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TECHNICAL AND COST ASPECTS OF RADIOACTIVE WASTES FROM DECOMMISSIONING

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The Co-operative Programme on Decommissioning

In 1985, the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) set up the International Co-operative Programme for the Exchange of Scientific and Technical Information concerning Nuclear Installation Decommissioning Projects.

The process of exchanging scientific and technical information between participating projects includes project descriptions and plans, data obtained from associated research and development activities, and data arising from the execution of decommissioning plans and operations. Special emphasis is put on the assessment of radioactivity inventories, dismantling techniques, remote operation, decontamination of dismantling materials and building structures, melting of metals, radioactive waste management, as well as on health and safety aspects. In addition, more detailed investigation is carried out on topical issues, such as decommissioning costs, decontamination for decommissioning, and recycling and reuse of materials arising from decommissioning.

In 1995, the OECD/NEA Co-operative Programme was renewed for a third five-year period. It has a wide and increasing range of projects, with growing interest from commercial size facilities in addition to the experimental or prototype plants, which were the original basis for the Programme. Today, 35 projects from 13 countries participate in the Programme, while active support is given by the EC, the IAEA and UNIPEDE.

The participating projects represent a wide variety of decommissioning projects, ranging from experimental or demonstration nuclear reactors and commercial scale power plants to pilot or full size fuel reprocessing plants. Apart from the differences that can be expected due to the type of nuclear plant, the major factors influencing any decommissioning project depend on the organisation, and on the economic, regulatory and other circumstances prevailing in each country and on each nuclear site specifically.

To date, the Programme has been successful in its exchange of technical knowledge and experience in decommissioning. Starting from this vast platform of international co-operation, the Programme now intends to put greather emphasis on the dissemination of their achievements to a wider audience, with the objectives to ensure that :

- best internationally accepted practices are employed,
- the knowledge and practical experience gained is brought to the attention of decision makers, influencing the regulatory climate in which decommissioning projects are undertaken,
- the achievements of the Programme are taken into account when regulations or standards are discussed and/or established at both national and international levels.

The present status and execution of the Co-operative Programme, the participating decommissioning projects and the results obtained in the main areas of interest are summarized in the paper.

Management of Radioactive Wastes from Decommissioning

When comparing to operational activities during the active life-time, significant volumes of materials are generated when nuclear facilities are decommissioned. According to existing regulations throughout the world, most of these materials should be classified as "low-level radioactive waste" and be removed to licensed disposal sites.

Such types of disposal sites mostly have limited capacity as well as limited public acceptability. On the other hand, the main constituents of the decommissioning materials produced are large fractions of steel, concrete and other valuable materials. Among others, these factors justify a management strategy for radioactive wastes from decommissioning, which includes due consideration for waste minimization, decontamination, recycling and reuse of valuable materials and unconditional release of other materials.

The paper reviews these important management issues as practised today in the various decommissioning projects. In particular, the country and site specific approaches are addressed which are currently implemented in order to achieve the objectives to reduce radioactive waste volumes for final disposal. Task Groups within the Co-operative Programme have examined existing and proposed standards and regulations to evaluate whether the regulatory environment is promoting options to recycle or reuse valuable materials rather than to restrict them, taking into account the health, environmental and socio-economic impacts associated with disposal as well as with replacement of scrap materials. They also collected and evaluated available practices and techniques for unconditional release measurements and examined the technical adequacy and cost-effectiveness of available decontamination techniques.

Decommissioning Costs

As part of the exchange mechanism within the Co-operative Programme, cost estimates and cost data from participating decommissioning projects were progressively reported and discussed, showing large variations in both cost figures and decommissioning activities considered. A Task Group on Decommissioning Costs was established in 1989 to identify the reasons for these variations and to develop a transparent cost matrix, including all tasks that may be considered in a decommissioning project, starting from the termination of the active operations of a nuclear facility up to achieving green field conditions on site. Recently, the activities of this Task Group were resumed considering the information from an increased number of projects and taking into account the industrial size of new projects now participating in the Co-operative Programme.

In co-operation with the EC and the IAEA, a standardized list of cost items for decommissioning activities was prepared, representing a new, uniform and more complete approach to decommissioning costs. It will be published and broadly distributed for general use. It might be reniewed after a period of three years when additional experience may be obtained.

The paper will focus on the main findings of the Task Group, in particular the cost estimates and the data for waste management, including disposal, and the influence of national waste management strategies on these costs.

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MANAGEMENT ROUTES FOR MATERIALS ARISING FROM THE DECOMMISSIONING OF A PWR REACTOR

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The paper will give a complete overview of the management of materials arising during the decommissioning of a small PWR reactor, the so-called BR3, situated at the Nuclear Research Center in Mol Belgium. The dismantling of a reactor produces a variety of obsolete materials. A part must be conditioned and disposed of safely if still radioactive or toxic; another part can be recycled in the nuclear industry even if it is still considered as radioactive material; a final part, the most important in volume and weight, can be recycled in the industry or disposed of as industrial waste if it can be free released with or without decontamination. The management of all the different types of materials produced by the dismantling operations must be performed safely and economically with minimisation of costs and of radioactive wastes as main objectives.

For a reactor, we produce mainly contaminated or/and activated materials. From the ALARA point of view, the critical nuclide is ^{60}Co as it dominates as strong γ emitter. Therefore, the radioactive wastes generated are classified as LLW (<2 mSv/h), MLW(<0.2 Sv/h) and HLW(>0.2 Sv/h) in function of the dose rate on the outer surface of the conditioned waste. However, from the disposal point of view, the so-called "critical nuclides" (e.g. presenting potential health hazards and long half-lives) must be considered. ONDRAF/NIRAS, the responsible governmental authority for waste management in Belgium, has established a complete list of critical nuclides to declare for each waste package. The most important nuclides for a reactor are: $^{63,59}Ni$, ^{137}Cs, ^{90}Sr and the α emitters such as Pu and Am. To determine the waste inventory of a specific package, we use the "fingerprint method" i.e. we use correlation factors between the easy to measure γ nuclides (^{60}Co and ^{137}Cs) and the difficult to measure isotopes (pure β or α emitters).

The radioactive dismantled materials that must be evacuated as radioactive waste are classified in function of the conditioning technique applied. For solid radwaste, we use mainly incineration for burnable LLW, supercompaction followed by cement embedding for LLW low density packages and direct cement embedding in 400 l drums for massive metallic pieces. This last technique is applied for LL to HLW materials. For liquid waste, we use mainly precipitation followed by bitumen embedding of the produced sludge in 200 l drums. For ion exchange resins, we use incineration for low active resins and cement or concrete embedding for medium and high active resins. All the conditioning operations are performed by Belgoprocess, a subsidiary of ONDRAF/NIRAS, which is located in Dessel next to SCK•CEN. The conditioned waste packages are temporary stored on the same site. For the moment, there is no disposal site available in Belgium.

A part of the radioactive dismantled materials can be recycled inside the nuclear world. We actually recycle low level radioactive metallic materials by sending them to a specialised nuclear foundry that can handle them safely and can reuse the melted materials for fabrication of metallic shieldings or containers for radioactive wastes.

For slightly contaminated or/and activated concrete, we examine the possibility to recycle the radioactive concrete as raw materials for the preparation of fresh grout or concrete. The idea is to replace partly or completely the inactive sand or gravel by radioactive materials. This grout or concrete can then be used for conditioning of radioactive wastes.

The major part of the dismantled materials can be free released and either reused unconditionally or disposed of as industrial waste. This requires the set up of free release limits in Bq/cm^2 for surface contaminated materials and in Bq/g for homogeneously massive contaminated or activated materials. Procedures and limits are being set by the Health Physics department under supervision of the competent authority. This procedure is still a "case by case" procedure and is applied currently for materials free release. One important aspect of this procedure is the "traceability aspect" that requires a continuous follow up of the dismantled pieces from the cutting operation to the final evacuation. For contaminated materials, we always consider the possibility to free release them by thorough decontamination. For metals, two processes are currently in use: the Wet Abrasive decontamination process for metallic pieces of simple geometry and a chemical process, the MEDOC process using cerium as strong oxidant, for pieces of complex geometry internally contaminated. These aggressive processes allow to remove completely the contaminated layer so that even high contaminated surfaces (several kBq/cm^2) can be free released after treatment. The generation of secondary wastes (abrasive sludge or chemical solution) is minimised by the use of recycling techniques: recycling of the abrasives in one case and regeneration with ozone of the reacted cerium salt in the other case. For concrete, to separate the radioactive part from the clean one, we mainly use scarifying or scabbling techniques for surface contaminated concrete and remotely operated jack hammer or diamond cable sawing for deep contaminated or activated concrete.

For high active pieces, the choice of the dismantling technique is not only governed by the technical aspects (thickness to cut, type of materials) and by the ALARA aspect (cutting underwater for HA pieces) but also by the amount and type of secondary waste produced. A comparison made during the cutting of the thermal shield of the reactor allowed to quantify all these aspects for 3 underwater cutting techniques: the plasma arc torch, the Electro Discharge Machining and the milling cutter. It appeared that mechanical cutting was the best compromise and was then selected for the further cutting of two sets of internals and for the cutting of the Reactor Pressure Vessel. The produced chips are easily collected, packaged and conditioned. For the dismantling of low active pieces, one important aspect is to minimise the contamination spreading during on site dismantling of the equipments; this is realised by selecting mainly mechanical cutting such as reciprocating saw, nibbler, hydraulic shear which do not produce airborne contamination. The large dismantled pieces can further be reduced inside a ventilated containment in which thermal techniques (oxygen burner, plasma arc torch) or dust producing techniques (grinder, circular saw) can be used. The filtration is performed using regenerable filters to protect the HEPA filters and avoids their frequent replacement. For equipments of complex geometry or for moderately thick equipments, we have tested underwater cutting with an abrasive water jet technique. We foresee to use this technique for the dismantling of the neutron shield tank (complex activated geometry) and for several vessels (steam generator, pressurizer, large vessels, ...) .

Finally, the evacuation routes for dismantled materials will be compared on the economic point of view with some discussion on the relative cost aspects.

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40 YEARS OF EXPERIENCE IN INCINERATION OF RADIOACTIVE WASTE IN BELGIUM

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Since the very beginning of the nuclear activities in Belgium, the incineration of radioactive waste was chosen as a suitable technique for achieving an optimal volume reduction of the produced waste quantities; several R&D projects were realised in this specific field and different facilities were erected and operated.

An experimental furnace "Evence Coppée" has been built in 1960 for treatment of LLW produced by the Belgian Research Center (CEN-SCK).
Regularly this furnace has been modified, improved and equipped with additional installations to obtain better combustion conditions and a more efficient gas cleaning system.

Based on the 35 years experience gained by the operation of the "Evence Coppée", a completely new industrial incineration installation has been designed in the nineties and commissioned in May 1995, in the frame of the erection of the Belgian Centralised Treatment / Conditioning Facility CILVA.
At the end of 1997, the new furnace has burnt 475 tons of solid waste and 270 tons of liquid waste.

Beside the conventional incineration process, a High Temperature Slagging Incinerator (HTSI) has been developed, constructed and operated for 10 years in the past. This installation was the combination of an incinerator and a melter producing melted granulated material instead of ashes, and provided experience in the incineration of hazardous waste, such as chlorinated organic compounds and waste with PCB content.

