THE DEVELOPMENT OF A SAFETY ASSESSMENT APPROACH AND ITS IMPLICATION ON THE ADVANCED NUCLEAR FUEL CYCLE

YONGSOO HWANG and CHUL-HYUNG KANG*
Korea Atomic Energy Research Institute
1045 Daedeok-daero, Yuseong, Daejeon, 305-352, Korea
*Corresponding author. E-mail : chkang@kaeri.re.kr

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The development of Advanced Nuclear Fuel Cycle (ANFC) technology is essential to meet the national mission for energy independence via a nuclear option in Korea. The action target is to develop environmentally friendly, cost-effective measures to reduce the burden of long term disposal. The proper scenarios regarding potential radionuclide release from a repository have been developed in this study based on the Advanced Korean Reference Disposal System (A-KRS). To predict safety for the various scenarios, a new assessment code based on the GoldSim software has also been developed. Deterministic analysis indicates an environmental benefit from the ANFC as long as the solid wastes from the ANFC act as a proper barrier.

KEYWORDS : Long Term Post Closure Safety Assessment, Pyro-processing, Advanced Nuclear Fuel Cycle, GoldSim, Reference Scenario, What if Scenario

1. INTRODUCTION

Until 2007, KAERI developed the Korean Reference Disposal System KRS [1], a generic repository concept for the direct disposal of spent nuclear fuel (SNF) in crystalline rock in Korea and its associated safety assessment [2-3]. From 2007, the goal has shifted to development of the Advanced Korean Reference Disposal System (A-KRS) [4], for the permanent disposal of low-level radioactive waste (LLW) and high-level radioactive waste (HLW) from the ANFC, illustrated in Figure 1. ANFC adopts three components of pyro-processing: electrolytic reduction, electro-winning, and electro-refining. The major concerns over the ANFC in terms of waste disposal are as follows:
(1) Reduction of an underground repository area,
(2) Reduction of annual individual doses, and
(3) Reduction of a controlled time period.

The development of the A-KRS promises a significant reduction of the repository area due to the removal of decay heat from a certain number of fission products such as Sr-90 and Cs-137. This should be a great help in the disposal of radioactive wastes in countries like Korea which suffer from a limited supply of high quality massive host rock. The second and third targets can be reviewed throughout intensive studies involving long term radiological post closure safety analysis. The safety is related with inventories, dissolution rates of waste forms, the soundness of engineered and natural barriers, and the characteristics of a biosphere.

Therefore, it is necessary to develop a certain series of scenarios and the corresponding mathematical and computational tools to evaluate those scenarios, along with a proper dataset. In this paper, the detailed approaches are summarized to understand the benefit of a certain fuel cycle concept as follows:
(1) Characteristics of radioactive wastes from the ANFC,
(2) Fundamental concepts of the A-KRS,
(3) Reference and alternative scenarios for safety assessment,
(4) Software development for safety assessment, and finally
(5) Deterministic safety assessment.

2. ADVANCED NUCLEAR FUEL CYCLE CONCEPT AND ITS WASTE ARISING

Waste from each step of the ANFC is generated, as shown in Figure 1. The recovery rates for specific nuclides at each step have not yet been finalized. However, based on the target recovery rates and associated waste generation rates as depicted in Figure 2, the inventories of radionuclides at a specific step of the ANFC are estimated. As shown in Figure 2, six waste streams are by-products of the ANFC. Subsequent reclassification produces five different streams which are ultimately disposed of into four different facilities and their corresponding inventories.

