Neutronic and Thermal Hydraulic Assessment of Fast Reactor Cooling by Super Critical Water with two-path coolant scheme

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Water-cooled reactors with super critical parameters of coolant, developed under program GENERATION IV /1/, are considered as perspective reactors which construction is supposed to be about 2030.

The international community develops concept of the SCWR (Super Critical Water Reactor) reactors with thermal and fast spectra of neutrons /2/. The reactor with fast spectrum of neutrons and tight lattice of pitch SCFR (Super Critical Fast Reactor) is developed for distant perspective.

In Russia the specified concept is called "WWER-SCWR "/3, 4/ according to which water-cooled reactors with super critical water reactor with thermal and fast spectra of neutrons are developed.

In developing projects of SCW reactors the concept of coolant one-pass via core is accepted according to which all heating of the coolant occurs at its movement via core from bottom to top (upcoming). Cause the value of coolant heating is great and equal about 230 ÷ 250 °C than even a small non-uniformity (or peaking factor) in energy release distribution on Fuel Elements (FE) will result in the big differences in coolant output temperatures and in temperatures of FE clad .

In a reactor with a fast spectrum of neutrons the density of coolant changes on height in ~ 10 times and a spectrum of neutrons - from thermal up to fast. In this case it is necessary to use the complex scheme of fuel enrichment profiling on core and introduction a blankets for deriving of required negative void reactivity effect. For reduction of these specified problems it is offered to use the double-pass scheme of cooling the core /4/.

In this report results of researches and pre-design studies of super critical water reactor with use of double-pass scheme of coolant circulation and fast spectra of neutrons are presented.

For a reactor with fast (or fast - resonant) spectrum of neutrons the comparative analysis is carries out on use of the double-pass scheme of coolant circulation with reference project with one circuit reactor installations WWER-SCWR with the one-pass scheme of circulation, which under many characteristics: parameters of reactor installations, FE design , FA, the fuel composition, materials, is close to Japanese project SCFR.

In a reactor with thermal spectrum of neutrons we take the most of designs and the technologies fulfilled in the WWER rectors, while essential simplification of the nuclear installation circulation (one-circuit) and increase of efficiency up to ~ 42 %.

1 Reactor with a fast - resonant spectrum of neutrons

1.1 Scheme of the reactor cooling

In this reactor design SCFR –2X (the Supercritical Fast Reactor with the double pass of coolant) we offered to use cooling system in which the core is divided on radius into the central and peripheral zones with approximately identical number of Fuel Assemblies (FA).

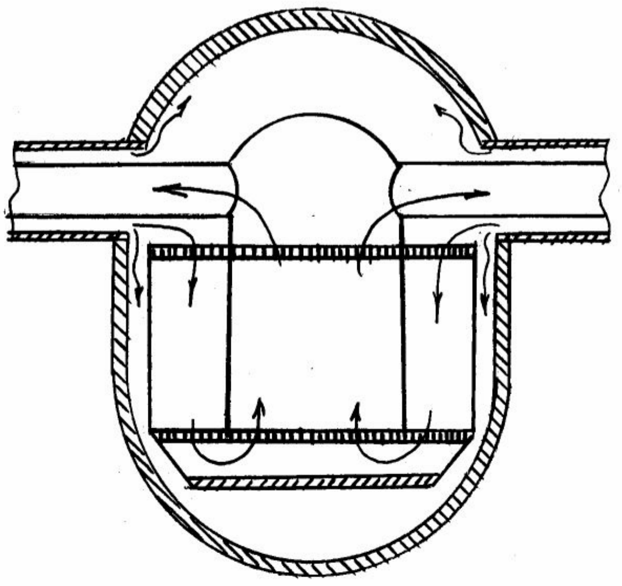


Figure 1 -The scheme of the SCFR –2X reactor cooling .

The peripheral zone is cooled by down coming coolant. In the bottom of the core in the so-called mixing chamber all streams from the peripheral FA are united and are fed to an input to central zone which is cooled by upcoming coolant movement **(fig. 1)**.

