BG0100151

SAFETY ASSESSMENT OF NOVI HAN RADIOACTIVE WASTE REPOSITORY FEATURES, PROBLEMS, RESULTS AND PERSPECTIVES

Mayia Mateeva

Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences

SUMMARY

This paper summarizes the work done and the achievements reached in the Novi Han radioactive waste repository safety assessment within the International Atomic Energy Agency Model Project "Increasing the Safety of Novi Han Radioactive Waste repository BUL4/005". The overall Safety Assessment has a wide context, but the work reported here relates only to the some details and results concerning the development and implementation of the appropriate methodology approach, model and computer code used for the calculations. Different steps and procedures are included for a better practical understanding of the obtained results during the safety assessment performance. The methodology approach is widely based on an international experience in safety analysis and implemented for evaluation computer code AMBER, which is one of the recommended from the safety assessment experts.

1. INTRODUCTION

The general aim of the safety assessment performance is to present an appropriate and applicable approach for safety assessment of a near surface repository for low and intermediate level radioactive waste in Novi Han. Application of the methodology should demonstrate how the Novi Han disposal facility would perform as a permanent facility. What is the environmental impact of the repository? Is the disposal facility is really safe for the population living in the region?

The next aim is to assist the interpretation of available data and to identify additional data needs concerning the repository and the repository site and also to demonstrate a framework which repository can conduct safety performance assessments, and which may be applicable to further repository performance safety assessments in Bulgaria.

The testing of the methodology and results of safety assessment could help guide any proposed site investigation and field sampling/monitoring programs by focusing the collation and analysis of existing data and the identification of important gaps in the data record. It can assist with the formulation of appropriate administrative and legislative controls by allowing the quantitative modeling of the effects of possible practices and regulatory control.

The safety assessment can assist in the optimization and improvement of the repository with respect to environmental and reducing of the radiological risk for the population living in the repository region.

The methodology and its demonstration for safety assessments of Novi Han disposal facility can allow for systematic consideration of possible effects, processes and events. It can also be used to choose the full range of parameters values to determine the important features.

2. METHODOLOGICAL APPROACH

The chosen methodological approach is clear, comprehensive, well documented and defensible. The safety assessment process has been devised incorporating a few general steps, which can provide a systematic, simple and flexible procedure for collection, reviewing interpretation and assessment of the repository site data and environmental impact. These steps schematic are shown in Figure 1.

