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ABOUT RELIABILITY ON VVER PRESSURE VESSEL NEUTRON FLUENCE CALCULATION

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ABSTRACT. *Reliability study of neutron fluence calculation for VVER pressure vessel has been carried out. The estimation of influence of the geometry approximation in calculation model and the choice of neutron cross section data file on the calculation results are presented.*

The neutron fluence determination on the reactor pressure vessel is an essential part of the Surveillance Program for ensuring save operation of the reactor unit during the lifetime limit.

The main tools for determining the neutron flux are the neutron transport calculations. The reliable assessment of the neutron flux and its responses in the vicinity of the reactor pressure vessel is a difficult problem for the reactor systems. It is caused by the complexity (multi-layer heterogeneous structures) of study media and its considerable extension (15-20 free mean paths). The adequate modelling of the neutron transport is restricted by the proximate description of the complicated interaction of the neutron with the nucleus (the cross section data libraries must be widely tested for such calculations), and limited abilities of the existing transport codes and the used computers..

In this paper the estimation of influence of: 1) the geometry approximation in the calculation model; and 2) the choice of the neutron cross section data files; on the calculation results are presented.

1. TORT APPLICATION IN RPV NEUTRON FLUX CALCULATIONS

The neutron flux values onto the Pressure Vessel(PV) of VVER-1000 and VVER-440 reactors, at the places (Table 1) important for the metal embrittlement surveillance, have been calculated by 3D code TORT [1] and synthesis method [2] by code DORT [3], both based on the discrete ordinate Sn method. The comparison of the results received by both methods has been applied for a demonstration of the reliability of synthesis method.

The neutron fission source is represented by the power efficiency, averaged over the plane cross section of each cassette. The axial dependence is taken into consideration. The library FLUNG [4] has been applied in all following flux calculations.

The comparison of the VVER-1000 flux values (Table 2) shows a good consistency within the limits of the solution accuracy. The differences diminish from 5% to 1% with enlarging the energy range limits.

These results are expected for VVER-1000, because the places of interest are far from the reactor core axial edges.

TABLE 1. Places for the Test Comparison

Reactor	N°	r, cm	θ,°	z, cm	Comments
VVER-1000	1	207.35	8	*96.0	on RPV, azim max, axial max
	2	207.35	30	96.0	on RPV, azim min, axial max
VVER-440	1	178.40	30	29.5	on RPV, azim max, weld 4
	2	192.45	30	29.5	beh. RPV, azim max, weld 4
	3	178.40	13	29.5	on RPV, azim min, weld 4
	4	192.45	13	29.5	beh. RPV, azim min, weld 4

* z=0 - core bottom

The more important conclusion is connected with the good consistency of the flux values for VVER-440 in the places near the core bottom, the weld seam 4 (Table 3).

In addition, it may be noted that the CPU time for the synthesis method calculations is about 18 times shorter than the TORT one. The accomplished comparison indicates that at reasonable cost the calculations necessary for the Metal Embrittlement Surveillance Program should be performed by the synthesis method.

This comparison, however, is not sufficient for estimating the inaccuracy of the neutron flux calculation results because it is based on one and the same calculation method of discrete ordinates.

TABLE 2. Neutron Fluxes Ratio for VVER-1000

Point	Energy range, Mev			
	>5.0	>3.0	>1.0	> 0.5
1	1.03	1.02	0.98	0.99
2	1.05	1.03	1.01	1.01

TABLE 3. Neutron Fluxes Ratio for VVER-440

Point	Energy range, Mev			
	>5.0	>3.0	>1.0	> 0.5
1	0.971	0.969	0.970	0.974
2	0.951	0.957	0.966	0.964
3	0.960	0.961	0.968	0.970
4	0.940	0.950	0.963	0.967

2. IRON SPHERE BENCHMARK

Calculation results in according to the Iron sphere benchmark [5] carried out in Check (SKODA) have been obtained. The ^{252}Cf fission source is placed in the centre of on iron sphere with outer radius of 25 cm. The spectral measurements of the neutron leakage from the sphere have been done using a proton-recoil detector and stilbene crystal spectrometer located at 1 m from the source. There are presented the flux responses above different energy limits. Comparisons of experimental leakage spectrum with the calculated ones obtained by the multigroup neutron cross sections based on ENDF/B-4 and ENDF/B-6 data have been carried out (Fig.1, Table 4).

In the energy region above 1 MeV, of the greatest interest for the neutron embrittlement of reactor vessel, the best consistency with the experiment has been obtained by the data from ENDF/B-6 in VITAMIN-E [5] group structure. The calculations performed by the DLC37F multigroup library (from ENDF/B-4) underestimate the experimental ones. This is in accordance with the another authors results [5,6]. The differences between the experimental and calculated by FENDL [7] (from MAAE) results show that the mentioned version is not good enough for transport calculation.

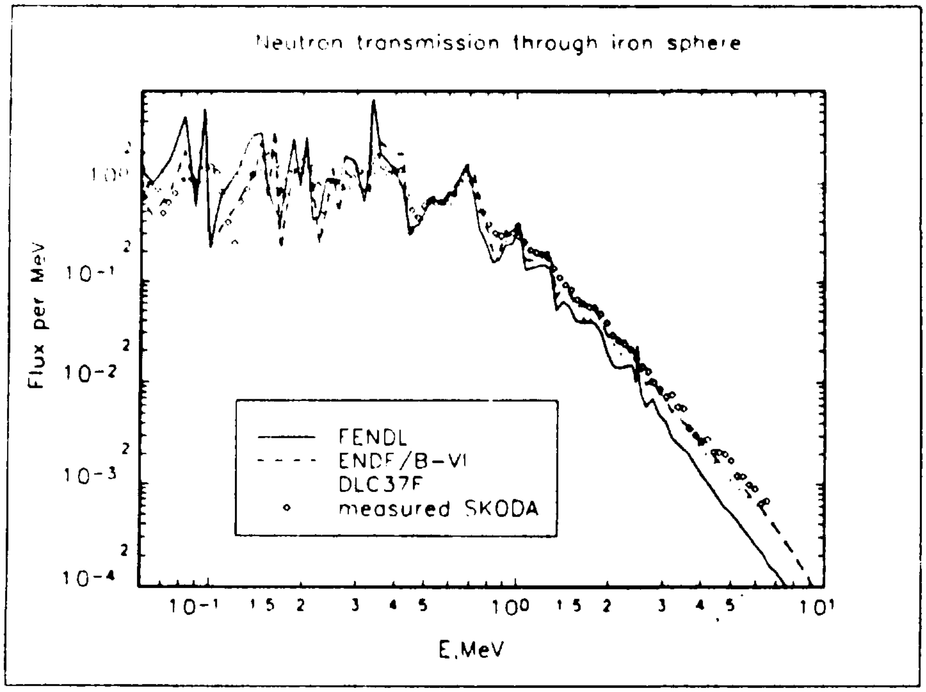

Fig. 1

TABLE 4. Ratio C/E of calculated to experimental SKODA flux responses

Above Energy, MeV	FENDL	DLC37F	ENDF/B-VI
0.1	1.0209	0.9980	1.1107
1.0	0.6412	0.8036	0.9169
2.0	0.5235	0.8152	0.8796
3.0	0.4007	0.7427	0.8021
4.0	0.4639	0.6969	0.7533

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