Safety Assessment for VLLW Disposal at the National Radioactive Waste Repository Mochovce in Slovakia - 13508

E. Biurrun*, A. Lázaro **, B. Haverkamp*, A. Miralles**
*DBE TECHNOLOGY GmbH, Eschenstr. 55, D-31224 Peine, Germany
biurrun@dbe.de; haverkamp@dbe.de
**Westinghouse Electric Spain SAR, Padilla 17, E-28006 Madrid, Spain, lazaroa@westinghous.com; miralla@westinghouse.com

ABSTRACT

Recent developments in the Slovak Republic have prompted the need to introduce the new category of very low level waste (VLLW) in the operation of the country's repository for low and intermediate level radioactive waste (LILW). By doing this, significant savings are expected to be achieved while disposing the waste resulting from early decommissioning of older, Soviet type reactors. To study the feasibility and the likely impact of such introduction, a project was launched and assigned in international competition to a German-Spanish consortium. The study confirmed by means of a safety assessment the feasibility of this waste category in the specific context of the Slovakian repository. Moreover, the advantages that such new waste category would render were stressed and the best option for enlargement of the repository, the construction of a module for LILW disposal within the limits of the existing repository, was identified.

INTRODUCTION

The National Radioactive Waste Repository (NRWR) at Mochovce, in Slovakia, currently receives operational LILW originating from the Water-Water Energy Reactors (WWER) 440-V213 of the Units 1 and 2 of the Bohunice V2 nuclear power plant (NPP) and of the Units 1 and 2 of the Mochovce NPP. In the near future the NRWR will also receive decommissioning waste from the Bohunice A1 NPP and from the Bohunice V1 NPP Units 1 and 2 (two WWER 440-V230 reactors). In addition, the operational and decommissioning waste amenable for surface disposal of all other four operating WWER 440-213 at Bohunice and Mochovce and that of the two reactors under construction at this last mentioned site, will also be disposed of in the NRWR. The waste received for disposal there is conditioned into 220 liter drums, which in turn are placed into a cubic-shaped standard Fiber Concrete Container (FCC), the void space around the drums is filled with cement mortar.

The decommissioning of the Bohunice A1 and V1 NPPs is to be carried out ahead of the original schedule as a condition of Slovakia's accession to the European Union (EU). The early decommissioning has a significant economic impact for Slovakia, so that the EU established a compensation fund to support Slovakia in its efforts: the Bohunice International Decommissioning Support Fund (BIDSF), which is managed by the European Bank for Reconstruction and Development (EBRD). The early NPP decommissioning has a significant impact on the planned disposal operations at the NRWR, and especially on the rate at which disposal vaults need to be constructed. Therefore, the Slovak organization for waste management and disposal, the Jadrová a vyraďovacia spoločnosť, a.s. (Javys), in cooperation with the EU launched as a first optimization step a *feasibility study for the enlargement of the Mochovce repository*.

The study included design efforts to optimize the use of the disposal space through the introduction of updated waste packages as well as the establishment of a dedicated disposal area for VLLW. The VLLW disposal was considered to be implemented at four prospective locations:

a) at the site of the Bohunice NPP,

b) at the site of the Mochovce NPP,

c) at the site of the current NRWR, a few kilometers away from this last NPP, as well as

d) at a yet to be determined site elsewhere in Slovakia.

The evaluation of alternatives to select the most suitable one was to be carried out on the basis of a multi-attribute analysis considering operational and long-term safety, environmental impact, public acceptance aspects, and costs.

THE EXPECTED WASTE INVENTORY

As a first step the waste inventory expected to arise from operation of the existing NPPs, from the two units currently under construction, as well as from NPP decommissioning was estimated, as such information was not available in Slovakia. The estimated inventory information included not only volumes of waste but also their expected radionuclide spectrum.

The assessment of expected waste volumes is necessary to determine the space needed to dispose of both types of waste: LILW packed in FCC's and emplaced in concrete disposal vaults (double rows) on the one hand and the disposal cell proposed for disposal of the VLLW but packed in different containers (big bags, 220-liter drums, etc.) The required result is the number of fiber concrete containers expected to arise, used in turn to determine the number of LILW disposal vaults that the repository must have, and the volume of VLLW expected to arise from all mentioned activities, used to determine the dimensions and amount of VLLW disposal cells needed.

