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SVBR-TYPE REACTORS WITHOUT ON-SITE REFUELING CONCEPT DEVELOPMENT AND SCALING FACTORS IN SAFETY.

(Year report)

IAEA CRP on “Development of Small Reactors Without On-Site Refuelling”

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INTRODUCTION

Development of advanced or new generation nuclear power technologies is directed on the solution of the following four main issues:

- improved safety, based on inherent reactor properties;
- economical efficiency;
- non-proliferation of nuclear materials;
- long-living radioactive waste management and long-term fuel supply.

Solution of the last problem is impossible without introduction in a nuclear fuel cycle technologies of fast spectrum reactors. In this sense development of fast spectrum reactor technologies in short-term and long-term prospect is predetermined by the scale of problems arising from recycling of radioactive wastes and is a strategic direction of high technologies development in nuclear power. Application of liquid metal coolants in such reactors is the natural solution determined by their properties to not moderate and poorly to absorb neutrons. The key restraining factors are dearness of already developed technology of fast sodium cooled reactors and their potential problems in safety connected to fire danger inherent in them.

The use of eutectic lead - bismuth alloy as the coolant of fast reactors allows to improve essentially economic parameters of fast reactors due to its unique properties:

- high temperature of boiling ~1700^{O}C;
- no active chemical interaction with water and air (fire safety);
- convenient from the engineering point of view melting temperature ~123.5 ^{O}C which on the one hand provides the minimal additional requirements to operation, and on the other hand allows to use actively "freezing" of the coolant for conservation and transportations of these reactors, including transportation with the loaded fuel. Thus the heavy coolant represents itself as an additional safety barrier and physical protection of nuclear materials against the non-authorized access.

Realization of the mentioned above opportunities and features is most fully realized for reactors of small and average capacity in a range from 10 to 100 MW (e).

Realization of extra long core lifetime (corresponding to continuous work on rated power during ~70000-150000 hours) with minimum reactivity loses and, accordingly, a high level of safety based on inherent properties of used materials is possible only for fast neutrons reactors in above mentioned power range.

Preliminary results of transportable reactor facilities SVBR-10 development are presented below. At the first stage most efforts were addressed on the following two primary goals:

- formation of the concept and requirements to transportable reactor installation with extra long core lifetime and without on-site refueling;
- development of reactor design providing the increased life time and reliability of the equipment for exception (or minimization) operations with unsealing of the primary circuit.

Presented below investigation is the year report for IAEA CRP I25001 "Small reactors without on-site refueling".

1 SCOPES AND CONCEPT OF FUEL AND REACTOR FACILIES TREATMENT

Reactor facilities SVBR-10 is intended for transformation of a nuclear energy in thermal power and supply the turbine generator of nuclear power plant with the steam of given parameters.

SVBR-10 can be used in nuclear power plants of low power (NPPLP) in conditions of the undeveloped infrastructure.

SVBR-10 is the basic part transportable reactor unit (TRU) - transportable module functionally ready to application, completely the factory manufactured and delivered to a NPP site (or taken out from NPP site) by water or another way. Transportable реакторные units with SVBR-10 can be applied in power units of nuclear power plants of low power for multiple purpose, such as electricity production, heat supply or sea water desalination. Depending on required capacity of the nuclear power plant, in its structure can be used single TRU, as well as several TRU joint at the NPP site into modular nuclear power steam generating facilities.

« The nuclear island » of NPPSP is formed on the basis of transportable реакторных units with SVBR-10, delivered on NPPSM site by sea or waterway. Structure of NPPSP includes constructions and replaced TRU with SVBR-10. Constructions of NPPSM are the property of the country - user (Customer). Transportable reactor units are similar to charged « nuclear batteries » and are delivered by a principle «Build — Own — Transfer in rent ». It means, that the Supplier transfers the Customer replaced TRU in rent for the term of, determined by the core lifetime and time fore reactor cooling for coolant freezing (~20 years). In NPPSP site one or the several TRU are simultaneously maintained.

