SAFETY ASSESSMENT AS AN INSTRUMENT FOR WASTE ACCEPTANCE CRITERIA DERIVATION

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Abstract. According to requirements of Russian Federation regulatory framework the substantiation of safety must be provided in the safety case report. One of the key parts of the safety case is the safety assessment. The safety assessment must be performed at all stages of a facility lifecycle starting from facility siting and development of conceptual design until the termination of the regulatory control usually linked to the period of potential radioactive impact.

The safety assessment performed at designing stage of the near-surface disposal facility for operational and post-closure period is presented here as a practical example. The main purpose of the safety assessment was a derivation of maximum total activity and permissible specific activity for considered radionuclides in L/ILW.

Safety assessment for the operational period was performed according to the SADRWMS and GSG-3 methodologies for normal operation, accidental and incidental situations. Performed calculations resulted in the doses that exceed the safety criteria for staff. Taking this into account permissible specific activity for considered radionuclides were re-calculated as acceptance criteria.

Safety assessment for the post-closure period was performed according to the ISAM methodology. Normal evolution scenario and alternative scenarios were considered. Obtained results exceed the admissible level of radionuclide concentration in ground. Based on proportion of resulted concentration to allowable concentration in ground the total permissible activity for each radionuclide was re-calculated.

After analysis of both operational and post-closure phases integrated waste acceptance criteria in terms of radionuclide activity were derived for considered near surface disposal facility.

Key word: safety assessment, safety case, waste acceptance criteria, disposal

1. Introduction

Life cycle of disposal facility goes through several stages, including interrelated operation and post-closure phases, and according to international practice it is assumed to distinguish between long-term (post-closure) safety assessment (LSA) and operational safety assessment (OSA). Operational and long-term safety assessments are widespread and admitted instruments for objective analysis, assessment of possible radiation impact of radioactive waste (RAW) disposal facility on human and the environment and decision making.

At the end of 1980th Back End of the Nuclear Fuel Cycle became one of the most significant problems of radiation safety for further nuclear energy development. LSA provides understanding of a facility behavior over a long period. The main purpose of LSA is estimation and analysis of radiological impact on human and environment due to radionuclides migration from the RAW disposal taking into consideration wide range of aspects – geological, chemical, physical, social and others. Our days widely used methodology was developed within the IAEA Co-ordinated Research Project Improvement of
Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities (ISAM) and then examined and illustrated within the Project on Application of Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities (ASAM). Later on it was integrated into Safety Case within the following IAEA Projects: Practical Illustration and Use of the Safety Case Concept in the Management of Near-Surface Disposal (PRISM), Practical Illustration and Use of the Safety Case Concept in the Management of Near-Surface Disposal Application (PRISMA). Result of these projects became a base for further development of IAEA Safety Standards, such as SSR-5, SSG-23, SSG-29 and etc. and regulatory documents in the Russian Federation NP-055-14, NP-058-14, NP-069-14 and etc.

In comparison with long term timeframes of RAW potential hazard, the operational period and operational safety previously considered as negligible. Only within the International Intercomparison and Harmonization Project On Demonstrating the Safety of Geological Disposal (GEOSAF) it was realized that operational period can significantly affect the long term safety of disposal facility. At the same time it was recognized in some countries that safety of disposal facility during operation can’t be demonstrated just by the references to radiation protection measures and emergency preparedness and response, but should be somehow numerically assessed and ensured in a systematic manner. In general operation of disposal facility is close enough to operation of storage facility and it seems to be possible to use the methodology developed within the IAEA project on Safety Assessment Driving Radioactive Waste Management Solutions (SADWRMS) and included into the IAEA General Safety Guide No.3 “The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste” (GSG-3). Similar safety documents are under development in the Russian Federation.

2. Practical example

For practical purposes one of real Near Surface Facilities for disposal of RAW of classes 3&4 was considered at design stage. The main purpose of the safety assessment was a derivation of Waste Acceptance Criteria (WAC). Usually only long term (post-closure) safety is considered for this purpose without taking into account operational period of disposal facility. In this research both operational and long-term safety assessment were taken into account.

