Disposal aspects of low and intermediate level Ignalina NPP decommissioning waste

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CONTENTS OF THE CD-ROM

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DISPOSAL ASPECTS OF LOW AND INTERMEDIATE LEVEL IGNALINA NPP DECOMMISSIONING WASTE

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Abstract

The paper presents the main principles, criteria and methods for estimating amounts of contaminated and activated radioactive waste generated during dismantling of technological installations at Ignalina NPP. The improved computer code "DECOM" enabled the recording of the necessary information, performing of the initial data processing and splitting of contaminated waste into different streams based on their dose rates. Rather detailed information about the rooms in controlled area is the bases for performing analysis of the possible waste generation during decommissioning of Ignalina NPP.

Activation modelling of the components of the RBMK-1500 reactor core was performed and preliminary specific activity limits for disposal in planned near surface repository in Lithuania, based on water pathway analysis, have been derived for packages of activated reactor components, such as the shielding and support plates of graphite stack.

1. SCOPE

As part of the coordinated research project "Disposal Aspects of Low and Intermediate Level Decommissioning Waste," the Lithuanian Energy Institute carried out the research project "Disposal Aspects of Low and Intermediate Level Ignalina NPP Decommissioning Waste". Specific areas of the work included:

- Assessment of the characteristics and inventory of the contaminated decommissioning waste streams at Ignalina NPP on a system-by-system basis.
- Calculation of the activation of some reactor components (conservative assumption of impurity quantities)
- Preliminary specification of packages for some activated waste to be disposed of in the planned near surface repository in Lithuania

2. INTRODUCTION

There is only one nuclear power plant in Lithuania – Ignalina NPP (INPP). It operates two similar units with a power rating of 1500 MW(e) and a present power level of about 1250 MW(e) each. They were commissioned (first grid connection) in 12/1983 and 08/1987 respectively and provided approximately 70-80% of the electricity produced in Lithuania. The original design lifetime was projected from 2010-2015. On 10 October, 2002 Seimas (Lithuanian Parliament) approved an updated national energy strategy that indicated that the first unit will be shutdown before the year 2005 (following this it was to be shutdown in December 2004) and the second unit in 2009, if funding for decommissioning is available from the EU and other donors. On 26 November, 2002, the Lithuanian government approved an immediate dismantling strategy for Unit 1.

Decommissioning of nuclear power plants is a long and complicated process that requires considerable funds. The preparation for this process also lasts a few years and in case of Ignalina nuclear power plant (Ignalina NPP) means the preparation for safe dismantling of a power plant, the treatment storage and disposal of operational radioactive waste, the storage of spent nuclear fuel, etc. In order to be able to plan the dismantling activities and to introduce radioactive waste processing technologies, storage facilities and repositories, it is necessary to have the preliminary data of the amount of radioactive waste generating during the decommissioning process of the plant, the radioactivity level, nuclide composition and other data.

This work presents the methodology and preliminary results of the assessment of contamination in technological installations by radionuclides, neutron induced radioactivity in some reactor components and the waste streams generating during future dismantlement process at Ignalina NPP. It also presents the preliminary analysis of the possibility of disposing activated reactor metallic components into the near surface repository.

3. CONTAMINATED WASTE

During operation of a nuclear power plant not only the reactor itself, but also other systems are being contaminated, such as the main circulation circuit, purification and cooling system, spent nuclear fuel storage pools and others. Their contamination by radioactive particles is due to the circulation of cooling agent (in case of Ignalina NPP – water) in these systems. The water itself is contaminated in the reactor area because of the activation, corrosion processes and defects in fuel cladding. In the case of forced water circulation, radioactive particles in various systems precipitate on the internal walls of system components.

For the assessment of closed systems equipment radioactive contamination, the modified computer code "LLWAA – Decom" (Belgium) was used. The code allows for the determination of the activity (Bq/m^2) of the deposits located on the system equipment inner surfaces, taking into account the coolant specific activity (Bq/m^3) and the construction data of system elements (construction materials, geometrical measurements, etc.). It allows also for the calculation of equipment contact dose rate (or the dose rate at the given distance, for example, in case of the presence of thermal insulation). The predicted dose rates can be compared to the measured values at INPP. A good agreement between the predicted and measured equipment dose rates constitutes the basis for the code validation, i.e. of the validation of the predicted deposited activities. Another possibility of validation was the measurement campaign carried out on steel samples removed from the MCC of Unit 1 during the 2002 maintenance outage.