The assessment of the radioactive waste inventory relies on three main elements:

- The general radioactive waste processing strategy.
- The available radiological information of the expected waste streams resulting from the operation and decommissioning of the Slovak nuclear power plants.
- The waste acceptance criteria for the LILW and VLLW categories.

The General Waste Processing Strategy

In assessing the waste inventory the following aspects of processing strategy were considered:

- Pre-treatment process according to the nature of the radioactive waste, i.e., decontamination for solid and evaporation for liquid waste.
- Treatment and conditioning process according to the waste nature, i.e., compaction and incineration for solid waste and cementation and bituminization for liquid waste.
- Waste type and number of waste packages and containers used.
- Immobilization process, i.e., the waste grouting with mortar in the waste packages.

The Radiological Information

In assessing the inventory the following radiological information was taken into account:

- Rates of production, types of waste arising during operation of the NPPs, and the total amounts expected during the full service life of the power plants

- Pre-decommissioning studies of WWER-440 NPPs and of the JE Bohunice A-1, a HWGCR nuclear demonstration plant decommissioned after an incident.

The total number of NPPs considered and the current status and projected service lives are detailed in Table I.

Table I: NPPs giving rise to the waste considered in the safety assessment

Nuclear Power Plant	Rated Power [MW(e)]	Reactor Type	Current Status
NPP Bohunice A-1	150	HWGCR	Decommissioning
NPP Bohunice V-1 (JE V-1)	2x440	WWER 400/230	Final shutdown
NPP Bohunice V-2 (JE V-2)	2x440	WWER 400/213	Oper. until 2020s
NPP Mochovce -1,2 (JE EMO-1,2)	2x440	WWER 400/213	Oper. until 2040s
NPP Mochovce -3,4 (JE EMO-3,4)	2x440	WWER 400/213	Under construction

The Waste Acceptance Criteria for LILW and VLLW

In assessing the radioactive waste inventory (volumes, masses) the following information on acceptance criteria was taken into account

- The current LILW acceptance criteria of the NRWR Mochovce, which were an input to the study.
- The VLLW acceptance criteria, derived in the feasibility study as summarized in the subsequent section.

Both sets of acceptance criteria allowed assigning most of the waste streams either to the category of LILW or to VLLW; specially the decommissioning waste that by volume is the largest waste fraction. Further information used to assess the waste inventory includes design information on the WWER reactors as, e.g., data on masses of pieces of equipment and contamination and activation of materials, etc. The results obtained for the estimated inventory are shown in Table II, subdivided into LILW and VLLW and into operational and decommissioning waste.

Table II: Volumes of radioactive waste estimated for the NRWR Mochovce

Total Waste Volume [m^3] estimated to result from NPP Operation and Decommissioning		
LILW	**VLLW**	**LILW & VLLW**
49343 (45.5%)	**59126** (54.5%)	**108469**

With these waste volumes, the modules for LILW disposal (the so-called double rows) and the space required for VLLW disposal shall have the following total capacities:

LILW Packages	**27,105 FCCs**
LILW Disposal Vaults	**7.5 Double Rows**
Total VLLW Volume	**67,706 m^3**

Also the spectrum of the waste expected to arise was estimated using available radiological information on the waste streams and the corresponding amounts of contaminated or activated materials. After applying the Specific Activity Limits for both waste categories the contents of radionuclides for the LILW and VLLW fractions were assessed as detailed in Table III.

