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# MOX LTA Fuel Cycle Analyses—Nuclear and Radiation Safety

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**A Russian Contribution to the Fissile Materials Disposition Program**

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Leipunski Institute for Physics and Power Engineering

*Joint U.S. / Russian Project to Update, Verify and Validate Reactor Design/Safety Computer Codes Associated with Weapons-Grade Plutonium Disposition in VVER Reactors*

# MOX LTA Fuel Cycle Analyses. Nuclear and Radiation Safety

**General Order 85B-99398V**

## (Report)

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Document issued according to **Work Release 02. P. 99-5a and 99-5b** consists of 2 parts that consider correspondingly problems of nuclear and radiation safety in transportation operations at a VVER-1000 NPP with 3 MOX LTAs particularly at Balakovo-4.

# Part 1

# Nuclear Safety

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## Abbreviation

BV - cooling pool (non-compact)
FA - fuel assembly
MOX - mixed uranium-plutonium fuel
NPP - nuclear power plant
UBV - compact cooling pool
UOX - uranium-oxide fuel
VVER - water cooled water moderated power reactor

## Abstract

Tasks of nuclear safety assurance for storage and transport of fresh mixed uranium-plutonium fuel of the VVER-1000 reactor are considered in the view of 3 MOX LTAs introduction into the core. The precise code MCU that realizes the Monte Carlo method is used for calculations.

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# 1. Introduction

Actually in Russia the principle of direct replacement of uranium fuel by MOX fuel is being applied while designing VVER-1000 fuel cycles with partial loading by weapons-grade MOX fuel. FA design is not noticeably modified. Particularly, in fuel cycle with 1/3 MOX core [1] MOX FA design is identical to that of advanced uranium fuel cycle with Zirconium guide tubes and space grids. Fuel pins with uranium-gadolinium fuel, tested in UOX FAs, are used in MOX FAs as burnable poison rods. Taking into account an equivalence of UOX and MOX FAs design, the use of the existing NPP equipment for MOX FAs transport and storage is considered in present studies.

Fresh MOX FAs in comparison with UOX ones possess the following features as applying to transport and storage tasks:

- availability of a set of plutonium isotopes with fission and absorption cross sections substantially different from U-235 ones,
- increased radioactivity of MOX fuel,
- eventual positive void effect if considerable Pu concentration in MOX fuel.

In present work the studies of nuclear safety in transportation and storage at VVER-1000 NPP with MOX fuel are performed. Initial data and safety criteria are formulated according to "Safety rules for nuclear fuel storage and transport at the nuclear engineering object" [2]. The calculations are performed by the precise code MCU-RFFI/A [3], using Monte Carlo method and neutron libraries formed on the base of estimated nuclear data. The results obtained for UOX and MOX fuel are compared.

## 2. Means for transport and storage of fresh fuel at VVER-1000 NPP

Fresh fuel assemblies are transported from the production plant to NPP by means of the railway in special package sets. The package set consists of two tubes positioned in parallel and welded on to the interlayer. The ends of the pipes have nozzles covered by removable lids. There is a wooden container in the each tube that holds the fuel assembly. The package set is intended for two FAs location.

Storage of uranium FAs at NPP Fresh Fuel Depository is performed either in package sets stacks or in decks or in the covers for fresh fuel. MOX fuel storage is supposed to perform in package sets avoiding an excessive radiation in comparison with other types of storage.

Fuel assembly transport from the fresh fuel depository to the reactor hall is performed in the covers for fresh fuel. The cover for fresh fuel is intended for transport of 18 fuel assemblies placed vertically. The fuel assemblies may be loaded from the cover directly into reactor or if necessary - into cooling pool.

The cooling pool is intended for holding the irradiated fuel assemblies for the time necessary for decrease of residual heat to the values, which allow transportation of spent fuel from NPP. According to the initial Technical project of VVER-1000 NPP, the FAs are placed in non-compact decks. The cooling pool sub-criticality is ensured by the choice of their positioning pitch in the water. This type of positioning of fuel assemblies provide 3 years storage of irradiated fuel assemblies and possibility for emergency unloading of the core at any moment. Recently cooling pools at some VVER-1000 units have been modernized (for example Balakovo 4) by means of the decks for close or compact fuel storage (SUHT) with the use of hexagonal tubes made of borated steel. Such technical decision, which realizes the principle of combined neutron "trap", increased the storage capacity of the decks' lattice 1.8 times and provided sub-criticality for uranium fuel in the accidental regimes while coolant density lowering.

# 3. Physical features of criticality task in fuel transport and storage equipment and brief description of the code MCU-RFFI/A

Keff ≈1.4 for physically big array without gaps of VVER-1000 FAs with 4.4% enrichment (the enrichment of make-up fuel) filled with water without absorber dissolved in it. Such large value of Keff requires special measures to provide nuclear safety of fuel storage and transport. These measures are realized in constructions of facilities to treat fresh and irradiated fuel and in organizational and technical means.

FAs are placed with a distance (gap between FA ~15cm) between them and in some cases absorbers are placed between FAs (storage SUHT) to provide sub-criticality in the means to treat the fuel. These peculiarities of FAs placement in the storage and transport facilities together with the necessity while calculating to change water density, boric acid concentration and geometry in accidental situations stand for physical peculiarity of the task, and distinguish them from traditional tasks of VVER core calculation.

Physical nature of the neutron moderation and diffusion in fuel transport and storage equipment is as follows:

- part of the neutrons that were born in FA (fission spectrum) moderates inside the FA, the other part (approximately equal to the ratio of gaps volumes between fuel assemblies and overall water volume) moderates in gaps without any noticeable absorption;
- the existence of strong scattering anisotropy in hydrogen leads to the predomination of fast neutron diffusion to the gap;
- thermal neutron density in the gap is significantly higher than the density in FA itself, this leads to predomination of thermal neutron diffusion from the gap to the fuel (it is also necessary to take into account scattering anisotropy correctly to calculate criticality in this case);
- the existence of the cover absorbers from boric steel (in SUHT) with strong absorption leads to the non-monotonous neutron distribution function near the FA-to-gap border.

Thus, the criticality calculations require, on the one hand, to take into account scattering anisotropy, which is common for neutron radiation protection tasks; on the other hand, to use detailed space and energy grid, which is common for cells calculations by means of precision methodologies.

The codes using Monte Carlo methods with libraries from estimated neutron data files meet these requirements.

MCU-RFFI/A [3] allows the user to calculate effective multiplication factor Keff and flux functionals in the neutron multiplying systems. The MCU-RFFI/A code is intended to solve the neutron transport equation by means of the Monte Carlo method using estimated nuclear data for systems with arbitrary three-dimensional geometry. The code allows the user to solve both homogeneous (criticality) and non-homogeneous (outer source) tasks. The code allows three-dimensional tasks with different border conditions: leakage through outer surface, white and mirror reflections, translational symmetry.

Subgroup approximation or Bondarenko f-factors are used to take into account cross-sections shielding in the region of unresolved resonances. Besides, pointwise description of cross-sections is allowed for the region of resolved resonances; in this case cross-sections of the most important nuclides are described by "infinite" number of points, because they are calculated by means of analytical formulae based on resonance parameters for each energy point during modeling. Such scheme allows the user to perform calculations at any temperature without preliminary preparation of cross-sections table.

The DLC/MCUDAT-1.0 neutron physic data base is the data support for the MCU-RFFI/A code. DLC/MCUDAT-1.0 includes:

BNAB/MCU - enlarged and modified version of 26-group constant system BNAB,

LIPAR - resonance parameters in the resolved resonance region,

TEPCON - multy-group cross-sections in the thermalization region,

VESTA - the library for modeling of neutron collisions with nucleus taking into account continuous change of neutron's energy in the thermalization region, that is given in the form of the probability tables obtained from scattering laws $S(\alpha,\beta)$.

DLC/MCUDAT-1.0 and the code are verified for main nuclides using the data of more than 400 integral benchmark experiments. The precision of Keff calculations is comparable to the experimental one.

The MCU-RFFI/A code allows one to calculate three-dimensional systems practically of any complication. The systems are described by means of combinatorial geometry as Boolean combinations of primitive bodies. The user has a choice of 13 types of bodies (cylinder, cone, sphere, parallelepiped and etc.). The possibility of using the symmetry of the system and lattices - that are generated by means of multiplication of some of the initial elements - makes it easier to describe the geometry and border conditions. The lattices may include heterogeneity.

The functionals are determined as flux integrals with given weight functions in registration zones, registration objects, and the system as a whole. The borders of registrational energy groups for integration over the energy are set by the user.

The following values are calculated:

- neutron multiplication factor (by the number of collisions, number of absorbtions, combined estimations),
- neutron flux density,
- nuclear reaction rates for separate nuclides and their mixture in the given space-energy intervals,
- few-group constant set for registration objects,
- effective fraction of delayed neutrons.

# 4. Initial data and approaches

Main characteristics of UOX and MOX FAs, used in VVER-1000 fuel cycles with 1/3 MOX core, are presented in [1]. Maximal enrichment on U-235 of UOX FAs is equal to 4.08%. Contents of $U^{235}$, $Pu^{239}$, $Pu^{240}$ and $Pu^{241}$ in MOX FAs equal correspondingly to 0.200%, 3.190%, 0.206% and 0.034%. Zr is used as a FA construction material, UOX fuel pin weight is 1.575 kg, MOX one is 1.600 kg. 6 and 18 pins with uranium-gadolinium fuel are placed correspondingly in UOX and MOX FAs.

According to the requirements of "Safety rules for nuclear fuel storage and transport at the nuclear engineering object" [2] the presented nuclear safety calculations were performed using the following conservatism:

- Fresh UOX and MOX FAs are considered without integrated burnable poisons in FAs;
- $Pu^{239}$ content is increased, $Pu^{240}$ one is decreased in comparison with [1] MOX FA;
- $Pu^{241}$ is taken into account by additional increase of $Pu^{239}$ content;
- Water distributions, leading to maximal effective multiplication factor in the means of transport and storage, are considered;
- Zero boron concentration in water is considered;
- As a rule infinite fuel systems without leakage are considered;
- Auxiliary steel construction elements with strong neutron absorption are not considered.

Taking into account the above-mentioned, the conservative options of UOX and MOX FAs design have been defined (See Table 4.1) and have been used in nuclear safety calculations for the means of transport and storage.

According to the "Safety rules..." [2] the effective multiplication factor in the means of transport and storage during normal operation and in accidental regimes is limited by the value 0.95.

Table 4.1. Main FA characteristics in nuclear safety calculations

| Characteristic | Value | |
|---|---|---|
| | UOX FA | MOX FA |
| Width across flats, mm | 234 | 234 |
| Number of fuel pins (rods) in FA, items | 312 | 312 |
| Number of uranium-gadolinium burnable poison rods in FA, items | - | - |
| Pitch between rod and rod, mm | 12.75 | 12.75 |
| Number of guide channels for control rods, items | 18 | 18 |
| Number of space grids, items | 15 | 15 |
| Material of space grids, guide channels and central tubes | Zr+1%Nb | Zr+1%Nb |
| Rod cladding diameter, mm | 9.1x0,69 | 9.1x0,69 |
| Diameter of guide channels, mm | 13.0x1.0 | 13.0x1.0 |
| Diameter of central tube, mm | 13.0x1.0 | 13.0x1.0 |
| Height of fuel column, mm | 3530 | 3530 |
| Contents of U-235 in the fuel, % | 4.400 | 0.200 |
| Contents of Pu-239 in the fuel, % | - | 4.200 |
| Contents of Pu-240 in the fuel, % | - | 0.175 |
| Contents of Pu-241 in the fuel, % | - | - |
| Mass of fuel in rod, kg | 1.460 * | 1.600 |

* U-235 mass in fuel rods corresponds to U-235 mass in UOX FAs described in [1].

# 5. Criticality in facilities for fuel storage and transport

## *5.1. Package set*

Calculational model of the package set is given in the figure 5.1. Cell vertical dimension is 550 мм, horizontal one is 620 мм. In axial direction cell size is infinite.

Table 5.1 presents Keff values for infinite lattice of FA (with UOX and MOX fuel) placed into package sets. Calculations have been performed for normal operation ("dry" package set) and for accidental regimes with cold water without boron both inside and outside the package set.