The paper presents "the Belgian Experience" accumulated year after year with the design and the operation of the here above mentioned facilities and demonstrates how the needs required today for a modern installation are met.

The paper covers the following aspects :

- Design particularities and description of the systems
- Operational results for different solid waste categories (bulk waste, precompacted waste, ion exchange resins, ...) and for different liquid waste categories(organic, aqueous, oil...)
- Required pretreatment of the waste
- Ashes conditioning; R&D projects performed in this field
- Flue gas cooling and purification system, design and efficiency
- Improvements made to solve specific problems
- Public acceptance of incineration.

REDUCTION OF THE RADIOATIVE WASTE VOLUME FROM BELGIAN NPP: REALITY, COSTS AND GOALS

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1. BELGIAN SITUATION

Electrabel, private company, is the country's biggest generator, accounting for 86% of electrical generation. It manages 7 PWRs on two nuclear power plants: 4 at Doel with a total capacity of 2,800 MWe, and 3 at Tihange with a capacity of 2,900 MWe. The total electricity generated up to the end of last year was 43,958 GWh of 55% of the total electricity generated in Belgium.

Radioactive waste management in Belgium was assigned to a federal body, ONDRAF/NIRAS in charge with the liability of the storage and final disposal of all radioactive waste. ONDRAF/NIRAS has an industrial subsidiary, BelgoProcess, which performs low-level waste processing and conditioning on the behalf of those waste producers that do not have the necessary facilities.

Waste management therefore includes processing and conditioning, transport, storage and final disposal either near surface or in geological formations depending on the category of waste concerned.

The ELECTRABEL's NPP produce two kind of low level waste: process and technological waste. Process waste such as filters, resines and boron concentrates are conditioned into cement, concrete or polymer at the plant, using facilities licensed and certified by ONDRAF/NIRAS. Technological waste, essentially produced during maintenance operation, is sorted and characterised on the site, than transported to the centralised treatment conditioning facility, at BelgoProcess, for incineration or compaction.

2. REDUCTION OF THE WASTE PRODUCTION

Up to 1983 Belgium used sea dumping for disposing its low-level radioactive waste. Since 1983, ONDRAF/NIRAS store the produced waste. It became thus necessary to reduce the amount of waste. In 1983 it was estimated that 150,000 m^3 of low-level waste would have to be stored at a surface site.
The reduction of waste must respect some basis principle such as ALARA or no dilution or no increase of the release of radioactivity into the environment.

To reduce the volume of conditioned waste, ONDRAF/NIRAS decided to use an unshielded 400 l metallic drum and to allow high dose rate (300 mSv/h contact).

One of the first actions taken by ELECTRABEL was to designate a special effluent manager on the site, able to have a global approach and to coordinate the different actors. Than come the formation of all the workers –internal and external.

Several actions have been taken after an in depth analysis such as maintenance works coordination, selective collect of liquid effluents, increasing of the number of filters conditioned in the same drum or package's forbidding in controlled area, use of reusable protection panels in place of paper or plastic.

The decreasing of the radioactive waste without any negative impact on the radioactive releases into the river and on the volume of the classic technological waste shows that the waste volume reduction is well a reality and not a myth.

3. IMPACT ON TARIFFS

Management by ONDRAF/NIRAS includes the operations of processing and conditioning outside of the plant, when necessary, as well as the transportation, storage and final disposal of the conditioned waste. The cost of this management appears to be inversely proportional to the reduction of waste production.

Because the fixed costs of BelgoProcess remain the same, the constant fall in the amount of waste to be processed brings about an increasing of the fee for the processing/conditioning of future waste. That means that the tariff was constantly changing, what is not easy to manage.

Due to the waste production reduction, BelgoProcess was forced to restructure its workforce and shed around 20% of jobs.

4. NEW CONTRACTUAL RELATIONS FOR WASTE PROCESSING/CONDITIONING

The ONDRAF/NIRAS and BelgoProcess's fixed costs for the duration of the contract are now spread on the waste producers using an objective distribution key –named a reservation capacity. Through this commitment the producers cover the fixed costs, ONDRAF/NIRAS and BelgoProcess can offer a guaranteed tariff to cover the variable.

For the local managers, i.e. the Doel and Tihange sites, the less waste they produce, the less they pay. Their company is, moreover, committed to covering the fixed costs of the waste processing and conditioning plants. In this the local manager also finds a motivation to continue with the waste reduction campaign.

The alternative processing/conditioning solutions and new techniques are easier to be economically analysed: return of investment can be calculated taking into account, all things being equal, the impact on the reservation of capacity applied in the next contract.

The same problem could come to light with regard to covering and meeting the costs associated with final waste disposal, and a similar approach is foreseen.

5. CONCLUSIONS

While, percentage-wise, Belgium has one of the highest nuclear energy generating capacities, it is still a very minor player with regard to radioactive waste production. Moreover, the Belgian surface disposal site will be less than 1/10th the size of the Soulaines site in France. Both these considerations lead us to believe that where radioactive waste is concerned, Belgium, thanks to its waste reduction policy, is operating at less than the critical threshold for an economic use of its processing installation's capacity. The amounts of low-level waste produced by the smaller countries are equivalent to those that the larger countries have not produced as a result of the waste reduction measures they have implemented.

Costs could also have been kept under control, not by adjusting installation capacity but by bringing the amount of waste in line with installation capacity: by that, we mean the processing of foreign waste, but until now, there is no agreement and this solution is not authorised by the Federal Authorities.

The volume reduction is well a reality but the associated cost reduction is well a myth.

DETERMINATION AND DECLARATION OF CRITICAL NUCLIDE INVENTORIES IN BELGIAN NPP RADWASTE STREAMS

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The Nuclear Power Plants managed by ELECTRABEL are located at the Doel (4 units) and the Tihange (3 units) sites and have a total capacity of 5700 MWe. All units are of the PWR type.

Taking into account the need for retrievability and reliability of all requested waste data, the operator ELECTRABEL has subcontracted a complete study to the engineering company TRACTEBEL ENERGY ENGINEERING (TEE) in order to elaborate a computer code for the determination of critical nuclides in the different waste streams. This program should guarantee retrievability and reliability of all information related to the waste packages produced at the NPP.

Two computer codes, LLWAA and DECL, have therefore been developed by TEE.

The first code (LLWAA ; Low Level Waste Activity Assessment code), enables to predict the global inventories and/or the scaling factors of the critical nuclides in the conditioned and in the non-conditioned waste generated by the operation of a PWR.

This code is site-specific as it takes into account the plant design characteristics and operating conditions. A version for BWR plants is under development.

The second code 'DECL' , deals mainly with the complete database management of each waste package produced in order to guarantee full retrievability.

LLWAA and DECL are implemented as an integrated software package called 'DECLARE' at the sites of Doel and Tihange.

Furthermore the LLWAA-code has been extended for the determination of the critical nuclides activities in ashes produced by incineration (LLWAA-Ashes) and for the assessment of the critical nuclides activities deposited on equipment of the nuclear auxiliary systems (LLWAA-Decom).

LLWAA-ASHES-code

The LLWAA-ASHES-code calculates the specific activities and the scaling factors of the critical nuclides in the ashes produced by the incineration of the combustible waste generated by the nuclear installations. This code takes into account the characteristics of the combustible waste (basic materials, specific weight, specific activities of the critical nuclides calculated by the LLWAA-code) and the operating conditions of the incinerator (temperature, waste mass reduction factors, critical nuclides volatility). This code has been validated against measurements for the major critical nuclides (^{14}C, ^{54}Mn, ^{58}Co, ^{60}Co, ^{59}Ni, ^{63}Ni, ^{94}Nb, ^{3}H, ^{90}Sr, ^{134}Cs, ^{137}Cs, U- and Pu-isotopes).

LLWAA-DECON-code

The costs of the future decommissioning of the NPP's appears to be highly dependent on the equipment contamination (pipework, valves, heat exchangers,...). Therefore a new software has been developed to assess the critical nuclides activities deposited on the equipment of the nuclear auxiliary circuits. This software takes into account the contamination in the streams of the systems (calculated by LLWAA), the operating conditions (fluid velocity, pH, temperature), the corrosion products characteristics (particulates diameter distribution) and the nuclides deposition/release rates on the equipment.

Main field of interest.

Above mentioned codes are full featured computer programs for use in radiological characterization of materials, packages and shipments. These programs operate under Windows™ environment. They were developed to provide a practice oriented and easy to operate system for retrieval of all information related to the waste packages.

Since there is a growing need for characterizing radioactive waste one can substantially benefit from these programs : a demonstration by the use of a portable PC can be organized during the meeting for those who are interested in the software packages.

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WASTE TRANSPORT AND STORAGE
Packaging Refused due to Failure in Fulfilling QC/QA Requirements

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The first Brazilian Nuclear Power Plant, Angra I, a PWR-Westinghouse design, 2 loops-626 MWe reactor in operation since 1981 has produced, among other, an average of 27,3 (200 liters) drums of spent resin per year during the first 10 years [1] (see Table 1). The second Brazilian NPP, a PWR-Siemens/KWU design, 1260 MWe reactor, is scheduled to start its operation by the end of 1999. The already existing and near future waste generation will increase either the demand for a final solution regarding the waste disposal or a hurry up for actions to guarantee a safe interim storage method and measures.

According to Federal Law, Brazilian Nuclear Energy Commission-CNEN is the governmental body responsible for the reception and final disposal of radioactive wastes in the whole country as well as for the adoption of regulations concerning waste management and disposal. Political and psycho-social aspects related to the subject of radwaste disposal (e.g. *Not In My Back Yard Syndrome*) have led Brazilian authorities to difficulties when decisions on waste management are to be taken. As a result of such situation, site selection criteria for repository have already been established while a final repository site has not been defined until now. One of the immediate consequences is that radwaste generated by Angra I NPP is still in a provisory facility at the plant's site.

IAEA recommends that the Waste Acceptance Criteria-WAC should be established in close connection with the development of the disposal route for the waste. WAC should be derived from considerations of both operational and those contained in the safety assessment. Since the decision for a Brazilian final repository site has not been taken, the WAC development as recommended by IAEA can not be adequately done. Nevertheless, IAEA TECDOC-560 [2] and 864 [3] provide guidance on the establishment of general WAC & specific packaging specifications and a method by which demonstration on how compliance with the criteria will be achieved. On the other hand, either as the future "owner" of the waste or as Competent Authority, CNEN must specify the minimum level of performance for operations that give the waste a suitable form for storage, transport and disposal. Such performance requirements include acceptance criteria that must be complied by the waste producer(s).

In order to save space at the temporary storage area, 70 concrete type A packagings had to be provided. The packagings would serve as containment for 200 liters metallic drums for storage Intermediate Level Waste-ILW generated by the operation of Angra I NPP. Based on a successfully designed and tested prototype, specifications were written, approved and released for serial production.

As part of the packaging certification process, a regulatory inspection as well as an audit were performed by CNEN. Inspection and audit focused mainly toward aspects as: a) Mechanical resistance, b) Packagings identification, c) Quality Assurance/Control activities, and d) General compliance with codes, standards and specifications.

Weaknesses were identified both in quality control and quality assurance activities. In particular: 1) Non-satisfactory results of mechanical tests, showing performance below the estimated in design documents/specification; 2) Poor identification of already produced packagings, which makes the capacity of data retrieval doubtful; and 3) Packaging production performed by the same sector and personnel as quality control and release for use, which reveal a lack of indoctrination or culture related to QC/QA activities.

