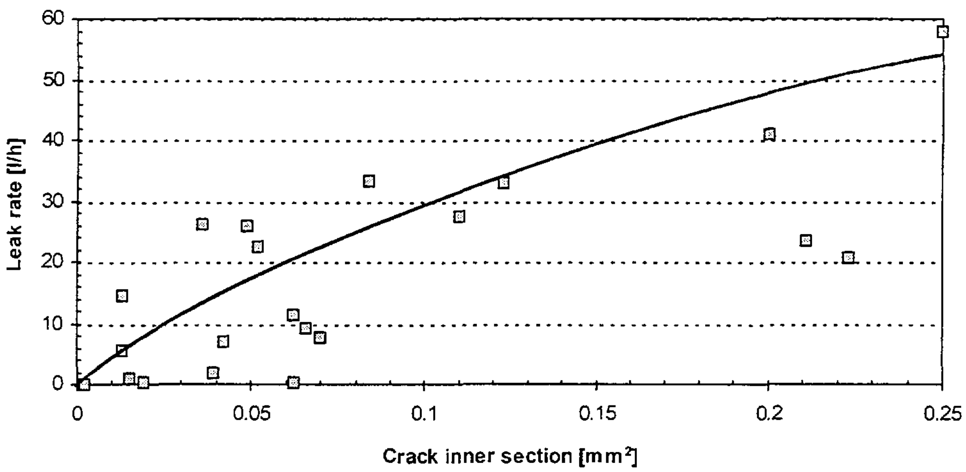

Fig.3. Relationship between leak rate and crack section on the tube inner surface for WWER 440 steam generator tubes

The formation of narrow through-wall cracks in the initial period does not necessarily lead to a primary-to-secondary leak because the leakage is determined primarily by the crack width on the tubes inner surface. In the experiments performed under primary and secondary circuit operating conditions of WWER 440 and 1000 (primary-to-secondary pressure difference of 7.8 and 9.4 MPa respectively), tube samples with axial cracks of 3-45 μm inner width and 1.3-14 mm inner length were investigated. No primary-to-secondary leak was determined at cracks of inner width 3-5 μm, whereas in 5-20 μm wide cracks the leakage was restricted (Tab.2,3). The leakage was blocked by clogging of cracks with particles either from the corrosion products or boron oxide compounds precipitating during expansion of the primary water through the cracks.

Table 2. Relation between cracks size and leakage

Number of tube specimens	Cracks		Leakage
	Inner width [μm]	Inner length [mm]	
8	1.3 - 5	1.4 - 6.0	without leakage
20	5 - 20	2.3 - 13.0	restricted leakage
5	20.0 - 86.0	1.4 - 14.0	leakage

Table 3. WWER 440 primary water particle sizes

Power plat	Operating time [hours]	Range of particle size [μm]	Average particle size [μm]
EDU	14 000	0.6 - 4.4	1
EBO	7 000 - 22 000	1 - 6	4.5 - 5.0
KKW	60 000	1 - 30	5.0
Experimental reactor water loop	11 000	0.5 - 30	7.3

The permissible leak rate value was determined using following assumptions :

- The leak occurs under normal conditions. The permissible crack lengths were calculated under loads which are defined as a threefold value of the primary-to-secondary pressure difference, i.e. $\Delta P = 3 \times 7.8 = 23.4$ MPa for WWER 440 and $\Delta P = 3 \times 9.4 = 28.2$ MPa for WWER 1000 steam generators.
- The leak rate can occur at the primary circuit pressure of 12.5 MPa for WWER 440 and 15.7 MPa for WWER 1000 attained under accident conditions, such as secondary side depressurization resulting from a main steam line or feedwater line break. Under these conditions the permissible crack length were evaluated, using a safety factor of $\sqrt{2}$.

On the basis of graphical evaluation using lower bound curve [3] of the measured values the permissible leak rates have been determined, as 8 $l.h^{-1}$ for crack length of 11.3 mm for WWER 440 and 10.2 mm for WWER 1000. Permissible crack lengths were determined by means of an instability criterion for given pressures using the conservative σ_f value and a relation taken from [4]. The obtained results are summarized in Tab.4. The leak rate value of 5 $l.h^{-1}$ is recommended to keep secondary water radioactivity below a sufficiently low level under operating conditions.