Table III: Estimated total activity content of the expected Slovak waste inventory

Total Activity Content			
NUCLIDE	**LILW [Bq]**	**VLLW [Bq]**	**VLLW [%]**
H-3	1.6E+13	4.3E+10	0.26%
C-14	3.7E+11	2.1E+10	5.34%
Ca-41	4.0E+10	2.6E+08	0.66%
Co-60	2.1E+14	4.2E+11	0.20%
Fe-55	9.7E+14	1.5E+12	0.15%
Tc99	2.3E+10	4.5E+07	0.20%
Ni-59	1.7E+13	1.3E+09	0.01%
Ni-63	2.4E+14	4.4E+11	0.18%
Cs-137	2.3E+14	1.9E+12	0.80%
Cs-134	4.4E+12	2.0E+10	0.44%
Nb-94	1.5E+13	1.8E+10	0.13%
Sr-90	3.2E+12	1.5E+09	0.05%
I-129	1.8E+11	1.7E+08	0.09%
Eu-152	3.1E+11	2.1E+10	6.36%
Cl-36	3.5E+10	2.3E+08	0.66%
Mn-54	2.0E+11	1.1E+07	0.01%
Pu-239	1.3E+11	3.7E+08	0.28%
Pu-238	1.6E+11	9.0E+08	0.58%
Am-241	2.0E+11	3.8E+09	1.84%
TOTAL	**1.71E+15**	**4.36E+12**	**0.3%**

THE WASTE ACCEPTANCE CRITERIA FOR VLLW

The waste acceptance criteria for VLLW include a set of specific activity limits (SALs) that must be fulfilled by all waste packages delivered for disposal. The activity limits for VLLW disposal result from a set of scenarios covering the *Operational Phase* on the one hand, and the period after *end of the Institutional Control Phase*, which follows the operational phase and lasts typically for 300 years, on the other hand. To compute the SALs the involved source term for a given scenario is considered as well as the critical individual and the dose limit or dose constraint applicable to this scenario, which might be different for different scenarios depending on their probability of occurrence. Moreover, depending upon the scenario and of the exposure individuals (workers, members of the public), direct radiation and also inhalation are taken into account.

The SAL for a given radionuclide is derived from the calculated dose received by the critical individual resulting from the exposure to one unit of specific activity (i.e., 1 Bq/g) of this radionuclide. The maximum acceptable specific activity for a given radionuclide results from dividing the regulated dose limit by the dose that results from exposure to one unit of specific activity of such radionuclide, as follows:

$$A_{i,j,\max}(t) = Au_i \times \frac{D_j(t)}{Du_{i,j}(t)}$$

In this equation is:

$Du_{i,j}(t)$: The dose [mSv] resulting from exposure to one unit of specific activity (1Bq/g at t=0) of the radionuclide "i" by all exposure pathways of the scenario "j" at an occurrence time t.

$D_j(t)$: The dose limit value [mSv] for the scenario "j" at the occurrence time t.

Au_i: The unit of specific activity of the radionuclide "i" at t=0 [1 Bq/g].

$A_{i,j,max}(t)$: The specific activity of the nuclide "i" at t=0 which produces a dose equal to the limit value for the "j" scenario at an occurrence time t (Bq/g).

The scenarios considered for the VLLW disposal operational phase include as Normal Operation Scenarios (NOS): *handling and disposal of a waste package* (NOS-D) and the *closure of a disposal section or lane* (NOS-C)[1]. For each scenario, the specific activity of each radionuclide was calculates that leads to just achieving the specified dose limit for workers during the mentioned operations in one lane. The Specific Activity Limits (SAL) for each radionuclide is the minimum of the results obtained for both scenarios.

$$SAL_{\text{Normal Operation Scenarios}}(\text{disposal lane})_i = \min(SAL_{NOS\text{-}D}, SAL_{NOS\text{-}C})$$

The specific activity limit for one disposal vault considers a factor of inhomogeneity F1, assumed equal to 3, which render the limit for a vault as a whole three times lower that the SAL value mentioned above for a single disposal lane. Equally, the global mean activity limit valid for the repository as a whole was set to one tenths of the SAL value above for a single lane, to take account of the larger inhomogeneity in the specific activity contents over the whole repository for VLLW (inhomogeneity factor F2 assumed equal to 3.

In addition, two different Accident Scenarios (AS) were considered during the operational phase: the *drop of a waste package during handling* (D) and a *fire affecting a waste package* (F). For each scenario the consequences to both *the operational staff* (workers, W) and to *members of the public* (P) were taken into account. For these scenarios the specific activities limits that lead to just reaching the dose limit considering a full disposal section or lane was obtained. The specific activity in one disposal section or lane is the minimum of the results of the four scenarios.