Deposition rate and release rate coefficients are the functions of fluid characteristics (velocity, temperature, Reynolds number), the system equipment characteristics (geometry, inner walls roughness, friction factors), and the characteristics or radioactive particles (its specific weight, diameter). As mentioned above, equipment contamination is concentrated in the surface layer. Contamination occurs due the contact with contaminated coolant. Only the fuel channels (MCC elements) located in the reactor core are contaminated, mostly due to the activation process in the core. The dose rates from different nuclides in reactor water and nuclides deposited on the inner wall of the elements have also been determined. Calculations show that the dose rate from MCC fluid is much smaller than dose rate from deposits. Coolant will be removed before the dismantling process. However, there is an opportunity to compare predicted and measured dose rates even for the systems filled with the coolant.

A detailed assessment of component radiological characteristics was performed for the five most contaminated Unit 1 systems of Ignalina NPP:

The contamination of the remaining system components is rather low. Due to the lack of radiological characterization data, the assessment the radiological characteristics of mentioned components is made conservatively based on existing radioactive measurement data for operational waste, categories of the rooms, etc. (Table 1)

For the assessment and grouping of radioactive waste at Ignalina NPP controlled area the computer code DECOM, developed while preparing the Preliminary decommissioning plan for decommissioning of Ignalina NPP by the efforts of both consortium NIS/SGN/SKB and Lithuanian Energy Institute (LEI), was used. The database of this code includes the data about 42 000 components (or their groups). Later on, this database was permanently complemented and adjusted by more detailed information about the installation data, as well as improving the software by LEI.

Estimated decommissioning waste streams for Ignalina NPP are presented in Figure 1 and Table 1.

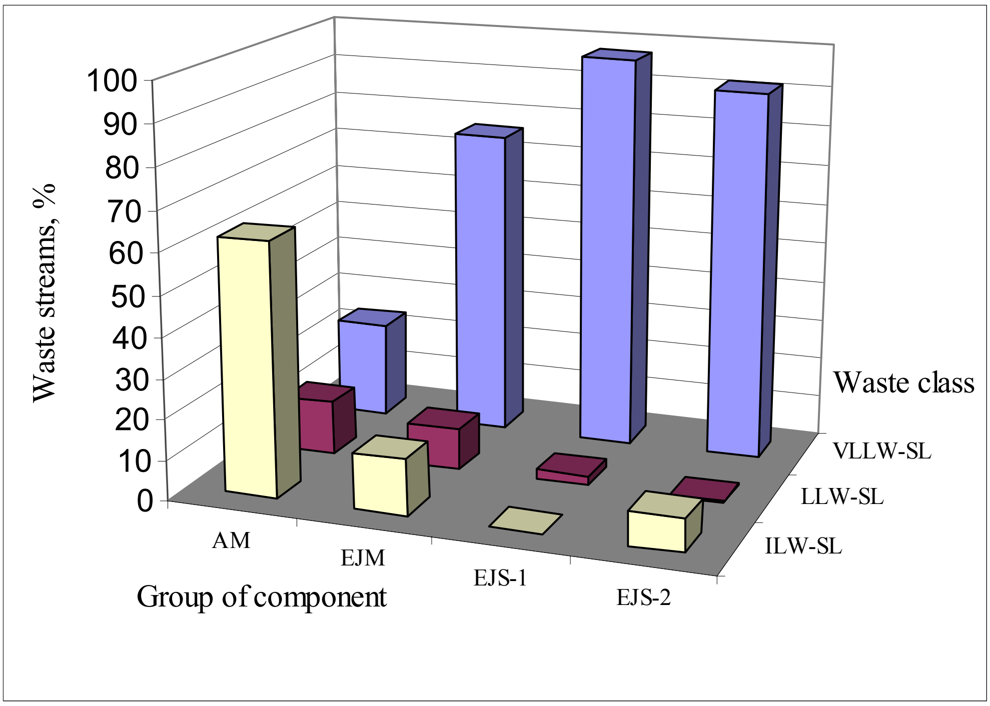

Fig. 1. Overall decommissioning waste streams for the whole Ignalina NPP at reactor final shut down. VLLW-SL – very low level waste short lived, LLW-SL – low level waste short lived, ILW-SL – intermediate level waste short lived.

