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Perspectives on Safety Assessment for a Potential HLW Disposal: Project SR-AKRS

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1. Introduction

The total system performance assessment for the Korean Repository System (KRS) concluded that the proposed disposal system meets the guideline of the annual individual dose rate at 10 mRem/yr for series of different scenarios to safety disposed of spent nuclear fuels from PWRs and CANDUs. The traditional probabilistic safety analysis with a certain limitation in both methodologies and data, is carried out for the study. Two main scenarios, a natural discharge and a small well one along with some alternative ones are assessed. The next phase research started from March in 2007 targets to extend the methodology development for the coupling processes, to analyze the pros and the cons of the advanced nuclear fuel cycle concepts such as the pyro process, to understand the phenomena at the geosphere and biosphere interface(GBI.), and to perform the qualify assurance over the entire safety analysis works. This new work, Project SR-AKRS, which stands for the Safety Report on the Advanced Korean Repository System will focus on the following substantial R&D works:

2. Story Telling

The first task is to extend the story telling on the nuclide migration; FEPs and scenario development. KAERI developed the so called KAERI FEP encyclopedia suitable for the disposal of spent nuclear fuel. The FEPs will be extended to include the potential disposal of HLW as well as other wastes from the pyro process. Then the new approach to express the interactions between FEPs will be applied. So far the Rock Engineering System (RES) is applied to identify the interaction which creates events and processes among features. This approach is effective to overview the top scale linkage among features. However, detailed interactions cannot be explicitly described by this RES. The Process Influence Diagram(PID) approach will be added to explain the detailed reactions among FEPs. The PID will not be used to cover the whole processes due to its complexity. Instead it will be used to describe the interaction in detailed levels. To visually illustrate the linkage the current CYPRUS software will be extended.

3. Understanding the couplings on THMCRG

The thermo hydraulic mechanical chemical radiological and gaseous coupling is a big challenge to demonstrate the understanding the actual processes in a proposed disposal system. In the previous study phase, KAERI developed the multi-dimensional probabilistic safety assessment code, MDPSA to assess the groundwater and radionuclide transfer and transport in a fractured porous media. The code based on the control volume theory can simultaneously solve the migration of both groundwater and a radionuclide. In the Project SR-AKRS it will be extensively extended to assess the groundwater and radionuclide transfer and transport with potential salt intrusion under thermal effect caused by the decay of radionuclides. The whole work will be done step by step.

Firstly, the new module will be constructed to predict the time dependent thermal profiles in a repository system. The change in temperature affects the viscosity, the density of groundwater as well as the stresses which in turn alter the aperture distributions. In 2007 the first two effects will be studied followed by the additional study over the stress and strain analysis in 2008.

Secondly the potential intrusion of salt water into fresh groundwater will be studied. The denser sea water changes the distribution of fresh groundwater. The effect is usually significant near a repository which practically uplifts the groundwater pathways. It implies that the traveling distance and associated traveling time with sea water intrusion will be shortened quite significantly. This fresh and salt water non-linear interaction will be carefully studied in 2008 when the appropriate new module is added into the MDPSA code.

Thirdly chemical alteration will be studied in detail. The creation and transport of a pseudo-colloid in a fracture surrounded by a porous medium which allow the matrix diffusion only for a solute will be studied based on the existing study for a single member decay chain. The current mathematical solution for a single member decay with a constant inlet concentration boundary condition will be extended to include the effect of the infinite member decay chain and the arbitrary inlet boundary condition. When the concentration at the inlet boundary is prescribed then we can have a Duhamel type solution with a proper canonical form for a decay chain. For the arbitrary flux inlet boundary condition, the similar solution form can be derived in the form of the Laplace transformed one. Then the numerical Laplace transform based on either Talbot or Green will be used to quantify the numbers. The second subject on the chemical transport is the change of the solubility, a diffusion coefficient, and the distribution coefficient in geologic media spatially as well as transiently. The temperature distribution will alter the diffusion coefficient by Einstein law and the distribution coefficients at least in the theory. When alpha and gamma radiolysis is significant, it affects the solubility of uranium in a matrix to alter its form from UO2 to U3O8. The change of the UO2 phase affects the release of other nuclides congruently dissolved with the uranium matrix. The extent of a redox front is a key to identify the confined domain for this alteration process. Once the region under influence is estimated the different chemical batch can be applied for that specific region. This can be combined with the MDPSA so that for a given time domain at a certain spatial domain the different distribution coefficients are to be applied.