After manufacturing and mounting of reactor facilities equipment in TRU necessary production tests are carried out, loading of the core and filling by coolant (LBE) are made. Then hot tests are carried out with shut downed core, and reactor facilities it is maintained before LBE coolant freezing. After that TRU is transported to NPPSP site and installed in its place of reactor building, protecting TRU from external influences. Further TRU is connected to supply and control systems of plant, so its start-up and commissioning can be made. Connection of TRU to auxiliary systems, start-up and commissioning of TRU are carried out without the other units shut down.

After connection of the new one, TRU with the wasted core is decommissioned from operation and is transferred into cooling mode up to primary coolant freezing. Reactor module has no refueling operations at the NPP site.

After coolant freezing TRU is transported to the country - supplier for the core refueling, necessary repair work and replacements of the equipment with limited life-time. The released place on site is used for installation next TRU.

During the decommissioning of NPP the last TRU after necessary cooling down is transported to the country – supplier. A radioactive waste on NPPSP site does not remain.

2 BASIC CHARACTERISTICS OF REACTOR FACILITIES

SVBR-10 has two circuit heat transfer from the core.

The primary coolant – lead – bismuth eutectic alloy (Pb – 44,5 %, Bi – 55,5 %).

Protective gas of the primary circuit – argon.

Secondary coolant – waters / steam.

Configuration of reactor – integrated (pool-type).

Basic characteristics SVBR-10 are resulted in table 2.1.

Table 2.1 - Basic characteristics SVBR-10

Parameter	Value
Electric capacity (gross), MW	12
Thermal capacity (nominal), MW	43,3
Steam capacity, tons/hour	56
Parameters of generated steam:	
- pressure, MPa	4,2
- temperature, °C	410
- pressure in the steam separator, MPa	4,6
Temperature of feedwater, °C	105
Temperature of the primary coolant, °C:	
- core outlet	480
- core inlet	320
Core dimensions D × H, m	1,086 × 0,9
Number of control rods,	31
including, transportation control rods.	3
Fuel: - typy	UO_2
- loading on U-235, kg	755
- average enrichment, at. %	18,7
Core lifetime, eff. hours	135 000
Operation time of the core, years	~17
Time between refueling, years	~20
Design service life of the irreplaceable equipment, years	60
Number of heat exchange loops	2
Number of heat exchanger modules	4
Number of I circuit pumps	4
Overall dimensions:	
- diameter, m;	3,100
- height (including control rod drive), m	7,100