Table 1. Estimated decommissioning waste streams for Ignalina NPP at reactor final shutdown

No	Group of components	Waste generated (mass proportion), %		
		VLLW-SL	LLW-SL	ILW-SL
1	DA (Detailed radiological analysis of systems installations)	24	13.4	62.6
2	EJM (Engineering judgment of system contamination based on maximum dose rates defined during scheduled maintenance and repair works in that system.	76	10.3	13.7
3	EJS-1 (Engineering judgment of the contamination of the installations located in the room, based on the maximum dose rate measurements of the operational waste collected in the room).	97.6	2.4	0
4	EJS-2 (Engineering judgment of the contamination of the installations located in the room, based on the maximum available contamination assigned to the room category).	91.6	0.3	8.1
5	**Total waste stream mass distribution, %**	**79.3**	**6.0**	**14.7**

An analysis of Ignalina NPP decommissioning waste streams shows that it is possible to expect that about 80% of the waste will be VLLW, which could be disposed of into licensed landfill repositories. It is necessary to keep in mind that the waste was split into groups according to the dose rate, which is included in the requirements for radioactive waste treatment at NPP before their disposal [1]. However, this is only a very rough estimation because waste acceptance criteria for landfill repository usually also includes limitations on some of the most important nuclides (especially ^{137}Cs). Therefore, in the future it will be necessary to assess the nuclide activity of waste and to apply them to the real waste acceptance criteria for the landfill facility. More detailed information on modelling aspects and obtained results is presented in [2, 3]

4. ACTIVATED WASTE

As soon as decommissioning work of the reactors of Ignalina NPP begins, the arising radioactive waste will also consist of construction materials of the reactor structure (graphite, concrete, metal parts). These materials are located in and near the reactor core and become radioactive due to the neutron irradiation during the NPP operation. The material composition, neutron flux density and energy distribution in different reactor zones are different, so are the neutron activation conditions. The reactor core (graphite stack, fuel channels tubes, etc.) is the most activated part of the reactor structure, whereas the biological shield (usually concrete and steel structure) is much less activated, as the neutron fluxes are relatively low.

The activity of reactor structure components depends on the initial chemical composition of the particular component material, time and other conditions (e. g. neutron flux) of neutron irradiation and time after final reactor shutdown.

The RBMK-1500 is a graphite-moderated, water-cooled reactor core having a design thermal power capacity of 4800 MW. A schematic section of an RBMK reactor vault is shown in Figure 2.

