PROGRESS IN LOW AND INTERMEDIATE LEVEL OPERATIONAL WASTE CHARACTERIZATION AND PREPARATION FOR DISPOSAL AT IGNALINA NPP

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ABSTRACT

In Lithuania about 70–80% of all electricity is generated at a single power station, Ignalina NPP, which has two RBMK-1500 type reactors. Units 1 and 2 will be closed by 2005 and 2010, respectively, taking into account the conditions of the long-term substantial financial assistance rendered by the European Union, G-7 countries and other states as well as international institutions.

The Government approved the Strategy on Radioactive Waste Management. Objectives of this strategy are to develop the radioactive waste management infrastructure based on modern technologies and provide for the set of practical actions that shall bring management of radioactive waste in Lithuania in compliance with radioactive waste management principles of IAEA and with good practices in force in European Union Member States.

SKB-SWECO International-Westinghouse Atom Joint Venture with participation of Lithuanian Energy Institute has prepared a reference design of a near surface repository for short-lived low and intermediate level waste. This reference design is applicable to the needs in Lithuania, considering its hydro-geological, climatic and other environmental conditions and is able to cover the expected needs in Lithuania for at least thirty years ahead.

Development of waste acceptance criteria is in practice an iterative process concerning characterization of existing waste, repository development, safety and environmental impact assessment etc. This paper describes the position in Lithuania with regard to the long-term management of low and intermediate level waste in the absence of finalized waste acceptance criteria and a near surface repository.

INTRODUCTION

In Lithuania about 70–80% of all electricity is generated at a single power station, Ignalina NPP, which has two RBMK-1500 type reactors. The closure of Ignalina NPP will take place in two stages. Units 1 and 2 will be closed by 2005 and 2010, respectively, taking into account the conditions of the long-term substantial financial assistance rendered by the European Union, G-7 countries and other states as well as international institutions.

Ignalina NPP is undertaking a program of decommissioning support projects, financed by grants from the International Ignalina Decommissioning Support Fund, administered by the European Bank for Reconstruction and Development. This program comprises also the implementation of investment projects in a number of pre-decommissioning facilities including the management of low and intermediate level waste (LILW).

The projects on the LILW management facilities (storages, treatment and conditioning facilities, modification of the existing operational waste management scheme) are of utmost importance. The existing operational waste shall be characterized and conditioned but first of all a LILW management system shall be modernized to be ready for the management of radioactive waste which will result from the Ignalina NPP decommissioning. It will take 5 years to implement the major projects aimed at the modernization of the LILW management and storage.

The Government approved the Strategy on Radioactive Waste Management in 2002 (Ref 1). Objectives of this strategy are to develop the radioactive waste management infrastructure based on modern technologies and provide for the set of practical actions that shall bring management of radioactive waste in Lithuania in compliance with radioactive waste management principles of IAEA and with good practices in force in European Union Member States.

State Enterprise Radioactive Waste Management Agency (hereafter referred to as "RATA") has been founded in 2001 implementing the resolution of the Government. The role of RATA is to provide Lithuania with safe, environmentally sound and publicly acceptable options for the long-term management of radioactive materials. The Government approved the RATA's activities program for 2002–2004. The main task of this program is to attain that RATA performs all functions on radioactive waste management established by the Law on Radioactive Waste Management (Ref 2). RATA's role also includes the provision of advice on, and where possible endorsement of, waste packaging proposals such that the resulting waste packages will be suitable for long-term management including disposal.

MANAGEMENT OF SOLID RADIOACTIVE WASTE

Ignalina NPP produces about 1100 m^3 of solid radioactive waste per year. Solid radioactive waste generated during plant operation is collected and segregated into different groups depending on dose rate and composition (combustible and incombustible) (Ref 3). Combustible 1^{st} group (low level) waste is for most part wastes resulting from normal plant operation. Average annual generated amount is 260 m^3. It consists of overalls, personal protection means, paper and rags (about 40-50%), filters with a wooden frame (15-20%), timber and wooden trestle (15-20%), plastic waste, protective equipment, plastic film and rubber items (15-20%).