The feed water cools all the pressure vessel. The supply and removal of the coolant are carried out via special designed pipes such as “ a pipe in a pipe ”, but separate pipe design is also possible.

Water of supercritical pressure at heating has no phase transitions. However it is possible to mark out a pseudo-critical point at 385 °C about which while change of temperature of water on 15°C its density changes in 2.5 times. Streams of the coolant in dowcoming and upcoming sites are offered to be divided at ~ 395 °C. In downcoming site the coolant will be heated up on 115 °C, its density will change in 3.5 times. In an upcoming site heating of the coolant will make 135 °C, the density will change in 2.2 times. Thus, the spectrum of neutrons on height in each part changes insignificantly, and will change mostly on radius, and in this case it is not required difficult profiling of fuel enrichment at the core for energy release alignment. So all details of Fuel Assemblies (FA) will work at twice smaller temperature differences .

While division of the core into two sites the through passage cross-section for the coolant decreases in 2 times and speed of the coolant respectively increases in 2 times which becomes equal 1.6 m/s on input in a peripheral zone and ~ 15 m/s on an output of the central zone.

Cause of reduction of the coolant flow rate (approximately in 10 times in comparison with WWER) its speed turns out small, power for pumping becomes insignificant (losses on friction will make ~ 0,8 MPa, power for pumping about ~ 2500 kW). At increase of coolant speed the factor of coolant heat exchange increases about in 2 times, that will lead to decrease in temperature of FE clad and to improvement of its durability and reliability .

1.2 A fuel cycle

The basic characteristics of the reactor installation are following:

Parameter	Value
Thermal power, MWt	3830

Electric power, MWt	1700
Coolant pressure, Mpa	25.0
Coefficient of efficiency, %	43.5
Coolant Temperature input / output ,°C	280/530
Coolant flow rate via reactor, tonn/hour	6750
Fuel element (FE) Clad material	Ni- alloy
Clad tube size, mm	∅10.7×0,55
Step of triangular fuel pin lattice, mm	12.0
Fuel	UO_2+PuO_2
Number of fuel pin in the FA, pieces	252
Number of covered FA in the core, pieces	241
Size of cover in a hexagonal lattice, mm	205
Material of FA cover	Ni-alloy
Thickness of FA cover, mm	2.25
Height of the core, m	3.76/3.38
Avearage power density in the core, Wt/cm^3	107
Average heat flux from fuel pin, Wt/cm	158

The fuel composition represents a mix of the WWER spent fuel and weapon grade plutonium. Such fuel loading essentially reduces required amount of weapon plutonium in comparison with an option with the depleted uranium. Besides additional volumes of the spent fuel do not collect at the storages .

At effective density of a mixed Uranium and Plutonium oxides of 9.3 g/cm^3 the density of weapon plutonium oxide makes 0.7 g/cm^3 and is identical in all te FA.

Fuel loading model of the core and FA are presented on the fig. 2, 3.

In the calculation model the central and peripheral zones were divided into four subzones on height with change of average parameters of the coolant, temperatures of fuel and an FE clad (received from preliminary computations (tab. 1)).

Neutronic calculations of a reactor basically carried out by the complex code WIMS-ACADEM developed for three-dimensional hexagonal geometry, in five group approach. Three groups fast and on one group of resonant and thermal neutrons have been allocated. The low energy power borders of groups were 1.35, 0.111 MeV, 9.12 keV, 4 and 0 eV. Group constants for each subzone of a reactor depending on value of burn-up were calculated by modified code WIMS-4D. Fuel cycle with 5 partial reload of core FA once in each calendar year is chosen. Reloading FA could be discharged as from central as peripheral zones, and between zones.

Results of fuel cycle calculation in comparison with usual WWER-SCR /4/ with the one-pass cooling scheme and blanket from depleted uranium are presented in the Table 3.