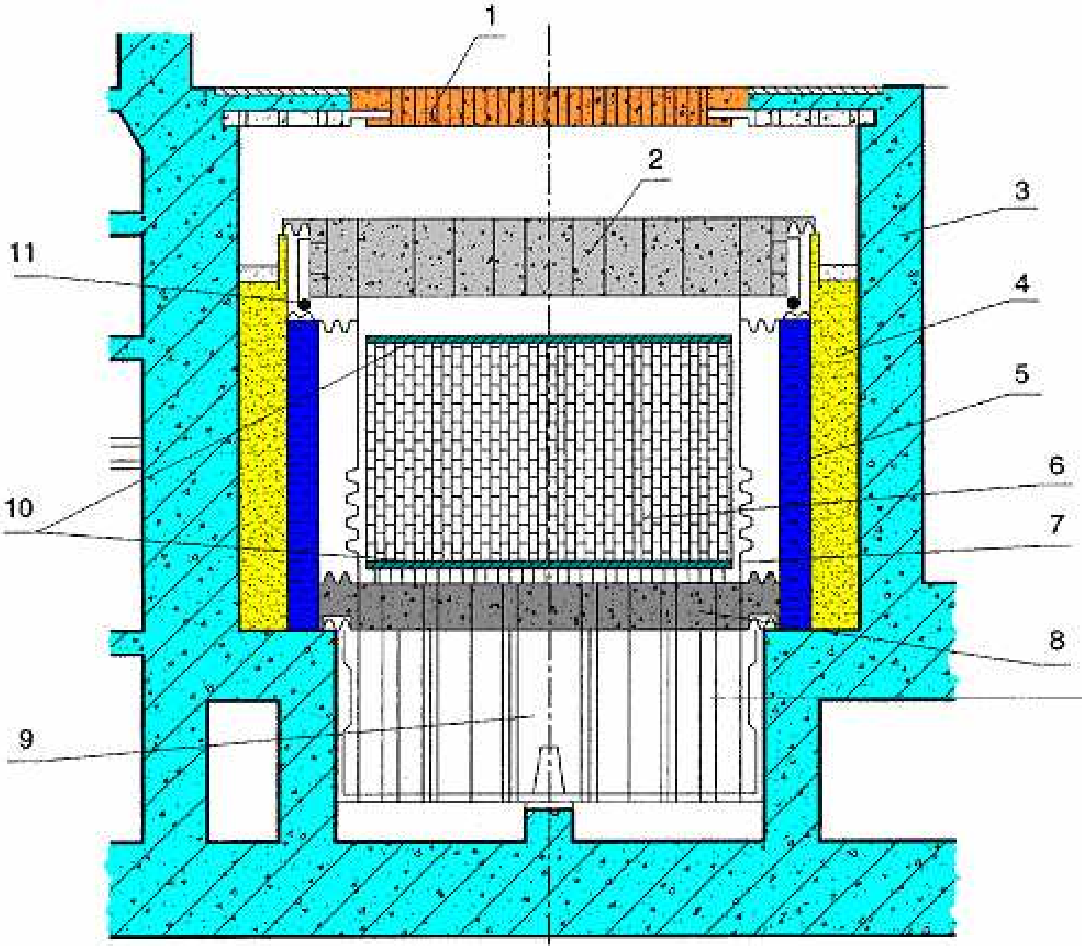

Fig. 2. Cross-section of the reactor vault [4].

1. top cover, removable floor of the central hall
2. top metal structure filled with serpentinite
3. concrete vault
4. sand cylinder
5. annular water tank
6. graphite stack
7. reactor vessel
8. bottom metal structure
9. reactor support plates
10. steel blocks
11. roller supports

The graphite structure consists of 2488 channels, and is made up of columns of bricks each with an axial hole for the channel tube. There are 2052 channels used for fuel, control rods and instruments with the remaining 436 channels around the edge of the core filled with graphite rods to act as the reflector. The entire stack is approximately 8 m high and has a diameter of about 14 m. The four rows of graphite columns at the outer edge make up the radial reflector (~ 1 m thick), and a 0.5 m thick layer at the top and bottom make up the end reflectors. Radial displacement of the graphite stack is prevented by 156 water cooled supporting tubes situated at the outer periphery. These tubes are welded to the lower support structure at their base but have the freedom to move in vertical guides at the top. The core is contained above and below by the biological shields and is radially surrounded by a nitrogen/helium gas blanket and water tank. The pressure of the environment is 0.49 kPa in the core cavity and this is lower than outside. Each column of bricks is independent, i.e., there is no keying system. A cup-cone arrangement is used to join and align the bricks end to end. There are steel bricks at the top and bottom of the column. The upper steel bricks are called shield plates and the bottom steel bricks are called support plates. At the base of the column the bricks are located on a spigot and at the top the column is located in line with a hole in the upper biological shield by means of a telescopic joint. The horizontal joints between bricks are staggered in adjacent channels to avoid any horizontal planes of weakness. The brick has a square cross-section and the central hole has a diameter of 0.114 m. The basic bricks are 0.6 m high although shorter bricks of 0.2, 0.3 or 0.5 m height are also used in parts of the stack. Initially there is very little clearance between the bricks, approximately 1 mm. The fuel cell assembly includes a zirconium pressure tube into which the fuel element assembly is inserted and through which the coolant flows. The pressure tube is located in the central hole of the brick by a system of graphite split rings. Each ring is alternately tight on the pressure tube or tight in the bore of the brick. To prevent oxidation of the graphite and to improve the thermal efficiency, the core is contained in 90% helium, 10 % nitrogen gas mixture. The slots in the graphite rings are aligned to allow the gas mixture to pass along the channel. In this study the model for numerical assessment of Ignalina NPP reactor construction materials neutron activation was developed based on conservative assumptions. Activity inventories were estimated only for graphite stack, channel tubes (to be more precise, for their middle parts, which are made of zirconium-niobium alloy E125), support and shielding plates.