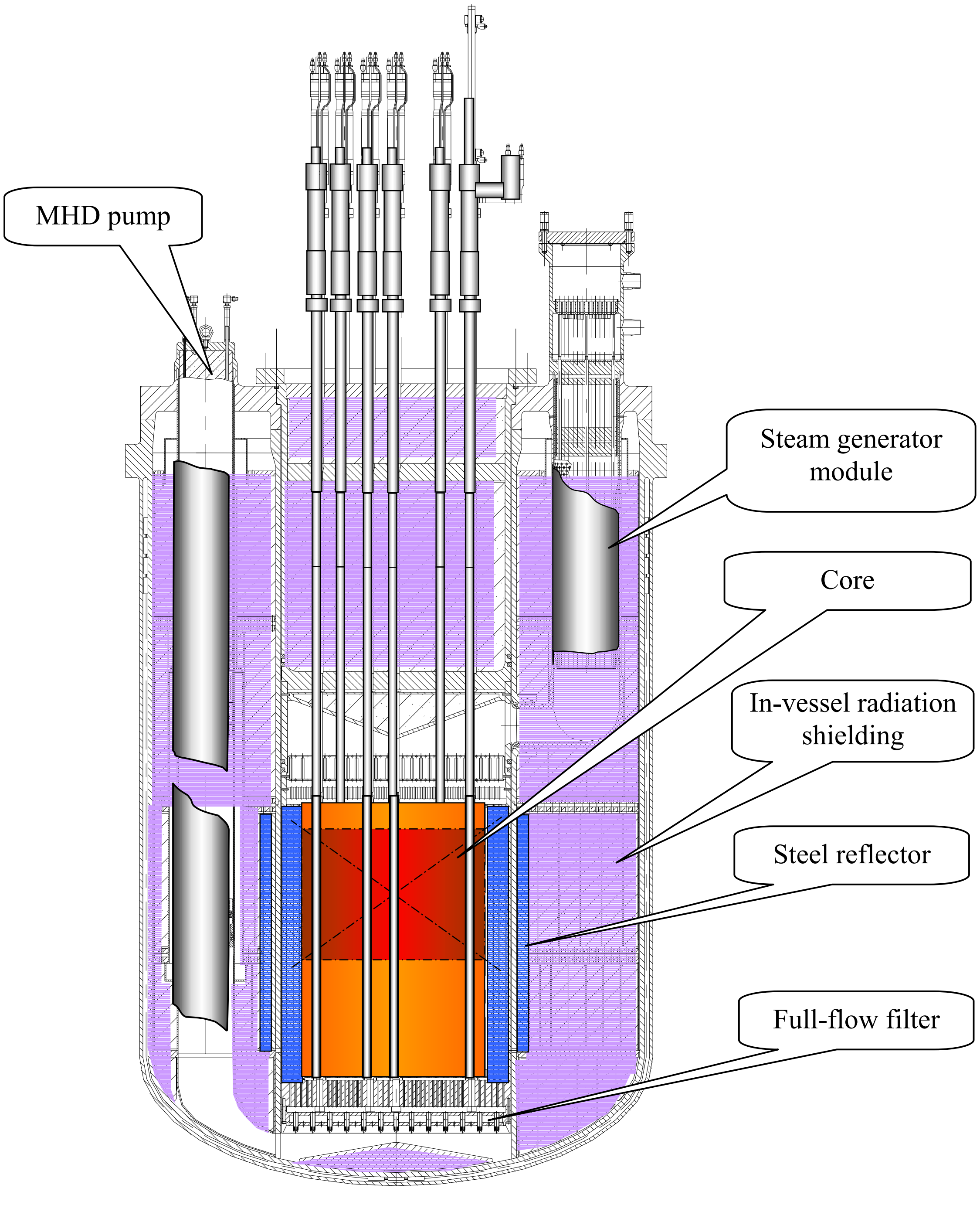

Fig. 2.1 General view of SVBR-10.

General view of SVBR-10 is presented in fig. 2.1. SVBR-10 reactor unit with auxiliary systems is placed inside TRU (fig. 2.2), representing a tight steel vessel. Water tank of Passive Heat Removal System (PHRS) provides emergency heat removal during long time plant and used as radiation shielding. Radioactive wastes storage is placed in PHRS tank also.