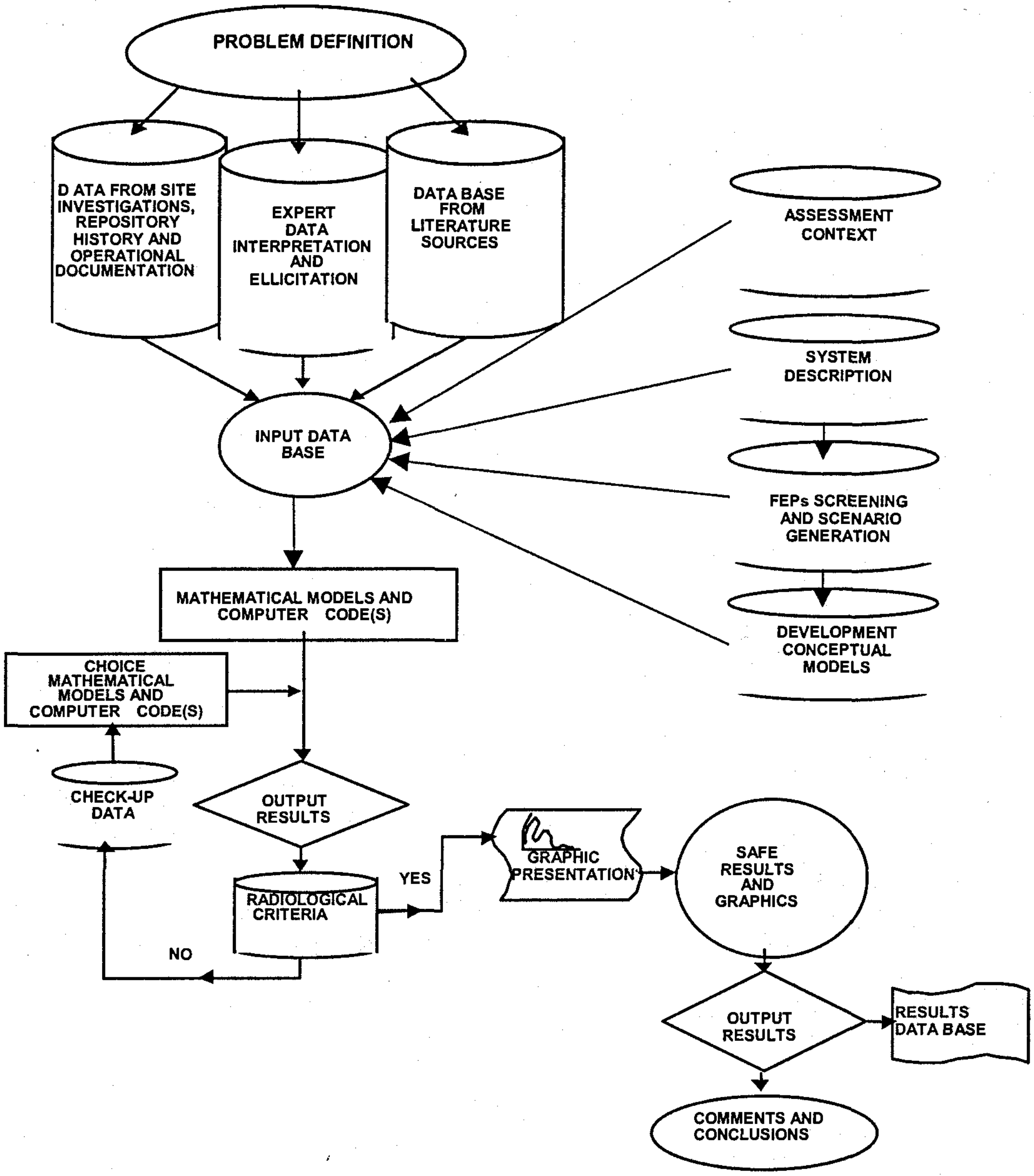

Figure 1. Information structure of the methodological approach

The key components used in the Novi Han safety assessment are recommended by the IAEA Coordinated Research Program for Improving long term Safety Assessment Methodologies for near surface radioactive waste disposal facilities (ISAM) [1] are:

- Assessment context;
- System description;
- Development and justification of scenarios;
- Formulation and implementation of models;
- Interpretation of the results.

An assessment context defines the scope and content of the analysis and provides information concerning the aims of performance assessment, radiological criteria, definition of the end points, regulatory requirements and exclusions, time scales of concern and data availability.

In relations of the system description this safety assessment consists of a full description of the disposal system (disposal facilities, type, state and radionuclide inventory of waste, available geological and hydrological situation and other specific parameters about geochemistry, social situation, prognosis of future actions of the society, habits and etc.).

The scenario development step has undertaken in conform to the concepts and approaches being developed for the ISAM program and other existing documents related to the definition of the Process system and the relevant features, processes and events (FEPs). This approach has been applied for the safety assessment of Novi Han repository for the first time. The general aim is to improve and help for better understanding of the role and procedures of generating a set of scenarios. This step also likely to be very important for the future analysts that wish to extend and supplement the analyses presented in this report.

The process of formulation and model implementation is presented in accordance with the concepts and assumptions taken in the scenario generation step. It is necessary to note that this procedure has been made at one more detailed and practical level. The actually conduct calculations with these models are produced.

The result of the consequence analyses is a set of numerical values that is ease to compare with the safety criteria.

Some scenarios have been omitted from these analyses because of insufficient specific information about some element of the whole system, or they are not so significant for the Novi han repository situation.

At finally in the revision has collected and revised models, scenarios, assumptions and parameters appropriately. The following conduction of the safety assessment using revisions has been able to determine their effect on the overall system safety.

The preliminary safety assessment results and very preliminary screening comparisons of the Novi Han inventory with internationally acceptable concentration limits for near surface repositories has been reported [2] previously. This safety assessment iteration was therefore the first site specific assessment with the real data about the radionuclide inventory of the disposal system.

As radiological protection criteria was used internationally accepted effective dose restriction of 1mSv/yr, excluding natural background radiation and medical procedures.

For the purpose of this safety assessment was proposed that annual individual effective dose to a number of a hypothetical critical group to be used as a primary measure of impact. Calculated doses can not be seen as predictions of future impacts, especially over the long time scales of interest. It is clear that missing or inadequate data needed for safety assessment is a normal situation. Where was necessary, existing site-specific data was supplemented by generic data available from the literature but it is very important that the study is able to identify the most important uncertainties and suggest ways in which they can be addressed during subsequent iterations of the safety assessment.

3. DISPOSAL SYSTEM DESCRIPTION

The part of this safety assessment that is related with the system description presents a lot of information concerning three general categories:

- Disposal facilities;
- Geological and hydrological situation;
- Biosphere.

Some characteristics about the disposal facilities and radionuclide inventory are shortly presented here. The Novi Han repository consists of disposal vaults for solid, biological and spent sources, one concrete trench for solid radioactive waste and four liquid waste storage tanks. They are multibarrier engineer facilities constructed from reinforced concrete covered outside and inside with bitumen insulation and the internal surface is covered also with stainless steel, and the external walls are mechanically stabilized with brick wall.

The type of radioactive waste and radionuclides are various as it is shown in the Figure2.