Table 4. Permissible leak rates

Power plant	Loading [MPa]	Coefficient of safety	Permissible crack length [mm]	Leak rate [$l.h^{-1}$]
WWER 440	$\Delta P = 7.8$	3	11.3	8
	$P_{ID} = 12.5$	$\sqrt{2}$	20.3	>> 10
WWER 1000	$\Delta P = 9.4$	3	10.2	8
	$P_{ID} = 15.7$	$\sqrt{2}$	20.1	>> 10

INTEGRITY OF HEAT EXCHANGE TUBES WITH THROUGH-WALL DEFECTS

The instability criterion of a straight tube with longitudinal through-wall defects is given by Hahn's relation [5]. For the evaluation of burst tests results this criterion has been used in following form:

$$P^* = \frac{P_a}{\sigma_f} \cdot \frac{R}{t} \approx \frac{1}{M} \qquad (1)$$

where: P^* is normalized burst pressure, P_a is measured burst pressure, σ_f is flow stress, R, t are mean radius and wall thickness of tube and M is the bulging factor and $\sigma_f = k(\sigma_y + \sigma_u)$, σ_y, σ_u are flow, yield and ultimate stress respectively.

The dependence of the bulging factor on the normalized length λ of an axial through-wall defect has been derived by Folias (relation 3), Erdogan (relation 4) and Hernalsteen (relation 5) respectively Tab.5. The normalized through-wall flaw length λ is given by relation

$$\lambda = \frac{2a}{\sqrt{Rt}} \qquad (2)$$

where $2a$ is the through-wall defect length.

The heat exchange tube integrity was evaluated by means of a burst test. Through-wall axial defects were prepared by electro-discharged machining (EDM) and the SCC method. For the burst test purposes the tubes were lined with a plastic bladder of 2 mm thickness. The relation between the normalized pressure for WWER 1000 steam generator tubes and the normalized flaw length are shown in Fig.4, together with a plot of the 1/M values. The value of the „k" constant in the expression for σ_f was determined as 0.522.

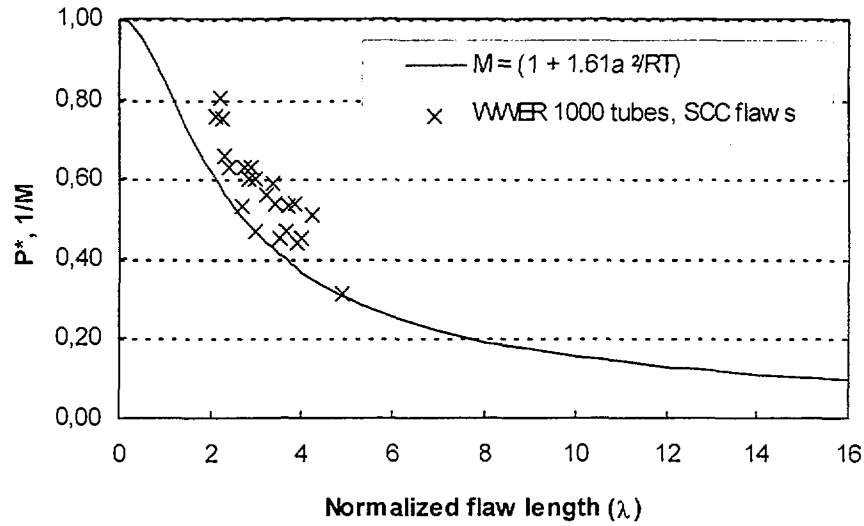

Fig.4. Burst test results on WWER tubes with SCC axial through-wall defects in free span

All the normalized burst pressure values lie above the curve determined by the Folias' relation which represents rather a lower bound curve of the test results. The larger scatter is caused probably by the uncertainty of the through-wall crack length determination . In fact, the crack usually does not represent a single defect, but rather a system of multiple cracks mutually separated on the tube outer surface by strips of material, their failure occurring only when the unstable crack propagation takes place during the burst test.

Table 5. Bulging factors M and constants k

Relation	M	k	Material	References
(3)	$(1 + 1.61a^2/Rt)^{1/2}$	0.58	Inconel	[4]
(4)	$0.614 + 0.386 \exp(-1.25\lambda) + 0.481\lambda$			[6]
(5)	$0.614 + 0.386 \exp(-1.15\lambda) + 0.436\lambda$	0.55	Inconel	[7]
	relation (3), (4)	0.522	08Kh18N10T	this paper

The Folias′ relation was used to establish the lengths L = 2a of axial through-wall cracks for the WWER 440 and 1000 steam generator tubes, permissible from the point of view of a controllable crack development under normal operating conditions, accident conditions e.g. secondary circuit depressurization, as well as under conditions of a threefold of the normal operating pressure value. The results are summarized in Tab. 6.