[1] Please note that in the proposed design the VLLW disposal vault consists of several sections or lanes

$$SAL_{\text{Accidental Scenarios}}\ (\text{disposal vault})_i = \min\ (SAL_{AS\text{-}DW}, SAL_{AS\text{-}DP}, SAL_{AS\text{-}FW,}\ SAL_{AS\text{-}FP})$$

The specific activity for the whole repository for VLLW was calculated considering an inhomogeneity factor F2 assumed equal to 10, as previously indicated for normal operation.

For the long-term safety, i.e., the period after the end of the Institutional Control Period the analysis considered exposures resulting from *human intrusion* (IS) into the repository as well as the chronic exposure that might result in the repository normal evolution through the *water pathway*. The intrusion scenarios included one scenario of chronic exposure to members of the public, *the residence or dwelling scenario* (IS-R), and one scenario of acute exposure to workers, *construction of a road*, which intrudes into the former disposal vaults (IS-RC). For these scenarios, the specific activity limit of each relevant radionuclide that leads to just achieving the dose limit of exposure for the critical individual considering the whole repository was calculated. The SAL for the repository as a whole is the minimum of the results of the two scenarios.

$$SAL_{\text{Intrusion Scenarios}}\ (\text{whole repository})_i = \min\ (SAL_{IS\text{-}RC}, SAL_{IS\text{-}R})$$

The limits corresponding to one disposal section or lane were obtained applying an inhomogeneity factor F2 assumed equal to 10, i.e. in a single lane the acceptable specific activity can be up to a factor 10 larger than the mean value for the whole VLLW repository, as was the case for the other SAL detailed above.

The dose limits taken into account are prescribed in the applicable Slovak regulations. For the operational period they are:

- Dose limits for the public (critical individual)
  - Normal operation: 1 mSv/y
  - Accident conditions: 5 mSv/y
- Doses to workers
  - Normal operation: 5 mSv/y
  - Accidental conditions: 50 mSv/y

For the long-term impact after the end of the institutional control period the dose limits are:

- Potential exposure resulting from human intrusion scenarios:
  - Road construction scenario: 1 mSv
  - Residence scenario: 1 mSv/y

The limit of specific activity per radionuclide for the whole repository, resulting from the analysis of all the considered scenarios, is the minimum value of the SALs for each scenario: normal operation, accident and inadvertent human intrusion.

$$SAL\ (\text{whole repository})_i = \min\ (SAL_{NOS,i}, SAL_{AS,i}, SAL_{IS,i})$$

The limits corresponding to the disposal lane are obtained applying an inhomogeneity factor F2 assumed equal to 10

$$SAL\ (\text{disposal unit})_i = SAL\ (\text{whole repository})_i \times F2$$

The resulting list of SAL is compiled in Table IV.

Table IV: Specific Activity Limits for VLLW disposal at Mochovce

NUCLIDE	Half-Life Time [y]	SAL in the Repository [Bq/g]	Maximum average specific activity in the repository [Bq/g]	Maximum specific activity in a disposal lane [Bq/g]
C-14	5.73E+03	5.34E+05	1000	10000
CA-41	1.03E+05	3.77E+04	1000	10000
NI-59	7.50E+04	6.56E+03	1000	10000
NI-63	9.60E+01	2.38E+06	1000	10000
SE-79	6.50E+04	4.55E+05	1000	10000
SR-90	2.91E+01	1.67E+04	1000	10000
MO-93	3.50E+03	1.28E+03	1000	10000
ZR-93	1.53E+06	9.08E+03	1000	10000
NB-94	2.03E+04	7.26E+00	10	100
TC-99	2.13E+05	2.38E+05	1000	10000
PD-107	6.50E+06	4.55E+06	1000	10000
SN-126	1.00E+05	2.59E+00	1	10
I-129	1.57E+07	6.42E+02	1000	10000
CS-135	2.30E+06	3.60E+05	1000	10000
CS-137	3.00E+01	6.01E+01	100	1000
SM-151	9.00E+01	6.76E+05	1000	10000
PU-238	8.77E+01	5.81E+01	100	1000
PU-239	2.41E+04	2.50E+01	10	100
AM-241	4.32E+02	6.41E+01	100	1000