ORIGEN-S computer code (SCALE 5 codes system) was used for the activation analysis [5]. The code considers radioactive disintegration and neutron absorption (capture and fission) and enables to identify isotopic content, activities and concentrations of neutron activated elements.

On the basis of the neutron fluxes measurements carried out in Ignalina NPP Unit 1 reactor core [6] and geometrical dimensions of particular reactor components, it was assumed that thermal neutron flux density remains constant during all irradiation period for all directions in selected component and is equal to:

- $3 \cdot 10^{13}$ n/(cm^2·s) in the active reactor core (fuel channels, graphite sleeves and blocks in the active core);
- $1 \cdot 10^{13}$ n/(cm^2·s) in the top reflector of the active core (graphite sleeves and blocks in the top reflector);
- $1.5 \cdot 10^{13}$ n/(cm^2·s) in the bottom reflector of the active core (graphite sleeves and blocks in the bottom reflector);
- $7.5 \cdot 10^{12}$ n/(cm^2·s) in the radial reflector of the active core (graphite blocks in the radial reflector);
- $6 \cdot 10^{10}$ n/(cm^2·s) in the upper steel bricks (shielding plates);
- $9 \cdot 10^{10}$ n/(cm^2·s) in the lower steel bricks (support plates).

Figure 3 shows the variation of total specific activities for different reactor components during the 150 year cooling period. It is seen that channel tubes have highest activity concentration during all modelled decay time period. Metal reactor parts have higher specific activities than the graphite parts (bricks and sleeves) at the time of shut down, but after ~ 15 cooling years they reach specific activity levels of active core graphite blocks and sleeves.

The highest activity concentrations for graphite parts are accumulated in active core blocks, sleeves and bottom reflector blocks for all modelled 150 years cooling period.