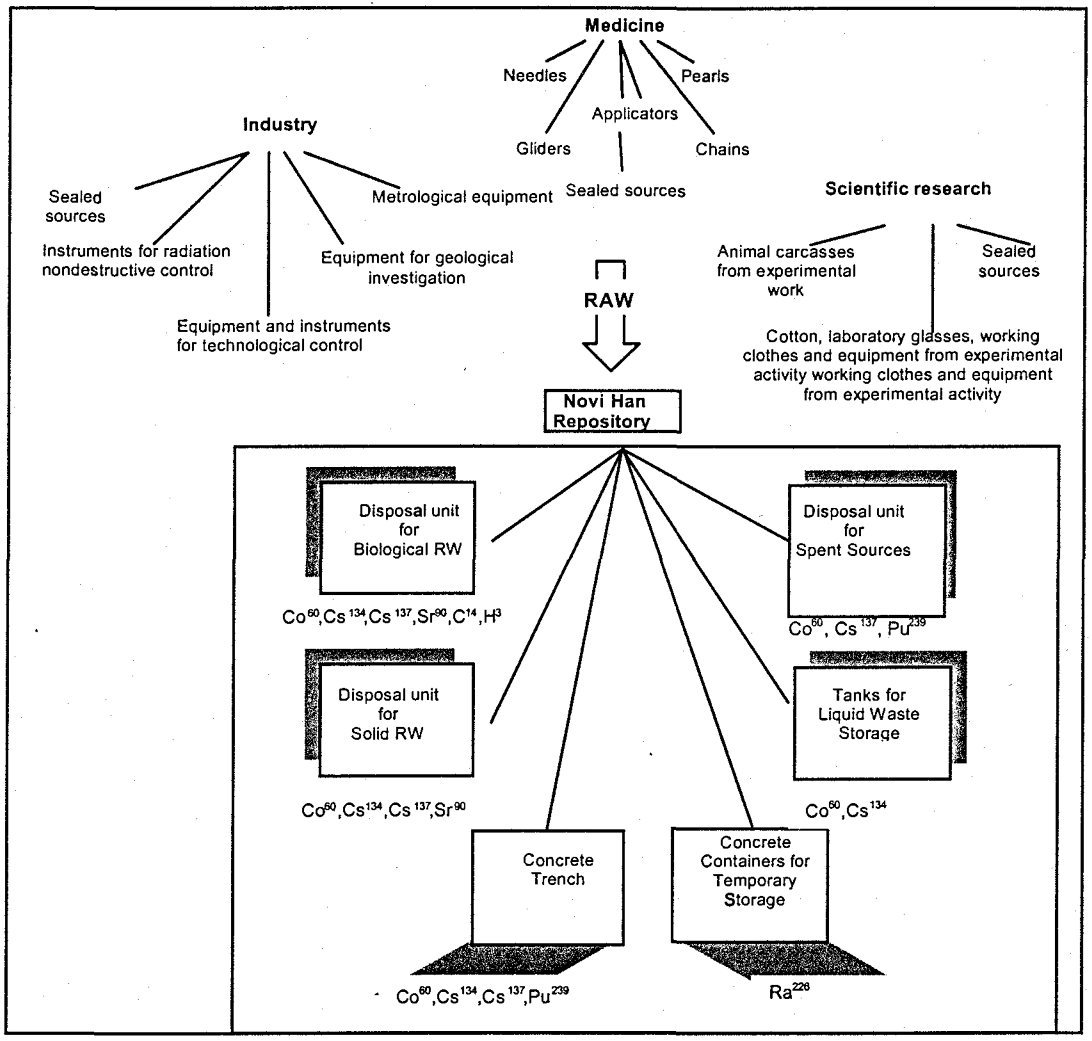

Figure 2. Structure of the repository disposal units and organization of the radioactive waste separation according to their type and activity

The used data concerning the chosen key radionuclide inventory in [Bq] in accordance with the repository documentation is:

- Disposal Unit for Solid Waste - ^{137}Cs - $22.76 \cdot 10^{10}$; ^{60}Co - $23.19 \cdot 10^{8}$; ^{90}Sr-$3.96 \cdot 10^{10}$;
- Disposal Unit for Biological Waste -^{137}Cs -$5.96 \cdot 10^{10}$; ^{60}Co - $2.23 \cdot 10^{8}$; ^{90}Sr-$9.32 \cdot 10^{9}$; ^{14}C -$1.084 \cdot 10^{11}$; ^{3}H -$9.5 \cdot 10^{9}$;
- Disposal Unit for Spent Sources -^{137}Cs -$2.73 \cdot 10^{13}$; ^{60}Co - $1.45 \cdot 10^{11}$; ^{239}Pu - $1.85 \cdot 10^{11}$; ^{226}Ra -$5.92 \cdot 10^{11}$; ^{241}Am -$2.268 \cdot 10^{10}$;
- Concrete Trench - ^{137}Cs - $4.14 \cdot 10^{11}$; ^{60}Co - $0.93 \cdot 10^{10}$; ^{239}Pu -$6.89 \cdot 10^{5}$;

All data related to the geological and hydrological situation are taken from the investigations carried out from the Geological Institute [3] and unavailable from the literature sources. For this safety assessment was assumed that the groundwater path from the facilities can follow a route through the weathered surface zone (unsaturated) and then through the phyllite-shist (saturated zone). The carefully study of the geografical and hydrological information about Novi Han repository place show that there is no evidence for observed hazardous atmospheric phenoma.