**Table 5.1 Multiplication factors in infinite grid of package set cells with UOX and MOX FAs**

| Regime | UOX FA | MOX FA |
|---|---|---|
| «Dry» package set | 0,827 (2) | 0,855(2) |
| Cold water both inside and outside the package set | 0.839 (2) | 0.848 (2) |

As it is seen from the presented results, nuclear safety is ensured under normal operation and principal accident.

Additionally the variant with infinite grid of package sets has been considered where cold water is found only inside the set and not outside. In this case the effective multiplication factor is equal to 1.063 (2) and 1.075 (2) correspondingly for UOX and MOX FAs. The limiting effective multiplication factor value ( Keff<0.95) in this case of water distribution is ensured for package sets location in 4 rows without limitation in width.

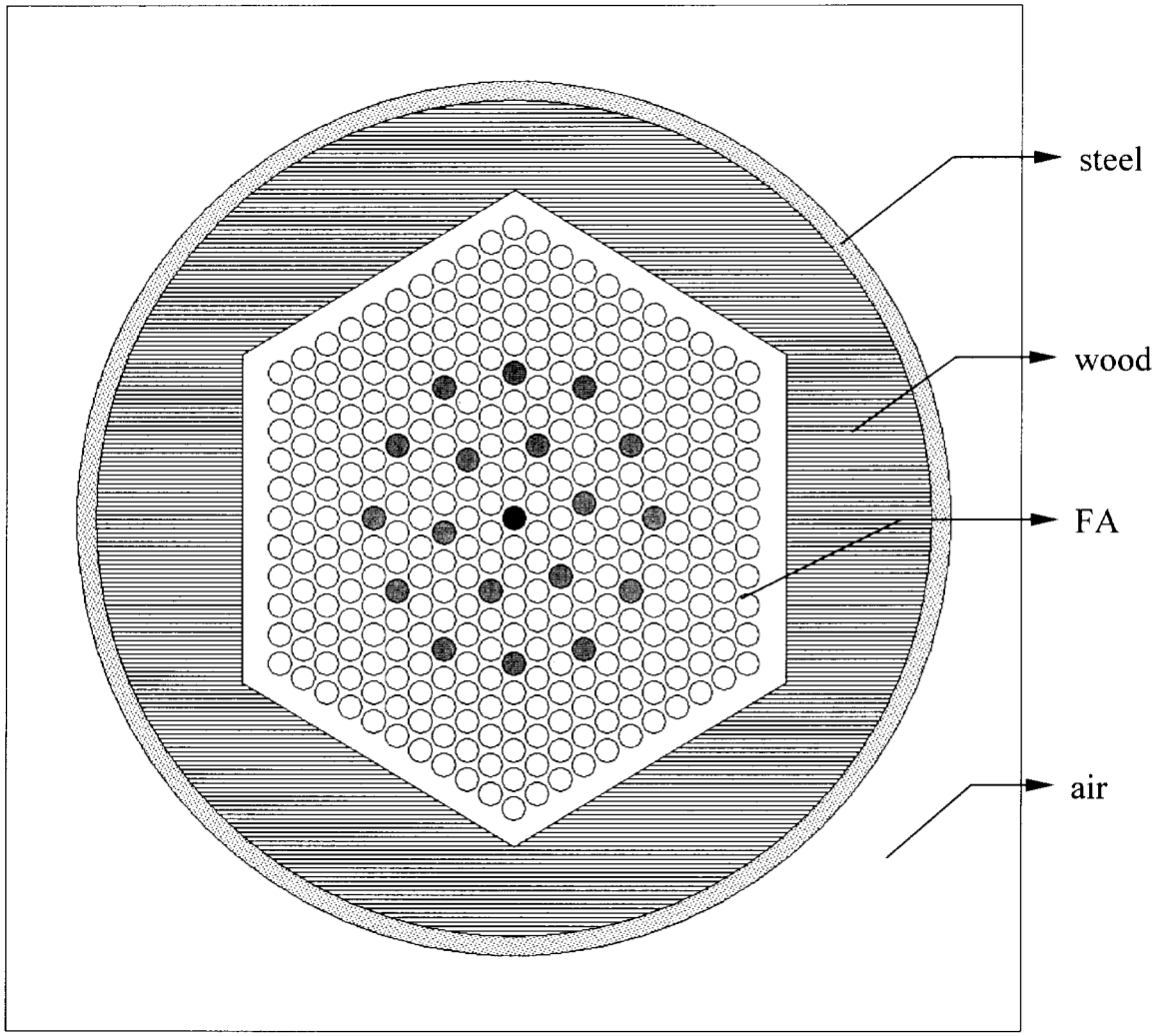



Fig. 5.1 Calculational model of package set cell

## *5.2. Cooling pool*

Calculational models for both closed (compact) and not closed (non-compact) cooling pool are given in figure 5.2. Infinite grid of cells has been considered in calculations to ensure, according to "Safety rules..." [2], conservative estimation of maximal design capacity of storage and of radial reflector influence. In axial direction a calculational model dimension has been supposed unlimited except of some cases mentioned in the text.

Dependence of the effective multiplication factor on placement pitch of UOX and MOX fuel assemblies in compact and non-compact cooling pools is given at the figure 5.3. Coolant density corresponds to a cold state, boron acid in current is absent. The curves concern UOX FAs, a set of points concerns MOX FAs. The results show that the effective multiplication factors of UOX and MOX Fas are close for all pitch values.

For design cell dimensions of compact and non-compact cooling pools (correspondingly 300 and 400 mm) and under normal operation conditions the effective multiplication factor is less then 0.95 both for UOX FA and MOX FA.

$Pu^{240}$ content significantly influences the MOX FA multiplication factor in a cooling pool. If it decreases from 4% till 0, the effective multiplication factor increases of about 4% but the limiting subcriticality value 5000 pcm is still ensured.

Figure 5.4 presents Keff values for accidental regimes with the decrease of water density. The curves concern Keff evolution for UOX FAs, corresponding dependencies for MOX FAs are described by a set of points. Dotted lines correspond to a non-compact cooling pool with and without axial neutron leakage. The presented results show that Keff increasing takes place in non-compact cooling pools for coolant densities lower than 0.7 $g/cm^3$ and Keff is over 0.95 for the densities lower than 0.5 $g/cm^3$. If no neutron leakage, Keff increasing continues till the density of about 0.1 $g/cm^3$ and after that it is seen the strong drop of Keff. For empty (without water) non-compact cooling pool the calculations give Keff = 0.689 for UOX FAs and Keff = 0.733 for MOX FAs. If axial neutron leakage is taken into account, Keff significantly decreases for low coolant densities.

So the subcriticality values meet requirements in a non-compact cooling pool without boron in coolant for the coolant densities in the interval of 0.5 – 1.0 $g/cm^3$ and also for the densities close to 0.

The curves and corresponding points in Fig.5.4 describe effective multiplication factor dependencies on water density for a compact cooling pool. The upper curve corresponds to the

case of water density decreasing in a whole pool, the lower one – to water density decreasing only inside of hexagonal tubes. The presented results show that Keff decreases in a compact cooling pool both for traditional uranium fuel and MOX fuel. If no water in a compact cooling pool, Keff values are 0.589 and 0.604 correspondingly for UOX and MOX FAs. So a cooling pool with compact decks of fuel storage ensures a significant capacity increase, meet subcriticality requirements for MOX and UOX FAs both under normal operation conditions and accidental situations with water density decreasing.

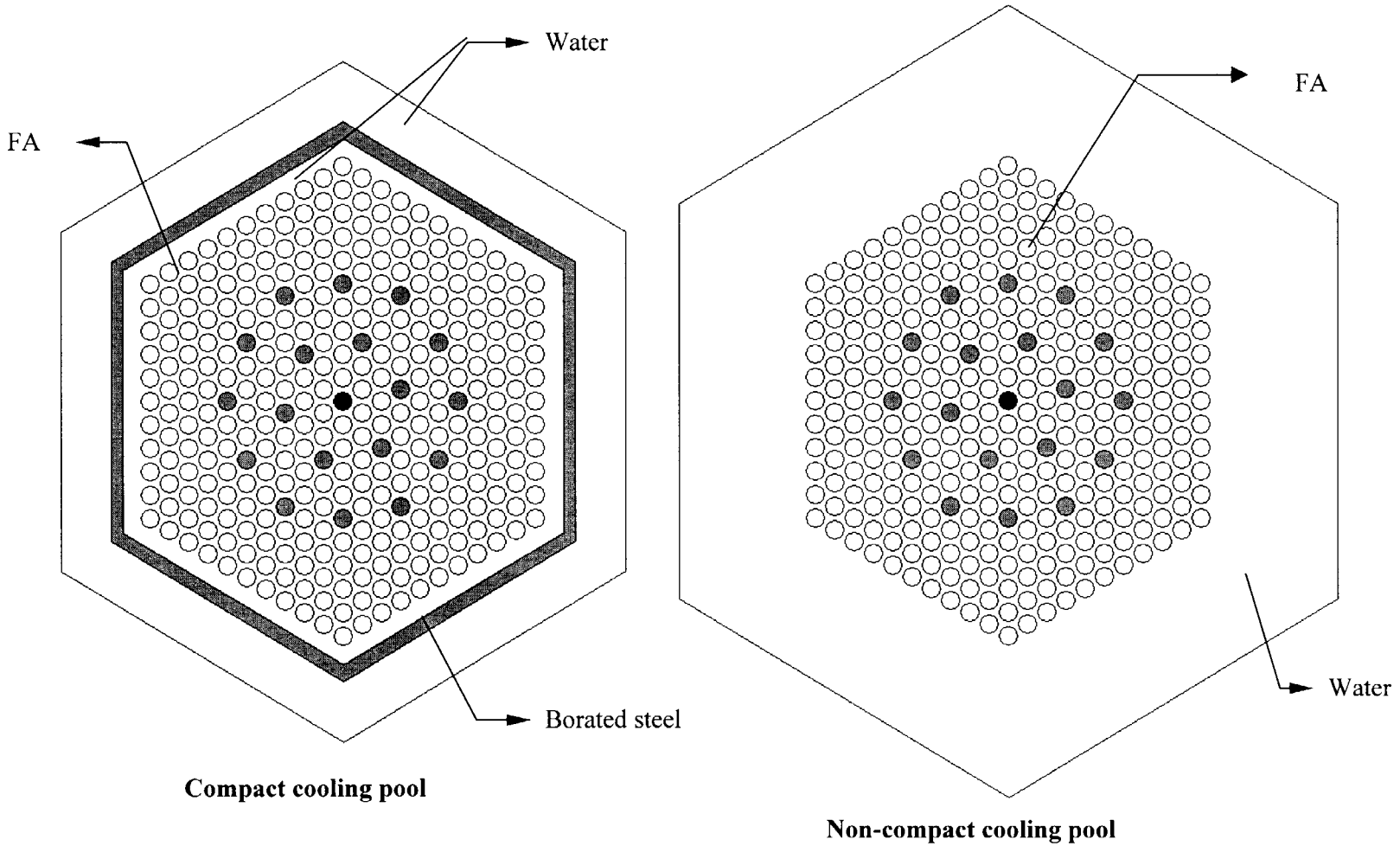



Fig. 5.2 Calculational cell models for compact and non-compact cooling pools

Fig.5.3. Dependency of effective multiplication factor on FA placement pitch in compact and non-compact cooling pools