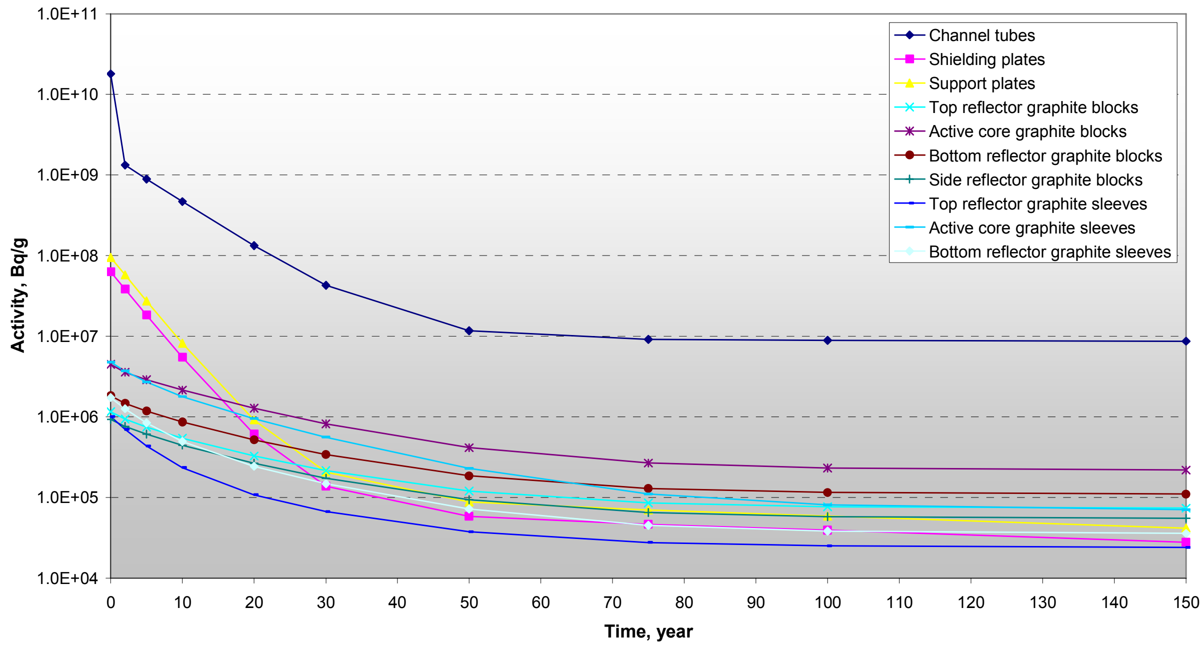

Fig. 3. The variation of specific activities in different components of INPP Unit1 reactor structure.

Total radionuclide activities in selected reactor components were estimated as the result of the specific activities and mass of each of the components. At the time of final reactor shutdown, the highest overall activity is accumulated in channel tubes. During the first 2 years of cooling the activity of channel tubes decreases significantly, then from this moment to ~ 50 years the decrease is not so high, and for the remaining ~ 100 years the activity concentration stays almost constant. However, the highest overall activity for the modelled time period up to 150 years still is induced in fuel channels. The total activities of metal shielding and support plates are higher than the activity of graphite blocks for the first ~ 10 years of cooling, and reaches the total activity level of graphite sleeves after approximately 30 years of cooling. More detailed information on modelling aspects and obtained results is presented in [7, 8].

5. PRELIMINARY SPECIFICATION OF THE PACKAGES FOR ACTIVATED WASTE TO BE DISPOSED OF IN THE NEAR SURFACE REPOSITORY

Preliminary specific activity limits have been derived for packages of activated reactor components, such as the shielding and support plates of graphite stacks. It was assumed that the wastes should be conditioned and disposed of in the near surface repository (NSR) that is planned to be constructed in Lithuania [9]. The derivation of the specific activity limits has been performed using IAEA recommended methodology [10], with respect to requirements of Lithuanian norms concerning radioactive waste management as well as radiation protection.

A waste-leaching scenario with relevant changes of water infiltration rate through the repository during analysed period is considered. Additionally for the task of derivation of activity limits the behaviour of waste form is taken into account, i. e. the container durability of 100 years and further uniform dissolution of activated metallic plates over a period of 100 years (1 % per year) is evaluated [11].

The migration of unit inventory (1 Bq of initial activity for each radionuclide) through the vault and vadose zone has been carried out using the DUST computer code [12] where finite difference method is employed to solve 1-D transport equation with processes of advection, dispersion and radioactive decay. The assessment of radionuclide transport in aquifer has been performed using GWSCREEN code [13] where 2-D dispersion modelling is implemented.

A potential exposure to local individual of critical group via ingestion of drinking water from well in aquifer has been evaluated. The well is installed at distance of 150 m from the edge of the vault (boundary of the repository). A drinking water consumption of 600 litres per year is supposed.

After modelling of ^{59}Ni, ^{63}Ni nuclides migration through the disposal system only impact of the ^{59}Ni radionuclide has been identified.

Due to rather short half-life (100 years), thick vadose zone (30 m) and high value of distribution coefficient in vadose zone (K_d = 0.3 m^3/kg) and in aquifer (K_d = 0.4 m^3/kg) the peak concentration value of ^{63}Ni radionuclide in aquifer well (receptor) is less than 10^{-150} orders of magnitude. Therefore the dose induced by ^{63}Ni is negligible. Hence, the preliminary activity limit for ^{59}Ni has been derived.