The available data concerning the geochemical situation suggests that the vaults be situated in relatively aggressive environment in relation to the concrete base, because the shallow ground water is of sulfate-hydrocarbonate-sodium-calcium type with total salt content 0,136 g/l and pH – 6,8.

There was not available data about sorption behavior and solubility limits in this safety assessment was used the generic literature data [4], [5], [6],[7].

The Biosphere is considered as temperate. However for the purposes of this assessment it was assumed that constant conditions and habits are maintained.

This safety assessment involve in consideration alternative patterns of future land use such as:

- Extension of the Novi Han village in the vicinity of the repository site;
- Development of a small family farming in the vicinity of the repository site;
- Development of individual farming practices at the site.

Further extension of the village near the disposal facility will increase the likelihood of intrusion event if the memory of the repository is lost.

4. SCENARIO DEVELOPMENT AND JUSTIFICATIONS

In this safety assessment scenarios are considered as a set of naturally occurring, human-induced, and waste- and repository – induced events and processes that represent realistic future changes to the repository, geologic and hydrologic systems that may affect the escape and transport of radionuclides from the disposal facility [8]. The procedure to identify scenarios defined in this way consists of the next steps:

- Compilation, adoption, or adaptation of a comprehensive list of events and processes that could affect the performance of the disposal system;
- Classification of the events and processes to aid in addressing completeness arguments;
- Screening events and processes to identify those that can be eliminated;
- Development a comprehensive and mutually exclusive scenarios by combining the events and processes that remain after screening;
- Screening scenarios to identify those of them that will have no bearing on the performance measure, so that consequence – modeling efforts can be focused on those that do it;
- Arranging scenarios that are selected into calculation cases for the safety assessment.

Within this safety assessment was done in details the identification of events and processes, development a comprehensive list of events and processes and their classification. The screening relevant FEPs and criteria was produced too. On this base was constructed the scenario analysis. The aim of this procedure is to construct an appropriate and flexible set of scenarios by combining and linking the events and processes, which survive the screening process. After this procedure were chosen the following scenarios that no intent to be comprehensive, not even to be mutually exclusive but they were generated to illustrate the consequences of key selected FEPs acting on the system:

- Leaching Scenario – SCE1 off-site scenario;

- Pu capsules Scenario – SCE2;
- Construction and Residence Scenario – SCE3 on site scenario.

The SCE1 scenario refers to off-site situations in the sense that the critical group is mainly located outside the disposal facility. For this assessment SCE1 was selected to represent a reasonable future evaluation of the disposal system and its surroundings. The following conditions were chosen for modeling:

- The institutional period and social memory of the site and sanitary zone is maintained for 300 years, after which the assumption that the memory of the existence of the facility is lost is made;
- The assumption that after the institutional control period, individual homes or cottages are built close to the repository site is made;
- It was assumed that individual homeowners should supplement their food intake from gardens, and rising a limited number of livestock. In addition is assumed that the individual gardens and cattle feed may become contaminated from contact with irrigation water from the spring supplemented by the aquifer;
- No credit for additional disposal of waste occurs at the facility;
- It is also assumed that no activity is removed;
- No physical disruption of the disposal structures by building activities.

Scenario SCE1 corresponds to using of contaminated water in the biosphere compartment at the interface with the geosphere, after leaching and migration of radionuclides outside of the repository through the geosphere. The interface between the geosphere and biosphere is spring intercepting the radioactive plume in the geosphere downstream the disposal was assumed.

The SCE2 scenario is selected because the Pu spent sources are disposed incorporated in ceramic capsules. There is not almost any data about structural and geochemical behavior of these capsules. It is clear that unaltered capsules should influence on the Pu migration from the repository and it was assumed that the sorption in the near field should be lower than is assumed in SCE1 scenario (no capsules considered). This scenario was chosen in accordance with the assessment context, because the vault for spent sources contains the high activity of Pu^{239} and also other long-lived radionuclides. Because of many assumptions made in the model this scenario is important to give some preliminary answers and proposals for the future of this disposal facility.