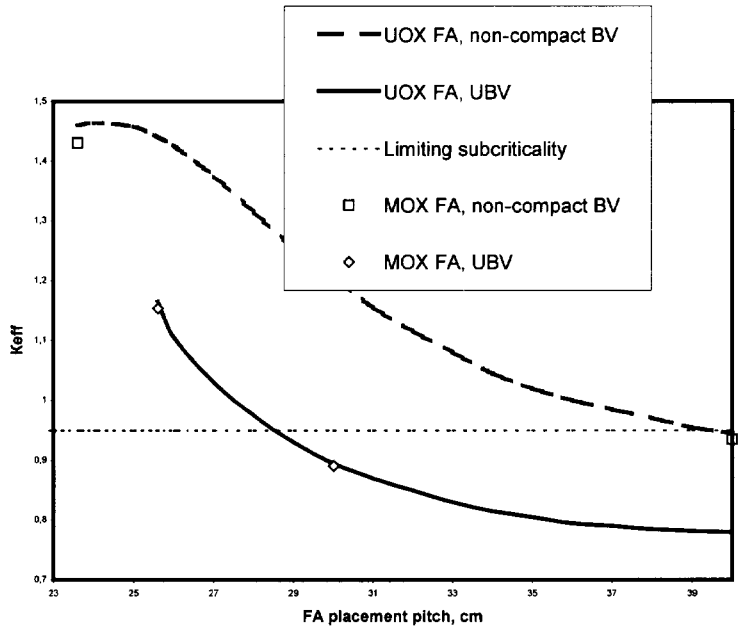
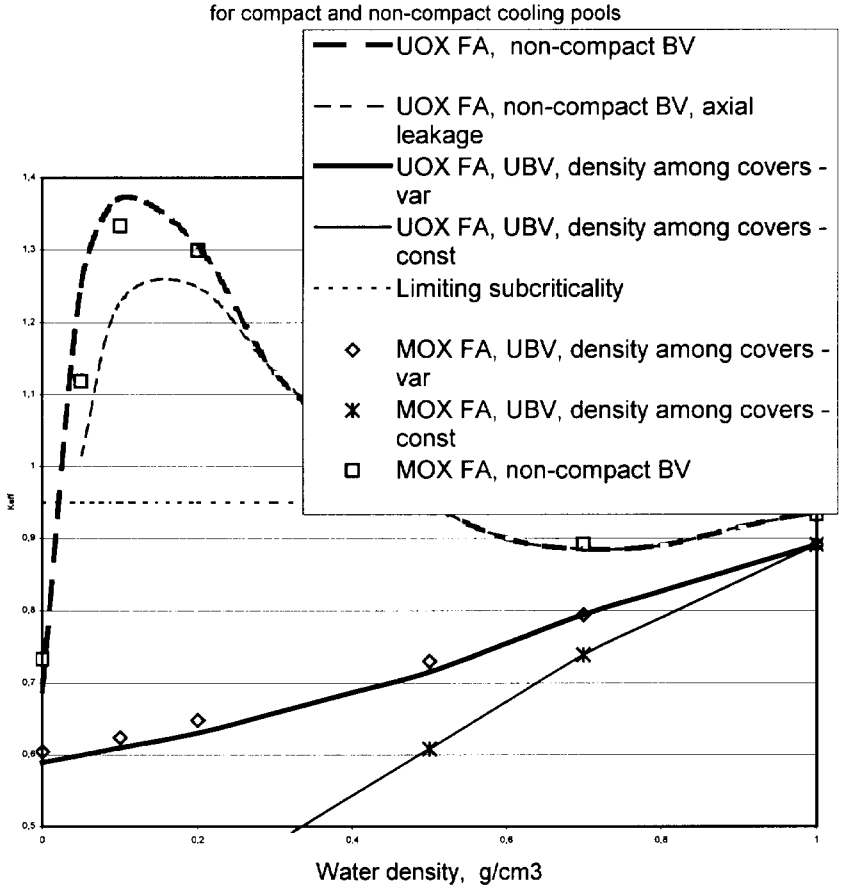


Fig. 5.4. Dependency of effective multiplication factor on water density in FA for compact and non-compact cooling pools



### *5.3. Cover for fresh fuel assemblies*

Cover for fresh FAs is intended for FAs storage and transport at NPPs. 18 fresh FAs are positioned in a cover with a pitch of 400 mm corresponding to the pitch in a non-compact cooling pool. So the results presented in p.5.2 for a non-compact cooling pool can be considered as conservative estimations for cover subcriticality. According to the presented in p.5.2 results, subcriticlity of a cover with MOX FAs is ensured for water densities about 0 and for water densities in the interval 0.5-1.0 $g/cm^3$. It is evident that if the finite radial dimension of cover is taken into account it will lead to widening of water density interval for which the limiting subcriticality is ensured. Corresponding calculations (if necessary) should be performed during future studies after definition of MOX fuel treatment technology.

## 6. 3 LTAs introduction in VVER-1000 and transportation problems

The presented calculations can be considered as conservative for the case of 3 MOX LTAs transportation. So it seems that no nuclear safety problems will appear for transportation of 3 fresh MOX LTAs at VVER-1000 NPP if the conservative case meets safety limitations.

# 7. Conclusion

The performed preliminary analysis of nuclear safety during transport and storage of MOX fuel shows that:

- Limiting subcriticality is ensured in an infinite grid of package sets for UOX and MOX FAs under normal operation and design accidents. For transport package sets placed in not more than 4 decks with no limitations for width the authorized value of subcriticality is ensured also in few probable accidental situation with adverse water density distribution.
- Cooling pools equipped with decks of non-compact storage of the fuel and covers for fresh FAs, according to conservative calculations, ensure the limiting subcriticality in the water density interval of 0.5-1.0 g/cm$^3$ and for the densities about 0.

- Cooling pools equipped with decks of close (compact) storage of the fuel, based on the hexagonal borated steel, ensure the limiting subcriticality in all considered normal operation and accidental conditions.
- 3 MOX LTAs will not cause any nuclear safety problems in the means of transportation and storage.

## 8. References

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# Part 2

# Radiation Safety

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# SUMMARY

This report considers radiation parameters of uranium and MOX fuel of VVER-1000 reactor with the aim to define necessity for modernization of transport-technological path equipment during supposed transition to MOX fuel based on weapons-grade plutonium.

Calculation models for equipment components and results of calculation estimations of radiation situation during transport-technological operations with fresh fuel are presented.

A comparative analysis of radiation parameters for two fuel types: uranium and mixed on the basis of weapons-grade plutonium.

It should be noted that all dose rate calculations were performed for case of full employment of any equipment by MOX FAs. In case of charging MOX FAs along with uranium FAs, such as three LTAs, the results obtained can be used as upper-range estimate of dose rates.

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# INTRODUCTION

At present transport-technological operations with fresh and spent nuclear fuel of serial VVER-1000 reactors are oriented to uranium fuel. Assumed utilization in VVER-1000 reactors of MOX fuel based on weapons-grade plutonium requires additional calculation studies on estimation of radiation situation in handling SA with both fresh and spent fuel. Major purpose of this kind studies - to estimate on the basis of two types fuel (uranium and mixed) comparison the necessity for changes in design of transport packing sets, shielding of transport technological path etc. in the case of MOX fuel utilization.

For this purpose calculation estimations were performed for dose rate levels when operating with fresh in different stages of SA passing along the transport-technological path - from input control to loading them to a reactor.

Dose rate levels were considered caused by neutron and gamma-radiation in the following cases:

- unshielded SA;
- SA in a sealed packing set;
- stacks of packing sets in fresh fuel storage;
- damaged SA in boxes;
- SA in jackets for transportation and storage of fresh SA;
- fresh SA in a boom of reloading machine.

It should be noted that all dose rate calculations were performed for case of full employment of any equipment by MOX FAs. In case of charging MOX FAs along with uranium FAs, such as three LTAs, the results obtained can be used as upper-range estimate of dose rates.

# 1. ANALYSIS OF SOLUTION RESULTS FOR TEST TASKS

Russian and American specialists performed calculations of test tasks, considering major peculiarities of fuel and geometry of SA and containers. A basic task of this work was a comparison of possibilities and, if necessary, subsequent correction of different methods and nuclear constant libraries used in assessing neutron and gamma-radiation source values from fresh fuel and corresponding equivalent dose rates.

Table 1 and 2 present calculation results for fresh $UO_2$ and ($U\hat{I}_2 + PuO_2$) fuel, obtained by Russian and American authors. Essential difference is observed in gamma-source intensity values. Probably, the American specialists didn't take into account decay product gamma-radiation for $^{238}U$: $^{238}U \rightarrow {}^{234m}Pa \rightarrow {}^{234}Pa$. A difference in neutron source power is small.

**Table 1.1**

**Fresh $UO_2$ fuel**

| Country | Russia | | USA |
|---|---|---|---|
| Method | CARE+ANISN, ABBN-90 26N +15G | CARE+KASKAD 26N+25G+Dγ(β) | SCALE/SAS2 27 N + 18 G (Rep. ORNL/TM-1999/207) |
| **Sources** | | | |
| Neutron | 5.727+3 | 5.870+3 | 5.730+3 |
| Gamma-radiation | 8.30E+9 *) | 8.78E+9 *) | 3.59E+9 |
| **Dose rates (SA without shielding) , [μkSv/h]** | | | |
| Neutrons | 0.43 | | 0.97 |
| Gamma-radiation | 20.8 *) | 59.5 | 3.69 |
| On surface | 21.2 *) | 59.9 | 4.66 |
| At 0.5 m | 3.17 *) | 8.27 | |
| At 1.0 m | 1.75 *) | 4.18 | |
| At 2.0 m | 0.91 *) | 1.58 | |
| **Dose rates (SA in container) , [μkSv/h]** | | | |
| Neutrons | 0.09 | | 0.080 |
| Gamma-radiation | 3.28 *) | | 0.082 |
| On surface | 3.37 *) | 7.39 | 0.162 |
| At 0.5 m | 0.84 *) | 1.85 | 0.039 |
| At 1.0 m | 0.46 *) | 1.05 | 0.022 |
| At 2.0 m | 0.23 *) | 0.46 | 0.010 |

The gamma-dose value on uranium fuel SA surface obtained by the American specialists has engaged our attention. It is much lower the value presented by the Russian specialists. It appears to be results of neglecting daughter products of $^{238}U$ and braking radiation. At the same time, twice as much value (as compared with similar data of other calculations) of neutron dose rate on SA surface for practically equal neutron sources is difficult to explain, especially as neutron dose rates on container surface (for the same source) are close.

In the case of MOX fuel for approximately equal sources, a neutron dose rate on SA surface was three times higher in American calculations. Rates between neutron and gamma-doses were different as well.

*) Without considering Dγ(β).

**Table 1.2**

**Fresh MOX fuel**

| Country | Russia | | USA |
|---|---|---|---|
| **Method** | **CARE+ANISN, ABBN-90 26N +15G** | **CARE+KASKAD 26N+25G+Dγ(β)** | **SCALE/SAS2 27 N + 18 G (Rep. ORNL/TM-1999/207)** |
| | **Sources** | | |
| Neutron | 2.04E+06 | 2.04E+06 | 2.10E+06 |
| Gamma-radiation | 5.64E+12 *) | 4.07E+12 *) | 1.53E+12 |
| | **Dose rates (SA without shielding) , [μkSv/h]** | | |
| Neutrons | 121 | 126 | 426.5 |
| Gamma-radiation | 133 *) | 259 | 430.0 |
| On surface | 254 *) | 385 | 856.5 |
| At 0.5 m | 42 *) | 53.2 | |
| At 1.0 m | 23 *) | 27.2 | |
| At 2.0 m | 12 *) | 10.3 | |
| | **Dose rates (SA in container) , [μkSv/h]** | | |
| Neutrons | 25.6 | | 29.0 |
| Gamma-radiation | 18.8 *) | | 10.6 |
| On surface | 44.4 *) | 64.6 | 39.6 |
| At 0.5 m | 12.9 *) | 15.5 | 9.39 |
| At 1.0 m | 8.7 *) | 8.6 | 5.20 |
| At 2.0 m | 5.9 *) | 3.8 | 2.35 |

Calculation results for heat release in fresh SA, obtained by different authors, are in rather good agreement to one another (Tables 1.3 and 1.4).

**Table 1.3**

**Heat release in fresh SA with $UO_2$ fuel, W**

| Country / Isotope | Russia | USA |
|---|---|---|
| U-234 | 2.930E-02 | 2.87E-02 |
| U-235 | 1.050E-03 | 1.05E-03 |
| U-238 | 3.253E-03 | 3.25E-03 |
| Sum | 3.360E-02 | 3.30E-02 |

*) Without considering Dγ(β).