Table 6. Critical lengths of axial through-wall crack

Pressure conditions	WWER 440		WWER 1000	
	P [MPa]	L [mm]	P [MPa]	L [mm]
Normal operation	7.8	41.9	9.4	36.6
Accident	12.5	26.1	15.7	21.9
Threefold pressure difference	23.4	13.8	28.2	12.0

TUBE PLUGGING LIMITS

Determination of the plugging limits is based on experimental measurement of the critical pressure at the rupture of a tube with a crack. The result is a relationship between the critical pressure and the tube wall thinning determined by means of ECT. In the case of WWER steam generators the tube wall thinning measurement can be performed only by means of a bobbin coil probe. Burst test were carried out at pressures up to 100 MPa and at temperatures of 290° and 320° C. The critical pressure was measured on tubes with both artificial corrosion defects, the maximum crack length 12 mm and defects prepared by EDM method, the maximum crack length 35 mm. Results of measurement of the normalized critical pressure vs. tube wall thinning for corrosion cracks are shown in Fig.5. As follows from the relationship between the experimentally determined normalized pressure and the tube wall thinning, the lower bound curve for measured values lies sufficiently, above the anticipated pressure load values. The permissible values for tube plugging have been derived for accident and normal conditions of WWER 440 and WWER 1000, i.e. for a secondary side depressurization and for threefold value of the primary-to-secondary pressure difference.

The resulting plugging limit for WWER 440 steam generator tubes involves the critical size of a defect, corrected by its growth during the period between two subsequent in-service inspections, and as well as by the accuracy of the ECT metod. The calculation of the plugging limit was based on the approach PWR steam generators, using the equation

$$D_p + D_E + D_G = D_C \qquad (6)$$

where D_p is the limit value of wall thinning for tube plugging, D_E - error of ECT measurement, D_G - tube wall thinning in the interval between two successive in-service inspections, D_C - critical wall thinning, determined for pressure loading under accident conditions. All values are in % of tube wall thickness.

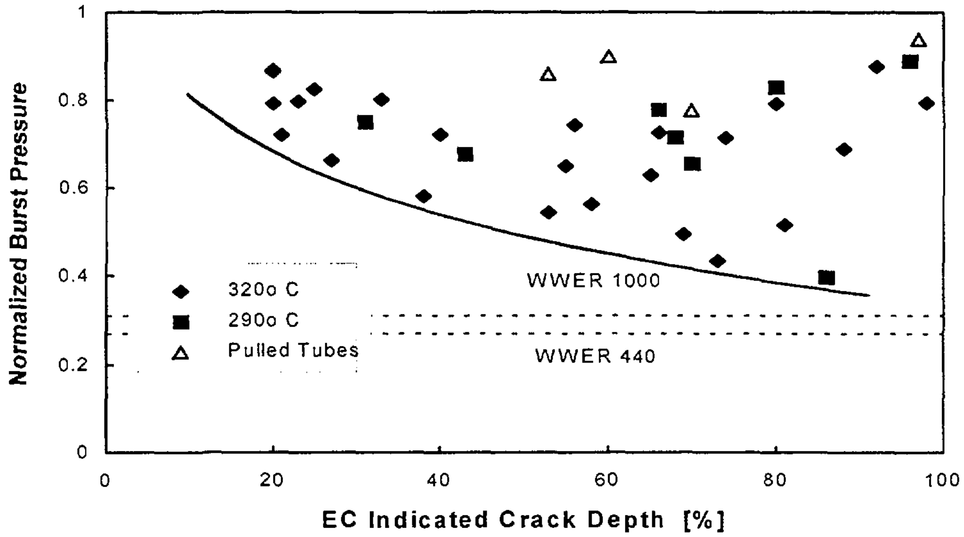

Fig.5 Normalized burst pressure vs. SCC - slot depth

As follows from the data in *Fig. 5*, for the normalized threefold operating pressure difference the critical WWER 440 tube wall thinning D_C is 100 %. The error of ECT measurement has been established as D_E = 15 %. The rate of the tube wall thinning has been determined as 0.58 % of the wall thickness per month [8]. This results in a wall thinning value of D_G = 17.4 % for a 3-year interval of in-service inspections. The resulting limit for steam generator tube plugging is then:

$$D_p = 100 - 17.4 - 15 = 67.6\,\%. \qquad (7)$$

The plugging limit value of 70 % has been suggested with respect to WWER 440 operating experiences.

ENVIRONMENT PARAMETER ASSESSMENT AND CORROSION DAMAGE PREDICTIONS

The results obtained on the damage of heat exchange tubes should have to be assessed from the point of view of possible consequences both for the reduction of the steam generator thermal capacity and the safety of primary and secondary circuit components. Although at present the corrosion damage of tubes is relatively small, for further operation of the WWER steam generators it is required to establish an evaluation system pointed to:

- prediction of the amount and rate of damaging with respect to the residual life-time;
- implementation of corrective measures, such as those optimizing the water regime parameters (molar proportion of anions and cations), the hideout return measurement and the condensate polishing operation;
- introduce water regimes in the WWER secondary circuit which would minimize corrosion and corrosion-erosion damages of materials and lower the amount of deposits in the circuit.

Basic approach to the assessment of corrosion damage is to determine the parameters of a local environment taking into consideration the preferential occurrence of damage in regions of tubes at the tube support plates, in thread holes of the WWER 440 collectors and in tube-collector joints of the WWER 1000 collectors.