ScenarioSCE3 is intrusion scenario that refers to on site situation – house construction and residence. The wastes are considered totally degraded, i.e. readily for multiple exposure if unearthed. It was assumed that as result of house construction the radioactive waste are excavated and distributed around the surface. Agricultural and farming activities are assumed to take place in the contaminated soil. Doses are calculated as result of external exposure, inhalation and ingestion. This scenario was divided in two parts as SCE3a scenario – construction where doses are calculated for a worker engaged in construction activities and SCE3b scenario – residence, where doses are calculated for an individual residence as result of ingestion, inhalation and external irradiation.

Scenario-generating FEPs that have been deferred for possible later consideration are presented below:

- Drilling Scenario;
- Subsidence Scenario;
- Bath-tubing Scenario;
- Biotic Intrusion Scenario;
- Erosion Scenario.

This safety analysis recommended that these scenarios should be considered in next assessment iterations only if there is a significant argument that they are likely to be important for decisions that associated with the disposal facility future.

Having chosen a relevant set of scenarios it is necessary to sort them and to present a range of possible combinations of exposure pathways and critical group assumptions that illustrate particular concerns about the facility. These assumptions are summarized in Table1.

Table1. Summary of pathways and critical group assumptions for the chosen scenarios

Scenario	Drinking Water	External Irradiation	Crop Consumption	Animal Product Consumption	Inhalation
SCE 1	x	x	x	x	x
SCE 2	x	x	x	x	x
SCE 3a		x			x
SCE 3б		x	x	x	x

5. MODEL FORMULATION AND IMPLEMENTATION

According to the developed approach the chosen scenarios must be organized into a form that is amenable to mathematical representation. A set of model-level assumptions (about dimensionally, features, events, processes, etc.) is needed for each scenario. These assumptions comprise the conceptual model. The conceptual for each scenario is then expressed in mathematical form as a group of algebraic and differential equations, which then need to be solved. These equations may be empirically and/or physically based, depending upon the level of understanding and available information concerning the processes. These equations and their associated parameters form the basis of the mathematical models. Solution of the mathematical models is usually achieved by implementing one or more computer tools using analytic and/or numerical techniques. For this safety assessment was used the re-running of deterministic tools with different conceptual and mathematical models and parameter values. It is important to note that in each case and scenario conceptual model identifies:

- The contaminant release media and mechanisms;
- The contaminant transport media and mechanisms;
- The primary and secondary biosphere receptors;
- The human exposure mechanisms.

Mathematical models by which the radiological impact of the scenarios can be quantitatively assessed are related to:

- The source term modeling as a means to evaluate the amount of radioactivity which is leached from the disposal and subsequently enters the geosphere;
- The geosphere modeling through the expression of the delay time, produced by the unsaturated zone, and through the advection-dispersion equation which governs the overall transport;
- The dose calculations with respect to the assessment of the internal and external exposures for the small farm system, construction of the house and residence on soil mixed with waste.

Models in this safety assessment are implemented in the computer code AMBER [9] that is a flexible software tool and allows the user to build compartment models to represent the migration and rate of contaminants in a system. AMBER overcomes key limitations associated with most existing compartment codes giving the user the flexibility to define:

- Any number of compartments;
- Any number of contaminants and associated decays;
- Any number of transfers between compartments;
- Algebraic expressions to represent physical processes operating between compartments;
- Algebraic expressions to represent the uptake of contaminants by humans and other output quantities of interest;
- Deterministic, probabilistic and time varying parameter values.

For this safety assessment AMBER was used for all calculations. For the on-site scenarios the computer code was used primarily as a spreadsheet to calculate doses (and associated disposal concentrations and amounts) for the relevant pathways. For the off-site scenarios it was used to simulate the migration of radionuclides from the disposal facility into the geosphere and thence into the biosphere by representing the system as a series of compartments linked by the transfer expressions.

The geosphere was sub-divided into several compartments and AMBER was used to provide an approximate solution to the 1D version of the 2D advection-dispersion equation. Advection was represented as an advective flux from the upstream to downstream compartments. Specifying two diffusive/dispersive exchanges between the upstream to downstream compartments represented dispersion. It was assumed to ignore the transverse dispersion and was made a conservative assumption that the concentration in the spring was derived from the leaching of waste across the entire disposal facility.

6. CONSEQUENCE ANALYSIS AND RESULTS

An initial set of calculations was conducted on each scenario using "high-end" values of parameters. This set of parameters is used as parameters producing the highest dose results. The primary parameters that have been investigated for the current safety assessment are:

- Sorption coefficients for near and far field;
- Aquifer hydraulic conductivity;
- Aquifer hydraulic gradient.

High-end parameters are considered as a combination of high conductivity and gradient and minimally values for sorption coefficients. Additionally the calculations with low-end parameters were conducted. This set of parameters refers to calculations involving low hydraulic conductivity and gradient and maximally values for the sorption coefficients.