Incombustible 1^{st} group waste consists of construction rubble (bricks, concrete) and other construction waste (35-40%), scrap metal (30-40%), insulation materials (15-20%), cable ends and other incombustible waste (5-10%). Average annual generated amount is 580 m^3.

2^{nd} group waste (intermediate level) consists mainly of parts, units and consumable materials resulting from maintenance work on equipment of reactor coolant circulation circuit, gas circuit and turbine. 70 m^3 of combustible and 110 m^3 of incombustible 2^{nd} group waste are generated annually on average. Composition and constituent percentages of this waste and the 1^{st} group waste are virtually identical, only 2^{nd} group waste is more contaminated. Ventilation filters are collected in the transport corridor and packaged in to shielded transport containers or alternatively wrapped in plastic film and collected with other waste. Graphite waste, found mixed in Group 2 non-combustible waste, consists of graphite rings disassembled from spent fuel channels and CPS channels. The graphite rings are removed from the channels prior to cutting by using a special device which breaks the ring into pieces. Graphite waste will be treated separately because of the high contents of long-lived nuclides (mainly ^{14}C).

Group 3 waste (long-lived), which is in general metallic (~90%), is produced in the Hot Cells and Cutting Facilities which are special assemblies used to cut long metallic pieces from reactor core and other high-active components, to make the waste items fit shielded transport containers used in transport of waste of this kind. To protect the inner surfaces of transport container against contamination, a PVC basket is used as a liner for Cutting Facility transport container.

Typical Group 3 waste items include central rods and tubes of spent fuel assemblies, sensors, joining pipes, Hot Cell filters, Control and Protection System (CPS) channels, CPS rods, fuel channels, additional absorbers, assembly channels, measuring channels, power detectors, emergency equipment, suspensions etc.

According to the Lithuanian strategy (Ref 1), it is foreseen to modernize the management and storage of solid short-lived and long-lived radioactive waste of Ignalina NPP in 2002-2009 and to:

- reduce both the total activity and volume of radioactive waste, for such purpose to implement best available technologies;
- implement the new classification system for radioactive waste according to the requirements (Ref 4);
- arrange a licensed landfill for disposal of very low level waste;
- retrieve, characterize, to fulfill the conditioning of and to send back to storage facilities the solid short-lived radioactive waste accumulated in existing storage facilities;
- retrieve, and characterize the solid long-lived radioactive waste accumulated in existing storage facilities, fulfill its proper treatment, install proper interim storage facilities for long-lived radioactive waste, and store the long-lived radioactive waste in storage facilities without final immobilization until the final management methods are decided;
- modernize the radioactive waste inventory record keeping system;
- strive for the direction of the radioactive waste for clearance as much as possible;
- perform investigations and prepare projects suggesting methodologies for calculation of conditional clearance levels and best management methods for materials with contamination exceeding unconditional clearance levels.

SKB-SWECO International-Westinghouse Atom Joint Venture with participation of Lithuanian Energy Institute (hereafter referred to as "LEI") has prepared a reference design of a near surface repository for short-lived LILW (Ref 5). This reference design is applicable to the needs in Lithuania, considering its hydro-geological, climatic and other environmental conditions and is able to cover the expected needs in Lithuania for at least thirty years ahead. The State Nuclear Power Safety Inspectorate (hereafter referred to as "VATESI") is drafting now the guide on siting of a near surface repository.

MANAGEMENT OF LIQUID RADIOACTIVE WASTE

The liquid waste is driven off and concentrated in an evaporation unit. The evaporator concentrate is fed to one of two bituminization units. The bitumen compound, having a salt concentration of about 40%, is transferred to the bituminized waste storage facility, which is an above ground concrete structure with 12 inside steel-coated canyons. It was determined that the existing storage facility may be acceptable for final disposal but further more detailed analysis is necessary (Ref 6). Now it is under the licensing process as storage facility. Ignalina NPP produces about 700 m^3 of bituminized waste per year.