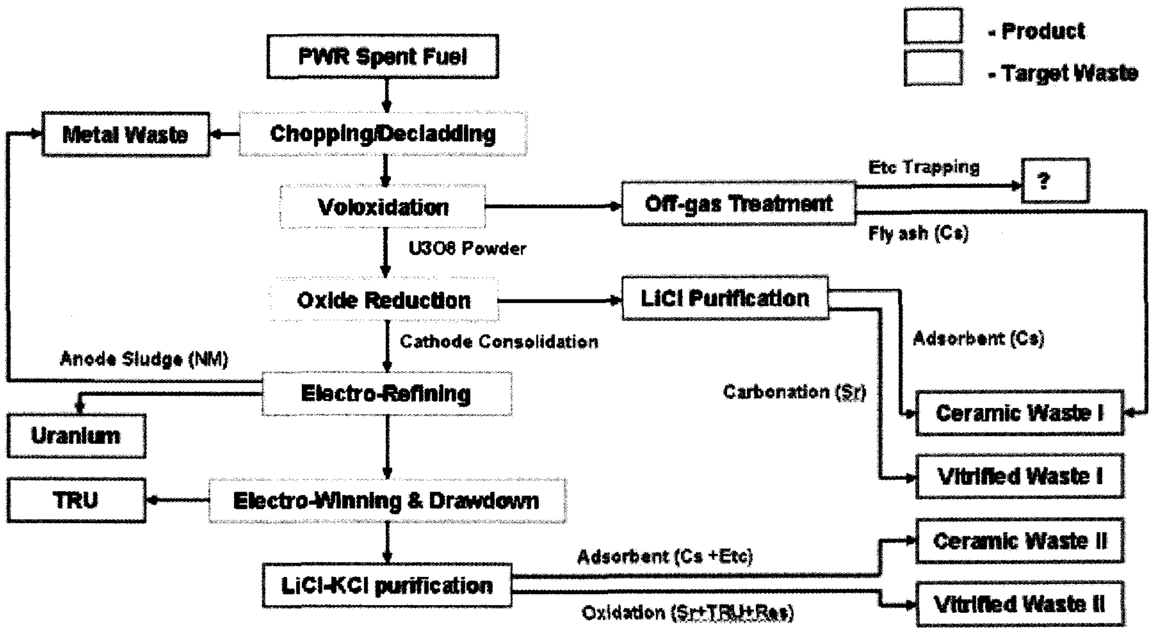

Fig. 1. Major Processes and Waste Arising in the ANFC

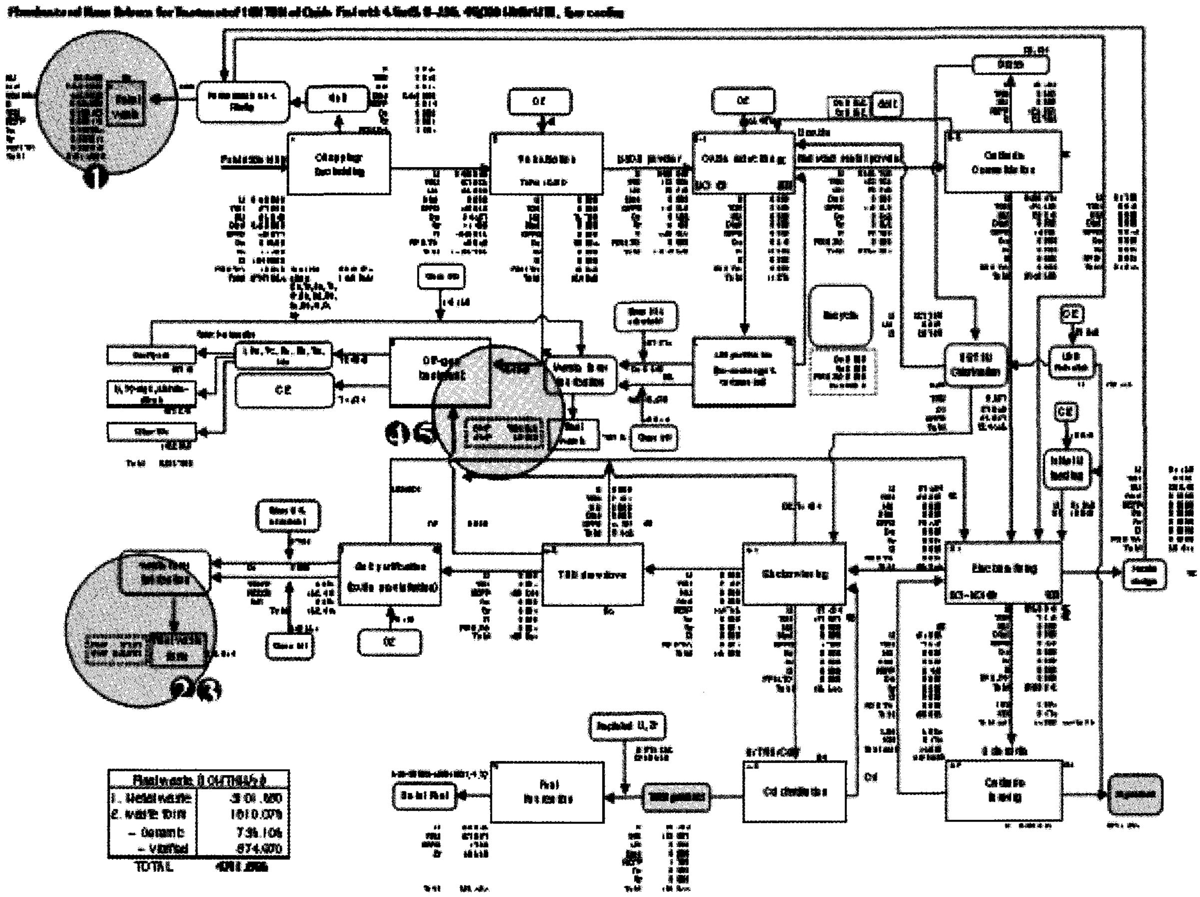

Fig. 2. Pyro-Process Material Balance

3. CURRENT DISPOSAL CONCEPT

The disposal concept of the A-KRS for radioactive wastes from the ANFC is under development. This paper estimates the safety of the A-KRS, which was developed at the end of 2008. The inventory information is based on a study that was completed in the spring of 2009. The current A-KRS has two floor repositories, one at the traditional 500 meter depth and the other rather shallow, at a 200 meter depth. Figures 3 to 6 illustrate the disposal of different waste streams from the ANFC: metals, vitrified HLW and intermediate-level radioactive waste (ILW), and ceramic LLW [4].

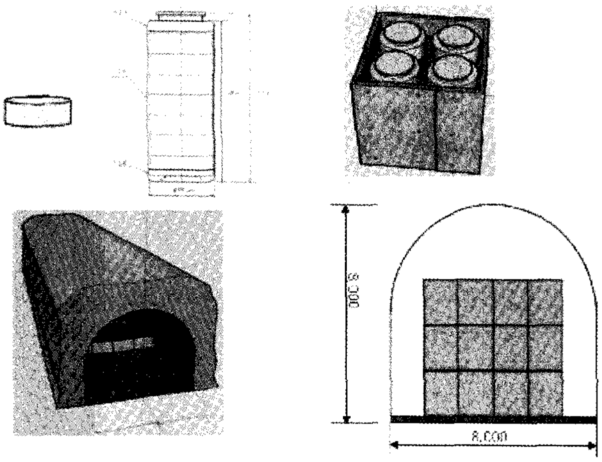

Fig. 3. Metal Waste Disposal Concept

4. SCENARIO DEVELOPMENT

The purpose of the scenario development and associated assessment code development is not just for the assessment of the ANFC. It aims at the option study of many prominent back-end fuel cycles such as direct disposal, conventional wet reprocessing, and the ANFC. The major distinctions among various scenarios are in the differences of inventories and the solid waste forms. Many others are identical for different waste streams. Various disposal technologies also create differences. Three types of disposal technologies are considered in the scenario and code development: KBS-3V that relies on the vertical borehole concept, the horizontal repository concept that is quite similar to the KBS-3H, and the silo concept that has already been implemented in a certain number of LLW repositories around the world.