Fourthly the corrosion of a waste canister is predicted. The commercial code like FRACMAN is updated to predict the transport of corroding species from a far field to the surface of a waste container via a fracture network. KAERI will develop the similar approach based on its analytic and numerical analysis capabilities. The amounts of potential corrosion agents such as Sulphate, Chloride, and Oxygen at the ambient condition will be assessed from field data of the K-URT. Once the sources are identified then their transport from an ambient environment to a waste container surface via a fracture network and a bentonite buffer can be predicted by analytic approach, and MDPSA and CONNECTFLOW codes. Two conceptual models can be applied one based on the cubic law relation between the hydraulic conductivity and a fracture aperture and the other assuming no relationship between two factors. In both cases we can predict the amount of the corroding species at the surface. There either copper or iron forms the chemical compounds with intruding impurities. But still it is not quite enough to predict a canister life time limited by the corrosion. The chemical reaction discussed above is good for the uniform corrosion not for the pitting corrosion which is the major driver to generate a hole by faster local corrosion. The experimental correlation between the rates of two corrosion mechanisms is to be applied for the problem. With local data from the K-URT one can then assess the canister life time limited by corrosion.

Finally update on the numerical schemes in the MDPSA is essential. The first subject is to extend the numerical library for a matrix solver. Presently only PCCG is implemented in the code. The PCCG code is an efficient, fast and in most cases robust solver. However when encountered by stiff problems, it sometimes fails to work properly. In such a case a more robust routine is to be used. The direct solver will be constructed to handle such a case. The second subject is to express the topology more accurately. At this stage the orthodox control volume approach is used in the MDPSA. By nature it allows only a cubical element which makes rather difficult to express the mountainous surface areas. To solve the problem the local coordinate system is applied with full mass conservation at the concerned elements. The appropriate section of the code is modified to handle the problem. The third subject is to simulate the non-linear equation system. So far only linear systems can be solved using the MDPSA. However to solve the salt water intrusion, the additional non-linear system solver is to be introduced. Two numerical approaches Newton-Rapson and Picardo will be considered for this purposed.

4. Performance Assessment

New features will be analyzed to check the safety of the proposed A-KRS. In the SR-KRS study the uncertainty in groundwater migration will be significantly studied. The CONNECTFLOW the main numerical analysis code for groundwater flow can assess the probabilistic safety analysis for different hydraulic permeabilities. The MDPSA can handle the PSA for major input data. In addition the so called response surface method will be applied to see the effect of key parameter values. Once the information on the groundwater is given, then the focus will be on the disposal of the HLW from the pyro process. The effect of the electro reduction, winning, and refining on the annual individual dose rate will be assessed in combination with the CANDU spent nuclear fuel. This study will give a certain guideline over the value of the pyro process. The analysis model will be based on the fleet of the KAERI codes from MASCOT-K, AMBER, to Goldsim. The effect of the GBI on the final dose rate will be fully assessed. Unlikely to the previous study where the most conservative dose conversion coefficients are derived, more realistic dose conversion factors are developed from the dynamic analysis.

5. QA and Others

In the previous phase study the CYPRUS code was developed, implemented and exported. In the SR-AKRS study the CYPRUS+ will be developed. Two near term targets are set; the first one is to develop the function of PID for the scenario development along with debugging. The second one is to develop the tailor made modules for the specific experiments such as information on fracture networks and the heater experiments. In parallel this new CYPRUS+ will merge with the independent information system which provides relevant information on radioactive waste disposal and nuclear fuel cycle strategy development.

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