The spent ion exchange resins and sludge of evaporator concentrates are currently being stored in the tank farm on site for liquid waste (Ref 7). Ignalina NPP produces about 280 m^3 of spent resins plus filter aid per year.

Liquid radioactive waste shall be solidified and waste forms shall be enclosed into suitable containers as required for storage, transport and disposal in the near surface repository. Spent resins and sludge will be cemented. Erection of the cementation facility of spent resins and sludge as well as new interim storage facility is now in phase of Basic Design review by competent authorities. This cementation facility is expected to be operational by 2004. The liquid waste will be mixed with binding agent in an optimum ratio by a continuous operating cementation unit and directly filled into 200 l drums. After capping the drums will be loaded into a storage container (8 drums per container) or, if necessary, this container will be put into a shielding container for transport. The throughput capacity of the cementation unit will be 450 m^3/year. The waste package interim storage building for storage time of 60 years will be designed for a capacity of minimum 10 years of cementation plant operation.

GENERIC WASTE ACCEPTANCE CRITERIA

General

Development of waste acceptance criteria is in practice an iterative process concerning characterization of existing waste, repository development, safety and environmental impact assessment etc. This section describes the position in Lithuania with regard to the long-term management of LILW in the absence of finalized waste acceptance requirements and near surface repository.

The Lithuanian regulation “Generic Waste Acceptance Criteria and Requirements for Waste Package Specifications” (Ref 8) has been drafted by Lithuanian Energy Institute together with SKB (Sweden) experts on behalf of RATA. The objective of this regulation is to set up the generic waste acceptance criteria and requirements for the waste package specification that are necessary to comply with for ensuring the minimization of any future re-conditioning needs associated with the disposal of short-lived LILW generated from the operation and decommissioning of Ignalina NPP and other radioactive waste that is transferred to Ignalina NPP for storage and/or conditioning. This regulation shall be applied to the conditioning of solid and immobilized short-lived LILW candidate for disposal in near surface repository. This regulation does not cover neither the conditioning of long-lived waste for disposal in geological repository, nor the conditioning and/or disposal of very low level radioactive waste.

In this stage the establishment of waste acceptance requirements for the disposal of waste packages is not possible, due to the absence of a near surface repository. Waste accepted for near surface disposal should conform to finalized waste acceptance requirements consistent with the operational and post-closure safety assessments for the repository. These waste acceptance requirements will be both repository and site specific. They will be specified by RATA and approved by VATESI.

Waste package specifications

The Waste Package Specifications must be used for determining the quality of packages produced. These specifications must comply with generic waste acceptance criteria and anticipate, in so far as it is possible, waste acceptance requirements, so as to minimize any future re-conditioning needs.

For each type of conditioned solid and immobilized short-lived LILW candidate for near-surface disposal Ignalina NPP must present a Waste Package Specification (hereafter referred to as "WPS") to RATA for approval before operating a waste conditioning facility. The main principle for a division of the waste into specific types must be that waste packages within specific waste type have similar properties, especially regarding waste form, packaging and origin. The standardized table of contents must be used for the data presented in the WPS.

In principle a specific waste description is required for all waste types. However, some odd types of waste comprise a small amount of packages and thus the normal procedure can be simplified, e.g. by additions to already existing WPSs. When big single component is to be brought to the near surface repository this component will be analyzed as specific waste type.

General properties

Only conditioned solid and immobilized waste shall be accepted for disposal. The waste form shall be essentially monolithic (i.e. an integrated single block of conditioned wastes) and uniform. Waste packages are not allowed to contain free liquid that, due to leakage, might exceed the limits established in Lithuanian regulations.

The design, geometry and dimensions of a waste package shall be in compliance with the systems for handling and transport and with the appropriate disposal part of the repository. According to the VATESI requirements, only certified containers shall be used. Full account shall be taken of the effects of waste form ageing during the storage period prior to disposal.