Radioactive waste will be acceptable for disposal as VLLW at the enlarged Mochovce repository if its specific activity does not exceed, on the declaration date, the mass activities in the fifth column of Table IV in the case of presence of a single radionuclide. In case that the waste contains a mixture of radionuclides additionally the following summation rule must be fulfilled:

$$\sum a_i / A_{i\,\max} \leq 1$$

In this equation is:

a_i: The actual specific activity of the radionuclide “i” in the disposal unit [Bq/g].

A_{imax}: The specific activity limit for the radionuclide “i” per disposal unit [Bq/g].

In determining the values detailed in Table IV the following rounding system was used: *if the calculated SAL value is between $3 \cdot 10^{x}$ and $3 \cdot 10^{x+1}$, the rounded value is set to $1 \cdot 10^{x+1}$ and, additionally, when the calculated SAL value is over 1000, the SAL value is set to 1000.*

SAFETY ANALYSIS FOR THE ENLARGED REPOSITORY

The safety analysis for a near-surface repository as the NRWR considers during the repository routine operations the radiation exposure resulting from direct radiation to workers and members of the public. Additionally, the exposures resulting from possible operational accidents through direct radiation and radionuclide inhalation for both the repository workers and members of the public are taken into account. These exposure values, determined by means of appropriate calculation models, are used to compute the specific activity limits. The safety of the repository, i.e., the compliance with the regulated level of protection against undue exposure is ensured during all the mentioned operational occurrences if (and only if) the SALs and the summation rule are fulfilled.

After closure, a radiological impact of the repository could result after inadvertent human intrusion as well as from exposure to radionuclides leached from the repository and transported via the groundwater. The exposure after intrusion could follow from external exposure, ingestion and inhalation. Possible exposure pathways due to radionuclide transport by groundwater are also external exposure, ingestion and inhalation. The safety of the repository in case of inadvertent human intrusion is also ensured by means of specific activity limits and the summation rule as previously described.

In contrast with this, the possible radiological consequences of exposure via the groundwater pathway in case of normal evolution and disturbed scenarios depend to a great extent from the total activity disposed of at a certain site. It is therefore important to determine for a given repository located in a specific environment the Admissible Maximum Activity (AMA) that can be disposed of there while still fulfilling the regulated protection objectives. For a given radionuclide the AMA is the maximum activity (Bq) that just matches the applicable dose limit or constraint due to exposure by all possible mechanisms. In case of decay chains, the AMA of the progenitor is evaluated considering the sum of the all doses produced by its progeny. As typically the dose peak cause by each radionuclide or radioactive chain is achieved at different times, the summation rule can usually be applied to a limited number of radionuclides at the same time, instead of applying it to the full set of radionuclides present in the repository.

For the repository normal evolution the basic assumption is that during the institutional control period and thereafter the behavior in the repository near field, the geosphere and the biosphere would be as planned and designed. The critical individual is supposed to be located at a point downstream from the repository. The dose constraint for this scenario is 0.1 mSv/yr.

As altered scenarios two different possible repository evolutions were considered: the so-called bathtub scenario and the unsealed clay scenario. In case of the former scenario as a consequence of the complete degradation of the clay cover after the institutional control period rainwater infiltrates through the cover and the waste until it reaches the bottom clay layer, which maintains its initially intended low permeability. The water then accumulates in the facility until it overflows the waste region and enters into the biosphere close to the surface. The critical individual is assumed to be located in a point next to the repository being exposed to soil contaminated with the contaminated water. The dose constraint for this scenario is 0.1 mSv/yr.