One of the most relevant points point identified during inspection/audit concerns the packaging resistance to compression. According to specifications, packagings shall demonstrate capacity to withstand 55 MPa (550 kg/cm^2). Tests results showed resistance no greater than 41,5 MPa. In spite of such results, 14 packagings were released for use. The detected weaknesses were considered as an indicator of failure to fulfilling QC/QA requirements. As a consequence, CNEN's inspection/audit team decided to witness the scheduled tests as well as to testify the construction of a packaging.

Fig. 1
Lost packaging

The production of the packaging #21 was testified by CNEN's staff. In order to increase the resistance to compression, the manufacturer decided to change the concrete composition by using an additive. It was noticed that the slump test failed to reach the specified values. After 3 trials the values were reached and production of package #21 initiated. In few minutes, workers got to the conclusion that concrete conditions would lead to the impossibility to meet the specifications' requirements. Taking into account such results, the manufacturer decided to abandon that packaging (see Fig. 1). CNEN, considering the identified weaknesses, decided not to accept the packages as a container until further tests could be performed. According to CNEN's decision, the packages could be used only as a shielding. In practice, it means that the first 20 produced packages are refused.

TABLE 1 – Inventory of Radioactive Wastes Generated by Angra I NPP/10 Years Operation
(Expressed in number of 200 liters drums)

YEAR	CARTRIDGE FILTER	EVAPORATOR CONCENTRATE	NON COMPRESSIBLE	SPENT RESIN	COMPRESSIBLE
1982	14	41	-	-	74
1983	17	14	06	-	272
1984	08	-	26	73	135
1985	10	23	32	60	116
1986	22	52	63	02	341
1987	11	129	111	-	138
1988	12	155	120	109	345
1989	08	116	31	01	203
1990	13	179	24	-	113
1991	03	68	09	28	86
TOTAL	118	777	422	273	1823

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REGULATORY ASPECTS AND ACTIVITIES IN THE FIELD OF RADIOACTIVE WASTE MANAGEMENT IN BULGARIA

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Bulgaria has been using nuclear power for electricity generation since the beginning of the 70's by operating the Kozloduy nuclear power plant with four WWER-440 and two WWER-1000 reactors. There is also a 2 MW research reactor of pool type that was commissioned in the 60's and is shut down at present.

The spent nuclear fuel (SNF) and radioactive waste (RAW) generated in Bulgaria and their safe and effective management are considered of big importance. During the last few year significant reduction of the RAW volume has been observed at the Kozloduy NPP. The construction of an on-site RAW treatment plant and storage facility is continuing at present. The plant is expected to be commissioned in 2000.

The Committee on the Use of Atomic Energy for Peaceful Purposes (CUAEPP) performs the state policy in the field of the use of atomic energy in the country and defines the requirements on the safe use of atomic energy, the order of accounting for, storage and transport of nuclear material. The CUAEPP is developing a national inventory of SNF and RAW. At the same time the upgrading of the ISUAE database is planned to be accomplished until the end of this year.

The national financial schemes for storage and disposal of RAW and decommissioning of nuclear facilities were defined by the new Regulations on the Management of the "Safe Management of RAW Fund" and the "Decommissioning of Nuclear Facilities Fund" that were approved by the Government at the beginning of 1999.

Important aspect in the field of RAW and SNF management is the National Strategy on Safe Management of Spent Nuclear Fuel and Radioactive Waste that has been developed and is currently under revision by all competent authorities. This document envisages the establishment of a new radioactive waste management organization, responsible for the transport, treatment, storage and disposal of the RAW.

The amendment and supplement of the national legislative programme is as well by elaboration of a comprehensive legislation on the safety of the RAW and SNF management. This includes the promulgation of an Act on Ratification of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management that was signed in September 1998.

The siting process aiming at construction of a National RAW repository in Bulgaria has been continued. The National Concept developed in 1993 is considered as a starting point defining 20 potential sites for disposal of RAW.

The main areas for further development in the field of RAW and SNF management could be defined as strengthening the regulatory activities, safe storage of the SNF and planning and elaboration of the relevant documentation on the NPP decommissioning.

The approval of the National Strategy on RAW and SNF Management and establishment of a complete national system, the implementation of a functional financial scheme and the amendment and supplement of the legislative framework are expected to facilitate the future activities in this area.

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SAFETY ASPECTS OF LOW LEVEL RADIOACTIVE WASTE STORAGE IN KOZLODUY NPP (BULGARIA)

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The safe storage of RAW is achieved by combination of several barriers between the radionuclides and the environment.

Kozloduy NPP has chosen the combination of natural barriers including engineering construction site with suitable geological, hydra-geological, morphology and other characteristics and engineering barriers including matrix with average physical and mechanical indices, package (container) with very high strength indices and shielding level, storage facility of high resistance level against ambient impacts.

Appropriate operational procedures and constant site and environmental monitoring supplement the design safety measures.

The place for storage facility construction is located on the guarded site of the NPP.

The facility site is located on the second non flooding terrace of the Danube river having altitude of 35 m, 4 km to the south of the Danube river midstream.

The geologic structure is of Pliocene and Quaternary materials. The total depth of the Pliocene is 100 m. The maximum level of the underground water is at level 29 m.

The probability for tornado is estimated to $9.177.10^{-6}$

The current analyses define specified for the site earthquake of 0.2 g intensity.

The latest studies specify that if dam wall breakage occurs ("Zelezni Vrata" dam) would appear a wave which will not be dangerous for the site. The maximum river level in this case would be 31.4 m, which is lower than the maximum river level in natural conditions.

The main pathway of the radionuclides existence at the conditioned low level RAW for storage is through the surface and underground water.

The only method of limiting the environment contamination, if having leaks of radioactive liquid from the engineering facility, is the choosing of geologic structures, ensuring low speed of radionuclides spreading.

The data from a study specifies that the horizontal spreading would be performed by the alluvial water level. The spreading is performed in two phases:

- spreading from the surface to the water level;
- horizontal spreading to the water level.

The average speed of a vertical spreading of ^{90}Sr (which is 3 to 30 times more movable from Cs) is 0.3 to 1.8 cm/y for the geologic site structure. The radionuclides would have reached the level of underground water after 350-400 years and more from the moment of their release. The conservative evaluation of the horizontal spreading indicates that a circular area of 1000 m radius contaminated with ^{90}Sr would have appeared after about 150 years, the concentration would have been 2.6%, compared to the initial one. The contaminated area radius, if it is of ^{137}Cs, would have been by the factor of 2 smaller, and the concentration not higher than 0.1%, compared to the initial one.

The treatment method of RAW comprises compaction of solid RAW, solidification of liquid RAW, packaging of compacted and cemented RAW in a reinforced concrete container.

It is expected the compacted RAW to be poured with a cemented mixture; also it is assumed that their immobilizing in a cemented matrix is an additional barrier.

The cemented matrix meets the requirements of OH 0185871-92:

- compression strength - not less than 3.5 MPa;
- resistance to thermal cycles;
- microbiology resistance;
- absence of free water;
- radionuclides' leachability - less than 1.10^{-3} g/cm^2/day;
- homogeneity.

It is considered that this ensures the matrix integrity and the decaying of some of the organic materials in the matrix would lead to forming calcium salts, which are not soluble and would exercise favorable influence.

The compacted solid RAW meet the requirements of OH 0285869-92. According to the performed analysis of the compacted drums' mass in the container, about 60 g gases can be generated annually. The cement matrix and the concrete have sufficient gas permeability; thus the gases shall pass into the atmosphere without breaking mechanically the matrix and the container.

The container is a reinforced concrete structure with cubic form. Its net volume is 5 m^3 and the gross volume is 7.41 m^3. The wall thickness is 14 cm at the base and 10 cm at the top. The container closes with a lid of 8 cm thickness. The mass of an empty container is 6t. and full is 20 t.

The container allows storage of 0.1 TBq (2.7 Ci) activity. The requirements to the container are specified in the regulation OH 0185755-92. To be proved the requirements to the container at licensing the container has passed a test program for:

corrosion resistance, concrete resistance to thermal cycles, reagents and microorganisms, water tightness, compression resistance, seismic resistance, drop test, fire resistance test, determination of radiation protection level.

The container meets the requirements for transport package type IP-III, according to IAEA Safety Series No 6, 1985.

The storage facility is a premise of 72 m length and 37 m width. Adjacent to the storage facility is located the premise for control and management. The floor and the walls are of epoxy coating. The foundation slab is of 1 m thickness and is calculated to bear load of four rows containers by height.

Special attention is paid to the roof and foundation slab waterproof. Bridge cranes using remote control stack the containers. The exact positioning of cranes is performed by bench marks and TV system. The containers are stacked in two areas. Each can take 960 containers.

Drainage system for collection water in the storage facility is provided. The floor of the facility is above the level of the site and the access of surface and underground water is eliminated. Feedwater is not foreseen for the storage facility. Any water on the floor is collected by the drainage system and is led to an underground tank which is dug into the floor of the facility. Then it is pumped in a tanker-trailer and is transported to the treatment facility. This avoids the falling of water from the storage facility into the ground.

The storage facility ventilation is performed by a natural aeration. Heating and conditioning is not provided. Considering extreme environmental conditions emergency roof fans are provided.

Analysis of the burnable load is carried out. The conditioned RAW are non burnable. Fire detection in the premises for remote control and supervision is provided.

The storage facility is located in the guarded NPP site to which trained personnel have access only. In addition, installation of signal-security system that to avoid the access to the radioactive materials of people without special permission, is envisaged.

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NEW ASPECTS IN BULGARIAN NATIONAL ELECTRIC COMPANY POLICY ON SAFE MANAGEMENT OF THE RADIOACTIVE WASTE AND SPENT NUCLEAR FUEL FROM KOZLODUY NUCLEAR POWER PLANT

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Since twenty five years Bulgaria successfuly uses nuclear energy to satisfy the domestic needs of electricity but still the country has not solved the problems with Radioactive Waste and Spent Nuclear Fuel which are stored now on the site of Kozloduy Nuclear Power Plant.

The new management of National Electric Company elaborated „Policy on Nuclear Fuel Cycle and Radioactive Waste" where in a short term aspect we consider to constract a waste treatment building and storage facility for radioactive waste and rerecking of the existing At-reactor pools for Spent nuclear fuel.

In a long term aspect we consider to constract a National Low and Intermediate level radioactive waste repository and new Awey from reactor dry spent fuel storage facility.

In order to deal effectively radioactive waste in Bulgaria a new Waste Management Organisation should be established along the lines of other countries successfully undertaking disposal tasks.
The model of this organisation is based on the „classical triangle" principal where the waste producer is NEK, regulatory body is Bulgarian Committee on the use of atomic energy for the peaceful purposes and Waste disposer will be the new Waste Management Organisation.

From the beguining of this year we started to cope money in National fund for safe management of radioactive waste which will be spent for treatment and disposal of radioactiwe wastes arising from all activities, including generation of electricity from Kozloduy NPP.

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THE BACK END OF THE FUEL CYCLE AND CANDU

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CANDU reactor operators have benefited from several advantages of the CANDU system and from AECL's experience, with regard to spent fuel handling, storage and disposal. AECL has over 20 years experience in development and application of short-term and medium-term storage and research and development for the disposal of used fuel. As a result of AECL's experience, short-term and medium-term storage and the associated handling of spent CANDU fuel are well-proven and economic and offer an extremely high degree of public and environmental protection. Both short-term (water-pool) and medium-term (dry canister) storage of CANDU fuel are comparable or lower in cost per unit of energy, than for other reactor systems. Both pool storage and dry spent fuel storage are fully proven, with many years of successful, safe operating experience.