Properties related to radioactivity

The isotopic composition and specific activity of individual isotopes in the waste form as well as the total activity of alpha, beta and gamma emitters in the waste package shall be determined. Measurements or calculations shall quantify the amount of radionuclides. Deviations in isotopic composition shall also be quantified (beyond those associated with the statistics of the radioactive decay phenomena) to be able to assess uncertainties in assay data. The activity assignment methodology must be evaluated, approved and included in the acceptance documentation. Efforts to the improvement of the old radionuclide assay system for updating of data and detection method are under discussion. It will enhance the declaration of activity inventories for individual waste packages.

The maximum acceptable activities $C_{i,\ max}$ must be derived from the inadvertent intrusion scenario, and maximum acceptable activities $A_{i,\ max}$ must be derived from the evolution (leaching) scenario. For the radionuclide contained in one waste package and belonging to the list of radionuclides, the activities C_i are divided by the acceptable maximum activities $C_{i,\ max}$. The sum of these ratios gives the figure X:

$$X = \sum_i \frac{C_i}{C_{i,max}} \qquad \text{(Eq. 1)}$$

The activities A_i are divided by the acceptable maximum activities $A_{i,\ max}$. The sum of these ratios gives the figure Y:

$$Y = \sum_i \frac{A_i}{A_{i,max}} \qquad \text{(Eq. 2)}$$

For a waste package to be candidate for near-surface disposal, X shall be <1 and Y shall ideally be <1. Nevertheless, waste packages for which X<1 and Y >1 but relatively close to 1 (e.g. 1< Y < 10) could be accepted provided that this can be compensated by other waste packages with Y<1, i.e. that the average Y figure for these waste packages remains below 1 (i.e. that globally, for all accepted waste packages, the radiological capacity of the repository site is not exceeded).

For long-lived alpha emitting nuclides additional limitations are applied: activity concentration shall be less than 4000 Bq/g in individual waste packages and an overall average activity concentration shall be less than 400 Bq/g per waste package.

The fissile mass of the waste package shall be limited in such a way that it can be exempted from the transport requirements applying to fissile material. The degree of cracking, swelling, shrinkage, gas evolution and the brittle-fracture phase change of the waste form as well as containers and container ancillary equipment, all constituting the waste package, shall be characterized in terms of its physical and chemical response to radiation fields.

The surface dose rate and surface contamination must be specified for ALARA handling and storage and must comply with the transport regulations and limits established in Lithuanian requirements.

Chemical properties

The ability of the waste form to resist solubility and leaching as well as the integrity of waste form and waste package shall be identified. The test procedures and choice of parameters must be coordinated with VATESI, Ministry of Environment and Radiation Protection Center.

The waste form chemical composition shall be determined to identify hazardous or toxic constituents. These constituents shall be minimized and/or avoided.

Wastes containing substances which are pyrophoric and materials with ignition temperatures of less than 60^{o}C shall not be accepted for disposal. If process knowledge indicates that waste constituents may be ignitable, this shall be verified by testing, and treatment methods shall be designed to eliminate this characteristic.

Reactive chemicals or metals shall not be accepted for disposal. Wastes which may contain potentially reactive materials shall be evaluated and treated to eliminate reactive characteristics.

Potential corrosivity of waste packages shall be assessed to show that the degradation of waste containers and solubilization of radionuclides in the wastes will not lead to unacceptable releases.

In addition to the hazard of explosion from the presence of reactive compounds, the Ignalina NPP must ensure that the hazard of explosion is not introduced through closed canisters or pressure vessels within waste containers. Provision must be made to identify and control such hazards in packaged wastes.

The potential for the generation of flammable gases, owing to the radiolysis of hydrogenous materials or the volatilization of hazardous or flammable gases because of volatile compounds in the waste substrate, including gas production from corrosion of metals such as iron and aluminum, shall be identified during the waste characterization process. Provision shall be made to preclude the accumulation of flammable gases to well below the lower explosive limit in any layer of confinement within the waste package.

Complexing agents and cellulose in the waste package shall be estimated and minimized as far as possible. Decomposition of organic waste shall be avoided as far as possible.