In case of the unsealed clay scenario it is assumed that after the complete degradation of the cover after the institutional control period rainwater infiltrates through the failed cover and the waste until it reaches the bottom clay layer, This bottom layer is assumed to have failed and thus to have a high permeability. In this situation, the water will not accumulate in the repository. The critical individual is supposed to be located at a point downstream from the repository. The dose constraint was set to 1 mSv/yr. The results obtained for all three mentioned scenarios and the Admissible Maximum Activity for the Mochovce site are shown in Table V below.

Table V Maximum Admissible Activities for the enlarged NRWR at Mochovce:

NUCLIDE	**Normal evolution scenario [0.1 mSv/y]**	**Unsealed clay scenario [1 mSv/yr]**	**Bathtub scenario [0.1 mSv/a]**	**Admissible Maximum Activity [Bq]**
	Activity [Bq]	**Activity [Bq]**	**Activity [Bq]**	
C-14	2.71E+11	2.02E+12	4.25E+13	2.71E+11
Ca-41	4.19E+14	3.42E+14	3.11E+15	3.42E+14
Ni-59	1.65E+16	1.06E+16	2.20E+15	2.20E+15
Ni-63	2.58E+34	3.32E+24	1.24E+15	1.24E+15
Se-79	6.07E+14	3.55E+14	8.28E+13	8.28E+13
Sr-90	3.59E+28	9.29E+19	3.19E+13	3.19E+13
Mo-93	4.52E+14	1.87E+14	1.38E+14	1.38E+14
Zr-93	5.44E+15	4.59E+15	2.25E+13	2.25E+13
Nb-94	9.11E+12	1.84E+12	1.08E+13	1.84E+12
Tc-99	3.26E+13	3.11E+13	6.91E+12	6.91E+12
Pd-107	1.80E+16	1.64E+16	9.54E+14	9.54E+14
Sn-126	2.51E+13	1.85E+13	1.51E+13	1.51E+13
I-129	5.82E+12	4.99E+12	1.56E+13	4.99E+12
Cs-135	1.34E+14	1.15E+14	1.20E+15	1.15E+14
Cs-137	3.50E+49	1.90E+37	1.28E+15	1.28E+15
Sm-151	1.10E+40	8.41E+28	2.41E+14	2.41E+14
Pu-238	5.03E+13	3.90E+13	1.03E+10	1.03E+10
Pu-239	1.36E+17	1.29E+15	1.65E+09	1.65E+09
Am-241	1.98E+17	1.86E+17	7.44E+08	7.44E+08

The Radiological Capacity (RC) of a repository to accept a given radionuclide is its Admissible Maximum Activity (AMA) if there is not a previous limitation to disposal of this radionuclide resulting from the application of the Specific Activity Limits SAL. When the specific activity limits holds then the RC for this radionuclide is irrelevant and it is not necessary to define it (NND). The radiological impact in the long-term by the groundwater pathway will be below prescribed constraints and thus acceptable whenever the disposed of activity of radionuclides were lower than its RC. The RC values allow overcome, for instance, uncertainties or variations on the expected radiological inventory.

To determine the RC for each radionuclide in a repository, it is necessary to assess the limitation of activity which would result from the WAC, using the total volume of waste planned to be disposed of at this site to compute total activity values, and then to compare these values with the AMAs. The Radiological Capacity results obtained for the enlarged NRWR at the current Mochovce site are included in Table VI.