**Table 1.4**

**Heat release in fresh SA with MOX fuel (4.2 %), W**

| Country / Isotope | Russia | USA |
|---|---|---|
| U-234 | 7.03E-04 | |
| U-235 | 5.02E-05 | |
| U-238 | 3.57E-03 | 3.57E-03 |
| Pu-236 | 3.64E-04 | |
| Pu-238 | 2.09E+00 | 2.09E+00 |
| Pu-239 | 3.33E+01 | 3.34E+01 |
| Pu-240 | 7.58E+00 | 7.57E+00 |
| Pu-241+U-237 | 1.08E-01 | 1.09E-01 |
| Pu-242 | 6.40E-04 | |
| Am-241 | 4.23E-01 | 4.22E-01 |
| Sum | 4.35E+01 | 4.36E+01 |

Maximum difference in calculation results was in the case of unshielded SA. In radiation transport through container shielding layers the differences are lower, and the difference in dose rate values is defined in this case mainly by difference in radiation source intensities, calculated by each researcher individually.

To fully clarify the reasons for results discrepancy, a detail comparison of all initial date, used by authors in calculations, is necessary:

- parameters of neutron and gamma-radiation sources (intensity, spectra etc.);
- sets of dose transformation factors of neutron and gamma quanta in corresponding equivalent dose rates etc.

# 2. CALCULATION ESTIMATION OF EQUIVALENT DOSE RATES FOR VARIOUS SITES OF THE TRANSPORT-TECHNOLOGICAL PATH

## 2.1. Initial Data

Two types of fuel are considered: $UO_2$ with 4.4 % enrichment by $^{235}U$ and ($UO_2 + PuO_2$) on the basis of weapons-grade plutonium with 4.2 % enrichment by plutonium.

The following plutonium isotope composition was adopted:

| $^{238}Pu$ | $^{239}Pu$ | $^{240}Pu$ | $^{241}Pu$ | $^{242}Pu$ | $^{241}Am$ |
|---|---|---|---|---|---|
| 0.02 | 93.94 | 5.81 | 0.18 | 0.03 | 0.02 |

$^{241}$Am content in plutonium corresponds to ~2 year stand after plutonium separation.

The same standing time was adopted for uranium and MOX fuel as well.

SA active part parameters used in further calculations are presented in Table 2.1, and fresh fuel integral radiation parameters - in Table 2.2.

**Table 2.1**

**SA active part parameters**

| | |
|---|---|
| Length, cm | 353 |
| "Flat-to-flat" dimension, cm | 23.4 |
| Fuel pin number | 312 |
| Cladding material | circonium |
| Outer cladding diameter, cm | 0.91 |
| Clad thickness, cm | 0.069 |
| Pellet diameter, cm | 0.753 |
| Central hole diameter, cm | 0.15 |
| $UO_2$ density, g/cm$^3$ | 10.5 |
| $UO_2$ mass, g | 4.95E+05 |

**Table 2.2**

**Actinide composition, radioactivity, heat release and power of neutron source and gamma- radiation in fresh fuel SA**

| Fuel type | $UO_2$ | MOX |
|---|---|---|
| $^{234}$U | 0.04% | 0.001% |
| $^{235}$U | 4.0% | 0.2% |
| $^{236}$U | | |
| $^{238}$U | 95.6% | 95.6% |
| $^{237}$Np + $^{233}$Pa | | |
| $^{238}$Pu | | 0.0008% |
| $^{239}$Pu | | 3.9% |
| $^{240}$Pu | | 0.24% |
| $^{241}$Pu + $^{237}$U | | 0.0076% |
| $^{242}$Pu | | 0.0013% |
| $^{241}$Am | | 0.00084% |
| Activity, Bq/SA | 4.38E+10 | 1.62E+14 |
| Heat release, W/SA | 3.35E-02 | 3.98E+01 |
| Gamma-source, quantum/s·SA | 1.01E+10 | 4.10E+12 |
| Neutron source, n/s·SA | 5.87E+03 | 2.04E+06 |

### 2.1.1. Neutron Sources

Calculation of neutron and γ-sources was carried out using CARE code [4], allowing the calculation of isotope kinetics for fuel element nuclei at both radioactive decay and neutron interactions.

Neutron source spectra were calculated in 26-group presentation of ABBN constant system [5], energy division of which is presented in Table 2.3. This table presents also corresponding division of the energy scale for γ-quanta, since in further calculations of (n,γ)-tasks capture γ-radiation is described just in this presentation.

**Table 2.3**

**Energy division of ABBN constant system**

| Group | Neutrons | | γ-quanta | |
|---|---|---|---|---|
| № | $E_{min}$ | $E_{max}$ | $E_{min}$, MeV | $E_{max}$, MeV |
| 1 | 6.5 MeV | 10.5 MeV | 9.0 | 11.0 |
| 2 | 4.0 | 6.5 | 7.0 | 9.0 |
| 3 | 2.5 | 4.0 | 5.5 | 7.0 |
| 4 | 1.4 | 2.5 | 4.5 | 5.5 |
| 5 | 0.8 | 1.4 | 3.5 | 4.5 |
| 6 | 0.4 | 0.8 | 2.5 | 3.5 |
| 7 | 0.2 | 0.4 | 1.75 | 2.5 |
| 8 | 100.0 keV | 0.2 | 1.25 | 1.75 |
| 9 | 46.5 | 100.0 keV | 0.75 | 1.25 |
| 10 | 21.5 | 46.5 | 0.35 | 0.75 |
| 11 | 10 | 21.5 | 0.15 | 0.35 |
| 12 | 4.65 | 10 | 0.08 | 0.15 |
| 13 | 2.15 | 4.65 | 0.04 | 0.08 |
| 14 | 1.0 | 2.15 | 0.02 | 0.04 |
| 15 | 465.0 eV | 1.0 | 0.015 | 0.02 |
| 16 | 215.0 | 465.0 eV | | |
| 17 | 100.0 | 215.0 | | |
| 18 | 46.5 | 100.0 | | |
| 19 | 21.5 | 46.5 | | |
| 20 | 10.0 | 21.5 | | |
| 21 | 4.65 | 10.0 | | |
| 22 | 2.15 | 4.65 | | |
| 23 | 1.0 | 2.15 | | |
| 24 | 0.465 | 1.0 | | |
| 25 | 0.215 | 0.465 | | |
| 26 | 0.0253 | | | |

When calculating neutron source, neutrons were taken into account forming at spontaneous actinide fission and in (α,n)-reaction with oxygen. Table 2.4 presents calculation results for spectrum and intensity of neutron sources in fresh uranium and MOX fuel. Radiation spectral parameters are given in relative units (normalized to 1).

Major contribution to a full intensity of neutron source in uranium fuel is introduced by $^{238}U$ (~91%) and $^{234}U$ (~8.6%), in MOX fuel - $^{240}Pu$ (~62%), $^{239}Pu$ (~34%) and $^{238}Pu$ (~3%).

**Table 2.4**

**Neutron radiation spectral composition, rel. units**

| Group № | $UO_2$ | | | MOX | | |
|---|---|---|---|---|---|---|
| | SF | (α,n) | SF+(α,n) | SF | (α,n) | SF+(α,n) |
| 1 | 1.38E-02 | 0.00E+00 | 1.25E-02 | 1.49E-02 | 0.00E+00 | 8.22E-03 |
| 2 | 7.88E-02 | 0.00E+00 | 7.14E-02 | 8.23E-02 | 3.40E-04 | 4.56E-02 |
| 3 | 1.71E-01 | 8.22E-02 | 1.63E-01 | 1.75E-01 | 8.89E-02 | 1.36E-01 |
| 4 | 2.66E-01 | 2.28E-01 | 2.63E-01 | 2.67E-01 | 2.17E-01 | 2.45E-01 |
| 5 | 2.09E-01 | 9.74E-02 | 1.99E-01 | 2.07E-01 | 9.90E-02 | 1.59E-01 |
| 6 | 1.51E-01 | 1.20E-01 | 1.48E-01 | 1.48E-01 | 1.24E-01 | 1.37E-01 |
| 7 | 6.74E-02 | 1.23E-01 | 7.26E-02 | 6.53E-02 | 1.24E-01 | 9.18E-02 |
| 8 | 2.67E-02 | 1.07E-01 | 3.43E-02 | 2.59E-02 | 1.08E-01 | 6.25E-02 |
| 9 | 1.06E-02 | 1.30E-01 | 2.18E-02 | 1.03E-02 | 1.28E-01 | 6.30E-02 |
| 10 | 3.46E-03 | 6.02E-02 | 8.78E-03 | 3.34E-03 | 5.94E-02 | 2.85E-02 |
| 11 | 1.11E-03 | 2.78E-02 | 3.61E-03 | 1.07E-03 | 2.74E-02 | 1.29E-02 |
| 12 | 3.53E-04 | 1.30E-02 | 1.53E-03 | 3.41E-04 | 1.28E-02 | 5.92E-03 |
| 13 | 1.12E-04 | 6.02E-03 | 6.65E-04 | 1.08E-04 | 5.94E-03 | 2.72E-03 |
| 14 | 3.55E-05 | 2.78E-03 | 2.93E-04 | 3.42E-05 | 2.74E-03 | 1.25E-03 |
| 15 | 1.12E-05 | 1.30E-03 | 1.32E-04 | 1.08E-05 | 1.28E-03 | 5.79E-04 |
| 16 | 3.55E-06 | 6.02E-04 | 5.96E-05 | 3.43E-06 | 5.94E-04 | 2.68E-04 |
| 17 | 1.12E-06 | 2.78E-04 | 2.71E-05 | 1.08E-06 | 2.74E-04 | 1.24E-04 |
| 18 | 3.55E-07 | 1.30E-04 | 1.25E-05 | 3.43E-07 | 1.28E-04 | 5.75E-05 |
| 19 | 1.12E-07 | 6.02E-05 | 5.74E-06 | 1.08E-07 | 5.94E-05 | 2.67E-05 |
| 20 | 3.55E-08 | 2.78E-05 | 2.64E-06 | 3.43E-08 | 2.74E-05 | 1.23E-05 |
| 21 | 1.12E-08 | 1.30E-05 | 1.22E-06 | 1.08E-08 | 1.28E-05 | 5.73E-06 |
| 22 | 3.55E-09 | 6.02E-06 | 5.67E-07 | 3.43E-09 | 5.94E-06 | 2.66E-06 |
| 23 | 1.12E-09 | 2.78E-06 | 2.61E-07 | 1.08E-09 | 2.74E-06 | 1.23E-06 |
| 24 | 3.55E-10 | 1.30E-06 | 1.22E-07 | 3.43E-10 | 1.28E-06 | 5.73E-07 |
| 25 | 1.12E-10 | 6.02E-07 | 5.65E-08 | 1.08E-10 | 5.94E-07 | 2.66E-07 |
| 26 | 4.98E-11 | 5.20E-07 | 4.87E-08 | 4.81E-11 | 5.13E-07 | 2.30E-07 |
| n/s·SA | 5.32E+03 | 5.49E+02 | 5.87E+03 | 1.13E+06 | 9.15E+05 | 2.04E+06 |

### 2.1.2. γ-radiation Sources

Preliminary calculations connected with determination of γ-dose rates on SA surface and at some distance from it have shown that the use of 15-group ABBN constant system for fresh fuel is not sufficient due to rough description of γ-quantum source spectrum in the energy range up to ~600 keV. Hence in calculations a 25-group constant system obtained from VITAMIN-C 36-group constant system was used.

Table 2.5 presents an energy division and spectral characteristics for the both fuel types.