Results obtained from the calculation with high-end set of parameters are summarized in Table 2 for whole repository concerning Leaching Scenario - SCE 1. Safety assessment results about individual disposal units are discussed below. Calculations with low-end set of parameters show lowest values of doses and no are presented here.

Table 2 Summaries of the Results from SCE1 – Whole Repository

	SCE 1 – Leaching Scenario, Peak Doses [Sv/y] and High-End Set of Parameters								
	Dtot	Dexternal	Ddust	Dingestion					
Whole Novi Han Repository	$^{90}Sr-4.8\cdot10^{-7}$ $^{3}H-2.04\cdot10^{-8}$ $^{14}C-1.47\cdot10^{-5}$ $^{239}Pu-1.6\cdot10^{-4}$ $^{231}Pa-2.5\cdot10^{-7}$ $^{226}Ra-5.2\cdot10^{-10}$ $^{235}U-2.2\cdot10^{-7}$ $^{227}Ac-4.41\cdot10^{-7}$ Pu>C>Sr>Ac> Pa>U>H>Ra	$^{239}Pu-1.4\cdot10^{-9}$ $^{235}U-9.9\cdot10^{-10}$ $^{227}Ac-6.8\cdot10^{-10}$ $^{231}Pa-1.14\cdot10^{-10}$ $^{90}Sr-2.23\cdot10^{-11}$ Pu>U>Ac>Pa>Sr	$^{239}Pu-9.6\cdot10^{-7}$ $^{227}Ac-5.8\cdot10^{-10}$ $^{231}Pa-8.3\cdot10^{-11}$ $^{235}U-1.9\cdot10^{-11}$ $^{14}C-1.3\cdot10^{-12}$ Pu>Ac>Pa>U>C	$^{239}Pu-9.6\cdot10^{-7}$, $^{14}C-1.3\cdot10^{-12}$, $^{90}Sr-4.8\cdot10^{-7}$, $^{227}Ac-5.8\cdot10^{-10}$, $^{235}U-1.9\cdot10^{-11}$ Pu>C>Sr>Ac>U					
				Drinking Water	Grain	Green vegetables	Root vegetables	meat	milk
				$^{239}Pu-9.6\cdot10^{-7}$ $^{14}C-1.3\cdot10^{-12}$ $^{90}Sr-4.8\cdot10^{-7}$ $^{227}Ac-5.8\cdot10^{-10}$ $^{235}U-1.9\cdot10^{-11}$ Pu>C>Sr>Ac>U	$^{239}Pu-1.3\cdot10^{-4}$ $^{14}C-1.2\cdot10^{-5}$ $^{90}Sr-3.5\cdot10^{-7}$ $^{227}Ac-3.1\cdot10^{-7}$ $^{235}U-1.6\cdot10^{-7}$ Pu>C>Sr>Ac>U	$^{239}Pu-6.99\cdot10^{-6}$ $^{14}C-8.1\cdot10^{-7}$ $^{90}Sr-2.2\cdot10^{-8}$ $^{227}Ac-1.8\cdot10^{-8}$ $^{235}U-8.9\cdot10^{-9}$ Pu>C>Sr>Ac>U	$^{239}Pu-2.3\cdot10^{-5}$ $^{14}C-2.4\cdot10^{-6}$ $^{90}Sr-6.4\cdot10^{-8}$ $^{227}Ac-5.7\cdot10^{-8}$ $^{235}U-2.9\cdot10^{-8}$ Pu>C>Sr>Ac>U	$^{14}C-5.7\cdot10^{-7}$ $^{90}Sr-3.2\cdot10^{-9}$ $^{239}Pu-1.7\cdot10^{-9}$ $^{227}Ac-2.8\cdot10^{-11}$ $^{235}U-1.7\cdot10^{-11}$ C>Sr>Pu>Ac>U	$^{14}C-1.5\cdot10^{-7}$ 90Sr-$3.9\cdot10^{-9}$ $^{235}U-7.6\cdot10^{-9}$ $^{239}Pu-6.5\cdot10^{-10}$ $^{231}Pa-4.8\cdot10^{-12}$ C>Sr>U>Pu>Pa

Results About Disposal Unit for Biological Waste

The total dose has two peaks. An early peak occurs in 80 years – $2{,}04 \cdot 10^{-8}$ Sv/y, followed by a second peak -$1{,}47 \cdot 10^{-5}$ Sv/y that occurs at 7.10^{3} years. Analyses of the peaks show that the early peak is the result of ^{3}H and ^{90}Sr. These radionuclides are released and rapidly transported owing to the assumptions of poor cover performance, high aquifer flow and low values of sorption coefficients. The second peak is due almost entirely to ^{14}C. Ingestion pathways dominate both of these peaks. The key contributors are consumption of crops in particular grain and root vegetables and contaminated drinking water. The doses from dust inhalation (contributed by ^{14}C) and external (contributed by ^{90}Sr) are less than 10^{-12} Sv/y. It is clear that the peak doses do not represent excessive dose for people.