Table VI: Radiological capacity of the enlarged NRWR at the Mochovce site

NUCLIDE	Admissible Maximum Activity [Bq]	Activity from WAC [Bq]	Radiological Capacity [Bq]	Limiting Condition
C-14	**2.71E+11**	6.81E+13	3E+11	Total activity
Ca-41	3.42E+14	**6.81E+13**	NND	Acceptance criteria
Ni-59	2.20E+15	**6.81E+13**	NND	Acceptance criteria
Ni-63	1.24E+15	**6.81E+13**	NND	Acceptance criteria
Se-79	8.28E+13	**6.81E+13**	NND	Acceptance criteria
Sr-90	**3.19E+13**	6.81E+13	3E+13	Total activity
Mo-93	1.38E+14	**6.81E+13**	NND	Acceptance criteria
Zr-93	**2.25E+13**	6.81E+13	2E+13	Total activity
Nb-94	1.84E+12	**6.81E+11**	NND	Acceptance criteria
Tc-99	**6.91E+12**	6.81E+13	7E+12	Total activity
Pd-107	9.54E+14	**6.81E+13**	NND	Acceptance criteria
Sn-126	1.51E+13	**6.81E+10**	NND	Acceptance criteria
I-129	**4.99E+12**	6.81E+13	5E+12	Total activity
Cs-135	1.15E+14	**6.81E+13**	NND	Acceptance criteria
Cs-137	1.28E+15	**6.81E+12**	NND	Acceptance criteria
Sm-151	2.41E+14	**6.81E+13**	NND	Acceptance criteria
Pu-238	**1.03E+10**	6.81E+12	1E+10	Total activity
Pu-239	**1.65E+09**	6.81E+11	2E+09	Total activity
Am-241	**7.44E+08**	6.81E+12	7E+08	Total activity

The last step of the safety assessment is the comparison with the projected inventory to be disposed of at the enlarged NRWR Mochovce in order to evaluate whether the considered repository design is appropriate to cover the disposal needs.

Considering the activity limits, either resulting as total activity limits derived from the long-term safety analysis (in Bq), or due to total activity limits derived from the SAL (in Bq/g), and taking into account the total volume of VLLW to be disposed of at the NRWR according to the design (68,052 m^3) and the mean density of this waste (assumed to be 1000 g/m^3), the ratio between the *expected radiological inventory* (ERI) and the mentioned *total activity limits* (TAL) applicable to this site is shown in Table VII:

Table VII: Ratio of Expected Radiological Inventory (ERI) to Total Activity Limit (TAL) for the NRWR Mochovce

VLLW Inventory		Ratio ERI/TAL for NRWR Mochovce
NUCLIDE	**Total Activity [Bq]**	
C-14	**2.10E+10**	0.078
Ca-41	**2.60E+08**	0.000001
Tc99	**4.50E+07**	0.000007
Ni-59	**1.30E+09**	0.000001
Ni-63	**4.40E+11**	0.00035
Cs-137	**1.90E+12**	0.0015
Nb-94	**1.80E+10**	0.0098
Sr-90	**1.50E+09**	0.000047
I-129	**1.70E+08**	0.000034
Pu-239	**3.70E+08**	0.22
Pu-238	**9.00E+08**	0.087
Am-241	**3.80E+09**	5.11

SUMMARY AND CONCLUSIONS

A comprehensive safety assessment was carried out in the framework of the feasibility study for the planned enlargement of the National Radioactive Waste Repository Mochovce in the Slovak Republic. A new category of VLLW was introduced and the corresponding waste acceptance criteria were developed.

As this information was not previously available, an assessment of the waste expected to arise from operation and decommissioning of the existing and planned NPPs in Slovakia was carried out. The expected waste inventory was then assigned to the two categories of LILW and VLLW. For the LILW it is planned to continue with the current practice of disposal in engineered vaults the primary waste packed into FCCs, the void space between FCC and waste being filled with mortar. For the VLLW a variety of packages can be used, and there is no requirement to immobilize solid waste.

Based on the expected waste quantities the space needed for disposal was estimated. All the forecasted volume of waste can be accommodated inside the current boundaries of the NRWR Mochovce site, and this variant of enlargement at the current site appears to be the most favorable one for a variety of reasons.

For the enlarged NRWR at the Mochovce site subsequently the radiological capacity was estimated. This is the maximum activity that the site can accommodate while still fulfilling the radiological constraints. The analysis showed that all the radionuclides present in the estimated inventory can be disposed of at the site with the exemption of Am-241. This specific radionuclide needs a more stringent SAL than originally estimated.

The tight limit for Americium results from the so-called bathtub scenario, in which the calculated radiation dose caused by Am-241 exceeds the dose constraint of 0.1 mSv/a. As for all the other scenarios the acceptable activity for this radionuclide is much higher, it is worthwhile to consider whether simple engineered measures can eliminate this particular effect. Alternatively, the Am-241-containing waste will have to be assigned to the LILW category and be disposed of with enhanced engineered barriers in the LILW vaults.