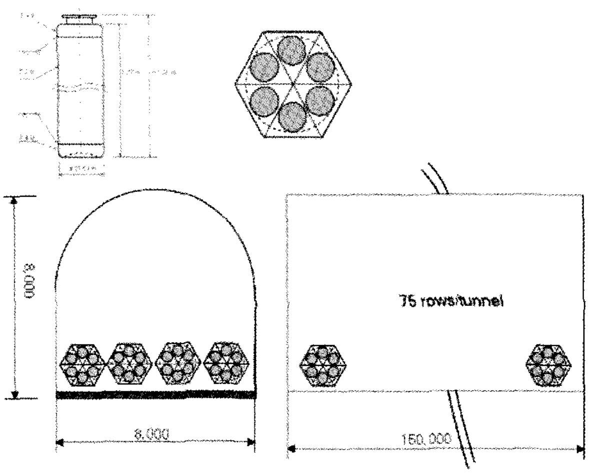

Fig. 5. Ceramic LLW Disposal Concept

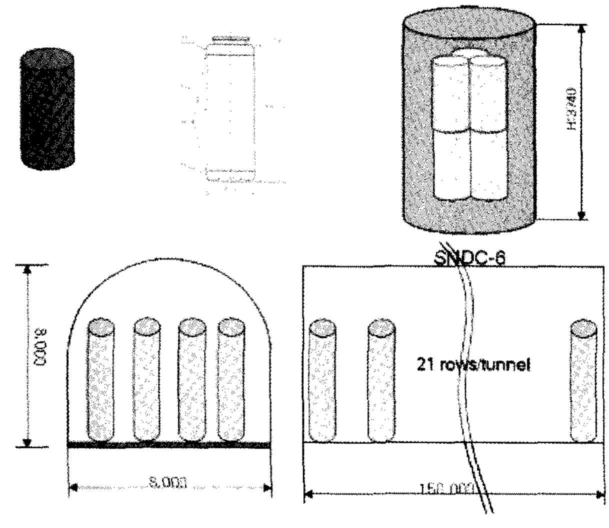

Fig. 4. Vitrified HLW Disposal Concept

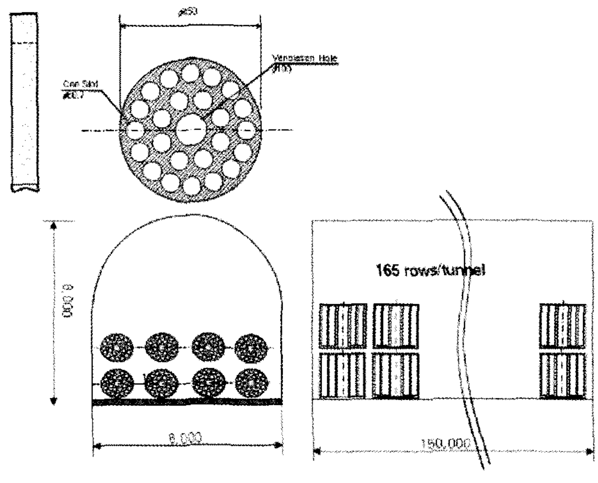

Fig. 6. Vitrified ILW Disposal Concept

4.1 Reference Scenario

There is no significant difference between the KRS and the A-KRS. The only difference is that for the A-KRS, there are two repository floors at 200 meter and 500 meter depths. The shallow floor is designed to host LLW by the construction of a silo or a tunnel. The deeper one will take care of HLW by the construction of a tunnel. When following the KRS for direct disposal, a waste container will be emplaced in a borehole surrounded by bentonite layers.

After emplacement of a waste container, intruding groundwater begins to corrode a waste container, eventually dissolves the solid radioactive materials, and it starts to transport them to a natural barrier by way of the bentonite layers and the surrounding excavated disturbed zones (EDZs). The thickness of the EDZs depends on the excavation methods: TBM or controlled blast. It is assumed to be 30 cm for the TBM method and 1 meter for the controlled blast method. In this scenario development, the average lifetime of a waste container is assumed to be 1,000 years. However, to be conservative, the initial failure of a waste container is allowed at a fraction of 0.1% of all containers to be disposed of. The effective area for the transport of radioactive materials between an EDZ and the adjacent natural rock is a key to quantifying the safety. In this study, for example, the effective transport area for a silo is set to be the cross sectional area of the silo multiplied by the frequency of a fracture along the silo height. Radionuclides entering a fracture migrate by advective and dispersive transport with matrix diffusion into an adjacent rock. In this study, the potential back diffusion is neglected. After passing through a fracture, radioactive materials flow into a major water conducting feature (MWCF) which eventually meets with the biosphere. For generic repository conditions, farming, fresh water fishing, and salt water fishing groups are affected by a repository. Detailed biosphere modeling is developed to predict the mass in each compartment such as soil layers, a river, a marine environment, and others.

4.2 What if Scenarios

4.2.1 Well Scenario

The well scenario is the most probable alternative scenario for a potential repository in Korea. In the future, residents near a repository are expected to drill a deep borehole to extract groundwater for drinking, irrigation, and other domestic uses. The groundwater contaminated by released radioactive wastes is assumed to be diluted by fresh groundwater coming into a borehole. For simplicity, the deep contaminated groundwater is assumed to flow into a well. The shallow fresh water then dilutes the contaminated water. The solution is applied to assess the volumetric flows from the two regions. The extracted water affects the daily life of residents. The modified farming group concept is applied for detailed calculations.

4.2.2 Earthquake Scenario

The earthquake scenario is a less probable scenario in Korea, where the geologic configuration is stable. However, if an earthquake occurs, its impact on the safety is significant. Two potential consequences happen as a result of an earthquake, both of which result in the enhancement of the groundwater velocity:

- the widening of a fracture aperture and
- the creation of a new pathway by a strong earthquake.