AECL's extensive R&D on the permanent disposal of spent-fuel has resulted in a defined concept for Canadian fuel disposal in crystalline rock. This concept was recently confirmed as "technically acceptable" by an independent environmental review panel. Thus, the Canadian program represents an international demonstration of the feasibility and safety of geological disposal of nuclear fuel waste. Much of the technology behind the Canadian concept can be adapted to permanent land-based disposal strategies chosen by other countries. In addition, the Canadian development has established a baseline for CANDU fuel permanent disposal costs. Canadian and international work has shown that the cost of permanent CANDU fuel disposal is comparable to the cost of LWR fuel disposal, per unit of electricity produced.

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PROJECTIONS FOR THE RADIOACTIVE WASTE MANAGEMENT IN CHILE

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After three decades of nuclear energy presence for peaceful purposes in Chile, low and medium activities waste arising from applications of radioisotopes which embraces the two Nuclear Research Centers (Lo Aguirre and La Reina) with one research reactor RECH-1 operating and belonging to the Chilean Comission for Nuclear Energy, CCHEN, and all those radioactive waste arising from application of nuclear energy in Industry, Medicine and Research at Universities, are processed in the Radioactive Waste Treatment Plant, PTDR, operated by the Radioactive Waste Management Unit, UGDR, in the Nuclear Research Center Lo Aguirre, CEN LA.

An average of 20 m^3/year of radioactive waste whose range of activity varies from some Bq to TBq is processed by UGDR. The composition of waste are principally spent sealed sources, compactible material; heterogeneous technologie waste solid, and in minor quantity liquid radioactive waste in aqueous and organic phase..

The radioactive waste management of these waste arising from nuclear application in the country, ends with the storage of conditioned waste in the facilities that UGDR operates since the 90's, with a good development level, to the point to be recognized by IAEA, as a Demonstration Center for Latin America and El Caribe, in the procedures and methods for radioactive waste before evacuation. Within the country, the radioactive waste management has become a Service to the national radioactive producers.

The challenge for the radioactive waste management context in Chile for the next century is to introduce in its Plan, the spent fuel that will arise from the research reactor RECH-1 in the next 15 years. These fuel elements, with a LEU bought to Russia, are being manufactured in Chile at the CEN LA by the Fuel Elements Plant, PEC, which is encharged of supporting the fuel elements to RECH-1 located in La Reina Nuclear Research Center (CEN LR).

According with the aims of the radioactive waste management of protecting the environment and safeguard people's health from ionizing radiations, the presentation of a National Plan for radioactive waste management in the near future, to the environmental authorities, which has to consider the spent fuel from RECH-1 , is one of the main tasks that Chilean Comission for Nuclear Energy (CCHEN), regulatory authority in nuclear and radioactive matters, has assigned to the Radioactive Waste Management Unit.

Current operation of the RECH-1 gives rise to a rate of 4 spent fuel elements per year, 50% burned-up, which should be managed in the country. This incoming will change the actual radioactive waste management concept in Chile, and different available options are being considered in a general way, due to the need from the very beginning of forseeing a storage facility for these wastes.

The benefit of RECH-1 operation, which is a pool reactor type with a nominal power of 5 MW, is placed on physics beam experiments, in core experiments, neutron irradiation, radioisotopes production and activation analysis . The fuel which RECH-1 has spent until now is being returned back to United States of America : a total of 28 HEU fuel elements fabricated by the United Kingdom Atomic Energy Authority (UKAEA) at Dounreay, Scotland, using Uranium enriched in the USA have been already transferred to Savannah

River Site in South Caroline (1), according with the Research Reactor Spent Fuel Acceptance Program of the USA. Another charge of 30 HEU spent fuel elements is projected to be returned back to USA in the year 2001.

The National Plan for waste management will propose the strategy of radioactive waste in the country and timescale for disposal of current low and intermediate activity short-lived waste produced in the radioisotope applications which are being conditioning and kept in temporary storage. The Plan, also will be addressed to propose infrastructure and to reach issues in those waste that have not been considered up to now: spent fuel.

The legal framework is being proposed in the Plan, under a Regulation for Radioactive and Nuclear Waste to complement the in force Nuclear Safety Law N° 18.302, in the definition of high activity waste, classification, responsabilities and destination of different kind of waste.

As a primordial item in the National Plan, the inventory of waste in the near future is being planning between UGDR and producers. A questionnary that covers until two next decades has been sent to them, from which is expected to have confident and basic information for characterization, which drives to a good classification from where strategy of destination and storage of radioactive waste should arise.

With regard to the spent fuel, the international experience (2, 3) has been considered. The Plan will follow criteria for the selection of the interim storage facility taking account the conditioning needs for an eventual disposal :

- Local conditions have to be considered :at the reactor (AR) whose lapsus will be short due its pool capacity, or reach to the alternative of improving its capacity. Away from the reactor on the same site (AFROS) will also be considered, and cost and time scale must be estimated, since spent fuel will arised from both research reactors, located in different research centers.
- Outside nuclear research center is another, but remote possibility since the nuclear power plant as not decided yet; but this option of energy is being currently considered in the context of alternative source for electric energy, due to change in weather that is affecting our daily life.
- Technical criteria associated to licensing: security and safety as hermeticity, monitoring, quality in design. Type of containers to be used and whether is necessary to be adapted to our spent fuel. Different scenarios must be considered with the feasibility of radiological and non radiological risks, comprising the biggest unacceptable local event.
- Public answer will be considered in the Plan, in the measure that the technical studies go in advance and make the necessity to share with a well reported public.

The National Plan for radioactive waste management will be based on our reality, economic and safety aspects, and addressed to decide between available technologies and associated criteria to joint an integrated solution for all kind of radioactive waste to be produced in the country.

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CHINA'S STATUS AND STRATEGY OF RADIOACTIVE WASTE MANAGEMENT

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China has a forty-year history of nuclear industry and nuclear technology application. Safe management of radioactive wastes has received high priority attached by the national agencies concerned. In 1992, the national policy on low-and intermediate-level radioactive waste disposal was focused on regional disposal. Afterwards, the principle of taking disposal as main body of radioactive waste management was formulated. China's strategy of radioactive waste management is: (1) to store high level radioactive waste in interim storage in a proper approach, to launch the studies of vitrification and deep geological disposal of high level liquid waste, to treat PWR spent fuel by using technological route of reprocessing; (2) to implement the policy of regional disposal, to make cement solidification of low-and intermediate-liquid waste prior to being sent to repository, to carry out bulk casting shallow land burial and hydraulic-fractured cement-solidification deep geological disposal in some special regions and under specific conditions, to treat low-and intermediate-level solid radioactive waste by means of cement-solidification, compression and volume reduction before being sent to repository; (3) to stabilizing the tailings from uranium mine and mill through reinforcing embankment and flood dam and plantation; (4) to develop or establish the laws, regulations, and standards on safe management of radioactive wastes. In establishing standards, in addition to following generic principle and requirements, emphasis should be placed on the following principles: safety first, economy, taking disposal of main body of radioactive waste management, introducing internationally advanced standards as most possible.

NATIONAL STRATEGY & PRACTICE FOR DISPOSAL OF RADIOACTIVE WASTE IN CHINA

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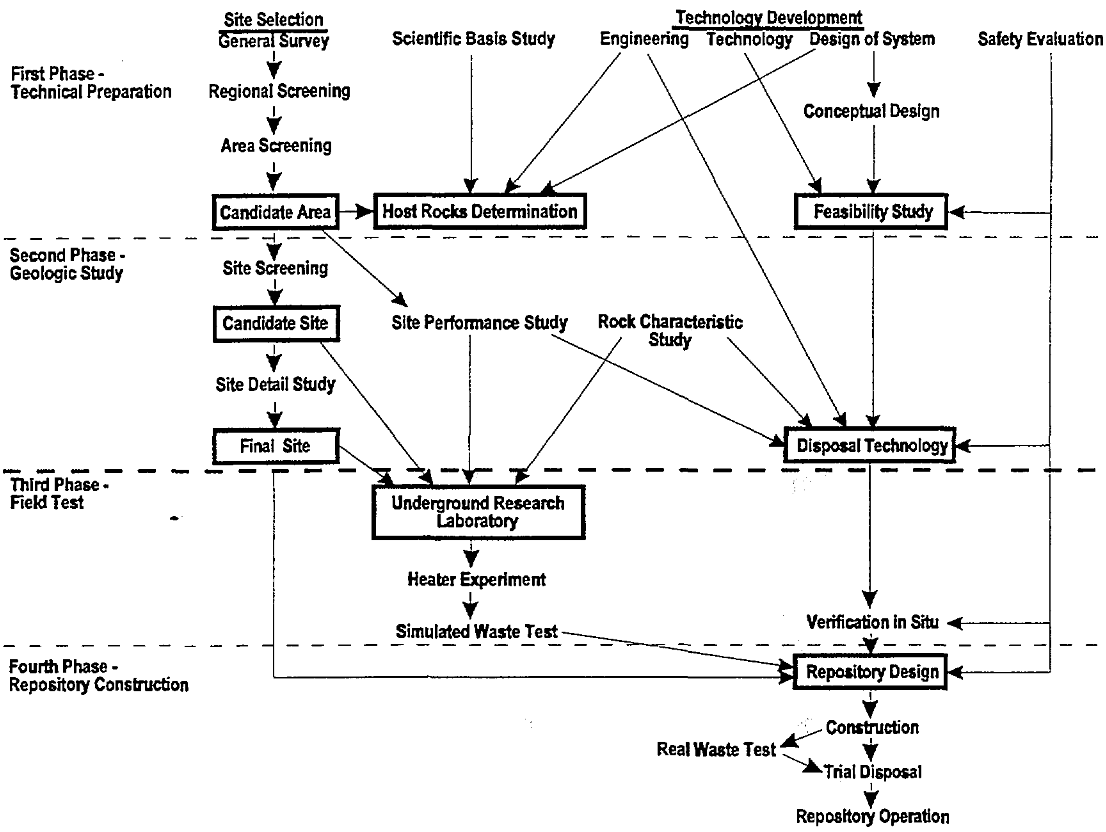
China needs to solve the problems arising from the disposal of radwastes accumulated for years in nuclear industry and has to accept the challenge on the disposal of radwastes generated from the nuclear power program. Accordingly, China, through China National Nuclear Corporation (CNNC) namely the former Ministry of Nuclear Industry and National Environmental Protection Agency (NEPA), established the SDC program for the disposal of HLW in 1985 and issued the policy of regional disposal for LLW & ILW in 1992.

SDC program, which is for the R & D of technology and for the construction of a national geologic repository related to the long-term isolation of HLW, was initiated in 1986. The program consists of four phases including the technical preparation phase, the geologic study phase, the field test phase and the repository construction phase. The final objective of program is to build a repository in the middle of 21 century.



R & D program for Deep Geologic Disposal of HLW (SDC program)

SDC program is being implemented at the first phase. The main activities at present include host rocks investigation, site screening and scientific basis study.