**Table 2.5**

**γ-radiation spectral composition, rel. units**

| Group № | $E_{min}$, MeV | $E_{max}$, MeV | $UO_2$ | MOX |
|---|---|---|---|---|
| 1 | 3.5 | 14.0 | 0.00E+00 | 0.00E+00 |
| 2 | 3.0 | 3.5 | 0.00E+00 | 0.00E+00 |
| 3 | 2.5 | 3.0 | 0.00E+00 | 1.32E-19 |
| 4 | 2.00 | 2.5 | 0.00E+00 | 0.00E+00 |
| 5 | 1.66 | 2.00 | 3.38E-04 | 7.99E-07 |
| 6 | 1.50 | 1.66 | 1.32E-04 | 3.12E-07 |
| 7 | 1.33 | 1.50 | 9.77E-05 | 2.31E-07 |
| 8 | 1.00 | 1.33 | 2.91E-03 | 6.87E-06 |
| 9 | 0.8 | 1.00 | 9.90E-04 | 2.34E-06 |
| 10 | 0.7 | 0.8 | 1.95E-03 | 4.60E-06 |
| 11 | 0.6 | 0.7 | 1.49E-04 | 3.52E-07 |
| 12 | 0.512 | 0.6 | 1.52E-04 | 3.59E-07 |
| 13 | 0.51 | 0.512 | 0.00E+00 | 3.00E-20 |
| 14 | 0.5 | 0.51 | 2.18E-05 | 1.77E-05 |
| 15 | 0.4 | 0.5 | 0.00E+00 | 1.44E-04 |
| 16 | 0.3 | 0.4 | 1.03E-04 | 3.76E-04 |
| 17 | 0.2 | 0.3 | 8.27E-03 | 4.63E-05 |
| 18 | 0.15 | 0.2 | 1.01E-01 | 1.22E-05 |
| 19 | 0.1 | 0.15 | 3.06E-02 | 9.89E-04 |
| 20 | 0.075 | 0.1 | 6.57E-02 | 1.23E-03 |
| 21 | 0.06 | 0.075 | 2.01E-02 | 4.64E-05 |
| 22 | 0.045 | 0.06 | 6.41E-03 | 1.06E-01 |
| 23 | 0.03 | 0.045 | 0.00E+00 | 1.12E-03 |
| 24 | 0.02 | 0.03 | 2.44E-02 | 8.86E-02 |
| 25 | 0.01 | 0.02 | 7.37E-01 | 8.01E-01 |
| Full intensity, quantum/s·SA | | | 8.78E+09 | 4.07E+12 |

Full intensity of γ-source in uranium fuel is formed due to $^{234}U$ (~47%), $^{235}U$ (~19%), $^{231}Th$ (~16%), $^{234}Th$ (~11%), $^{238}U$ (~5%) and $^{234m}Pa$ (~1%). $^{239}Pu$ (~42%), $^{240}Pu$ (~24%), $^{238}Pu$ (~6.4%) and $^{241}Am$ (~3%) contributing mainly to intensity in MOX fuel.

### 2.1.3. Breaking (X) Radiation Sources

An availability of braking radiation is a characteristic feature of fuel containing $^{238}U$ due to short-lived daughter product of its decay - $^{234m}Pa$. Maximum energy of $^{234m}Pa$ β-particles is ~2.3 MeV, therefore, X-radiation accompanying β-particle braking in the medium is much more hard, as compared with other β-emitters. Table 2.6 presents in 25-group presentation the calculated braking radiation spectra for uranium and MOX fuel.

**Table 2.6**

**Braking radiation spectral composition, rel. units**

| Group № | $UO_2$ | MOX |
|---|---|---|
| 1...4 | 0.00E+00 | 0.00E+00 |
| 5 | 1.11E-04 | 4.85E-06 |
| 6 | 2.72E-04 | 1.18E-05 |
| 7 | 7.20E-04 | 3.14E-05 |
| 8 | 4.58E-03 | 2.00E-04 |
| 9 | 8.11E-03 | 3.54E-04 |
| 10 | 7.29E-03 | 3.18E-04 |
| 11 | 1.11E-02 | 4.82E-04 |
| 12 | 1.41E-02 | 6.15E-04 |
| 13 | 3.96E-04 | 1.73E-05 |
| 14 | 2.03E-03 | 8.86E-05 |
| 15 | 2.60E-02 | 1.13E-03 |
| 16 | 4.10E-02 | 1.79E-03 |
| 17 | 7.43E-02 | 3.24E-03 |
| 18 | 6.26E-02 | 2.73E-03 |
| 19 | 9.68E-02 | 4.20E-03 |
| 20 | 7.43E-02 | 3.22E-03 |
| 21 | 6.02E-02 | 2.60E-03 |
| 22 | 7.97E-02 | 3.44E-03 |
| 23 | 1.15E-01 | 4.97E-03 |
| 24 | 1.18E-01 | 5.10E-03 |
| 25 | 2.03E-01 | 9.65E-01 |
| quantum/s·SAC | 1.32E+09 | 3.32E+10 |

A source intensity in uranium fuel is defined by $^{234m}Pa$ (~98%), and in MOX fuel - by $^{241}Pu$ (~96%), whereas $^{234m}Pa$ fraction is ~4.6% only, however, as is seen from the table, the whole plutonium X-radiation is concentrated in soft spectrum part, and therefore is practically completely absorbed in SA structure material. The X-rays radiation intensity defined by $^{234m}Pa$, is determined only by $^{238}U$ isotope content in fuel and thus creates the same dose rate on SA surface as in the case of uranium fuel.

### 2.1.4. Calculation Methods and Software

A task list on radiation situation estimation when handling fuel encompasses practically all stages of SA passing through the transport-technological path, including SA storage in racks. Thus, in the case of fresh fuel it is necessary to solve tasks connected with equivalent dose rates in the following cases:

1. Unshielded SA (input control).
2. Packing set with 2 SA.
3. Stack from 3×5 packing sets.
4. Jacket with 18 fresh SA.
5. The same jacket with 7 fresh SA.
6. One SA with fresh MOX fuel in reloading machine boom.

A large scope of calculations led to the use of both "exact" calculation methods and engineering approximate calculations.

Calculation performance with the use of well known kinetic codes, for example, **DOT, DORT, TWODANT** or **RADUGA** and **KASKAD** Russian codes is convenient in the case of axis-symmetrical or reduced to this geometry tasks 1, 4, 5 and 6. Complex in geometrical relation tasks 2 and 3 can be solved using Monte Carlo methods, however, this is more labour consuming, and dose rates for these cases are estimated by the source superposition method.

### *2.1.4.1. Kinetic Calculation Methods*

As "exact" were used kinetic calculation methods , realized in the codes **ANISN** [6], **KASKAD** [7] and **DOT-3.5** [8]. All two-dimensional calculations were performed in $S_8P_1$-approximation.

In solving (n,γ)-tasks, **ABBN-93** constant system [5] was used, and in tasks with γ-sources - **VITAMIN-C**. To prepare macroscopic cross-sections for material mixtures, **CONSYST** complex [5] was used in the case of ABBN constants, and **GIP** code - in the case of **VITAMIN-C**.

"Dose coefficients" [9] used in equivalent dose rate calculations are presented in Table 2.7.

**Table 2.7**

**Dose coefficients, (μk·Sv/h)/(1/s· cm$^2$)**

| Group № | Neutrons | γ-quanta | |
|---|---|---|---|
| | | ABBN | VITAMIN-C |
| 1 | 2.01E+00 | 9.47E-02 | 8.60E-02 |
| 2 | 1.90E+00 | 7.97E-02 | 4.41E-02 |
| 3 | 1.72E+00 | 6.66E-02 | 3.96E-02 |
| 4 | 1.57E+00 | 5.72E-02 | 3.47E-02 |
| 5 | 1.55E+00 | 4.96E-02 | 3.02E-02 |
| 6 | 1.47E+00 | 4.13E-02 | 2.73E-02 |
| 7 | 1.32E+00 | 3.31E-02 | 2.53E-02 |
| 8 | 1.01E+00 | 2.61E-02 | 2.20E-02 |
| 9 | 6.16E-01 | 1.95E-02 | 1.83E-02 |
| 10 | 3.75E-01 | 1.24E-02 | 1.60E-02 |
| 11 | 2.17E-01 | 6.14E-03 | 1.44E-02 |
| 12 | 1.02E-01 | 3.15E-03 | 1.28E-02 |
| 13 | 6.05E-02 | 2.94E-03 | 1.19E-02 |
| 14 | 5.82E-02 | 9.69E-03 | 1.18E-02 |
| 15 | 6.35E-02 | 3.12E-02 | 1.08E-02 |
| 16 | 7.02E-02 | | 8.75E-03 |
| 17 | 7.33E-02 | | 6.31E-03 |
| 18 | 7.47E-02 | | 4.40E-03 |
| 19 | 7.54E-02 | | 3.31E-03 |

| Group № | Neutrons | γ-quanta | |
|---|---|---|---|
| | | ABBN | VITAMIN-C |
| 20 | 7.56E-02 | | 2.73E-03 |
| 21 | 7.56E-02 | | 2.64E-03 |
| 22 | 7.56E-02 | | 2.97E-03 |
| 23 | 7.56E-02 | | 4.73E-03 |
| 24 | 7.56E-02 | | 1.43E-02 |
| 25 | 7.56E-02 | | 3.12E-02 |
| 26 | 6.38E-02 | | |
| 27 | 4.65E-02 | | |
| 28 | 3.74E-02 | | |

### *2.1.4.2. Approximate Calculation of Fuel Radiation Parameters*

For prompt and preliminary estimations of radiation situation when handling fresh and spent fuel, approximate calculations play an important role. Besides, they are useful for complex geometry cases, when estimations are performed using a superposition principle for elementary source radiation fields. An approximate method uncertainty directly depends on shield thickness: for example, since a shield thickness of containers for fresh fuel doesn't exceed several centimeters, one can expect an acceptable for practical purposes uncertainty.

To estimate radiation parameters of fresh SA with uranium and MOX fuel, the **FRESH** calculation module was developed, allowing the calculation of radioactivity, neutron and gamma-radiation source power, radiation heat release power, as well neutron and gamma-radiation dose on SA surface and at a distance from it.

Radioactivity is calculated using the formula:

$$a = \sum_i \lambda_i \cdot N_i \ , \text{Bq} \qquad (1)$$

where

$\lambda_i = \frac{\ln 2}{T_{1/2}}$ - decay constant for i-th radionuclide, 1/s;

$T_{1/2}$ - half-life, s;

$N_i = 6.02 \cdot 10^{23} \cdot \frac{M_i}{A_i}$ - radionuclide nuclei number;

$M_i$ - radionuclide mass in SA, g;

$A_i$ - atomic mass, amu.

A radiation heat release is calculated as:

$$H = 1.6 \cdot 10^{-13} \cdot \sum_i a_i \cdot h_i \ , \text{W} \qquad (2)$$

where $h_i$ - total disintegration energy, MeV.

We underline that the formula (2) estimates an emitted, but not absorbed in SA energy; energy carry-over with gamma-radiation leaving SA is not taken into account. Nevertheless, formula (2) uncertainty is small (less 1%) due to a low gamma-radiation

fraction in total uranium decay energy and high absorption of a rather soft radiation of uranium and transuranium elements by the fuel composition.

Neutron source power calculation is performed, taking into account spontaneous fission and creation of neutrons according to $O^{18}(\alpha,n)Ne^{21}$ reaction:

$$Q_n = \sum_i a_i \cdot \varepsilon_i \text{ , n/s} \quad (3)$$

where $\varepsilon_i$ - neutron yield per i-th nuclide decay.

Average neutron energy was adopted equal to 2 MeV .

The gamma-radiation source power is calculated in a similar way. Summing is performed by energy groups (index j) and radionuclides (index i):

$$Q_\gamma = \sum_i a_i \cdot \sum_j \varepsilon_{i,j} \text{ , quantum/s ,} \quad (4)$$

were $\varepsilon_{i,j}$ - gamma-quantum yield of j-th group per i-th radionuclide decay.

Neutron and gamma-radiation dose rate is calculated using analytical expressions, in which the radiation passage is described by functions with exponential nucleus. When calculating gamma-radiation dose rate, the effect of dissipation is taken into account using accumulation factors. In estimation of neutron dose rate, the presentation on relaxation length is used.

Neutron dose rate is calculated by the following formula:

$$D_n = H_n \cdot \frac{q_n}{\Sigma_s} \cdot G \quad (5)$$

where

$H_n$ - dose factor, (μkSv/h) / (n·s$^{-1}$·cm$^{-2}$);

$q$ - neutron source density in SA, n·s$^{-1}$·cm$^{-2}$;

$\Sigma_s$ - attenuation factor in the source (value inversely related to relaxation length), cm$^{-1}$;

$G$ - dimensionless attenuation function.