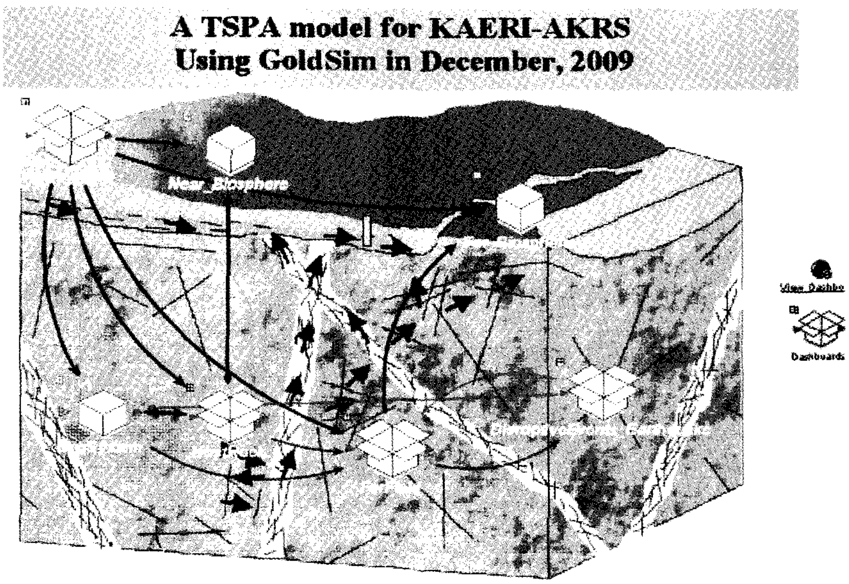

Fig. 7. Overview of the Transport Code

4.2.3 Early Failure of EBS

In normal conditions, functions of an engineered barrier system (EBS) are expected to last for a fairly long time. However, a certain EBS may experience the early loss of its function. In such a case, the EBS is no longer effective as a barrier.

5. ASSESSMENT CODE DEVELOPMENT

To develop the long term post closure radiological safety assessment code for potential repositories such as the A-KRS and KRS, GoldSim software [5], which is the main total system performance assessment (TSPA) code development tool for the Yucca Mountain Project (YMP) and many European cases, is used. For the code development, the following items are considered as top priorities:

(1) To develop a hierarchy of "Containers" in GoldSim to describe key physical barriers such as engineered, natural, and biosphere barriers,
(2) To develop a GoldSim container for an alternative scenario assessment linked with proper containers of the reference scenario assessment,
(3) To create an input data container which includes data globally used in the model and to set up localized data in each container, and
(4) To develop a graphic user interface "Dashboard".

Here the "Container" is a folder system, and the "Dashboard" is a graphic interface used in GoldSim. New code called Transport was developed for the assessment, as illustrated in Figure 7.

5.1 Engineered Barrier

Engineered barriers, such as vertical and horizontal emplacement, and silos, are modeled according to their disposal mechanisms. Potential release of radionuclides from a waste form is classified into four mechanisms:

- Congruent release,
- Constant release fraction,
- Constant release fraction per unit area, and
- Instantaneous release fraction (IRF).

Detailed modeling approaches are listed in the following section.

5.2 Natural Barrier

The natural barriers are modeled using the concepts of a porous medium, a fracture surrounded by a rock, and a fracture network. In this model development, a fracture element is used to represent a deep crystalline rock and a porous medium element is applied to describe an MWCF which connects a deep fracture system with a biosphere.

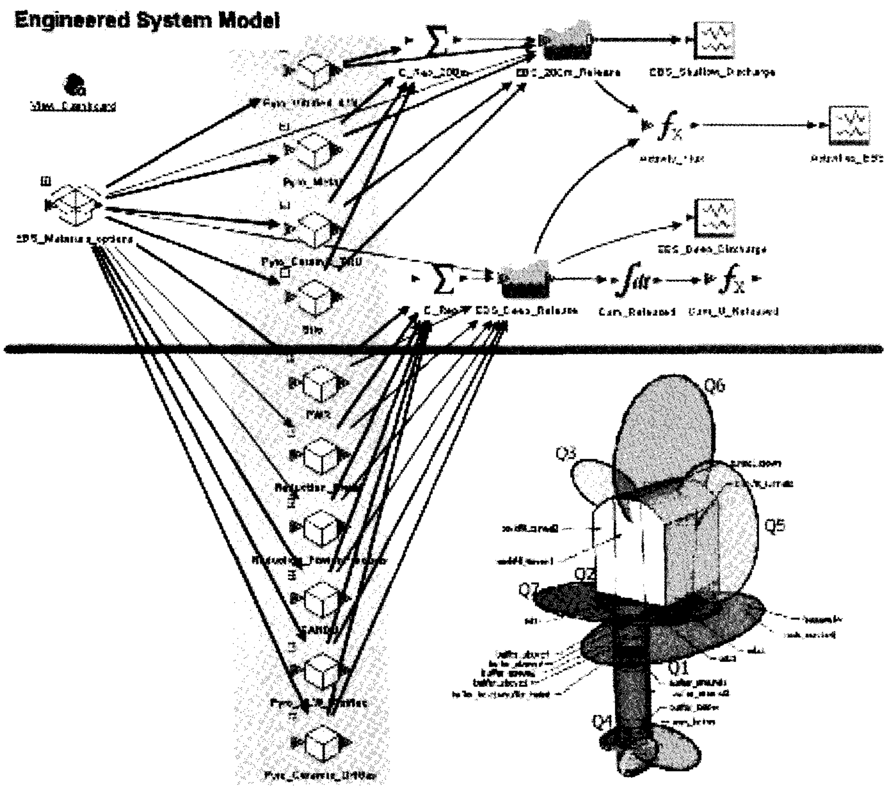

Fig. 8. Schematic View of an EBS for Different Waste Stream

5.3 Biosphere

To avoid the difficulty of the land transport of a cask to a repository, a generic repository is assumed to be located at a coastal area. Therefore, three major recipient groups are identified for an analysis:

- A farming group
- A marine water fishing group, and
- A fresh water fishing group.