• Host Rocks Investigation Granite and tuff are two kinds of geologic media which are studied most and can be considered as host rocks for a repository based on the conditions of geological, nature, economy and waste generation. Tuff distributes widely in the east of China and more than 400 exposed granite with surface of 25 km^2 mainly located in the northwest & south of China.

• Site screening The area screening has almost completed and two candidate areas in Gansu province, northwest of country are being investigated in detail.

• Scientific Basis Study The studies are now focus on the chemical behavior of actinides in the ground water containing in geologic media, sorption & diffusion of actinides in bentonite and the properties of bentonite. Complex constant of actinides and mechanical property & thermal conductivity of bentonite have been obtained based on the study.

Regional disposal was determined as the national policy for LLW & ILW. The main points of policy include that solidification is demanded for existing liquid waste as early as possible but the duration of interim storage is not longer than five years for new produced waste. CNNC is responsible for the siting, construction and operation of repositories but the approvals of environmental impact assessment, inspection of disposal activities are performed by NEPA.

By the end of 1990s two kinds of repository named "Northwest", "Beilong" will be operated at the northwest and south of China. Four or five repositories will be constructed in accordance with the national program.

• Northwest repository is located at dry and sparsely populated area. The total capacity is 200,000 m^3 of waste but its first phase is 20,000 m^3. In Northwest repository, wastes will be disposed in the underground concrete vaults with cover of 2 meters. The construction of repository has completed and approved by CNNC on October of 1998. It is expected that Northwest repository could be operated at the beginning of 1999.

• Beilong repository is situated at wet, rainy area, nearby the Daya Bay nuclear power plant. It can dispose 240,000 m^3 of waste but its first phase is 14,500 m^3. The vault has a cover of 5 meters at above ground partially. The construction of repository will be finished in 1999.

• Along with the development of nuclear energy, siting of repository is being conducted at the east area. Zhejiang repository, which is the third one, is under consideration for the disposal of wastes from Qinshen nuclear power plant.

This paper presents technical overview on the performance of SDC as well as experience related to the design and construction of repository for LLW & ILW.

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DEEP GEOLOGICAL DISPOSAL OF HIGH LEVEL RADIOACTIVE WASTE IN CHINA: BACKGRROUND AND STATUS 1998

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China is now facing the challenge of how to safely dispose of nuclear waste. The low and intermediate level waste will be isolated by near surface disposal method or underground disposal method, but the spent fuel in China will be reprocessed first, followed by vitrification and final geological disposal.

On the Chinese mainland, there are 2 nuclear power plants (NPPs) in operation: the Qinshan NPP the Daya Bay NPP with total capacity of 2100 MW. In the next 5 years 4 more NPPs will be built, their estimated capacity will be 6,600 MW. It is deduced that the accumulated spent fuel will be 1,000 tons by 2010, while 2,000 tons by 2015.

In China, the work related to radioactive waste disposal is managed by the China National Nuclear Corporation (CNNC), which is responsible for the transport of high level waste and spent fuels, reprocessing of spent fuels, vitrification of liquid HLW and final disposal. A Coordination Expert Group was organized for the geological disposal of HLW in 1986, which is composed of experts from Beijing Research Institute of Uranium Geology (BRIUG), Beijing Institute of Nuclear Engineering (BINE), China Institute of Atomic Energy (CIAE) and China Institute for Radiation Protection (CIRP). The group is responsible for R&D program, research work related to site selection, site characterization, repository design, environmental assessment, safety analysis and performance assessment.

In 1985, CNNC proposed an R&D program called DGD program for the Deep Geological Disposal of HLW. The Program is divided into 4 phases: 1) technical preparation phase; 2) geological study phase; 3) in situ test phase; and 4) repository construction phase. The objective of the program is to build a granite-hosted national geological repository in 2040, which can dispose of vitrified waste, transuranic waste and small amount of CANDU spent fuel.

Since 1985, the following work related with high level radioactive waste disposal has been conducted:

- Site screening and site characterization for HLW repository;
- Site screening and prefeasibility study of underground research laboratory site;
- Laboratory experiment on radionuclide migration;
- Study on natural analogies;
- Study on buffer /backfill materials;

- Study on speciation of transuranic elements in solutions;
- Study on models for safety and environmental assessments;
- Study on methodology of performance assessment;

In the past two years, the R&D of high level radioactive waste disposal in China are concentrated on the following fields: preliminary site screening and siting evaluation, radionuclide migration study, buffer/backfill material study, natural analog study and performance assessment of geological disposal system.

Site screening and site evaluation has been the key activity of China's HLW disposal. On the basis of previous nationwide screening, the Beishan area, located in northwest China's Gansu province, is considered as the most potential candidate area for China's geological repository. With rare inhabitants, barren low hills, little precipitation and large evaporation, the Gobi desert Beishan area is of no economic development prospect. The candidate rock is granite in which a geological repository will be built at a depth about 500--1000 meters. The potential granite bodies include Jiujin, Xingchang and Qianhongquan et al.. 1:200,000 scale geological mapping and hydrogeological inves-tigation are carrying out in the area, preliminary results show that the Beishan area is of stable crust structure without active faults, and the groundwater system of the area is of low permeability and low velocity. Satellite image processing, Geographical Information System (GIS) technology and ground geophysical survey are also used to evaluate the suitability of the Beishan area. The candidate host rock investigation in China reveals that granite is the most suitable rock for China's geological repository. In the period of 1999-2000, an International Atomic Energy Agency's Technical Cooperation Project entitled "Siting and site characterization study for China's high level radioactive waste" will be carried out, 1:50,000 scale surface geological, geophysical and hrdrogeological investigation will be conducted, 2 deep bore holes (600 meters deep for each) will be drilled in order to understand the deep geological environment in the saturated zone.

Radionuclide migration studies will help us understand how the nuclides transport through engineering barriers (waste canister and buffer materials) and natural barriers (geological formation such as granite and shale). The absorption and diffusion experiments of Pu-239, Am-241, Tc-99 and Sr-90 on bentonite and the Beishan granite samples are going on. A low oxygen glove box has been installed, and it can provide low oxygen environment (with oxygen concentration less than 5 ppm) for experiment. Experiments under simulated repository conditions has been paid particular attention. An installation, named as RADMIG, simulating the repository conditions has been constructed, in which experiment under T=100℃, P=5 MPa, Eh< -200mV can be carried out. Actinide geochemistry and colloidal-actinide reactions are also studied.

Bentonite is considered as the best buffer and backfill material for deep geological repositories, while China is rich in bentonite resources. After nationwide investigation and screening of bentonite deposits in China, the Gaomiaozi bentonite deposits in Xinhe county, Inner Mongolia Autonomous Region is considered as the best deposit which can provide enough high-quality bentonite for the potential repository. The Gaomiaozi deposit has a bentonite reserve of 127 million tons, the montmorillonite content can reach as much as 63.77%--80.92%. A systematic test on the bentonite is under way, including mineralogy, physical and mechanical properties, thermal properties, geochemical properties and radiation stability.

A hydrothermal granite-type uranium deposit in south China's Hunan province is used for natural analogue studies. Chemical composition, stable isotopes and uranium-series radionuclides of groundwater samples were analyzed, and the results indicated that the diffusion of uranium, thorium and rare earth elements resulting from water-rock interaction is very limited.

Performance assessment (PA) of geological disposal system is at a very beginning stage. Only some literature about performance assessment has been investigated and a plan for Chinese PA is under discussion..

Although China has made much progress in the geological disposal of high level waste, there is still a long way to go. In the 9^{th}-5-year-plan(1996-2000), our effort is concentrated on site screening and evaluation, buffer materials, experiment on migration of radionuclides under repository conditions, and performance assessment studies.

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CROATIAN RADIOACTIVE WASTE MANAGEMENT PROGRAM - CURRENT STATUS

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Croatia has responsibility of developing radioactive waste management program partly due to co-ownership of Krsko nuclear power plant (Slovenia) and partly because of its own medical and industrial radioactive waste. The total amount of generated radioactive waste in Croatia (80 m^3, 6 m^3/year) is stored in temporary storages located at two national research institutes, while radioactive waste from Krsko remains in temporary storage on site. Croatian Power Utility (HEP) and Hazardous Waste Management Agency (APO) coordinate the work regarding decommissioning, spent fuel management and low and intermediate level radioactive waste (LILRW) management in Croatia. So far, the majority of work has been done in developing LILRW management program. To be more precise, the furthest progress has been made in the following areas:

a) Site selection process

Table 1. shows chronological order of work conducted so far together with planned activities. At the present moment, two proposed sites (Fig.1) for final LILRW repository are going through parlamental procedure of formal approval. Approval should be expected in the next couple of months, followed by detailed site-specific data collection activities.

Table 1. Main events during site selection process

year	event
1988.-91.	SITE SELECTION METHODOLOGY AND SCREENING CRITERIA DEFINED/APPROVED (CROATIA 56.538 km^2)
1993.	7 POTENTIAL REGIONS (100-600 km^2) SELECTED
1993.-94.	POTENTIAL REGION SCREENING PROCESS FINISHED
1994.	34 POTENTIAL SITES (2-20 km^2) CHOSEN
1995.-96.	SELECTED SITES COMPARISON
1997.	4 PREFERRED SITES (2-10 km^2) SELECTED FOR SITE CHARACTERIZATION 2 SITES ABANDONED DUE TO POLITICAL AND PUBLIC PRESSURE
1998.	REMAINING 2 SITES WAITING FOR GOVERNMENT APPROVAL PRIOR TO ACTUAL CHARACTERIZATION
1999.	DETAILED SITE CHARACTERIZATION
2002.	FINAL DISPOSAL SITE (15-20 ha) SELECTED
2023.	CLOSURE OF KRSKO NPP - cca 8000 m^3 OF LILRW + 10000 m^3 OF DECOMMISSION WASTE

b) Disposal facility design project

The first phase of the project has been completed with two concepts considered for future repository: near surface disposal in a concrete structure and subsurface disposal (tunnel drift). Both concepts are to be evaluated at the proposed sites once they are accepted.

c) Safety assessment

Preliminary safety assessment study has been made based on preliminary design project and generic parameters of the two potential disposal sites (Moslavačka Gora, Trgovska Gora). This ended the first phase of establishing an iterative safety assessment process, which will be improved as more site-specific data are collected from the site characterization phase of the disposal project.

d) Public acceptance

Public acceptance of radioactive waste disposal, especially site selection has been shown as a sensitive issue resulting in removal of two potential disposal sites from further investigations. Even though the public can theoretically accept radioactive waste disposal activities as environmental protection related, the “NIMBY effect” seems to be stronger. This has mainly been derived from both lack of information and doubts about available given information. In order to mitigate this attitude number of activities have been taken. Among them are: issuing publications, brochures and regular bulletin "APO-News"; recording video tapes; organizing conferences, round tables and lectures, tours for particular groups (experts, journalists) to facilities of interest in Croatia and abroad; informing mass media; establishment of Information (Visitor) Centre.

Figure 1. Preferred sites selected for final LILRW repository

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DEEP GEOLOGICAL RADIOACTIVE WASTE DISPOSAL IN GERMANY LESSONS LEARNED AND FUTURE PERSPECTIVES

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ABSTRACT

As far back as in the seventies a fully developed, integrated concept for closing the nuclear fuel cycle was agreed upon in Germany between those days' Federal Government and the electricity utilities. In the twenty years elapsed since then it was further developed as necessary to permanently fit the state-of-the-art of science and technology. For management of spent fuel the concept currently considers two equivalent alternatives: fuel direct disposal or reprocessing and Pu recycling in thermal reactors. Interim storage of spent fuel and of vitrified HLW from reprocessing, necessary to allow for decay heat generation to decrease, is carried out in centralized installations. German regulations direct that resulting radioactive wastes of all types be disposed of exclusively in deep geological repositories.