The 21-group spectra of neutrons for the top and bottom subzones in the center of central zone (Fig. 4a) and in the middle area of peripheral zones (fig. 4б) are received. For the chosen three energy intervals: fast neutrons between 10 MeV ≥ E≥ 0.1 MeV; resonant between 0.1 MeV≥ E≥ 1.01 eV and thermal lower E = 1.01 эВ relative group neutron fluxes (fig. 4a, б) and fission neutrons ($\acute{\varepsilon}_{fis} = \nu_f * \Sigma_{f}, * \Phi$) (fig. 5) in shares from their total value for inlet and outlet parts of the core are calculated.

From the presented results of calculations (fig. 4, 5) it is visible, that:

- Spectra of neutrons not strongly change on height of the core;
- 60 % fissions are caused by neutrons of intermediate energy, 30 % by fast, only 10 % - by thermal neutrons.

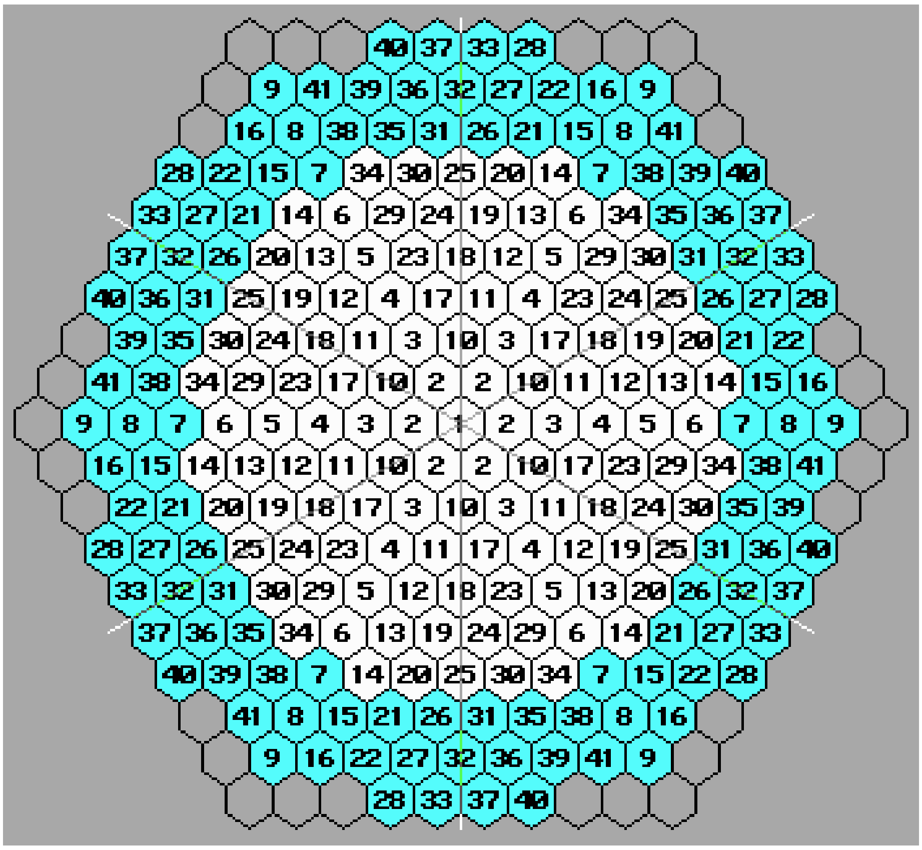


Figure 2 - the Cartogram of the core.