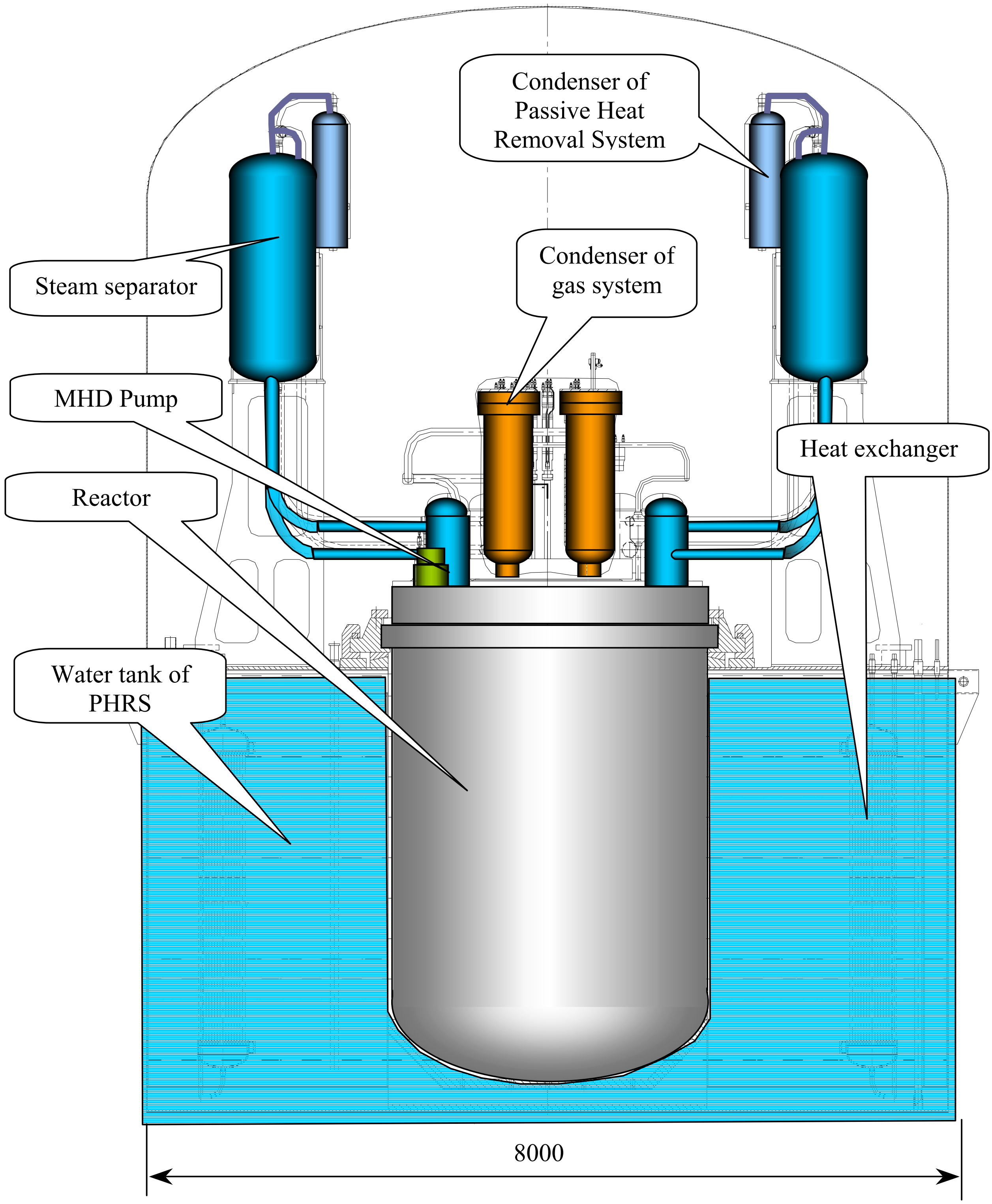

Fig. 2.2 General view of TRU

Control rods system includes three types of control rods (fig.2.3). At the present stage the system contains 22 rods compensating reactivity loss during core life time (KC), two control rods for reactor power control (PC) and four emergency control rods (A3) (fig. 2.3).