Gamma-radiation dose rate is calculated in a similar way:

$$D_\gamma = \sum_{j=1}^{16} H_j^\gamma \cdot \frac{q_j^\gamma}{\mu_j^{\gamma s}} G_j^\gamma \cdot B_j^\gamma \text{ ,} \quad (6)$$

where

$H_j^\gamma$ - dose factor, (μkSv/h) / (quantum·s$^{-1}$·cm$^{-2}$);

$q_j^\gamma$ - gamma-quantum source density in SA, quantum·s$^{-1}$·cm$^{-2}$;

$\mu_j^{\gamma s}$ - efficient attenuation factor in the source, cm$^{-1}$;

$G_j^\gamma$ - dimensionless attenuation function;

$B_j^\gamma$ - dose factor of scattered radiation accumulation.

In its turn, the dimensionless attenuation function G is calculated according to the following expression:

$$G = \frac{\alpha_{ef}}{\pi} \cdot (F_{cur}(\beta, \chi) - F_{cur}(\beta, \chi + \varphi)), \quad (7)$$

where

$\alpha_{ef} = \exp(\chi) \cdot F_{fl}(\alpha, \chi)$ - effective angle, radian;

$\alpha = \arcsin(\frac{R_s}{R_s + R_d})$ - the angle, under which a source element is seen, radian;

$R_s$ - source radius, cm;

$R_d$ - the distance between source boundary and detector, cm;

$\chi = \sum_i \mu_i^p \cdot t_i^p$ - the optical shield thickness, free path lengths;

$\mu_i^p$ - attenuation factor in i-th shielding layer, $cm^{-1}$;

$t_i^p$ - i-th shielding layer thickness, $cm^{-1}$;

$\beta = \operatorname{arctg}(\frac{H_s}{R_d})$ - the angle under which a top of cylindrical source is seen, radian;

$H_s$ - a half-height of source, cm;

$F_{fl}(\alpha, \chi) = \int_0^{\alpha} \exp(-\frac{\chi}{\cos \vartheta}) \cdot d\vartheta$ - integral flux secant;

$F_{cur}(\beta, \chi) = \int_0^{\beta} \cos \vartheta \cdot \exp(-\frac{\chi}{\cos \vartheta}) \cdot d\vartheta$ - integral current secant.

**FRESH** calculation module was written in VISUAL BASIC algorithmic language and built into working book of EXCEL system. Electronic tables of the working book contain physical constants, initial data and calculation results. The code provides simultaneous calculation of several SA variants, geometrically identical, but with different composition of fuel fission components. This provides a possibility of checking calculation results with known data, for example, to compare SA with uranium-plutonium and uranium fuel.

To estimate uncertainty of results obtained using the methodology presented, they were compared with experimental estimations for dose rates from fresh SA without shielding and the results of corresponding calculations by **KASKAD** code. Besides, the calculation results for one SA in a packing set, obtained by **FRESH** code, were compared with two-dimensional (**DOT-3.5**) calculations. Maximum results difference is in the range from -35 to +60 %.

## 2.2. Calculation Estimations of Equivalent Dose Rates

### 2.2.1. Unshielded SA

A calculation model, somewhat simplified as compared with real SA structure, is presented in Fig. 1 in (R, Z)-geometry. The model is symmetrical in relation to Z-axis. On all remaining surfaces – vacuum boundary conditions.

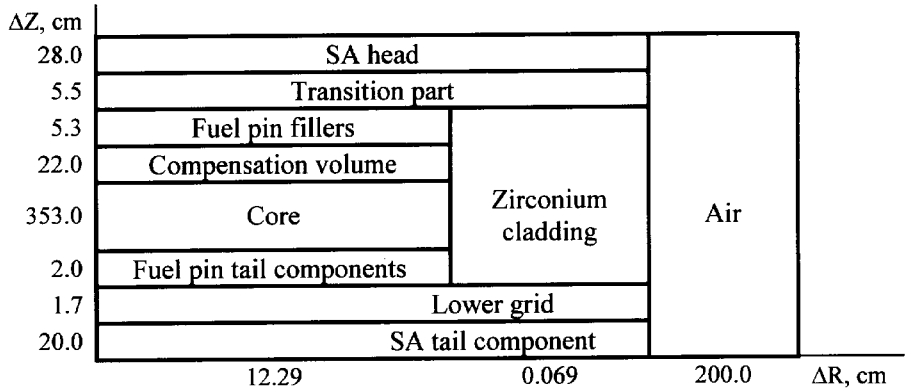



**Fig. 1. Calculation (R, Z)-model of SA without shielding**

Equivalent dose rates were estimated on SA surface and at a distance of 10, 50, 100 and 200 cm from it. In this case the following radiation components introducing to dose rate value were taken into account: source neutrons; multiplication neutrons, capture γ-radiation, γ-radiation of actinides and braking radiation.

The results of calculations performed using KASKAD code are presented in Table 2.8.

**Table 2.8**

**Equivalent dose rate from unshielded SA with fresh fuel, μkSv/h**

| Distance from SA surface, cm | $D_n$ | $D_{n,\gamma}$ | $D_\gamma$ | $D_{\gamma(\beta)}$ | $\Sigma_\gamma$ | $D_\Sigma$ |
|---|---|---|---|---|---|---|
| **$UO_2$** | | | | | | |
| 0 | 4.10-1 | 7.49-4 | 3.30+1 | 2.65+1 | 5.95+1 | 5.99+1 |
| 10 | 1.78-1 | 3.22-4 | 1.40+1 | 1.11+1 | 2.51+1 | 2.53+1 |
| 50 | 5.76-2 | 1.05-4 | 4.54+0 | 3.57+0 | 8.11+0 | 8.27+0 |
| 100 | 2.92-2 | 5.33-5 | 2.32+0 | 1.83+0 | 4.15+0 | 4.18+0 |
| 200 | 1.11-2 | 2.06-5 | 8.82-1 | 6.91-1 | 1.57+0 | 1.58+0 |
| **MOX** | | | | | | |
| 0 | 1.26+2 | 2.17-1 | 2.30+2 | 2.90+1 | 2.59+2 | 3.85+2 |
| 10 | 5.47+1 | 9.32-2 | 9.72+1 | 1.21+1 | 1.09+2 | 1.64+2 |
| 50 | 1.77+1 | 3.03-2 | 3.15+1 | 3.92+0 | 3.55+1 | 5.32+1 |
| 100 | 9.00+0 | 1.54-2 | 1.62+1 | 2.00+0 | 1.82+1 | 2.72+1 |
| 200 | 3.42+0 | 5.98-3 | 6.16+0 | 7.56-1 | 6.92+0 | 1.03+1 |

Here:

$D_n$ - the dose rate, defined by the neutron source (spontaneous actinide fission + (α,n)-reaction on oxygen);

$D_{n,\gamma}$, $D_\gamma$, $D_{\gamma(\beta)}$ - dose rates due to capture irradiation, actinide γ-radiation, and braking (X) radiation, correspondingly;

$\Sigma_\gamma$ - the dose rate from all components of X and γ-radiation;

$D_\Sigma$ - the equivalent dose full rate value.

### 2.2.2. SA Storage in Packing Sets for Fresh Fuel

A set consists of two steel tubes with wood inserts, forming a hexahedral cell for SA location. Tube ends are limited by flanges and removable caps.

The corresponding to this design calculation model for one tube with SA is presented in Fig. 2. The composition is symmetrical in relation to Z-axis.

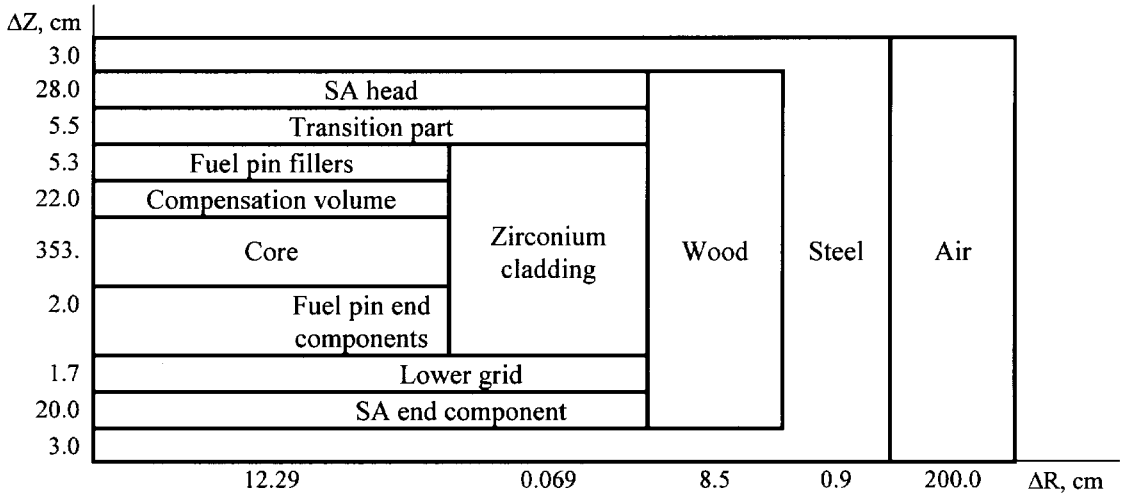



**Fig. 2. Calculation (R, Z)-model of SA without shielding**

The calculations were performed using DOT-3.5 code. The results are in Table 2.9.

**Table 2.9**

**Packing set with one SA. Dose rates, μkSv/h**

| Fuel type | Distance from SA surface, cm | | | | |
|---|---|---|---|---|---|
| | 0 | 10 | 50 | 100 | 200 |
| $UO_2$ | 7.39E+00 | 4.57E+00 | 1.85E+00 | 1.05E+00 | 4.62E-01 |
| MOX | 6.46E+01 | 3.92E+01 | 1.55E+01 | 8.62E+00 | 3.76E+00 |

Using the data from Table 2.9, dose rate distributions were obtained by solution superposition method in a direction perpendicular to the line connecting SA axes for two cases, shown in Fig. 3: I - in the plane passing through the centre between SA (R=0 corresponds to a level of tangent to container tube surfaces) and II – in the plane passing through axis of one of container tubes.

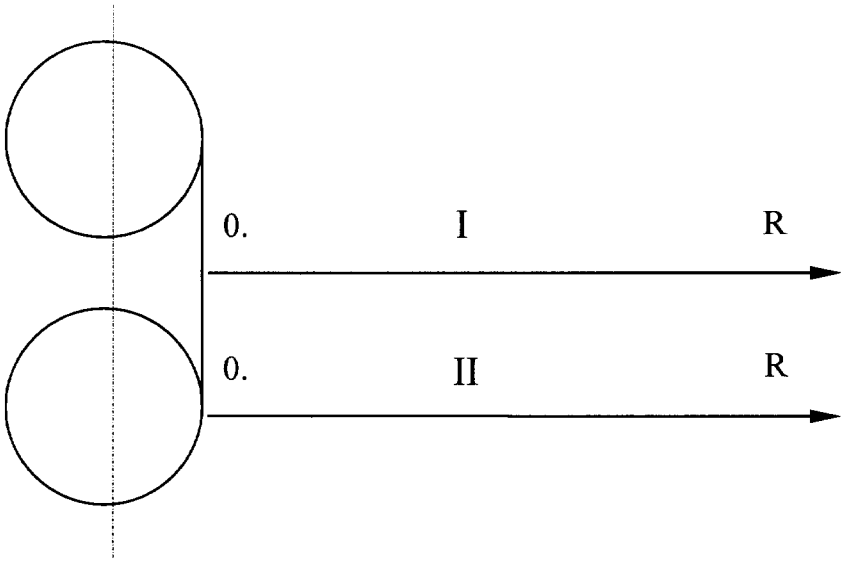



**Fig. 3. Location of axes of dose rate radial distribution from two SA in a packing set**

The calculation results are presented in Tables 2.10 and 2.11.