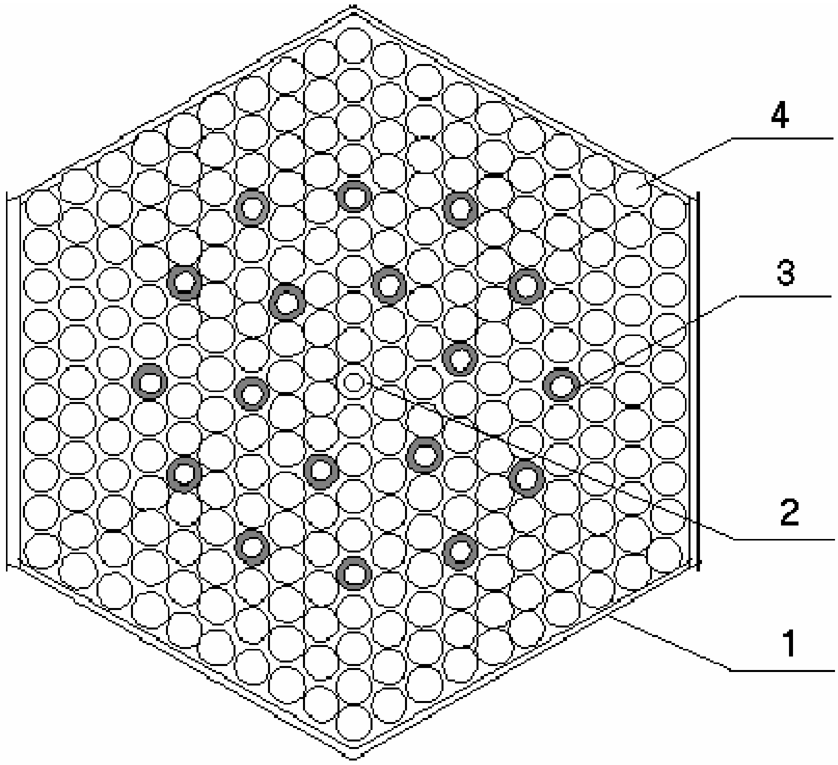


Figure 3 – Fuel Assembly Cross-section: 1 - a cover of 2.25 mm thickness; 2 - the central tube with a size ∅10,7 mm × 1 mm; 3 - 18 directing tubes for absorber control rods in size ∅ 10,7 mm ×0,55 mm; 4 - 252 Fuel elements, clad size of ∅10,7 mm × 0,55 mm, with a step of 12 mm. A construction material of all elements is a Nickel alloy.

Table 1. – Variation of chosen thermal-hydraulic parameters used in the FA calculation model by height of the core.

Lengths of parts, cm	Coolant density, g/cm^3	Temperature, °C		
		Coolant	Clad	Fuel
	Periphery part			
108	0.76	290	300	600
120	0.68	340	352	720
80	0.45	370	385	860
68	0.3	388	405	990
	Central part			
68	0.2	403	430	1000
80	0.15	420	465	1050
120	0.12	450	530	1100
108	0.095	513	575	1080

Similar neutronic calculations were carried out by Monte-Carlo code MCNP-4A with burn-up modules / 5/. Neutron spectrum of upcoming part of the core is presented at the Fig.6.
Variation of Pu concentrations is presented on the Fig.7.

1.3 Efficiency of control rods, reactivity coefficients and fuel breeding factor

For estimation of control rod (CR) efficiency the 4 following conditions of a reactor are considered:

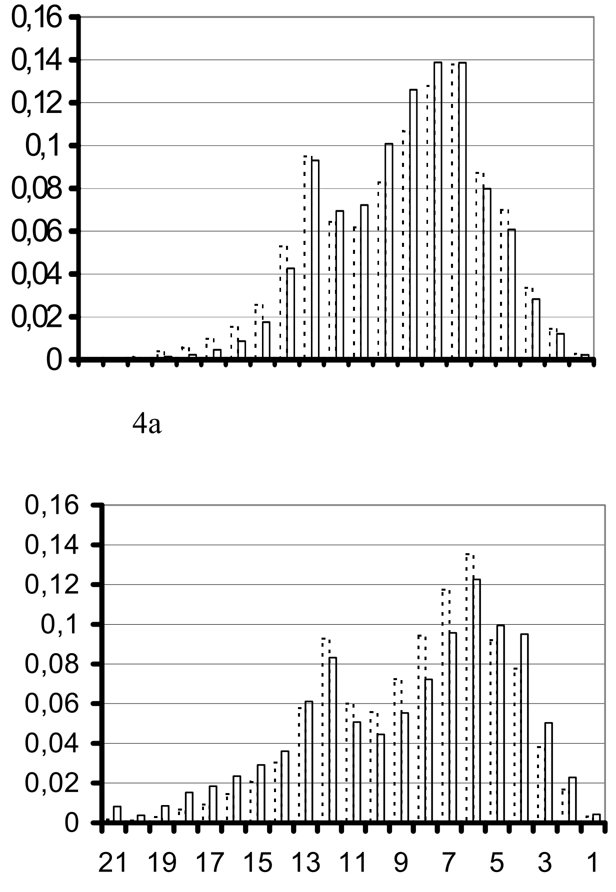
- Operation on the nominal power $N = N_{nom}$;
- Operation on the minimally controlled level at which all the core is filled with a feedwater with temperature 280 °C, pressure 25 MPa;
- Empty core at which in the core, and also in reflectors, there are only steam with a density of 0.09 g/cm3;
- Flooding of all reactor by cold water at temperature 20 °C and pressure 10^{-5}MPa.