Results About Disposal Unit for Solid Waste

The peak of the total dose occurs in early 200 years of the institutional control as result of ^{90}Sr - $4{,}2 \cdot 10^{-7}$ Sv/y and ^{134}Cs - $0{,}49 \cdot 10^{-20}$ Sv/y. The key contributors are ingestion pathways – consumption of crops (peak doses from grain ^{90}Sr - $3{,}0 \cdot 10^{-7}$ Sv/y and from root vegetables ^{90}Sr - $5{,}5 \cdot 10^{-8}$ Sv/y and drinking water ^{90}Sr - $2{,}43 \cdot 10^{-8}$). The doses from dust inhalation (contributed by ^{90}Sr – $7{,}5 \cdot 10^{-14}$ Sv/y) and external (contributed by ^{90}Sr - $1{,}8 \cdot 10^{-11}$ Sv/y). The peak doses do not represent excessive dose for people.

Results About Concrete Trench

A peak of the total dose occurs at around of 30000 years – $1{,}06 \cdot 10^{-7}$ Sv/y as result of ^{239}Pu. Analyses of the peak show that the ingestion pathways are dominated in particular consumption of grain – $8{,}06 \cdot 10^{-8}$ Sv/y, root vegetables - $1{,}46 \cdot 10^{-8}$ Sv/y and drinking water $6{,}42 \cdot 10^{-9}$ Sv/y. This peak occurs a long time after institutional control, but is much less than accepted radiological criteria. The dose from dust inhalation (^{239}Pu) is about 10^{-10} Sv/y and dose from external irradiation is smaller.

Results About Disposal Unit for Spent Sources

Results obtained from the calculation show that the peak total dose occurs at 10^{+5} years – $1{,}57 \cdot 10^{-5}$ Sv/y. The key contributor is ^{239}Pu and ingestion pathways are main contributors – consumption of grain – $4{,}12 \cdot 10^{-5}$ Sv/y, root vegetables - $7{,}57 \cdot 10^{-6}$ Sv/y, green vegetables - $2{,}3 \cdot 10^{-6}$ Sv/y and drinking water - $3{,}29 \cdot 10^{-6}$ Sv/y. This peak is greatest in comparison with the other disposal units but the main reason is the highest activity of the disposed ^{239}Pu. Nevertheless the peak dose does not represent a high potential dose to people.

SCE2 – Pu capsules Scenario

Calculation for SCE 2 Scenario that have been produced only for the disposal unit for spent sources, because the highest activity of disposed ^{239}Pu incorporated in ceramic capsules. The results show that the peak doses values from ^{239}Pu are lowest than in SCE1 scenario for the both cases with high-end set of parameters - $5{,}93 \cdot 10^{-6}$ Sv/y and with a low-end of parameters - $1{,}03 \cdot 10^{-11}$Sv/y. The result are reasonable, because in this scenario has been assumed an additional credit for the Pu release in near field – higher sorption coefficient in the disposal facility is used.

SCE3 - Construction and Residence Scenario

The results obtained from calculation of intrusion scenarios Scenario 3a – excavation and construction and Scenario 3b – residence are summarized in Table 3. Results have been reported in terms of annual individual dose. For each disposal unit the key contributors to the total dose are identified, as are the key pathways - inhalation, ingestion and external irradiation.

Table 3. Summarized Results of the Peak Doses [Sv/y], Radionuclides and Pathways for Intrusion Scenarios

	Scenario 3a - Construction	Scenario 3б - Residence
Disposal Unit for Biological Waste	Peak Dose – $5.0\cdot10^{-2}$ $^{137}Cs>^{60}Co>^{90}Sr$ ^{137}Cs – inhalation ^{137}Cs, 60Co - external irradiation	Peak Dose -$1.2\cdot10^{-3}$ $^{14}C>^{90}Sr>^{137}Cs$ ^{14}C – ingestion ^{14}C – dust inhalation ^{137}Cs - external irradiation
Disposal Unit for Solid Waste	Peak Dose – 0.11 $^{137}Cs>^{60}Co>^{90}Sr$ ^{137}Cs, ^{90}Sr – inhalation ^{137}Cs - external irradiation	Peak Dose -$3.14\cdot10^{-7}$ $^{90}Sr>^{137}Cs>^{60}Co$ ^{90}Sr – ingestion ^{137}Cs – dust inhalation ^{137}Cs - external irradiation
Concrete Trench	Peak Dose – 0.12 $^{137}Cs>^{60}Co>^{239}Pu$ ^{137}Cs, ^{239}Pu– inhalation ^{137}Cs, ^{60}Co - external irradiation	Peak Dose -$2.7\cdot10^{-7}$ $^{137}Cs>^{239}Pu>^{60}Co$ ^{137}Cs, ^{60}Co– ingestion ^{239}Pu – dust inhalation ^{137}Cs - external irradiation
Disposal Unit for Spent Sources	Peak Dose – 9.5 $^{137}Cs>^{60}Co>^{239}Pu>^{226}Ra$ ^{239}Pu, ^{226}Ra, ^{210}Pb– inhalation ^{137}Cs, ^{60}Co, ^{226}Ra, - external irradiation	Peak Dose -0.10 $^{226}Ra>^{239}Pu>^{210}Pb$ ^{226}Ra, ^{210}Pb– ingestion ^{239}Pu– dust inhalation ^{226}Ra- external irradiation