Physical properties

The permeability of a waste form and package shall be sufficiently high to allow gases to be vented, but low enough to restrict water migration and release of radionuclides. Porosity of the waste form shall be as low as reasonably achievable to improve the microstructure and to minimize the release of radionuclides.

Slurry type wastes shall be homogeneously dispersed to an extent that radionuclides never occur in such concentrations that the above mentioned radiological properties will be affected to an unacceptable extent. During packaging and grouting of solid waste, components etc., the active material shall be emplaced in the package in such a way that the activity becomes distributed throughout the package as homogeneously as possible.

Voidage in a waste package shall be minimized to ensure that wastes are immobilized and do not affect other properties, such as strength and permeability.

Mechanical properties

The mechanical strength of the waste package against external stresses such as pressure, strain, bending and impact, shall exceed a minimum for handling, storage and disposal and be sufficient to preclude unacceptable releases of radionuclides during foreseeable incidents and accidents. The waste form shall have a structure and homogeneity that is in compliance with this requirement. As example for transport, the waste package shall be designed so they can be safely stacked to a height of six packages and be capable of withstanding a compressive load equivalent to 13 kPa multiplied by the vertically projected area of the package. Necessary tests for demonstrating ability to withstand accident conditions shall be coordinated with VATESI, Ministry of Environment and Radiation Protection Center.

The structure and volume of the waste form shall be such that it does not deteriorate to an extent, leading to unforeseen release. Examples of such processes are swelling of the waste form under pressure build-up and decreasing of mechanical strength caused by changes in temperature.

Thermal properties

Waste package shall be able to withstand an external fire and be handled without causing fragmentation. The activity release from the waste package as a result of an external fire shall not exceed safety limits.

The design of the waste package shall take into account temperatures as low as $-40^{o}C$. Thermal cycling shall not cause instability or a significant reduction in the strength of the waste package. Necessary tests for demonstrating ability to withstand thermal cycling must be coordinated with VATESI, Ministry of Environment and Radiation Protection Center.

Quality assurance

Ignalina NPP is responsible for the quality assurance related to the waste packages in accordance with the requirements set up by VATESI (Ref 4) while RATA is responsible for control measures related to the near surface repository and its barrier functions. RATA is also responsible for the necessary co-ordination between RATA and NPP in this respect.

The WPS is essential in the quality assurance of waste packages. An important aspect of a WPS is documentation of how the quality control procedures are in compliance with the derived functional requirements. Quality control procedures are mainly based on process control where Ignalina NPP controls and monitors critical conditioning parameters. The WPS shall also define the parameter values, including variations. What to measure shall be derived from WPS and translated to instructions used by Ignalina NPP.

RATA permanently controls the quality assurance related to the waste packages. Auditing of the waste management system is incorporated in the Ignalina NPP internal auditing program prepared in accordance with the requirements set up by VATESI (Ref 4).

DERIVATION OF ACTIVITY LIMITS FOR DIFFERENT TYPE WASTE PACKAGE SUITABLE FOR DISPOSAL

On the basis of the document (Ref 4) RATA has initiated and is coordinating the development of waste package activity limits for Lithuania's near surface repository.

LEI has performed calculations based on existing repository concept and national radiation protection standards to estimate acceptable activity limits for different type waste packages. The calculations are based on methodological recommendations as given in relevant IAEA studies (Ref 9–12). The worldwide accepted computer code (Ref 13) has been used.

Reference design of the Lithuania's near surface repository (Ref 5) is based on hill-type repository concept since it was found to be preferable from the point of cost and since it would be saturated and percolated much less and much later than a repository located below ground water level. The basic design should be a vault-type construction similar to Rokkasho (Japan) or El Cabril (Spain) type, but founded on a low-permeable bed surrounded by erosion-protected, low-permeable engineering barriers. The proposed design comprises multi barrier system consisting of:

- Waste matrix;
- Waste packaging;
- Disposal container outer concrete;
- Backfill between disposal containers;
- Disposal vault outer concrete;
- Clay based soil barrier;
- Natural ground.