For the specified calculation states we derived the values of start reactivity excess, required amount of control rod assemblies for its compensations and for shutting reactor to the subcritical state with K_{eff}= 0.98 (tab. 2).

From data of the Tab. 2 it is clear, that at a core flooding by cold water it is necessary to place CR assemblies in the 216 FA from total 241 FA (except for 25 FA of peripheral lines). For the established mode of stationary overloads we carried out calculations of a water void of the core on the beginning and the end of campaign, reactivity coefficients $\partial\rho/\partial x$, where x are density, temperature of the coolant (with account of density variation), temperature of fuel and steam percentage accordingly (Tab. 3).

The conversion ratio is determined as the ration of total amount of fissile nucleus (U5 + Pu9 + Pu41) in discharged and in fresh fuel, is equal 1.013 in central zone, and 0.853 in peripheral zone and averaged over reactor is 0.933.

On the basis of these results it is possible to make a conclusion that at the suggested cooling scheme of the reactor neutron leakage from the core decreases, a spectrum of neutrons in a reactor is fast - resonant, that together with a use of a MOX-fuel composition (spent fuel + weapon Pu) results to essentially smaller enrichment of fuel and to negative void reactivity coefficient.



4a

4b

Figure 4 - The share of a group neutron flux in the center (a) and in a peripheral zone (б):
☐- Core Top, - ☐- core bottom.
j- -is a group number

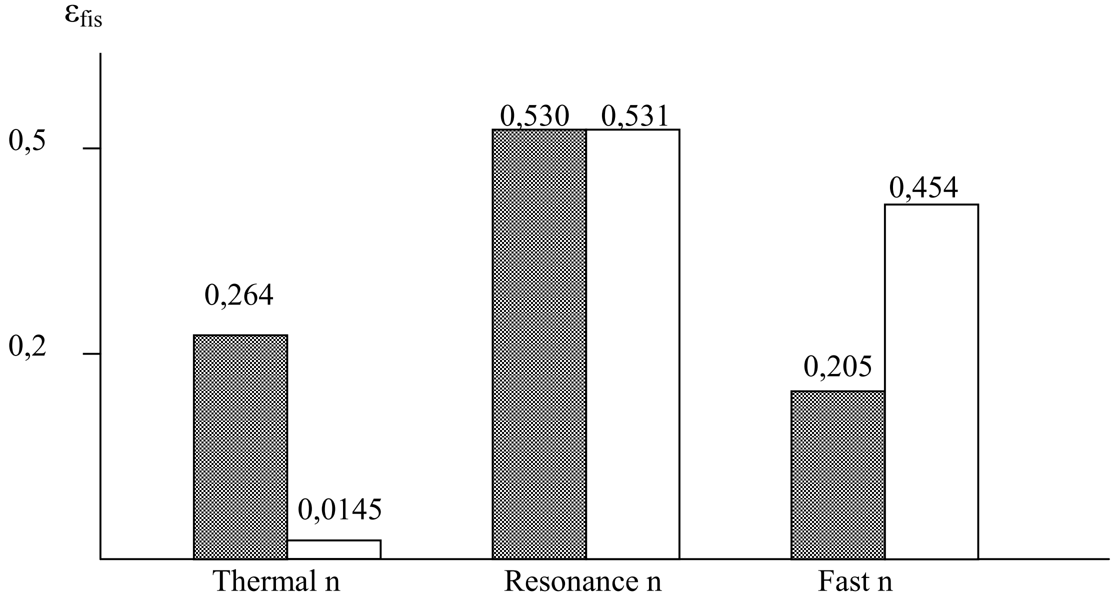


Figure 5 - the Relative contribution ($\acute{\varepsilon}_{fis}$) for neutrons of different energy groups in total amount of fission on the inlet (■), and outlet (□) part of the core

Table 2 - Overcriticality and required number of control rods for its compensation.