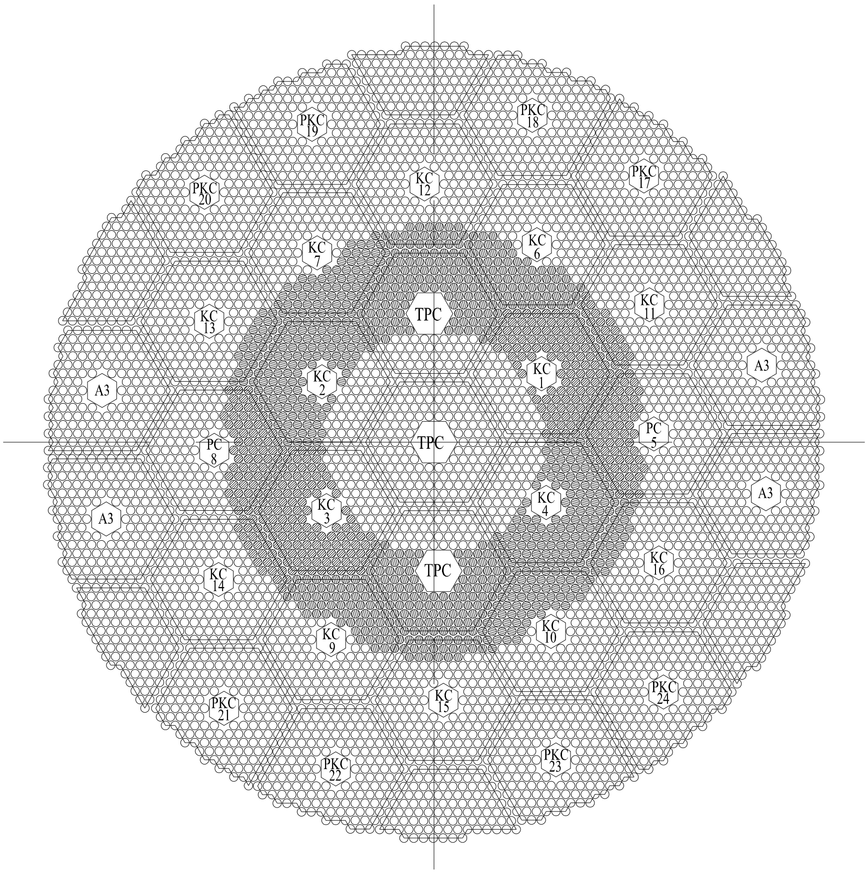

Fig. 2.3. SVBR-10 core arrangement

Absorbing material of control rods – enriched (to 80%) boron carbide.

The core is surrounded by steel reflector. Reflector thickness in radial direction is 250 mm.

Loading of uranium - 235 is G_5 = 755 kg, full loading of uranium – 4046 kg. Three groups of fuel rods are used for profiling of the heat release field in radial direction of the core. Enrichment in radial zones grows from the center to periphery. Number of fuel rods and enrichment of fuel in them are resulted below:

- enrichment 16 %- 432 fuel rods;
- enrichment 17,2 % - 1164 fuel rods;
- enrichment 19,7 % - 3768 fuel rods.

Average enrichment in the core is 18,7 %.

Value of $\beta_{эфф}$ is:

- in the beginning of core life time 0,007245;
- at the end of core life time 0,006185.

Results of neutronics calculations are presented in table 2.2.

Table 2.2 SVBR-10 core characteristics during operation

Parameter	Value
Change of reactivity $\Delta\rho(T)$, % ($\beta_{эфф}$)	7,08 (9,78)
Average quantity fission products in fuel rods, g/fuel rod	45,3
The maximal burn-up, % heavy atoms	8,83
The maximal non-uniformity of the heat release field (it is realized at the moment of time t = 48000 eff. hours) Kr_{max} Kv_{max}	 1,276 1,570
The maximal neutron flux for neutrons of all energies φ_{max}, n / (sm^2·s) (it is realized near to the end of core life time)	$7{,}35 * 10^{14}$
Maximal fluence of neutrons ($E_n \geq 0{,}1$ MeV), n/sm^2	$2{,}0 * 10^{23}$
The maximal damaging doze for fuel rod cladding, dpa	68,7