At present there exist three possibilities how to identify and model the environment in a crevice: (1) chemical analysis of surface deposits in crevices (qualitative method) and evaluation of blowdowns and hideout returns; (2) thermodynamic analysis based on the modeling of concentration processes, e.g. using the MULTEQ Code; (3) experimental modeling based on the formation of identical thermodynamic and chemical conditions in a model crevice.

The MULTEQ Code has proved to be a suitable tool for the identification of environment in crevices for several reasons: the code has been validated in several independent organizations and used for evaluation of the water environment at nuclear power plants; calculations performed for WWER steam generators have proved a quantitative agreement of results with the microanalysis of deposits on the inner surface of crevices, as well as conformity of model cases with engineering expectations; the code enables to perform sensitivity analysis for simulation of different chemical regimes and thermodynamic conditions.

Calculations performed in this way will serve to determine the composition and pH_T of the model environment for the stress corrosion cracking or pitting corrosion tests under conditions of a crevice. The first phase of experiments will be aimed at investigation of the influences of initial contents and mutual interactions of the present components on the level of environment concentration, the resulting pH_T values and the deposit composition. The measured values will be compared with in-process measurements of the bulk and local parameters.

During the second phase tests of SCC will be carried out and parameters of the crack initiation and propagation will be determined. The tests will be aimed at examination of changes of the water environment composition, especially the content of aggressive impurities, and of the level of stress. From the calculations and from the experimental and in-process measurements the damage mechanism will be predicted in correlation with the bulk parameters of the water environment as well as with the calculated stress values. The proposed procedure of assessment will proceed in following steps:

- Calculation of temperatures on the surface of heat exchange tubes with emphasis on the points of their contact with the tube support plate.

- Modeling of the environment in a particular crevice with following outputs: pH of the crevice environment, composition of the crevice environment, determination of solid and liquid phases; execution of a sensitivity analysis of the crevice environment taking into account thermodynamic conditions in the crevice; definition of conditions in crevices during an outage; determination of a model environment for autoclave tests.
- Tests of corrosion cracking in the given environment with following outputs: time to initiation in dependence on the stress for the given environment; threshold value of the stress intensity factor for the given environment; crack growth kinetics in form of a relation between the crack growth rate and the stress intensity factor taking into account the geometry of the tube.
- Variation of models of damage for the given environments. On the basis of such model it will be determined whether the damage can be initiated at a particular location and, if appropriate, whether the initiated defect can propagate and when it attains a critical size.

On the basis of a variance analysis the likely time to the incidence of a damage of a non-through defect type as well as to the incidence of a through-wall defect will be determined. Through monitoring of the parameters of the blowdown and hideout return chemistry environments differing from normal operating conditions will be identified and then for these individual events models of damage will be applied using the above mentioned procedures. If such conditions take place during the steam generator operation, the estimate of the remaining service life will be corrected taking into account the time-defined occurrence of such effects.

CONCLUSION

The permissible value of primary to secondary leak rate $8 l.h^{-1}$ was evaluated with respect to permissible through-wall defect size of WWER 440 and 1000 steam generator tubes. The conservative value of 5 l/h was suggested as the leak rate limit for WWER-440 nuclear power plants. The formation of narrow through-wall cracks in the initial period does not necessarily lead to a primary-to-secondary leak because the leakage is determined primarily by the crack width on the tubes inner surface. The leakage through the cracks is reduced if the size of corrosion product and other compound particles correspond with the through-wall crack width.

The validity of the criterion of instability for WWER tubes with through-wall cracks has been experimentally confirmed. The permissible size of axial through-wall cracks is 13,8 and 12,0 mm respectively for the threefold value of the primary-to-secondary pressure deference in WWER 440 and 1000 steam generators.

The crack lengths observed on pulled tubes were in all cases lower than the established critical crack lengths. The preliminary statistical evaluation of the ECT in-service inspection data proved that the crack growth is also sufficiently low. Thus the leakage can be monitored for a long period under operating conditions. Therefore the leak before break approach could be applied with a sufficient margin of safety.

The defected tubes with wall thinning above 70% rupture under burst test condition at a critical pressure of 35 MPa, which is higher than the threefold value of primary to secondary pressure difference 23.4 MPa for WWER 440 and 28.2 MPa for WWER 1000.

As shown by experimental studies, the use of permissible thinning of 70% for the heat exchange tube plugging does not lead to an unstable crack growth and consequently to an uncontrollable radioactive leakage. This corresponds to experience hitherto obtained at WWER 440 nuclear power plants whose number of damaged tubes is very low.

An expert system of WWER steam generators has been suggested with the main goal to predict the corrosion damage rate, the residual life-time and the life-time extension. The assessment is based on the evaluation of local environment parameters by MULTEQ Code combined with SCC experiments.

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