**Table 2.10**

**Superposition on centre between SA (I). Dose rates, μkSv/h**

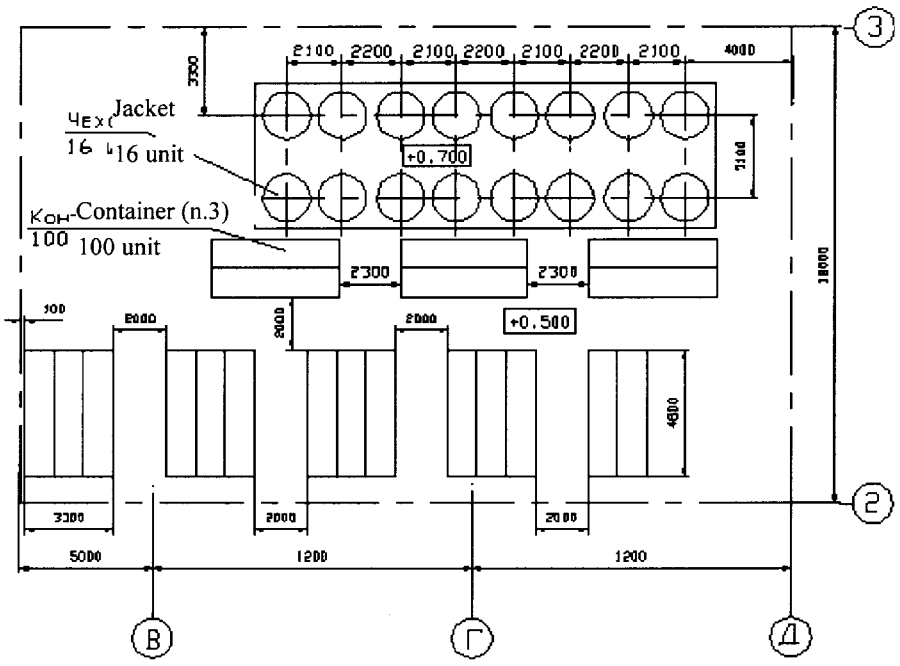
| | Distance from surface tangent to tubes , cm | | | |
|---|---|---|---|---|
| Fuel type | 0 | 50 | 100 | 200 |
| $UO_2$ | 8.27E+00 | 3.43E+00 | 2.07E+00 | 9.17E-01 |
| MOX | 7.07E+01 | 2.86E+01 | 1.71E+01 | 7.45E+00 |

**Table 2.11**

**Superposition by on axis of one SA (II). Dose rates, μkSv/h**

| | Distance from surface tangent to tubes , cm | | | |
|---|---|---|---|---|
| Fuel type | 0 | 50 | 100 | 200 |
| $UO_2$ | 9.45E+00 | 3.29E+00 | 1.97E+00 | 9.21E-01 |
| MOX | 8.40E+01 | 2.75E+01 | 1.63E+01 | 7.48E+00 |

The data obtained were used in estimating radiation situation in fresh fuel storage rooms when piling packing sets. Fig. 4 shows a scheme for location of packing sets in racks of the fresh fuel storage.




Прим Notes:
1. В2. Jacket capacity – 18 SA
2. В3. Container capacity – 2 SA
3. К4. Number of container tiers in a stack - 5

**Fig. 4. A scheme of location of jackets and containers with fresh SA in fresh fuel unit**

Neutron and γ-radiation dose rate at storage of fresh SA in packing containers was calculated by the source superposition method (of individual SA). The contributions of SA screened by other SA was not taken into account. The calculations were carried out for passages between stacks. The results of estimated in this way equivalent dose rates are presented in Table 2.12.

As follows from the data presented, when storing SA in packing sets located in stacks, a ratio of neutron and gamma dose rates for MOX- and uranium SA is ~5.

**Table 2.12**

**Equivalent dose rates when storing fresh fuel in stacks from packing sets, μkSv/h**

| Fuel type | $UO_2$ | MOX |
|---|---|---|
| **In the middle of passage; from one of nearest SA** | | |
| Gamma-radiation dose rate | 2.18 | 6.46 |
| Neutron radiation dose rate | 0.0119 | 3.57 |
| Total dose rate | 2.19 | 10.3 |
| **In the middle of passage; from all SA (from the left and right)** | | |
| Gamma-radiation dose rate | 40.4 | 120 |
| Neutron radiation dose rate | 0.221 | 66.4 |
| Total dose rate | 40.7 | 186 |
| **At the boundary of passage; from all SA (from the left and right)** | | |
| Gamma-radiation dose rate | 47.8 | 142 |
| Neutron radiation dose rate | 0.261 | 78.5 |
| Total dose rate | 48.1 | 220 |

Note: 1 stack = 5 layers × 3 sets × 2 SA = 30 SA. A passage between stacks is 2 m.

### 2.2.3. SA in a Box

Dose rate estimations were carried out using a semi-empirical method by **FRESH** code. A radial shielding thickness was chosen with account of wall and tubes to be 1 cm.

Total neutron and gamma-radiation dose rate as a function of distance from the centre is shown in Fig. 5. A ratio of total dose rate between weapons-grade MOX fuel and uranium fuel is ~5.

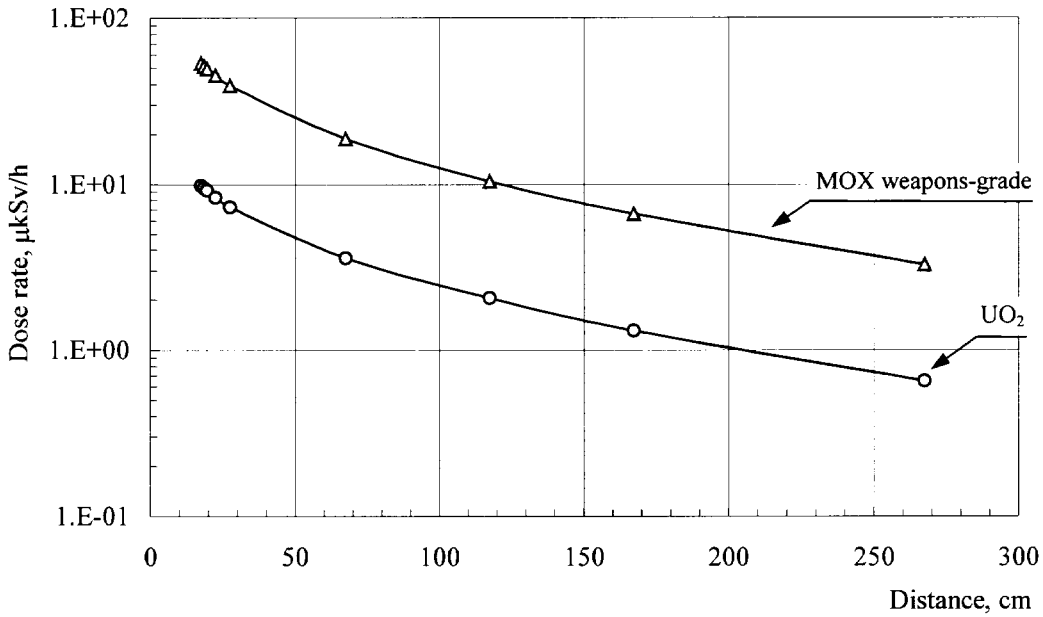
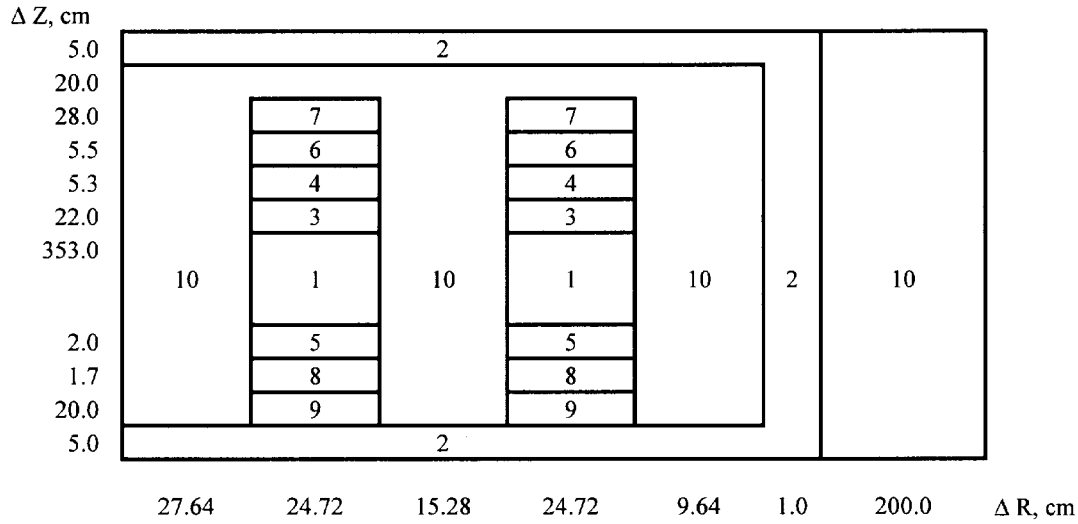



**Fig. 5. Spatial distribution of equivalent dose rate from fresh SA in a box as a function of distance from the centre**

### 2.2.4. Jacket for SA Transportation to Reactor Hall and Their Temporary Storing in the Fresh Fuel Storage

A calculation model in two-dimensional (R, Z)-geometry is presented in Fig. 6.




Notations:

1 - Core
2 - Jacket steel
3 - Compensation volume
4 - Fuel pin fillers
5 - Fuel pin end
6 - Transition part
7 - SA head
8 - Lower grid
9 - SA end component
10 - Air

**Fig. 6. Calculation (R,Z)-model of jacked for fresh SA**

Two variants of jacket utilization were considered - with full loading (18 SA) and with 7 SA in outer row.

As seen from Fig. 6, SA location in a jacket is modelled by two layers, in which 6 and 12 SA (full jacket loading) are mixed homogeneously. Table 2.13 presents radial distributions of equivalent dose rate values up to a distance of 2 m from jacket surface for the case of its full loading.

**Table 2.13**

**A jacket with 18 SA. Dose rates, μkSv/h**

| | Distance from SA surface, cm | | | | |
|---|---|---|---|---|---|
| Fuel type | 0 | 10 | 50 | 100 | 200 |
| $UO_2$ | 1.77E+01 | 1.52E+01 | 9.67E+00 | 6.64E+00 | 3.28E+00 |
| MOX | 2.85E+02 | 2.42E+02 | 1.44E+02 | 9.32E+01 | 4.47E+01 |

Similar data are presented in Table 2.14 for the case of homogeneous mixing 7 fresh SA in outer jacket row.

**Table 2.14**

**A jacket with 18 SA. Dose rates, μkSv/h**

| Fuel type | Distance from SA surface, cm | | | | |
|---|---|---|---|---|---|
| | 0 | 10 | 50 | 100 | 200 |
| $UO_2$ | 1.00E+01 | 8.62E+00 | 5.46E+00 | 3.75E+00 | 1.86E+00 |
| MOX | 1.25E+02 | 1.06E+02 | 6.29E+01 | 4.11E+01 | 2.10E+01 |

This model doesn't reflect a real SA distribution in a jacket, since 5 cells of outer row remain unoccupied. Besides, outer row SA defining a dose rate beyond the jacket are at different distance from outer surface.

The calculations have shown that the irregularity of dose field by angle, conditioned by location of 18 SA in jacket in hexahedral lattice, on a jacket circle does not exceed 10 %, and at some distance - essentially less. When locating in the same jacket 7 SA only, a practically the same dose rate will be observed in the most unfavourable case, when 3 of 7 SA are located on one outer side and a dose rate is measured from this side.

### 2.2.5. Fresh SA Reloading

Reloading is carried out by a reloading machine, which grips SA with the help of a boom. The boom includes two coaxially located tubes ∅30×0.9 and ∅41×1.5 cm. The reloading can be either "dry" or under water layer.

Fig. 7 presents the corresponding calculation (R,Z)-model of SA in reloading machine boom. Notations 1-10 in Fig. 7 are analogous to notations in Fig. 6. Material 11 models water, and 12 - boom tube steel. The composition is symmetric about the Z-axis.