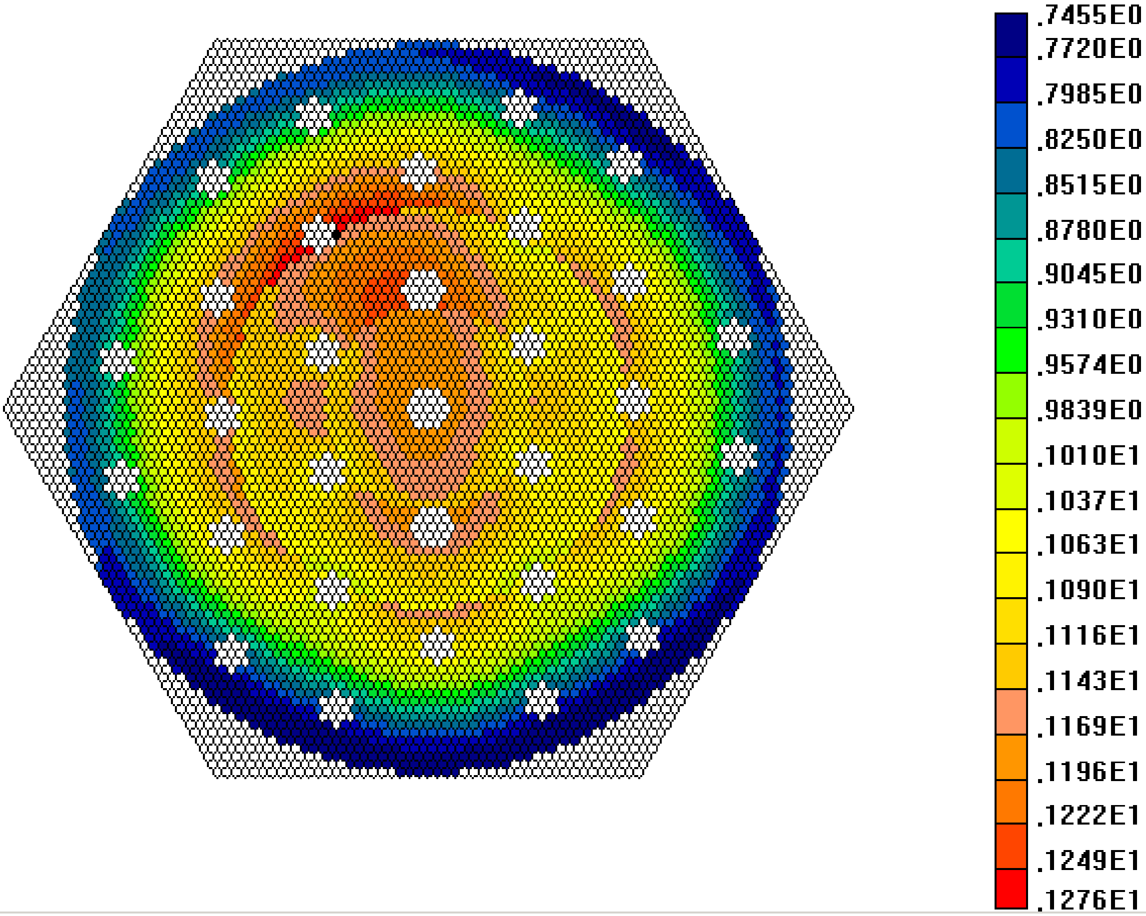

Fig. 2.4 Non-uniformity of integrated in height heat release fields at the moment of time when Kr_{max} is realized (a black point – fuel rod with Kr_{max})

Results of reactivity feedback calculations are submitted in table 2.3.

Table 2.3 Components temperature reactivity feedback

Component	Value, 10^{-5} 1/K
Doppler feedback in temperature interval from 200°C up to 700°C	-0,86--0,47
Axial expansion of the fuel	-0,21
Expansion of cladding	-0,0022
Expansion of the coolant	-0,046
Expansion of the bottom lattice of fuel rods	-0,18
Expansion of the top lattice of fuel rods	-0,43
Expansion of radial reflector	-0,15

All reactivity coefficients are negative, selected control rods system meets all safety requirements and provide reactivity compensation during core burn up, core shut down by two independent control rods groups, including failure of the most efficient control rod.

Radiation shielding of TRU was selected according to the following requirements:

- the irradiation of the personnel and the population at normal operation and emergencies should not exceed specifications /4, 5/, namely:
  a) doses for direct radiation to the public during normal operation and incident conditions is <0.1mSv/a;
  b) operational staff doses during normal operation and incident conditions - 5mSv/a for individual effective dose;
- during transportation of TRU radiation conditions should meet the requirements / 6 / and / 7/, namely:
  a) the maximum radiation level at any point on any external surface of a package under exclusive use shall not exceed 10 mSv/h.
- after plant decommissioning (evacuation of all TRU) a great bulk of concrete (the reactor shaft, building) can be released from the regulating control (i.e. requirements of radiating safety are not distributed to the given source), as according to recommendations of / 8 / and of IAEA / 9/under any conditions of its use the annual effective doze will not exceed 10mkSv.

Results of activation dozes computation are presented in fig. 2.5. Radiation shielding meets established requirements and, thus, one of the key radiation shielding design target is achieved.