At present, there are in the country three centralized interim storage facilities for spent fuel and vitrified HLW, as well as several facilities for LLW and ILW storage at power plants and other locations. A pilot conditioning facility for encapsulating spent fuel and/or HLW for final disposal is now ready to be commissioned. Substantial progress has been achieved in realization of HLW disposal, including demonstration of all the needed technology and fabrication of a significant part of the equipment.

With regard to deep geological disposal of low and intermediate-level waste, Germany has worldwide unique experience. The Asse salt mine was used as experimental repository for some 10 years in the late sixties and seventies. After serving since then as an underground research facility it is now being backfilled and sealed. The Morsleben deep geological repository was in operation for more than 25 years, until September 1998.

WASTE DISPOSAL CONCEPT

The basis of spent fuel management and waste disposal in Germany was laid down by a decision of the Council of heads of the Federal Government and the Governments of the Federal States adopted on September 28, 1979. This direction commanded operators of nuclear power plants to provide assurance for a period of six years in advance as to where the spent fuel will be stored and/or processed and as to how the radioactive waste will be later disposed of.

In Germany, radioactive waste disposal has always been considered as belonging into the sphere of responsibility of the State. Consequently, by the fourth amendment to the Atomic Energy Act of August 30, 1976 the administration under Chancellor Helmut Schmidt explicitly assigned this task to the Federal Government. As for the rest of waste management, disposal costs were to be charged to the waste producers according to the „polluter pays" principle. To this aim, the Atomic Energy Act includes two repository financing mechanisms ensuring that sufficient funding is available as needed. The Ordinance on Advanced Payments rules how waste producers pay for site characterization and repository construction, on account of the total expenditure for waste disposal. In addition, the Act allows for disposal fees covering repository operational costs to be charged in the future to producers upon waste delivery. The utilities, on their part, make financial provisions in their accounts to cover future disposal costs from today's electricity sales revenues.

The knowledge basis and the technology for deep geological disposal is basically available in the country. Besides the know-how obtained with the Asse experimental repository, operation of the Morsleben repository for decades provided long industrial experience in deep geological disposal of LLW and ILW. In addition, most of the technological systems and components for disposal of vitrified HLW and conditioned spent fuel have been developed and demonstrated, mainly in 1:1 scale and under repository conditions. The facilities already available or under construction as part of the German disposal concept include:

- The Morsleben repository for low and intermediate-level waste
- The Gorleben exploration mine.
- The planned Konrad repository for non heat-generating waste

The present paper focus on the operational experience gained in operating the Morsleben repository and in developing the Gorleben exploration mine.

MORSLEBEN REPOSITORY

The Morsleben repository is located in the federal state of Saxony-Anhalt. At the site, potassium was mined until the early twenties. Thereafter rock salt mining went on until 1969, leaving open cavities with a volume of approx. 10 million m^3. In 1970 the nuclear power plant operator of the former German Democratic Republic bought the mine to convert it into a low-level and intermediate-level radioactive waste repository. After a licensing procedure, waste disposal started in 1978 in rock cavities below the 500 m horizon. In the wake of German reunification in 1990, Morsleben became a Federal Facility under the authority of the Federal Office for Radiation Protection BfS, site operation was assigned to DBE.

In this worldwide pioneer deep geological repository different categories of solid LLW and ILW as well as sealed radiation sources were routinely disposed of until a stop on governmental and court decisions in September 1998. The most use technology was stacking of LLW packed in drums in chambers. Waste with higher activity content, delivered to the repository in shielding overpacks, was discharged into closed chambers below a drift through shielding lock systems. Waste was disposed of on the basis of contracts between waste producers and the Federal Government. Ownership of the waste passed over to the Federal Government upon delivery, the producers paid a fee to settle all costs.

As of end of October 1998, the radioactive waste disposed at Morsleben amounted to:

- 36,752 m^3 radioactive waste and
- 6,621 sealed radiation sources.

The Morsleben repository operation license, originally valid until June 30, 2000, was later extended for five more years, but without enlarging the volume or the activity of the waste to be disposed of at the site. The waste total activity acceptable at the site amounts to $1.0 \cdot 10^{16}$ Bq for ß and γ emitting nuclides and $1.0 \cdot 10^{13}$ Bq for α-emitters. Since the valid license did not cover repository decommissioning, a licensing procedure for this activity was initiated in 1992.

Pursuant to ruling by an administrative court commanding to cease waste disposal in one new field, waste disposal at Morsleben was completely stopped pending further decisions on September 25, 1998 by order of BfS.

The full paper will include more details about the operational experience gained at Morsleben.

GORLEBEN EXPLORATION MINE

After an extensive siting process, the salt dome near the village of Gorleben was selected in 1977 as a candidate site for a deep geological repository. This decision was made by the Government of Lower Saxony on behalf of the Federal Government under Chancellor Helmut Schmidt. The site is being surveyed since 1979 to confirm its suitability to host a repository for all kinds of radioactive waste, specially for HLW. A comprehensive exploration program is being carried out, initially from the surface, and more recently also from underground. This survey is aimed at providing all information about the geological and hydrogeological conditions in and around the salt dome needed for a final suitability statement and the repository licensing procedure.

Surface survey using geophysical non penetrating methods as well as exploration boreholes yielded data on the cap rock, the salt dome flanks, and the surrounding strata. This information confirmed in principle the suitability of the salt dome to host a repository as stated in a extensive interim report by the responsible governmental bodies in 1983. Preparatory work for underground exploration started in 1982, excavation of two shafts in the central part of the dome was initiated in 1986.

After reaching the anticipated repository level of about 840 m below surface, construction of the exploratory mine continued with excavating the mine infrastructure rooms, which are currently being equipped. The rock volume excavated hitherto amounts to approx. 200,000 m^3. Among others, refurbishment of shaft 1 is being carried out by installing the hoisting machine that will be in service in the repository if the exploratory mine is turned into a final disposal facility. Drifts and crosscuts are being driven to explore a first prospective waste disposal areas. The survey work includes mapping of strata crossed, evaluation of drilling cores, and non destructive methods as seismic profiles, borehole radar and geoelectric scanning, etc.

All this work, now in course, will provide information on the dome's geological structure and on the space required for waste disposal, which, obviously, cannot be obtained by survey from the surface. If the underground exploration yields positive results the exploratory mine can be turned into a final repository. According to official statements by BfS the facility could start accepting waste at 2012 the earliest. But the future licensing and operation of the repository is highly dependent on consensus between the main waste producers and the Government.

The full paper will include more details about the experience gained in developing the Gorleben exploration mine.

LESSONS LEARNED AND PERSPECTIVES

To be agreed upon

IAEA-SM-357/17

RADIOACTIVE WASTE TREATMENT TECHNOLOGY AT CZECH NPPS

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A basic aim of radwaste handling is its isolation from the environment. Radwaste from Dukovany NPP normal operation, with the exception of the high-level radwaste, is stored after a specific treatment in the disposal facility of the Dukovany NPP. The disposal of radioactive wastes in this facility is the optimum variant of fulfilling the basic aim - isolation of wastes from the environment untill the essential part of radioactivity decays. This period is determined for approx. 300 years corresponding to 10 half-time decays of the dominant radionuclides (^{137}Cs, ^{90}Sr).

Minimizing principle

An important principle in the system of radwaste handling is minimizing of radwastes volume. The minimizing is a process leading to a status that the volume of treated radwaste stored in the disposal facility is to be as small as possible. This process starts with the NPP's technology and its modifications, continues in working procedures and broad adherence of them, and finishes with the reduction factor of the treatment process and the stored drums configuration. The minimizing of wastes may be also understood as an effort for deposited waste mass to be as low as possible, as this is often a factor determining storage charges. Reduction of the waste volumes is of economic, ecological and political sense. It is also a very important issue in the field of public relations.

Classification of Dukovany NPP radwaste

Waste classification from the technological point of view:
Solid waste:

- compactible/combustible
- non-combustible
- wood
- flammable but unfit for combustion
- large metal objects

Wet waste:

- resins
- other sorbents
- sludges

Liquid waste:

- waste waters
- oils and solvents

Gases

Dukovany NPP uses treatment technologies as follows:

- bitumenisation of concentrated waste water (Concentrates)
- low pressure pressing of LLRW
- high pressure compacting of LLRW
- oil decontamination (washing by demineralized water)
- various metals decontamination
- air filtration

Application of the above mentioned technologies is the subject of the lecture. Cementation and calcination were abandoned.

Following other technologies are prepared for implementation:

- dewatering of resins in high integrity containers
- solidification of sludges
- recycling of boric acid

Annex 1: Radioactive waste classification at Dukovany NNP

Number	Category	Waste characteristics	Source	Basic technology of treatment	Alternative solution	One-year Production (t/y)	Approximate specific activity (MBq/kg)	Dominant Isotopes
1	compactible/combustible	condemned personal protective aids, decontamination and cleaning clothes, packing materials, paper, PE sheets	the biggest part originates during unit inspections and repairs	high-pressure compaction	combustion	20	1-2	60-Co 58-Co 110m-Ag 54-Mn
2	non-combustible	glass, wires, cans, metal particles, ceramics, filters	mainly during inspections and repairs	high-pressure compaction	disposal without treatment in suitable shells	3	1-2	--"--
3	wood	wooden transport packages, pallets, scaffold flooring, planks	contingent origination, air-conditioning filters replacement	high-pressure compaction	combustion	1	0,1	--"--
4	flammable but unfit for combustion	PVC, PTFE (teflon) - foils, sealing materials	previously extensively used materials in RCA	high-pressure compaction	low-pressure compaction	1	0-1	--"--
5	large metal objects	structural material of carbon and stainless steel	extensive reconstructions	disposal without treatment or decontamination and recycling	melting	10	0,01	--"--
6	resins	condemned purification station fillings	regular substance replacement, contingent leakages during technological operations	insertion into HIC	bitumenisation or fixation by other organic binders	30	100	137-Cs 60-Co 90-Sr
7	other sorbents	active coal, vapex (perlite), zeolites	--"--	insertion into HIC	combustion or cementing	1	not given	--"--
8	sludge	sediments in tanks, mixture of organic and unorganic substances of non-standard composition	floor washing and cleaning, dust from material separation and abrasion, crystallization beyond design basis	insertion into HIC	cementing or combustion	5	10	--"--
9	waste waters	usually diluted solutions of chemical unorganic substances containing impurities	uncontrolled leakages, sampling, laboratory waters, spilling of liquids	concentrate bitumenisation	cementing	350	1	137-Cs 134-Cs 60-Co 58-Co 90-Sr

10	oils and solvents	depreciated lubricants, solution residues and scintillators	filling exchange, laboratories, elimination of non-applicable and contaminated liquids	combustion	washing by demineralized water	2	0,001	58-Co 60-Co
11	ash, fly ash, slag	residues after combustion and melting	external incinerator, melting furnace	insertion into HIC	cementing	0	not given	no given
12	gases	aerial particles with radioactive aerosols and gases	release from equipment filled with radioactive medium	filtration	storage of compressed gas till holdup of radioisotopes part	not given	0-1	41-Ar 133-Xe 135-Xe 138-Xe 133m-Xe 135m-Xe 87-Kr 88-Kr 85m-Kr 85-Kr 14-C 3-H 135-I 134-I 131-I 132-I 133-I 129-I 58-Co 60-Co

IAEA-SM-357/18P

EXPERIMENTAL RESEARCH OF THE BEHAVIOUR OF BENTONITE FOR USAGE IN SEALING HIGH RADIOACTIVE WASTE DISPOSAL

XA9952242

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The main component of sealing barriers used for high radioactive waste disposal is supposed to be bentonite because of its natural origin, low water permeability and high swelling pressure.