7. INTERPRETATION OF RESULTS

It is necessary to interpret results because the consequence analysis does not exactly show the conclusions and recommendations.

There is conservatism in the assessment in that no credit is taken for the other engineered barriers, except the concrete. The assessment also assumed that all cesium present is ^{137}Cs whereas a mixture of ^{137}Cs and ^{134}Cs is known to be present. It is necessary to note that more detailed knowledge of the inventory at the site in particular about ^{3}H, ^{226} Ra and ^{14}C could improve the assessment.

Also it is necessary to obtain a more specific data about geochemical behavior and the effects from the interactions between ^{239}Pu and ceramic material of capsules.

In this safety assessment was used a simple, generic, biosphere model, which is considered to be conservative. However, an understanding of the local biosphere characteristics and evolution of potential future habits in the repository region would allow a more representative biosphere model to be constructed.

The safety assessment results obtained for the SCE1 and SCE2 scenarios are lowest than the accepted radiological criteria. These results may be interpreted as a condition that at the current time the repository does not represent a potential risk for population living in the region and using water from the spring supplied from the aquifer. It is necessary to add that the strong control preventing the access in the repository site exists now but we need of decisions from the competent national authorities on the post closure period.

As human intrusion has been identified as the most important pathway for potential radiation exposures at the site a more comprehensive evaluation of appropriate site specific exposure scenario and the possibility to minimize the radiological risk by varying the institutional control must be discussed in future work.

The doses arising from intrusion scenarios at the site indicate that in the absence of institutional control, intrusion into the spent sources disposal vault could result in high doses, generally result from ^{137}Cs over the next 300 years for the excavating worker. That is means that effective management control of the repository site is needed during this period. The dose due from ^{239}Pu is also higher than accepted radiological criteria and this dose is almost constant for a long time scale. The removal of the spent sources from the site would reduce the risk from intrusion, but the doses for the individuals which will do that are very high and are contributed generally from the external irradiation.

The results obtained from calculation of Residence scenario after 500 years – assumed intrusion time show that the individual doses are lower than accepted criteria for all disposal vaults, except the disposal unit for spent sources.

These result suggest that the regulators in Bulgaria have to consider the situation and take a decisions for justifications the national legislation with the relevant time scale institutional control of the Novi Han radioactive waste repository.

REFERENCES

[1] International Atomic Energy Agency, ISAM, The International Program for Improving Long Term Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities: Objectives, Content and Work Program, Final version, IAEA, Vienna, (1997)

[2] Mateeva M., Preliminary Safety Assessment of Novi Han Radioactive Waste Repository IAEA Technical Co-operation Project BUL/4/005 – report

[3] Report – geological and hydrological conditions of the Novi Han repository site "Инженерно-геоложки и хидрогеоложки условия на хранилището за радиоактивни отпадъци в Лозенската планина (Нови хан)" – Отчет за извънпланова договорна задача с ИЯИЯЕ при БАН за периода от 1 април 1991 до 1 септември 1991г, БАН – Лаборатория по Геотехника, София, 1991

[4] Stenhouse M.J., "Sortion databases for Crystalline, marl and Bentonite for performance Assessment, NAGRA, July 1995

[5] EPRI, "Yucca Mountain Total System Performance Assessment," Phase 3",Electric Power Research Institute, TR-107191, Palo Alto, 1996

[6] Brady P.V., and M.W.Kozak, "Geochemical Engineering of Low-level Radioactive Waste in Cementitious Environments', waste Management, Vol.15, pp , 293-391,1995

[7] Krupka K.M., and R.J.Feeney, "Effects of Radionuclide concentrations by Cement/ Groundwater Iteractions in support of performance assessment of low-level radioactive waste Disposal Facilities', NUREG/CR-6377, U.S.Nuclear Regulatory Commission, 1998

[8] Cranwell R.M., Guzowski R.V., Campbell J.E., and Ortiz N.R., 1990, 'Risk Methodology gor Geologic Disposl, of radioactive Waste; Scenario selection procedure, "NUREG/CR-1667, SAND80-1429, Sandia National Laboratory

[9] Andy Brice, AMBER 3.1 – reference Guide, September 1996