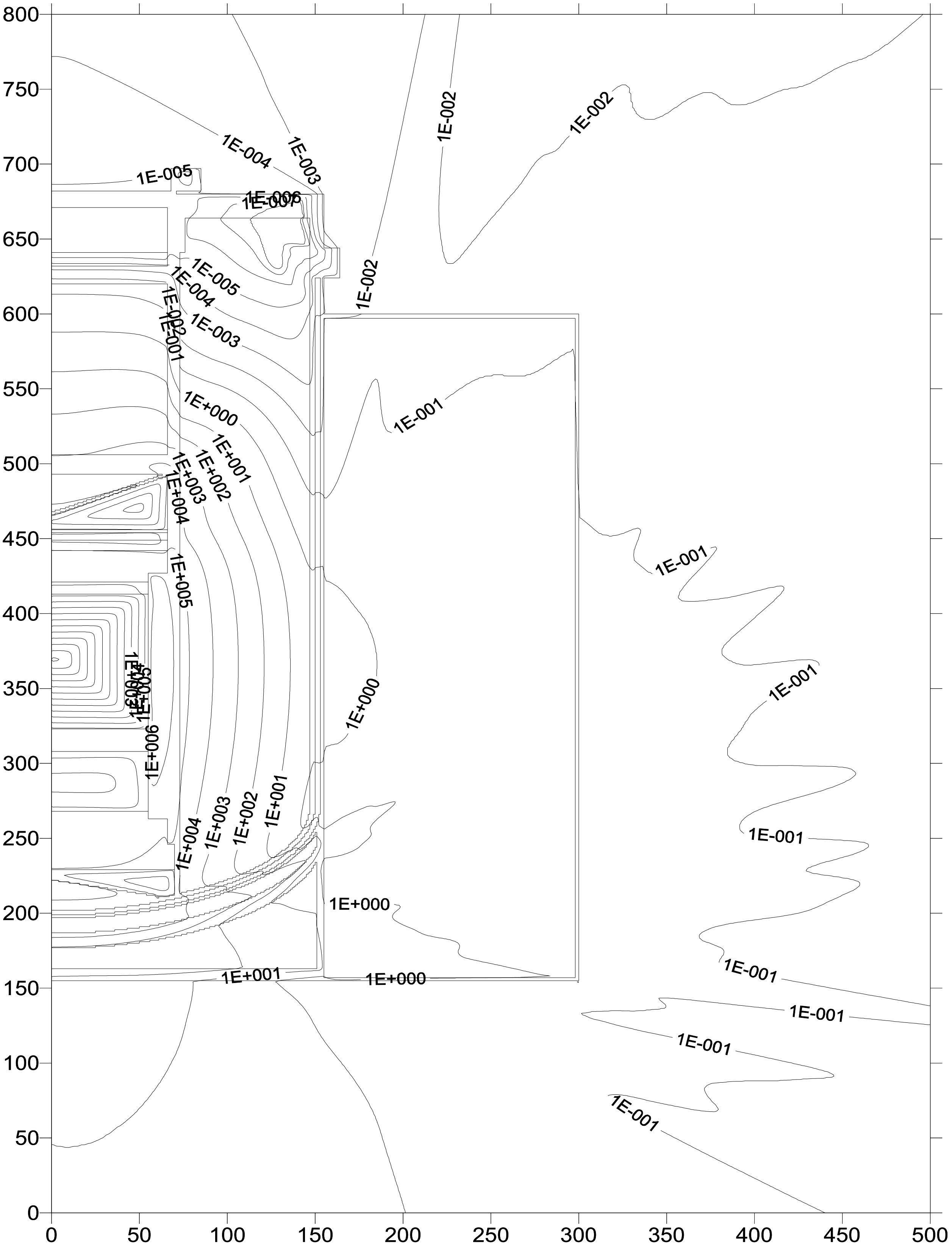

Figure 2.5 – Effective doze of activation radiation, mSv/hour

Dependence of a residual heat in the core at the end of core life time (after 150 000 hours of irradiation) from the time after reactor shutdown is presented on fig. 2.5.