The undrained shear strength and the uniaxial compression strength were among the tested properties of compacted industrial Na bentonite produced in the Czech Republic [1].

The undrained shear strength was tested in the shearbox of internal dimensions 84.5 mm x 84.5 mm on plan. Before shear testing the bentonite sample was compacted into the shearbox placed in the press. Then the box was fixed in the shearbox apparatus.

Each test of 4 samples was carried out for normal stresses 50, 100, 150 and 200 kPa. Shear stresses and strains were electronically measured and recorded. The test results were the peak shear strengths for each normal stress and the calculated peak values of friction angle and cohesion. The tests were divided into groups differing in the water content (10 - 40%) and the initial compaction pressure (300 - 1500 kPa).

The dependence of the peak values of friction angle and cohesion on the water content and the initial compaction pressures is displayed in Fig. 1 and 2. It can be seen that for the water content lower than approx. 20% (saturation ratio ≤0.5), the bentonite has high values of friction angle, approx. 35°. Then the friction angle drops. For the initial compaction pressures higher than 300 kPa, the differences in the friction angle are small. For the water content lower than 10 - 15%, the cohesion is low (≤20 kPa). Then it grows up to 30 - 40% of water content and subsequently it drops again. With higher values of the initial compaction pressure the cohesion considerably increases and its maximum is connected with lower water content due to the effect of pore water suction.

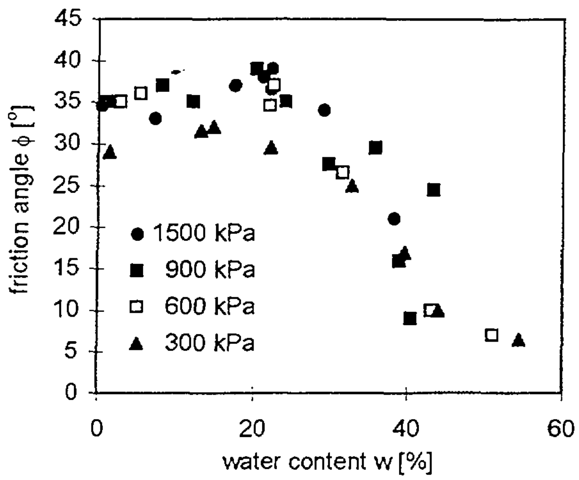

Fig. 1 Friction angle-water content relationship

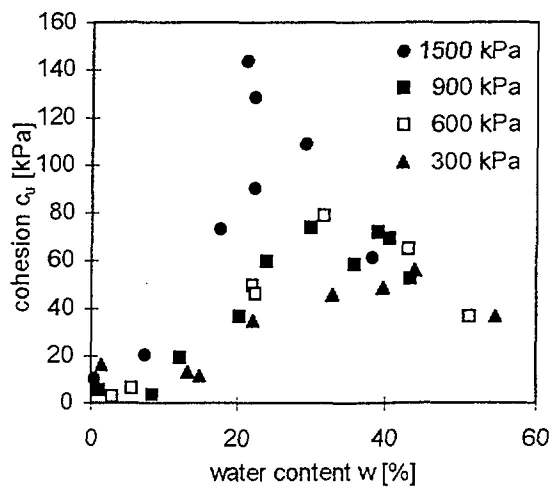

Fig. 2 Cohesion-water content relationship

The samples for the uniaxial compression test were prepared in cylindric moulds. The samples were 38mm in diameter and 76mm high (ratio 1: 2). They were prepared by compaction in the moulds in three layers. Then the moulds were dismantled and the samples were placed into the triaxial apparatus.

The tests were carried out as the triaxial shear test with zero chamber pressure. Vertical deformations of samples were measured as dependent on increasing vertical stress. The results of these tests were the uniaxial compression strengths and the "deformability modules" (including plastic deformations). The measured parameters depend on the water content in bentonite and on the initial compaction pressure. Eight groups of tests were carried out differing in the water content (7 - 45%). The initial compaction pressures were 300, 600, 900, 1200 and 1500 kPa.

Test results can be seen in Fig 3. The maximum values of uniaxial compression strength were reached for the water content of approx. 30%. It corresponds to the results calculated on the base of shearbox tests as the vertical stress for zero (active) chamber pressure (Fig 4).

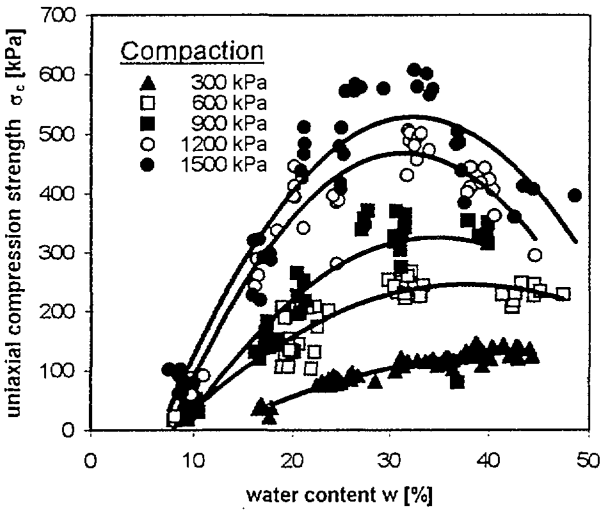

Fig. 3 Uniaxial compression strength -water content relationship

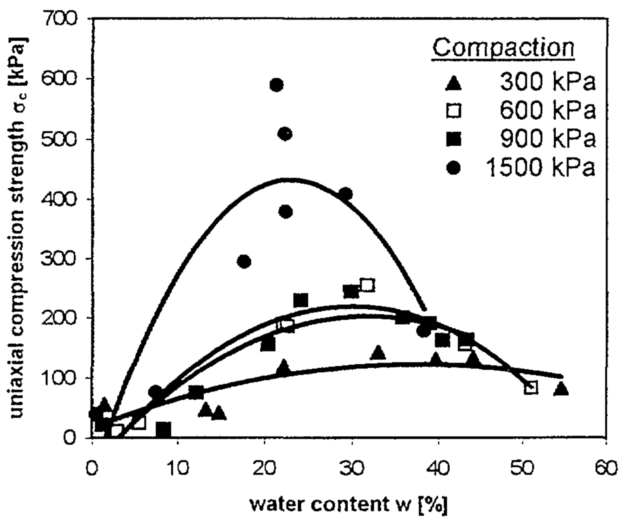

Fig. 4 Calculated uniaxial compression strength-water content relationship

With increasing compaction pressure the maximum value of uniaxial compression strength is connected with decreasing water content as the result of two antagonistic effects of the pore water suction. With decreasing water content the water suction and cohesion should increase, but the necessary higher compaction of the samples with more adhesive particles needs a higher pressure (collapse of the sample structure after watering). The trend has been confirmed, the optimum water content in the bentonite prefabricated blocks compacted by 70 MPa pressure being approx. 5%.

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IAEA-SM-357/20P

STUDIES ON THE CONDITIONING METHODS OF SPENT TRI-BUTYL PHOSPHATE/KEROSENE AND ITS DEGRADATION PRODUCT IN DIFFERENT MATRICES

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XA9952243

Abstract

The destruction of spent TBP/Kerosene (odourless Kerosene (OK)) with potassium permanganate have been investigated. The aim of this work deals with immobilization methods of spent TBP/OK and its degradation product(DP). into different matrices. The matrices used are Portland cement, cement with some additives, silica fume (SF), treated fly ash (TFA), epoxy resin and plain epoxy. Various physical and chemical properties of the conditioned waste have been investigated to avoid premature disintegration of the waste form which could lead to exposure of the radionuclides to the environment. The most important parameters affecting the properties of the solidified waste packages which investigated are, compressive strength, water resistance, chemical resistance, thermal stability, radiological stability and also the leaching rates and diffusion coefficient of the radionuclides $^{152\&154}Eu$, and ^{181}Hf from the solidified waste forms.

The results obtained indicate that, the addition of 5 wt % of silica fume or treated fly ash epoxy resin to Portland cement and plain epoxy resin improves the compressive strength of solidified waste forms as compared with plain cement. Solidified waste forms with degraded product are more resistant for immersion in sea water or tap water and γ-irradiation at high doses while solidified waste forms with TBP/ OK are more resistant to swelling and dissolution in acid, base and salt. Leaching rates and diffusion coefficient of radionuclides $^{152\&154}Eu$, and ^{181}Hf waste forms containing TBP/OK are less than these with degraded product than plan ordinary portland cement. The decrease in leaching rates and diffusion coefficient of the radionuclides was found takes, the follouing sequence ;

Plain ***OPC > OPC+5% SF > OPC+5% TFA > OPC+5% epoxy > Epoxy***

WETTING OF BITUMINIZED ION-EXCHANGERS UNDER SIMULATED REPOSITORY CONDITIONS

XA9952244

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In Finland, the number of nuclear power units is four; at Olkiluoto, two BWR units designed by Asea-Atom and at Loviisa, two PWR units VVR 440. The units generate about 30 % of the Finnish demand for electricity. The radioactive waste that accumulates during the operation of a nuclear power plant is divided into two separate waste streams; highly active spent fuel and operational waste. In accordance with the target schedule set by the Council of State in 1983, preparations have been made relating to the direct disposal of spent fuel from the Finnish nuclear power plants deep into the Finnish bedrock. The final disposal site will be chosen in the year 2000 and the necessary facilities will be constructed during the second decade of the next century. The actual final disposal operations are scheduled to begin around 2020. At present there are four candidate locations. According to the present plans the spent nuclear fuel of the Finnish nuclear power plants will be transferred after interim storage of tens of years, to the final disposal site where it will be encapsulated and disposed of in final repository constructed into the bedrock at a depth of 500 meters. The final disposal of nuclear waste in Finland is trusted to Posiva Oy.

Operational waste includes medium active process water filtering masses and low active waste, mainly produced in maintenance and repair work. Low active waste, such as plastic covers, used tools, protective clothing and towels are compressed into drums or steel containers. Liquid radioactive waste, such as medium active process water filtering masses is solidified. Bitumen has been chosen as an immobilisation agent for the wet wastes at Olkiluoto Power Plant. At Loviisa Power Plant the decision on solidification agent has not been made yet. The tightly packed operational waste is finally disposed of in the VLJ Repository at Olkiluoto since 1992, and also at Loviisa since 1997.