Taken near surface disposal facility is a concrete vault with dimensions (length, width, height) - 150 × 25 × 7 m. Annual planned capacity is 1100 m$^3$ of RAW. The whole capacity of the disposal facility is 22000 m$^3$ according to design. The operational time is supposed to be at least 20 years. It is planned to place solid conditioned RAW in special concrete NZC containers. After placing containers in NSF, filling free space by clay powder is assumed to be performed. The composition of waste radionuclides include: U-238, Cs-137, Sr-90, Co-60. For preliminary calculations maximum values of specific activity of considered radionuclides as for RAW of the third class according to Russian legislation were used as an input.

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1 According to the Governmental Decree No1069...

2 DERIVATION OF ACTIVITY LIMITS FOR THE DISPOSAL OF RADIOACTIVE WASTE IN NEAR SURFACE DISPOSAL FACILITIES. IAEA, VIENNA, 2003. IAEA-TECDOC-1380

3 $10^{10}$ Bk/kg for $\beta$-radionuclides, $10^9$ Bk/kg for $\alpha$-radionuclides, $10^8$ Bk/kg for transuranic radionuclides
2.1. Long-term safety assessment

LSA includes calculation of radiation exposure on the population and the environment caused by the possible withdrawal of radionuclides from the waste packages and their migration beyond the safety barriers of disposal facility into the environment after the closure. Calculations were performed for the maximal period of RAW potential hazard.

As safety indicators the values of specific activities in ground water on the sanitary protection zone border were chosen. The following assumptions were made: NSF to be constructed, commissioned, operated and finally isolated in accordance with the design; security, environmental monitoring and physical control are supposed to be provided during the period of active institutional control (first 50 - 100 years after the closure); structural integrity of disposal will be preserved; NSF territory can’t be used by people for living and farming work during the period of passive control (next 300 years). Normal evolution scenario and alternative scenarios were considered when performing LSA.

Normal evolution scenario assumes that radionuclides from the waste matrix migrate through containers, clay backfill and concrete wall of disposal vault into the environment. It was assumed that the concrete does not change its strength and filtration properties during first 100 years. After 300 years since vault construction, concrete permeability corresponds approximately to the permeability of sand.

In the period from 100 to 300 years, migration of radionuclides through concrete is due to convection and diffusion processes, and over 300 years, is determined primarily by convection. Migration of radionuclides through clay backfill is defined by diffusion process. After migration through the safety barriers radionuclides get into the unsaturated zone and further, by filtering with precipitation in the ground aquifer.

The migration of radionuclides in the aquifer is due to convective transport, taking into account the physico-chemical processes (adsorption, ion exchange, etc.) and molecular diffusion and hydrodispersion, which will be the scattering factor. As the alternative scenarios considered "raising the groundwater level". This scenario consider changes in the hydrogeological conditions at the site through the placement of the disposal 300 years, despite the fact that the groundwater level rises above the base of the disposal. Because of the degradation of engineering barriers in the system barriers will be enhanced permeability zones ("filtration box"). Conservatively assumed that 100% of radionuclides are in the liquid phase and can migrate with the flow of groundwater to drain, as in the normal evolution scenario.

On the basis of the developed conceptual and mathematical models calculations using Ecolego software tool have been conducted. During the LSA uncertainty and sensitivity analysis were also carried out.

2.2. Operational safety assessment

Main aims of OSA for pre-closure waste management were evaluating of hazards and radioactive impact on workers, population and the environment.

An individual dose rate for worker equal to 20 m/Sv, and for population – 0,1 m/Sv were used as safety criteria. For the environment – air, water and ground concentration (for accidents and incidents) were used as safety criterion. According to the facility design following workers are involved into operation of near surface disposal facility during its operational
period: hoistman, slinger, dosimetrist, controller. The NSF is operated in a shift-operation mode two times per week. Based on climate statistic it was supposed that 20% of working days have adverse weather conditions that is why works at these days will be missed. Total amount of operation modes per year was supposed as 80, average numbers of containers per one mode is 8. There are 3 configurations of radioactive waste into NZC container: 100% of Co-60, 10% of Sr-90 + 90% of Cs-137 and 100% of U-238. The container value is 1.5 m$^3$, wall thickness is 10 cm of concrete. For calculation it was supposed that at each position works one employee. Next step of OSA was development of normal operation, incidents and accidents scenarios. NZC protection uptakes α-β-radiation, that was a reason for Sr-90 and U-238 exclusion from further consideration in normal operation scenarios. Radiation impact for normal operation is due to external γ radiation of Co-60 and Cs-137. However, in incident and accident scenarios consideration α-β-radiation may have a serious impact due to internal exposure. As most dangerous accident scenario was considered NZC drop with waste release. For each scenarios were developed conceptual and mathematical models. Based on these models were calculated doses for workers and population. Dose calculation with consideration direct and scattered radiation.