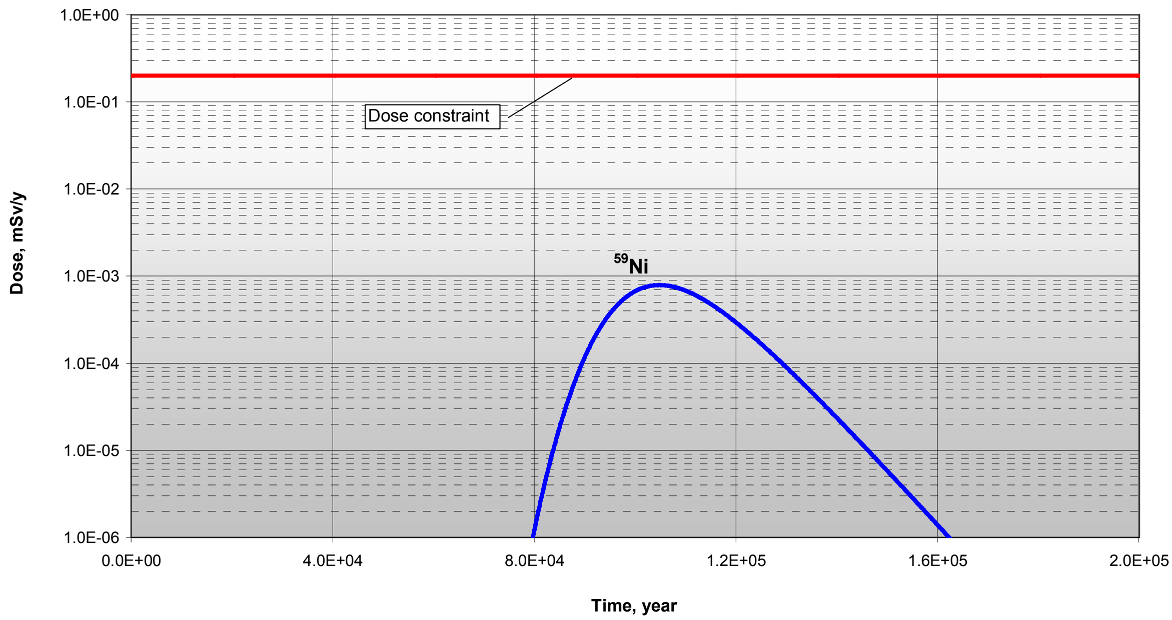

Fig. 4. Expected doses from radionuclide releases to groundwater pathway.

The preliminary specific activity limit of $2.08 \cdot 10^5$ Bq/g has been assessed for ^{59}Ni radionuclide. After comparison of this value to specific concentration values for ^{59}Ni radionuclide that equal to $6.45 \cdot 10^2$ Bq/g for shielding plates and $9.66 \cdot 10^2$ Bq/g for support plates at the time of reactor shutdown, it was concluded that it could be possible to dispose of the activated metallic radioactive wastes into the Lithuanian near surface repository. The expected doses of ^{59}Ni radionuclide releases from the vault to groundwater pathway are provided in the Figure 4. As the graph shows the maximum value of potential dose is two orders of magnitude lower than dose constraint of 0.2 mSv per year established in Lithuania.

This analysis provides only a very preliminary estimate because, according to Lithuanian regulations [14] and IAEA methodology [10], the activity limits in the case of disposal system evolution (groundwater pathway) as well as the case of inadvertent human intrusion should be evaluated.

6. CONCLUSIONS

(1) Analysis of Ignalina NPP decommissioning waste streams based on dose rate criteria shows that it is possible to expect that about 80 % of the waste will be VLLW that could be disposed of into licensed landfill repositories.
(2) The highest overall activity for the modelled time period up to 150 years is induced in fuel channels. Total activities of metal shielding and support plates are higher than the activity of graphite blocks for the first ~ 10 years of cooling and reaches total activity level of graphite sleeves approximately after 30 years of cooling.
(3) Fuel channel tubes have the highest specific activity during all modelled (150 years) decay time period. Metal reactor parts have higher specific activities than the graphite parts (bricks and sleeves) at the time of shut down, but after ~ 15 cooling years they reach specific activity levels of active core graphite blocks and sleeves.
(4) Under a conservative estimation of the radionuclide inventory of activated shielding and support plates intended to be disposed of in the vault of the planned near surface repository, the resulting dose to the member of critical group should be two orders of magnitude below the dose constraint defined by Lithuanian norms.

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