State	ΔK, %	$N_{\text{FA with CR}}$
N_{nom}	1.26	12
Minimally controlled level	7.265	120
Void	-2.26	–
Cold	13.679	216

Table 3 - Physical characteristics of a fuel cycle at various scheme of heat-removal.

Characteristics	Value	
	One-pass	Two-pass
Initial load of weapon-grade Pu into the core, tones	15.68	9.47
Loading of the mixed fuel in one FA, kg	598	560.6
Loading weapon Pu in one FA, kg	67.88	39.3
Part of FA overloads	1/5	1/5
Excess Reactivity reserve on campaign, %	1.5	1.26
Duration of an inter reloading interval , eff. Days	250	300
Amount of FA types /fuel of different enrichment in the core	4/16	1/1
Average energy release of discharge FA, MWt*day/kg. H.n	33.3	39.79
Maximal peaking factor of power density by FA /by the core volume	1.22/2.33	1.46/2.19
Loading of fissile isotopes ^{235}U, ^{239}Pu, ^{241}Pu, tones/year	2.65	2.34 (Weapon Pu = 1.89)
Discharge of fissile isotopes tones/year	2.48	2.18
Breeding ratio	0.936	0.933
Void reactivity effect in the beginning / end campaign, %	0.2/0.562	-5.88/-3.64
Reactivity coefficient for $N = N_{nominal}$ on the beginning / end of an interreloading interval:		
$\partial\rho/\partial\gamma\cdot10^{-2}$, cm^3/kg	1.64/-0.548	8.03/5.20
$\partial\rho/\partial T_T\cdot10^{-4}$, 1/°C	-1.09/0.646	-2.40/-1.9
$\partial\rho/\partial T_{fuel}\cdot10^{-5}$, 1/°C	-1.85/-1.77	-1.50/-1.45
$\partial\rho/\partial S_п\cdot10^{-4}$, 1/% steam	-	-6.0/-4.0

2 Reference reactor with a thermal spectrum of neutrons

Design and the sizes of the reactor vessel , in-vessel elements, cartogram core loading, FA and FE sizes is supposed to be maximally close to the WWER-1000 reactor . The basic characteristics of a reactor are as following:

Power, MWt	
Electric	1200
Thermal	2700
The coolant:	
Pressure, MPa	25
Temperature on an input / output, °C	280/510

Flowrate, tonnes/hour	5440
Height / equivalent diameter of the core ,m	3,55/3,16
Number of FA, pieces	163

Conclusions

The suggested double-pass scheme of coolant circulation with supercritical pressure in water-cooled reactors with fast - resonant and thermal spectra of neutrons allow us to realize some advantages in comparison with another similar projects of NPPs. At realization of the specified scheme of coolant circulation we got such an advantages as:

- temperature difference at which FA constructive elements are working decrease in 2 times;
- speed of coolant flow increases in 2 times, but owing to the small flowrates, related to supercritical water use, this speed is even lower, than in the WWER, thus the heat exchange coefficient will increase and the temperature of FE clad to be reduced;
- Required core non-uniformity of energy release distribution could be ensured without complex profiling of fuel enrichment;
- the coolant heating via height on the core upcoming site decreases in 2 times that will result in reduction of non-uniformity in coolant temperature distribution between FA;
- Negative feedback on key parameters are ensured to be negative: to temperature and density of the coolant, temperature of fuel, void effect (without use of additional measures – such as blanket introduction , solid moderator for a reactor with a fast spectrum of neutrons);
- Only small excess of reactivity is required on burn-up and the most difficult modes of operation (flooding by cold water), and this excess can be provided by regular means - an arrangement of absorbing control rods in 2/3 FA.