Operational safety assessment included uncertainties analysis. Uncertainties of time of procedures may have affection on workers doses during all operational period up to 225%, uncertainties of workers location relatively to containers – up to 210% and with both uncertainties – up to 315%.

3. SA results and WAC derivation

Preliminary endpoint results of LSA excess of the safety criteria. Particular, calculations shows exceeding of specific activity in water on the sanitary protection zone border for radionuclide U-238 (3.0 Bq/kg according to national requirements for drinking water) when the initial value of the activity in RAW is $10^9$ Bq/kg. For safe disposal initial specific activity of U-238 in a container was recalculated for WAC development. After recalculation following initial activity of radionuclides were obtained: $U-238 - 3,0\cdot10^5$ Bq/kg; $Cs-137, Sr-90 and Co-60 – $10^9$ Bq/kg (no additional limitation). Preliminary OSA endpoint results also exided the safety criteria - maximum allowable dose for workers –but for other than in LSA radionuclides. Dose for public satisfy the safety criteria for normal operation, incident and accident situations. Specific activity for WAC development were re-calculated based on OSA results for 3 RAW composition: Co-60 (100%) – $8,94\cdot10^7$ Bq/kg; Sr-90 (10%)+Cs-137(90%) – $6,53\cdot10^6$ Bq/kg; $U-238 – 10^9$ Bq/kg (no additional limitation). OSA and LSA have resulted to different activity restriction. Integrated consideration of Waste Acceptance Criteria for both LSA and OSA together gives following results:

Co-60 (100%) – $8,94\cdot10^7$ Bq/kg (based on OSA, no LSA additional limitation); Sr-90 (10%)+Cs-137(90%) – $6,53\cdot10^6$ Bq/kg (based on OSA, without LSA additional limitation); $U-238 3,0\cdot10^5$ Bq/kg (based on LSA, no OSA additional limitation). The research result shows that just operational either just long-term safety assessment separately is insufficient for determining those WAC parameters as radionuclide waste composition and there acceptable specific activities.

4. Conclusion

In general LSA and OSA have similar structure and algorithm. However, scenarios, instruments, assumptions and models are different. The main impact on WAC from LSA results is caused by such factors as radionuclides half-life, engineered and natural safety barriers retardation properties and the migration characteristic of radionuclides. Long-lived
alpha and beta emitting radionuclides, such as uranium and transuranic elements, C-14 and Cl-36 have the most impact on safety in long periods. It should be noted that carbon and chlorine are neutral migrants that is practically not adsorbed by engineering barriers materials and host rocks.

In case OSA the following factors appeared to be crucial: RAW management system, equipment, number of workers and their qualification, safety culture. Gamma-emitting radionuclides play the most critical role when considering normal operation. Alpha and beta emitting radionuclides mainly have no any negative impact during normal operation, while their presence may have a significant radioactive impact in case incidents and accidents.

According to WAC derivation the results show necessity of both operational and long-term safety assessment to be carried out on the integrated approach basis. This works concerns just radionuclide waste composition and there acceptable specific activities WAC parameters, but there is sharp difference in WAC derivation results with separate consideration from OSA or LSA standpoint. However, that is just fewer part of parameters and other parameters derivation needs further researches based on the integrated approach.

Moreover, an integrated approach seems to be essential for other tasks, such as: development and justification of technical, technological and organizational solutions of disposal; development and justification of limits and conditions of safe operation and closure of disposal; development and support of measures aimed at improving the safety of workers, the population and the environment; justification for changes in the design of disposal etc.