The engineered low-permeable clay material should be smectitic and of sedimentary origin or clay moraine. The thickness of top and bottom clay bed should be of 1.5 m. No inspection galleries are proposed but the design principle allows for inclusion of such units in the vault system. Gas release is through gas-permeable joints in the concrete. The disposal capacity for the reference repository design has been set to 100 000 m^3.

The site for Lithuania's near surface repository is still not selected as well as detailed vault system layout is not defined. The new waste management and conditioning technologies are in selection or design preparation stages. The activity limits for waste package suitable for different type of waste and considering not specified repository site conditions have been defined conservatively without giving credits for such repository barriers:

- Waste matrix;
- Waste packaging;
- Natural ground.

For estimation of acceptable activity limits for waste package two main scenarios were considered – normal evolution of repository barriers (waste leaching) and inadvertent human intrusion (road construction).

The exposure constrain for intrusion scenario was individual effective dose of 1 mSv/y. Scenario assumes that intrusion occurs immediately after the end of institutional control period (i.e. 300 y after closure of repository) with probability of 1. In calculations it was assumed that excavation time is relatively short in comparison to half-life of nuclides considered. For this reason no activity decay during exposure time is considered. In human dose calculations it was assumed that excavated activity that's contributing exposure is evenly distributed within certain volume of the upper soil surface. Dose to human is resulted by sum of exposures from inadvertent ingestion of contaminated soil, inhalation of contaminated dust and direct irradiation from contaminated soil. Concerning waste dilution it was assumed that exposure contributing material comes from excavated radioactive waste, which is evenly mixed only with material surrounding waste within waste vault group. The repository cap material and material in-between waste vault groups are removed out without mixing with content of waste vault group. These materials are removed prior intrusion into waste compartment or are covered by deeper layer excavations and thus do not dilute radioactive waste. Such an approach represents rather conservative dose estimation and gives lower acceptable activity concentration limits.

The exposure constrain for waste leaching scenario was individual effective dose of 0.2 mSv/y. As the repository site is still not selected and repository layout is not defined, for activity release estimation purpose the nuclides leaching was modeled from one waste vault group (9 vaults). It was assumed that radionuclides released from the one waste vault group are captured by aquifer, which represents 10 m deep water bearing horizon below the repository. Human exposure results from ingestion of contaminated groundwater from a well, which is drilled into aquifer directly below the vault group. The DUST-MS code was used to calculate migration of nuclides thought barriers of repository and nuclides release into aquifer. The nuclides release from waste forms was not modeled as the intention was to have results which could cover all potential waste conditioning technologies will be used. For this reason the calculation results shall be interpreted as conservative especially for short lived and well-sorbed nuclides. It was assumed that water is flowing evenly distributed between disposal containers and the surrounding backfill. Radionuclides in the waste are dissolved in the surrounding water. From there nuclides by diffusion and advection can be transported through barriers into aquifer. The radionuclide transport and release from repository is delayed due to sorption of surrounding material and barriers. For estimation of influence of barriers degradation on radionuclides release a several water flow scenarios were considered. In conservative case it was assumed that barriers retains its design properties only within period of active institutional control (first 100 y) and completely degrade by the end of institutional control period (i.e. 300 years). The second one scenario assumes that barriers are intact over whole institutional control period; however no credits are given for barriers after 500 years after repository closure.

The defined activity limits have to be used for design purpose of waste treatment and conditioning facilities. The activity limits later on have to be updated taking into consideration different waste conditioning technologies and different waste type package features as well as development in Lithuanian repository sitting and design.

Before starting operation of LILW treatment and conditioning facilities the WPS will be examined by RATA for compliance of the waste packages with disposal concept and generic waste acceptance criteria as well as the adequacy of control actions. The examination of WPS by RATA will be documented in a report. If the WPS conforms the disposal concept and generic waste acceptance criteria, RATA will issue for Ignalina NPP the Letter of Comfort.

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