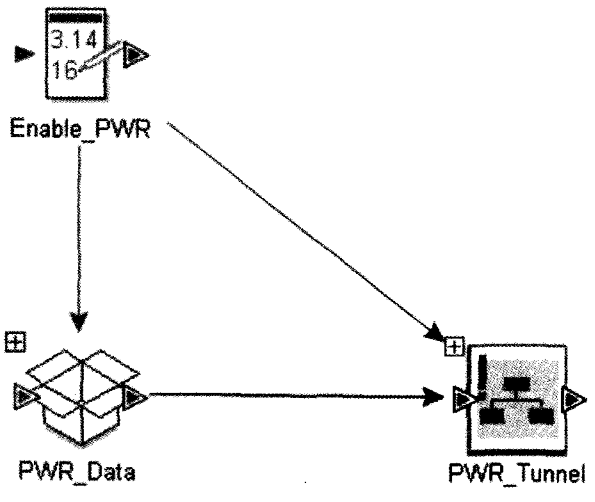

Fig. 9. Special Selector to Simulate the PWR SNF Direct Disposal

6. DETAILED MODELING

The detailed code development focuses on near field modeling. First, when the code is developed for radionuclide transport using GoldSim, it is necessary to develop a

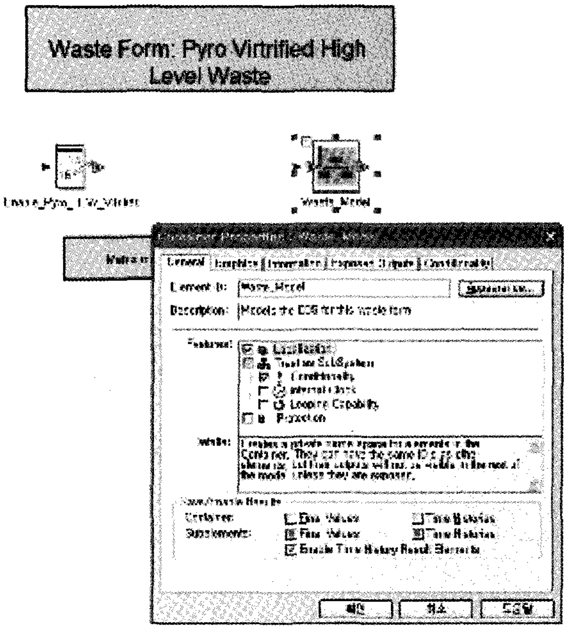

Fig. 10. Special Selector to Simulate the ANFC

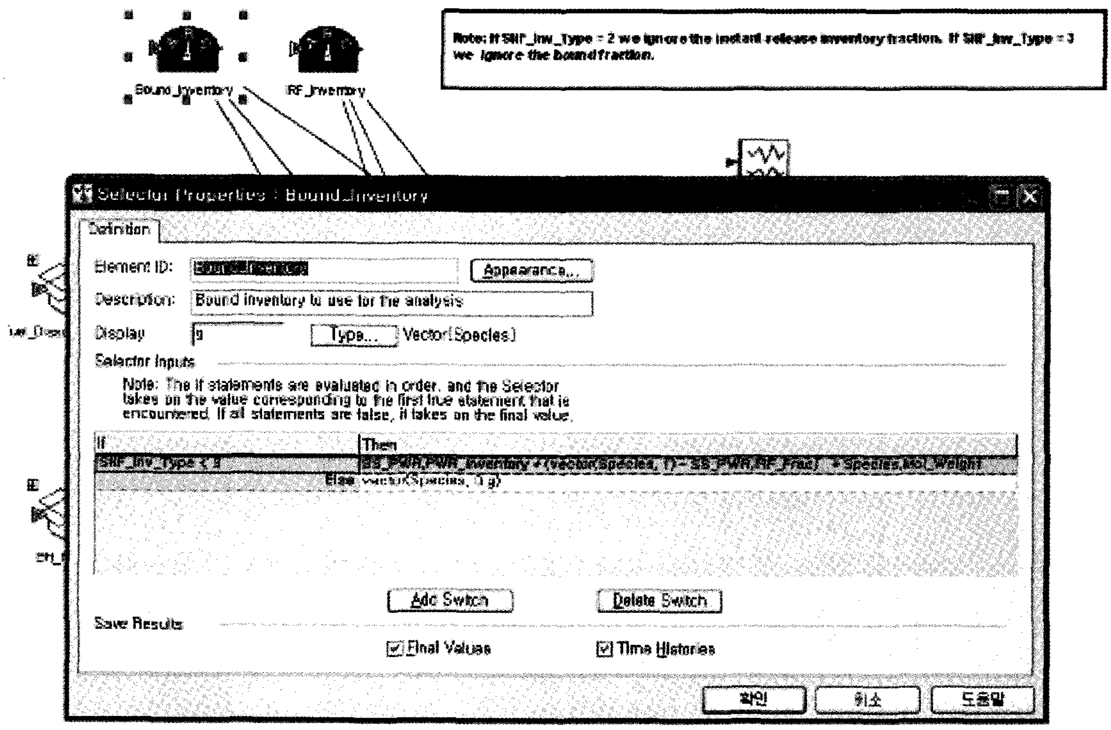

Fig. 11. Module for Congruent Release

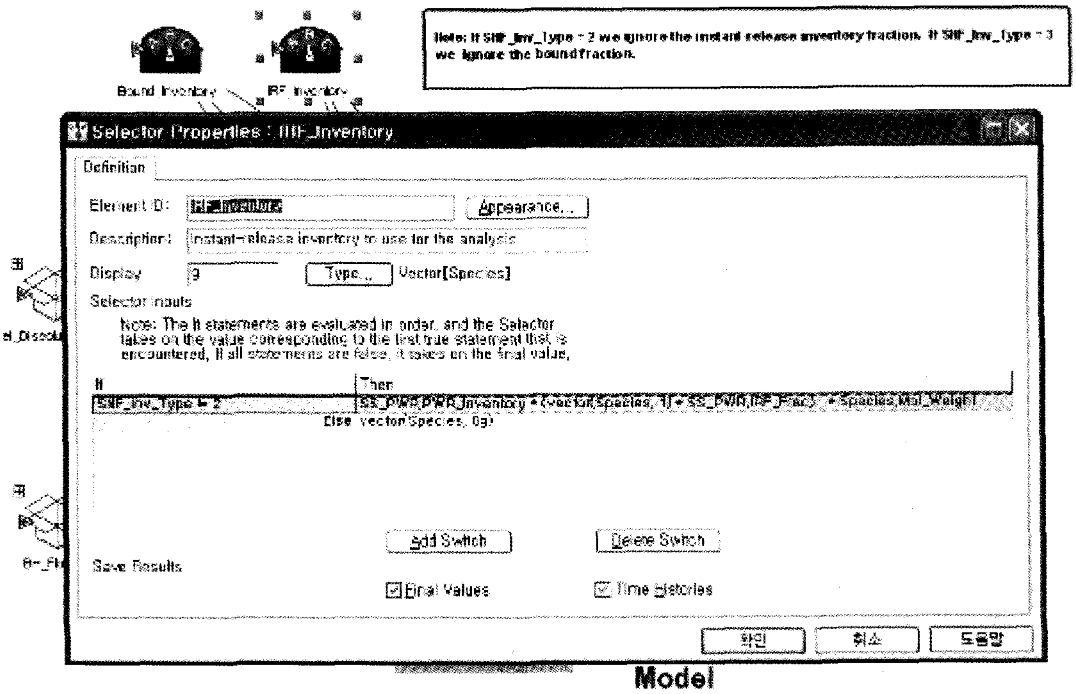

Fig. 12. Module for Instantaneous Release

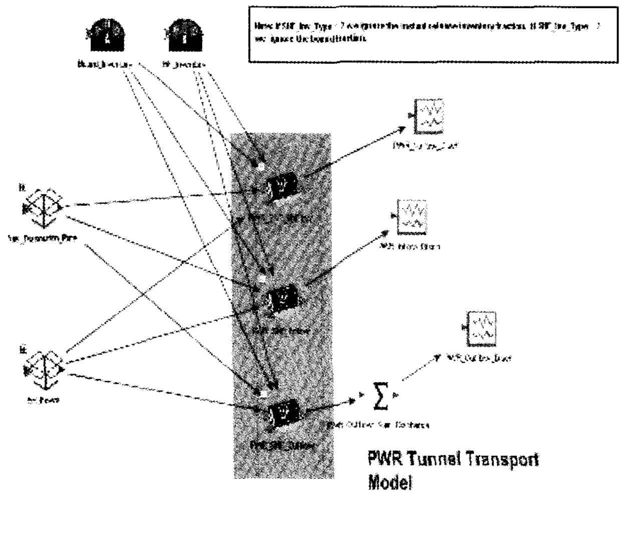

Fig. 13. Groundwater Flow Conditions

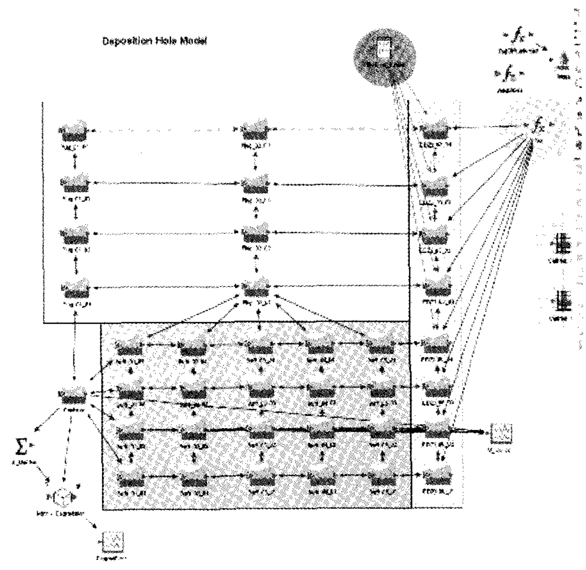

Fig. 14. Detailed Compartment for Vertical Emplacement

container, EBS_Materials. It includes information required to represent material properties globally used to describe an EBS.

Second, for each type of repository, vertical and horizontal emplacement, or silos, detailed features of the emplacement are modeled to simulate the potential release of radionuclides in an EBS, as illustrated in Figure 8.

Third, types of waste for assessment are selected by the special "Selector" function of GoldSim, as depicted in Figures 9 and 10. By this function, the Transport code assesses the safety of direct disposal or the ANFC.

Fourth, proper programming to model the dissolution of a radionuclide from a waste form is needed. The Transport code was developed to simulate four different release mechanisms: a congruent release as shown in