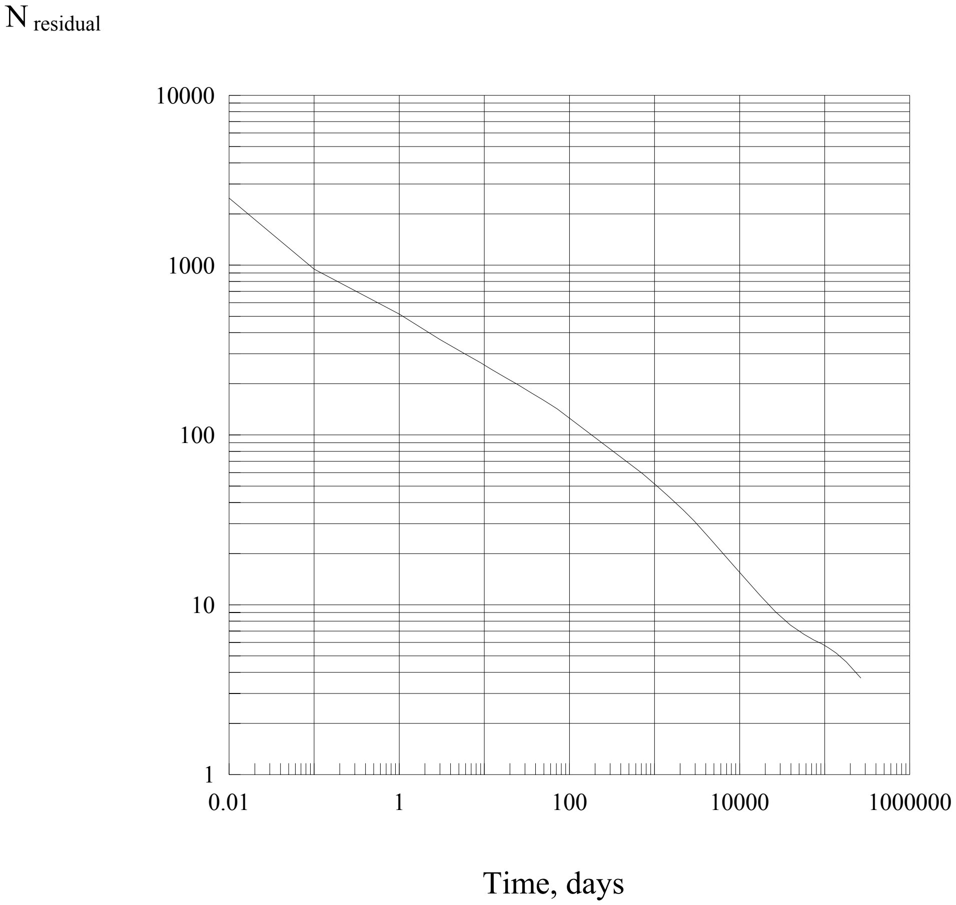

Fig. 2.5 Dependence of the core residual heat from time.

At present time heat loses from the reactor vessel during long term cooling down are estimated at the level about 10-12 kW. It means that primary coolant will start solidification not earlier then after 1-1,5 year of cooling down. Coolant solidification in the core will occur after 1,5-2 year. Such issue can decrease economical profits from the concept of reactor transportation only after full primary coolant solidification. This question needs in more detailed investigation, as influence significantly on safety and non-proliferation strategy.

Estimation of activity of PHRS water of tank was made in the following conservative assumptions:

- water is not removable during all core life time;
- corrosion of stainless steel is constant value ~ $1.68 \cdot 10^{-4}$ g / ($m^2 \cdot h$);
- content of Co-59 in steel was accepted 0.07 % weight.;
- products of corrosion presents in water in the dissolved form.

Activity of water is estimated by Fe-55 ($T_{1/2}$=2.7 years)-$5.46 \cdot 10^3$ Bk/l and Co-60 ($T_{1/2}$=5.27 years)-$1.0 \cdot 10^2$ Bk/l.

Thus, water of the PHRS tank can be determined as low level radioactive wastes.

SUMMARIES AND CONCLUSION

1. Preliminary concept of transportable reactor unit SVBR-10 without on-site refueling of 10 MW(e) power is developed. Improved safety and non-proliferation issues of reactor unit are provided by lead-bismuth eutectic use as the primary coolant and fast spectrum core design selection.
2. Preliminary core layout is selected. Control rod system selected and provide improved safety by use of control rods with low efficiency (each rod efficiency is less than β_{eff}).
3. Necessary data for further transient assessments and accidents simulation are obtained from neutronics calculations.
4. Radiological hazards are estimated and found that use of low power units can provide improved radiological conditions at NPP site after decommissioning, i.e. reactor buildings and auxiliary systems can be potentially released from operational control immediately after transportable reactor units evacuation.
5. Extra long core life time and tight equipment arrangement makes difficult to achieve full core “freezing” in the primary coolant in reasonable time, less than 1-1.5 year. This issue is addressed for further additional investigations.

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