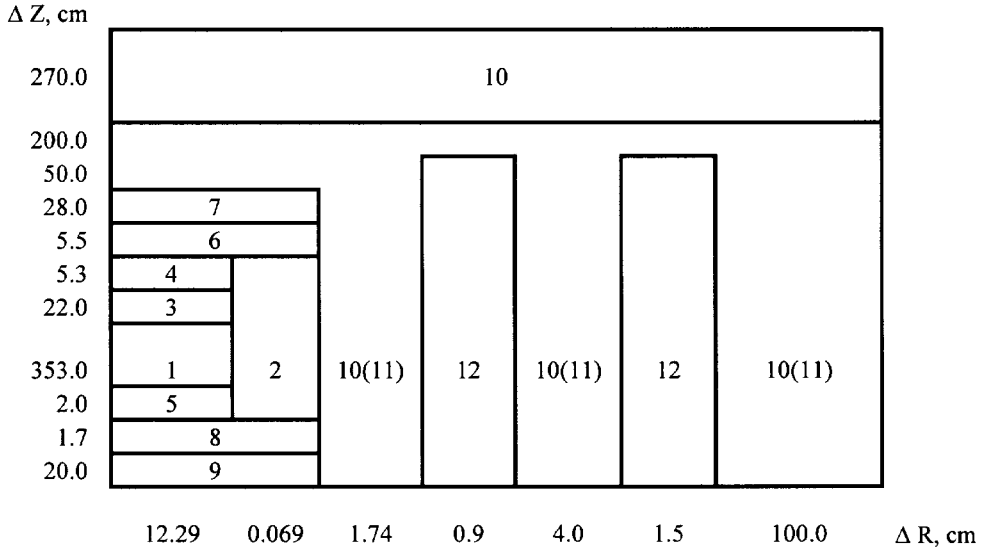



**Fig. 7. Calculation (R,Z)-model for SA in reloading machine boom**

It was required to determine equivalent dose rate values at a level of reactor room floor, at a distance of 0.5 m and 1.0 m from it and at a level of reloading machine bridge (~2 m from the floor). In the case of SA transportation under water layer (2.5 m of water above SA head), it was needed to determine a dose rate at the water surface level.

The calculations have shown that in the case when SA is in the reloading machine boom, dose rate values at the distances indicated are negligible, independently of fuel type and environment (air or water).

# CONCLUSION

From these calculation investigations the following conclusions can be drawn regarding radiation safety on fresh MOX fuel handling at NPPs with the VVER-1000 reactors:

1. Gamma and neutron dose rates at possible locations of inplant personnel in the course of process operations with fresh MOX fuel are determined.

2. On the basis of assessment of personnel individual doses the preliminary conclusion has been drawn that in the context of radiation safety the reconstruction of fuel transfer equipment is minimum when converting to MOX fuel. Radiation safety problems on MOX fuel handling can be solved through the execution of managerial and insignificant technology activity: reduction of process operation period, installation of shields, and so on.

3. Final conclusion about workscope of reconstruction of all fuel transfer equipment at NPP in converting to MOX fuel can be made after consideration of collective personnel doses including doses accumulated in process operation with spent MOX fuel.

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**Review comments on *MOX LTA Fuel Cycle Analyses, Nuclear and Radiation Safety* by D. T. Ingersoll and R. T. Santoro**

**General Comments**

***Review of Part 1: Nuclear Safety***

1. This section of the report (Part 1) is a duplication of the material contained in ORNL/SUB/00-85B99398V-3, *Shipping Cask Studies With MOX Fuel*, May 2001. Nevertheless, the reviewers offer the following comments. The work appears to be technically correct and is presented in a consistent and logical order.
2. Page 3, Abstract: Suggested wording, "VVER-1000 reactor is considered with respect to the introduction of three MOX LTAs into the core. The Monte Carlo code MCU-RFFI/A was used for the calculations."
3. Page 3, paragraph 1, line 3: Change "realizes" to "utilizes."
4. Page 5, paragraph 1, line 2: Change "loading by" to "loading of."
5. Page 5, paragraph 2, line 2: Change "applying" to "applied."
6. Page 5, paragraph 3, line 4: Change "precise" to "accurate."
7. Page 5, Introduction, consider the possible wording that follows:

   In Russia, the replacement of uranium fuel with MOX fuel is being evaluated as part of the design of VVER-1000 fuel cycles containing partial loading of weapon grade MOX.

   When compared with UOX FAs, MOX FAs have the following features that are relevant to transport and storage.

   - the fission and absorption cross sections for plutonium isotopes are substantially different than those for $^{235}U$.
   - MOX fuel is more radioactive.
   - there is a possibility for positive void effects when large Pu concentrations occur in MOX fuel.

   The calculations were performed using the Monte Carlo code MCU-RFFI/A [3].
8. Page 6, paragraph 1, line 4: Change "in the each" to "in each."
9. Page 6, paragraph 2, line 2: Change "supposed to perform" to "restricted to."
10. Page 6, paragraph 4, line 5: Change "provide" to "provides."
11. Page 6, paragraph 4, line 11-12: Change "regimes while coolant density lowering" to "lowering of coolant density."
12. Page 6, paragraph 1: Suggested wording, delete the last sentence beginning "The package set is intended to....."
13. Page 6, paragraph 2: Suggested wording, "Storing the MOX fuel in these special package sets reduces the external radiation levels."
14. Page 6, paragraph 3: Suggested wording, "Fuel assemblies are transported from the fresh fuel depository to the reactor hall in containers (covers) that hold eighteen assemblies. The FAs can be loaded from the container directly into the reactor or, if necessary into the cooling pool."
15. Page 7, paragraph 3, bullet 4: Change "non-monotonous" to "rapidly varying."
16. Page 7, paragraph 4, line 4: Change "precision" to "accurate."
17. Page 7: Suggested wording: Change follows.

    **3. Physical features of the criticality task in fuel……..**

    Keff is ~1.4 for physically large VVER-1000 FAs containing no gaps with 4.45 enrichment (i.e., the enrichment of the make-up fuel) that are filled with water containing no dissolved absorber. Such large values for Keff, require special measures to assure nuclear safety during transport and storage. The measures must be achieved in the construction of storage facilities that handle fresh

and irradiated fuel. In addition, there must administrative and technical controls placed on these potentially hazardous systems.

FAs are positioned with ~15 cm separation distance including, in some case, absorbers between the assemblies (storage SUHT) to assure sub-criticality of the FAs during storage or processing.. The potential dangers that may occur with FA placement during storage and transport require careful analysis of water density changes, boric acid concentration variations and geometry that could lead to accident situations are the principle requirements of this task. These analyses are different from the more traditional tasks associated with VVER core calculations.

Neutron moderation and diffusion in fuel transport and storage equipment have the following problems:

- some of the neutrons that are born in the FA (fission spectrum) moderate inside the FA, while other of the neutrons (approximately equal to the ratio of the gap volumes between fuel assemblies to the overall water volume) moderate in the gaps without significant absorption;
- the existence of strong anisotropic scattering in hydrogen leads to excessive neutron moderation in the gaps;
- the thermal neutron density in the gaps is significantly higher than the thermal neutron density in the FA itself, leading to large thermal neutron diffusion from the gap to the to the fuel. It is necessary to carefully account for the scattering anisotropy in the criticality calculations for these cases;
- the existence of borated steel (in SUHT) in the cover absorbers leads to non-monotonic neutron distributions near the FA-gap border.

Because of these, criticality calculations require that the anisotropic scattering be accounted for (a common problem in radiation protection calculations) and that the FAs and associated equipment be carefully modeled for cell calculations.

The Monte Carlo code MCU-RFFI/A [3] was used to calculate the multiplication factor, Keff, and the flux functionals in the neutron multiplying system. The code solves the neutron transport equations in arbitrary three-dimensional geometries using cross-section data specifically tailored for these kinds of problems. The code allows the user to solve both homogeneous (criticality) and non-homogeneous (outer source) problems. The code allows for different boundary conditions, leakage through the outer surface of the geometry, white and mirror reflection and translational geometry.

Subgroup (Bonderenko f-factors) are used to account for cross-section shielding in the unresolved resonance regions in the cross-section data. Pointwise cross-section data are used in the unresolved resonance region with the most important cross sections described by and “infinite” number of points that are determined from analytic formulae based on each energy point. This scheme allows the user to perform calculations at any temperature without preliminary preparation of the cross-section tables.

The DLC/MCUDAT-1.0 neutron physics data base is used and includes

| | |
|---|---|
| BNAB/MCU | an enlarged and modified version of the 26-group constant system BNAB, |
| LIPAR | resonance parameters in the resolved resonance region, |
| TEPCON | multigroup cross-sections for the thermal region, and |
| VESTA | the library for modeling neutron-nucleus collisions that account for the continuous change in neutron energy in the thermal region. These data are given in the form of probability tables obtained from $S(\alpha,\beta)$ scattering laws. |

MCU-RFFI/A and DLC/MCUDAT-1.0 have been verified against 400 integral benchmark experiments. Good agreement among-to-measured, C/M, values for Keff. was achieved.

The MCI-RFFU/A code incorporates combinatorial geometry that allows for detailed three-dimensional modeling. The user has a choice of thirteen types of bodies (cylinder, cone, sphere, parallelepiped, etc.). The use of symmetry of systems and lattices makes it easier to describe geometries and boundary conditions including heterogeneity.

Functionals are determined as flux integrals with given weight functions in registration zones, registration objects and the system as a whole. Energy group boundaries are set by the user to integrate over energy.

The following values are calculated

- neutron multiplication factor (by number of collisions, number of absorptions, combined estimates),
- neutron flux density,
- reaction rates for individual nuclides and mixtures in the specified space-energy intervals,
- few-group constant sets for registration objects, and
- the effective fraction of delayed neutrons.

18. Page 8, paragraph 1, line 5: Change “muly-group” to “multi-group.”
19. Page 11, Table 5.1: Change “,” to “.” in first row numbers.
20. Page 11, paragraph 4, line 5: Change “ϑeff” to “Keff.”
21. Page 11, paragraph 4: No details are provided on which to base the conclusion that four rows of FAs will meet the criterion.
22. Page 13, paragraph 2, line 2: Change “at the firure 5.3” to “in the figure 5.3.”
23. Page 13, paragraph 2, line 3: Change “acid in current” to “acid in coolant.”
24. Page 13, paragraph 2, line 5: Change “Fas” to “FAs.”
25. Page 13, paragraph 4, line 2: Change “4% till” to “4% to.”
26. Page 20, bullet 1: Keff values are only presented for the cases of an infinite array of FAs. Some basis for concluding that a four-row stack is acceptable should have been provided.
27. Page 20, bullet 2: Given the rapid rise in Keff for a slight increase in coolant density near 0 density, this option for FA storage should not be considered unless water ingress can be ensured.

*Review of Part 2: Radiation Safety*

1. Page 4, paragraph 1, line 4: The term “SA” is not defined. Same as “FA”?
2. Page 4, paragraph 6, line 1: Change “$UI_2$” to “$UO_2$.”
3. Page 5, Table 1.1: The term $D\gamma(\beta)$ is not defined. Also, the unit “μkSv/hr” is unknown. Is this intended to by mSv/hr, that is, $10^{-3}$ Sv/hr?
4. Page 6, Table 1.2: There are very large unexplained differences in three calculations, especially for gamma rays.
5. Page 7, paragraph 2, line 2: Change “date” to “data.”
6. Page 8, Table 2.1: Change “circonium” to “zirconium.”
7. Page 11, Section 2.1.3: Change “breaking” to “bremsstrahlung.”
8. Page 13, paragraph 2, line 1: Change “kinetic” to “transport.”
9. Page 13, heading 2.1.4.1: Change “kinetic” to “transport.”
10. Page 13, paragraph 3, line 1: Change “kinetic” to “transport.”
11. Page 18, Fig. 2 caption: Include “in packing set” to distinguish from Fig. 1 caption.
12. Page 24, Fig. 6 caption: Change “jacked” to “jacket.”
13. Page 24, Table 2.14 heading: Change “18 SA” to “13 SA.”

14. Page 25, paragraph 2: Conclusions regarding angular variations are not supported by any of the data presented because all presented data are from (R,Z) calculations that are angle-independent.
15. Page 27, bullet 2: Giving a specific dose impact would have been more informative, such as "dose increases ranging from a factor of 3 to 8 were observed for MOX assemblies relative to UOX."
16. Page 28: References need to be renumbered consecutively from 1 through 9.

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