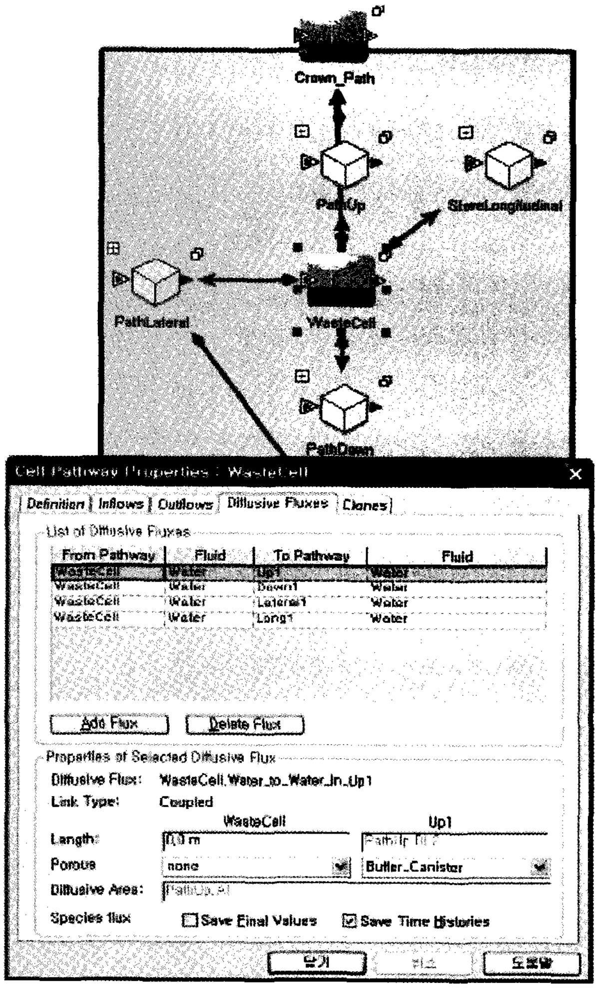

Fig. 15. Detailed Compartment for Horizontal Emplacement

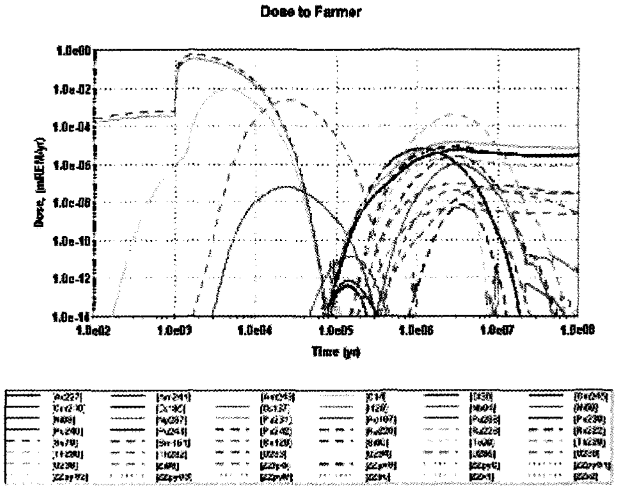

Fig. 16. Annual Individual Dose by Direct Disposal

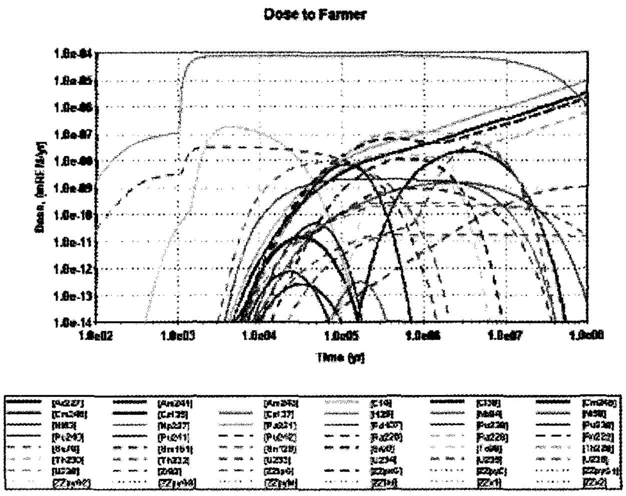

Fig. 17. Annual Individual Dose from Pyro-Process Waste with the Congruent Release

Figure 11, an instantaneous release for SNF as illustrated in Figure 12, a constant release rate per unit waste form and a constant release rate per unit mass.

Fifth, the groundwater migration condition is modeled in three different ways, as depicted in Figure 13: incoming condition, out-bounding condition and diffusion dominant case. Based on the real repository condition the proper option is to be chosen.

Finally, detailed modeling for the EBS compartments is done inside the Transport code. For example, for a vertical emplacement as demonstrated in Figure 14, the introduction of specific compartments for bentonite layers, EDZ, and the backfilled tunnel is pursued. For a horizontal emplacement, modeling of the upward, downward, horizontal and lateral compartments is performed as shown in Figure 15.

7. DETERMINISTIC SAFETY ASSESSMENT

The inventory used for the assessment is based on the SNF from 24 PWRs to be constructed by 2016 whose enrichment, burn-up, and cooling time after discharge are 4.5%, 4,500 MWD/MTU, and 5 years, respectively. This means that 26,000 MTU will be processed via pyro-processing. All details related to this data are from a related research report [4]. Figure 16 illustrates the annual individual dose from direct disposal. The maximum peak occurs due to the higher release of certain fission products such as I-129 that follows instantaneous release. After a long time since emplacement, TRUs affect the doses due to their long half lives and high sorption coefficients.

The effect of pyro-processing on disposal comes from two factors, the removal of TRUs and the dissolution rate of a waste form. Eventually, most of TRUs separated from pyro-processing are used for the manufacture of Sodium

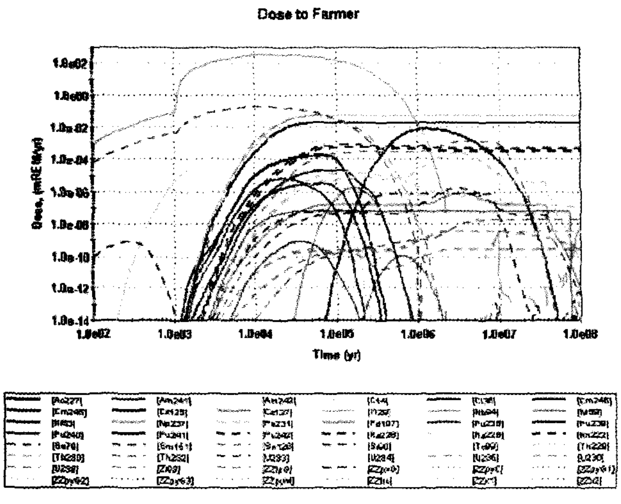

Fig. 18. Annual Individual Dose from Pyro-Process Waste with the Dissolution Rate of 10^{-5} 1/yr

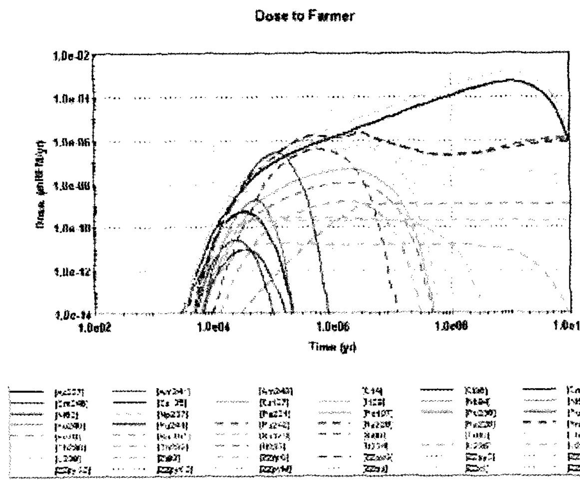

Fig. 20. Annual Individual Dose from Pyro-Process Metal Waste with the Dissolution Rate of 10^{-12} 1/yr