References

1. A Technology Roadmap for Generation IV Nuclear Energy Systems. Technology Road Map Report, September 23, 2002.
2. Oka Yoshiaki. Review of Temperature Water and Steam Cooled Reactor Concepts - The First International Symposium on Supercritical Water-cooled Reactors, Design and Technology [SCR-2000], Rep. № 104, November 6-9, 2000, The University of Tokyo, JAPAN.
3. Yu.Baranaev, P.Kirillov, V.Poplavsky, V.Sharapov Nuclear reactor on super critical pressure water Atomnaya Energia 2004, v. 96, #. 5, p. 374 – 380.
4. Glebov A.P., Klushin A.V. Reactor with a fast - resonant neutron spectrum, cooled by water of supercritical pressure with a double –path scheme of coolant movement. Atomnaya Energia, 2006; - v. 100; - iss.#. 5.
5. V.F.Ukraintsev, V.V.Kolesov Transfer of Nuclear Data Uncertainties to the Uncertainties of Fuel Characteristics by Interval Calculation. In: Proceeding of the WONDER-2006, Cadarache, France, p.54

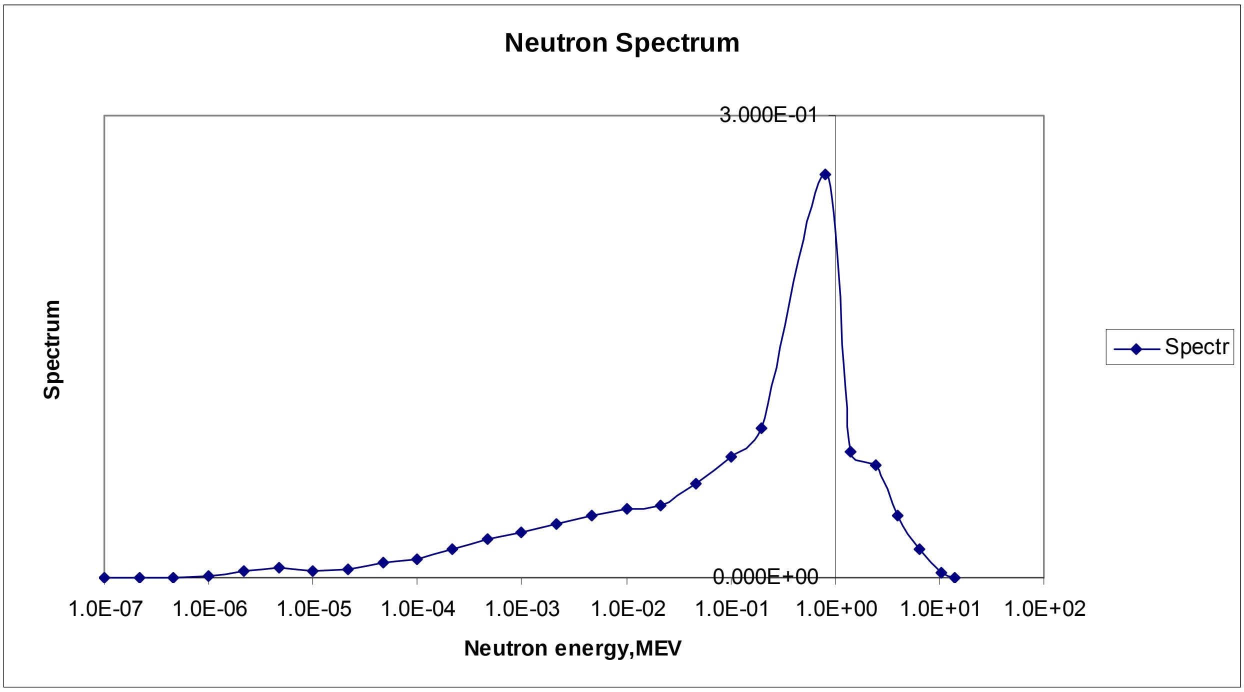


Figure 6. Neutron spectrum of the SCFR –2X on the upcoming part