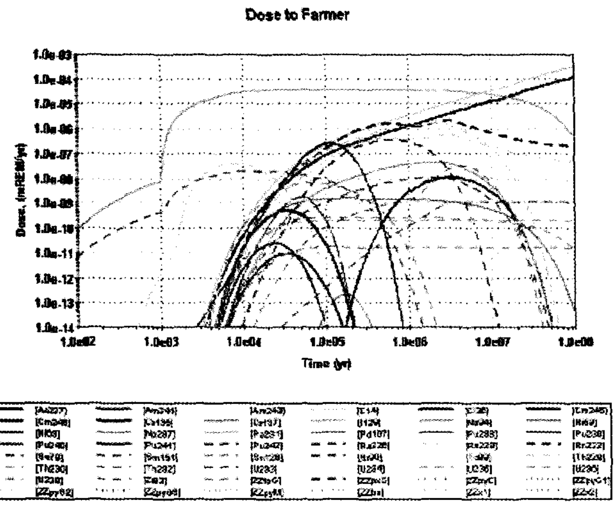

Fig. 19. Annual Individual Dose from Pyro-Process Waste with the Dissolution Rate of 10^{-12} 1/yr

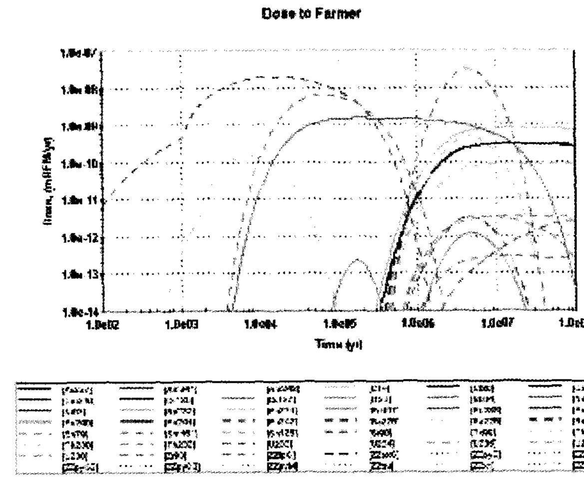

Fig. 21. Annual Individual Dose from Pyro-Process Vitrified HLW with the Dissolution Rate of 10^{-12} 1/yr

Fast Reactor (SFR) fuels. If not, they are assumed to be disposed of. If disposed of, then like in the case of direct disposal, illustrated in Figure 17, after longtime emplacement, the effect of TRUs will create a certain impact to the biosphere. If all TRUs are recycled for SFRs, the concern over potential environmental impact after hundred thousand years since emplacement will be vanished.

The characteristics of dissolution rates for waste solid forms from pyro-processing are not known. If the dissolution rate is quite high, such as 10^{-5} 1/yr, its annual individual dose is quite high, as illustrated in Figure 18. However, when the dissolution rate of a waste form from pyro-processing is equal to the level of the congruent release rate for PWR SNF, 10^{-12} 1/yr, the contribution from the highly soluble fission products such as I-129 and Se-79, etc. significantly decreases, as shown in Figure 19. Those are two major contributions of pyro-processing in reducing potential doses for the public.

Figures 20-22 illustrate the annual individual doses for different waste forms from pyro-processing. As shown in Figure 20, the annual individual doses from metal waste are not negligible over a long period of time. This is due to the limitation in the current A-KRS design and the low dissolution rate. Once the current A-KRS is modified, the impact will be minimized. Figure 21 shows the annual individual doses from a vitrified HLW. As noted, the major contributors are Se-79 and a small portion of TRUs. If the dissolution rate remains at the same level as that from the congruent release for direct disposal, Se-79 will be mostly decayed out at a geologic medium so that its

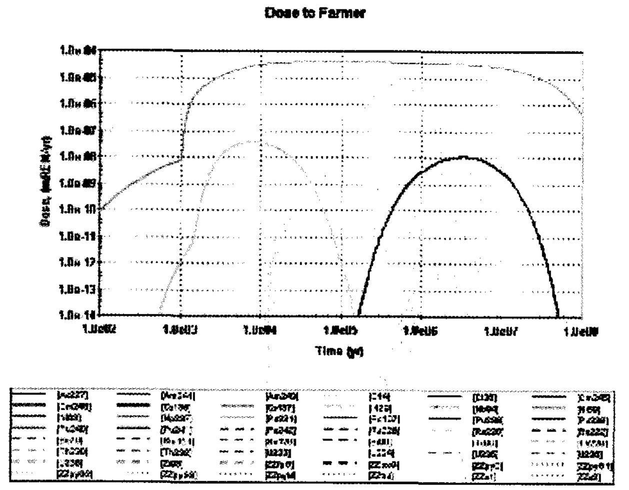

Fig. 22. Annual Individual Dose from Pyro-Process Ceramic ILW with the Dissolution Rate of 10^{-12} 1/yr

contribution to the biosphere will become negligible. Figure 22 represents the annual individual doses by ceramic waste whose contributors are I-129, Cs-137 and C-14. As previously pointed out, the dissolution rates are the key factors for the doses from this type of waste.

8. CONCLUSIONS

In this summary paper, a systematic approach to assess the long term post closure radiological safety of a potential A-KRS repository emplaced in a crystalline rock is presented. To specifically assess the safety for the reference and alternative scenarios, a code called Transport is developed based on the GoldSim software. A deterministic study is performed and compared with that for direct disposal.

Results indicate if the TRUs are recycled and the dissolution rates of waste forms are rationally low, pyro-processing will give future environmental benefits.

Further studies on different repository design concepts and different dissolution mechanisms along with probabilistic safety assessment are recommended. In addition, code verification and validation is proposed for near-term research.

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