Properties of bituminized spent ion-exchange resins from Olkiluoto Power Plant have been studied by VTT Chemical Technology since the late 70´s. The current interest lies on wetted product as a diffusion barrier. The purpose of the research just in progress is to produce experimentally justified parameter values characterising barrier properties of the bituminized waste product. These parameter values are valuable input data for the next safety analysis of the VLJ-Repository. It was found that normal leach tests are not adequate for assessment of long-term barrier properties of the product. Most leach test and water absorption tests have been performed at room temperature. However, the rate of water uptake and swelling has been found to be much faster at low temperatures [1]. The chosen temperature in this study is 5 - 8 °C and the equilibration media is simulated concrete water to simulate the conditions in the silo. One of the aims is to study the wetting process of the product in microscale. By microscopic examination it is possible to gain more information about the behaviour of bituminized ion-exchange product as a function of time.

In the preparation of the test specimens, a mixture of ratio 1:1 bitumen and dried granular ion exchangers was made at about 140 °C. The mixture was cast in a specimen holder (id. 40 mm and height 40 mm). At the bottom of the holder a Teflon plug ensured the centred positioning of the cast sample. The thickness of the specimen was about 20 mm. All together 20 specimens for the diffusion test and 30 specimens for the microscopic examination were cast. The diffusion samples and identical microscopic samples were equilibrated with simulated concrete water in a same plastic barrel isolated from atmospheric CO_2. The equilibration solution was changed periodically. The new and old solutions were analysed and the important cations and anions were determined by IC chromatography. Also the pH of the solutions was measured. In connection of the exchange of equilibration solutions, all the samples were weighed for the follow-up of the water uptake and three microscopic samples were picked up for an examination. The microscope used in these measurements was a stereo microscope Wild M 8. The microscopic examination revealed that the range of the wetted front was after six water exchanges (481 d) a liner function of the square root of time. The presented range of values in Figure 1 are an average of more than 20 independent measurements.

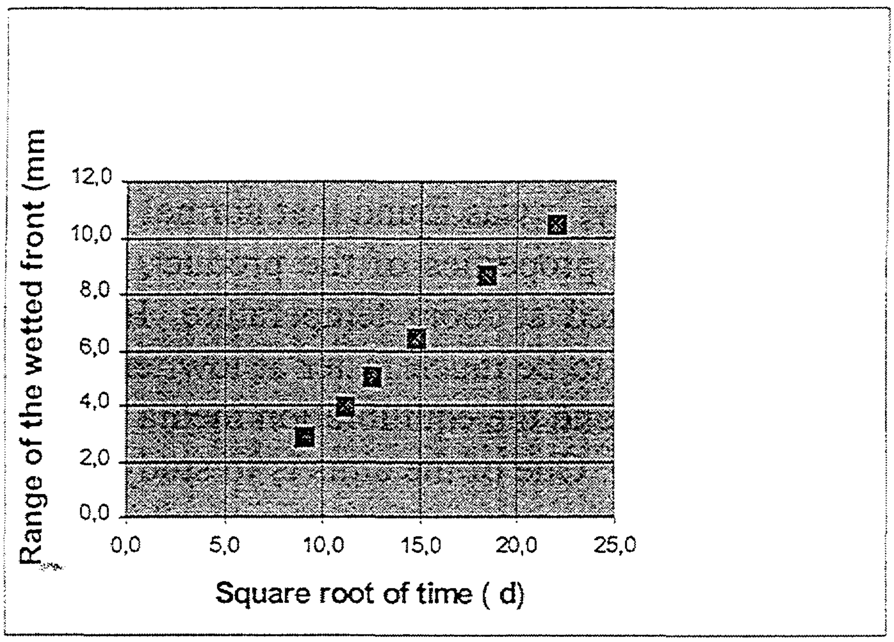

Figure 1. Range of the wetted front from the bottom of the samples as a function of square root of time.

ACKNOWLEDGEMENTS

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XA9952245

IAEA-SM-357/22

MANAGEMENT OF THE FUEL CYCLE BACK-END THE ELECTRICITE DE FRANCE' STRATEGY

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The nuclear fuel cycle operation have effects in the long term and the implemented solutions take time to produce their effects. A fuel cycle back-end strategy needs then to be defined from the very beginning of a program for nuclear electricity production. The consequences of any rupture in the government policies are generally a source of problems and then, the Utilities have to elect strategies permitting adaptation to that eventual changes.

Thanks to a stable French Government attitude, for more than 25 years, sustaining the development of reprocessing and recycling, ELECTRICITE DE FRANCE had the benefit of a well established nuclear context.

Nevertheless, the reprocessing-recycling strategy was experiencing new justifications from the mid 80's. The initial reason for that policy was the need to preserve fissile material resources and breeder reactors were then the solution to develop. When the abundance and associated low costs of natural uranium occured, the economial and strategical interest of that reactors lead to postpone their construction. Then, the preservation of existing industrial facilities of the back-end in order to maintain the possibility to cope with some major change in the uranium market was the main reason to go on reprocessing and to develop MOX fuel.

We consider now that inventories of used fuel and produced plutonium are kept mastered by reprocessing and recycling. Since 850t among the 1150t annually unloaded used fuel are reprocessed, the total quantity of used fuel is only increasing annually by 300t. All the produced plutonium is recycled in PWR and since less plutonium is produced in EDF'reactors, plutonium is progressively concentrated in a limited number of used fuel assembles with a better quality regarding non-proliferation.

Our present policy basis is now founded on the plutonium management and then remains adaptable to any future change-including the eventual turn back to fast-breeder reactors.

Then, the advent in 1997 of a new majority in France, with the support of Green Party, had no dramatic consequences for fuel management but delais in issuing some pending decisions.

A common study made with the support of ELECTRICITE DE FRANCE and the main French Nuclear industry representatives to convince the participants that no short term decision had to be taken and, also, to comfort the Authority that the long term future was not jeopardized by implemented operations.

ELECTRICITE DE FRANCE is implied in the developments to be conducted up to 2006 and required by the 1991's law :

- solutions for separation and transmutation of long lived active products
- possibilities of deep disposal (retrievable or not) - including fuel direct disposal
- various solutions for long term provisional storage of wastes and fuel assemblies.

We have to be well informed about the results of researches in order to bring the Authority the economic aspects of the choices. The Government choice will have to be guided by environmental, political and sociological criterias but also by economy and energy availability.

The allotted time is very short to be sure that, in 2006, all the necessary technical results shall be available.

XA9952246

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SPENT FUEL TRANSPORTATION - LESSONS DRAWN AND WAYS FORWARD: EUROPEAN UTILITIES PROSPECTIVE

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This paper presents the position of European utilities concerning the issue of contamination on spent fuel flasks which has resulted in a mediatic crisis and a halt in spent fuel transportation in may 1998. These events gave rise in three different types of consequences in European countries :

- In some countries, like UK, it was assessed that the available non fixed flask contamination data were consistent with the known effects of flask weeping, with insignificant effect, and that the use of covered rail-wagons for fuel movements precludes casual access to the surface of the flasks. On the basis of this evidence, no special actions were required and fuel transports continued.
- In other countries, like France and Belgium (under temporary specific requirement), after introduction of substantial improvements concerning technical measures, contamination control measurement and documentation, transports have resumed under closer supervision and have since shown a good record.
- In two countries, Germany and Switzerland, required improvements have been implemented but the fuel transports remain stopped due to purely political reasons aiming to harm nuclear production.

It was recognised by all relevant national competent authorities that this contamination issue had no consequences. There was never any leakage from inside of the flask found. Existing procedures, based on the IAEA standards, along with improvements to demonstrate full compliance through more stringent QA application rules, are well adapted to ensure safe fuel transportation.
The paper highlights some aspects concerning the status and regulatory background of fuel transportation, the contamination issue, the origin and consequences, and the lessons drawn by utilities to restore public confidence and recommendations on how to proceed in the future.

Status of spent fuel transportation: stringent international rules apply

Spent fuel assemblies are shipped in special flasks which are tested to severe accidents. The IAEA safety standards (1996) take into account the ICRP 60 and the current spent fuel flasks comply with these regulations. The flasks are carried on dedicated railway wagons, under the protection of a locked canopy or cover and with a closed drain well under the wagon to collect residual water. Each year, several hundred journeys transporting spent fuel are safely carried out in the world without incident.

The transportation organisation - a fully traceable and auditable regulated system:

- The utility has the final responsibility to guarantee that the transport criteria are met within the approved design certificate of the flask.
- The consignor (or shipper) certifies that the contents of the consignment are in proper condition.
- The carrier performs the transport operations, mainly with railroad transport companies.
- A competent authority is in charge of ensuring full application of the radioactive material transport regulation.
- Quality assurance programmes are implemented.
- Transport of radioactive material comply with Euratom requirements and material accountability rules.

The meaning of the contamination limit of 4 Bq/cm^2

For specified parameters, such as surface contamination, derived limits are defined as precautionary standard under pessimistic assumptions. The limit for the non fixed contamination on accessible surface is a cleanness threshold for early warning and does not present any health hazard. It is recognised in the IAEA regulations that non fixed contamination can rise during transport such that it may exceed the derived limits at the end of the journey, without inducing any significant hazard.
For fuel flasks, a "skirt" is used to avoid contact with contaminated water in cooling ponds. The dedicated wagons are provided with a "canopy" or a locked cover which precludes any contact with the external surface of the flasks. So there was never any danger to the public.

When leaving the plant after being monitored in compliance with regulation, these preventive measures ensure that the accessible part of the railway-car are clean. Any surface which could be accessible to the public during transport is then under the contamination limit. When arriving at the dedicated receipt facility, the flask is then handled by specialised workers according to specific procedures and in controlled areas.
The non fixed contamination, which could then naturally occur during transport, would be limited to non accessible surface, on the flask itself or in the railway car drip pan, and should not be considered as an incident as long as it cannot jeopardise in any way the public health and safety.
These comments concerning non accessibility, along with a better explanation of the origin of the 4 Bq/cm^2 limit, should be included in the future ST2 in order to clarify the existing IAA standards.

4/ The contamination of spent nuclear fuel flasks in Europe:

Since the beginning regulatory controls were made and recorded at departure and no contamination was found, with current procedures at that time.
Since early 1990s a few results were found above this 4 Bq/cm^2 limit on the flasks, under the canopy, or in a few occasions on accessible wagon floor service area, at their arrival at the reprocessor or sometimes at the plant. These contamination events affected only limited surfaces and, in the majority of cases, were not in accessible areas. The containment of the flasks was never in question. Remedial measures were implemented - prevention, decontamination of the flasks, QA procedures,

measurements... - but the problem was not really solved. There missed a formal information in general at management level and towards railway companies, due to the fact that it did not involve any health hazard. Beside, there was no formal reporting requirement to the regulatory authorities.

5/ The causes of non fixed contamination: (1) the sweating phenomenon and possible rise of non fixed contamination on the flasks during transport; (2) droplets or residual contaminated water ("hot particles") may result in localised contamination on the flask or in the closed well area under the wagon; (3) the transfer of contamination during handling on the service area of the wagon. The contamination issue on the accessible surfaces of the wagon which, although minor, occurred in few cases, is of particular concern. The rigor of prevention, decontamination and monitoring procedures has been strengthened.
The consequences: all the monitoring campaigns have confirmed the absence of any radiological impact, both for the public and rail workers.

6/ The lessons drawn and ways forward
To improve the existing procedures, information and good practices were exchanged between utilities, carriers and reprocessors teams.
- **Preventive measures:** to avoid contamination during loading in the cooling pool with focus on administrative and technical means, like protection of the flask surface against contact with the pool water, and at the reprocessor with a thorough cleaning.
- **Enhancement of monitoring and cleaning procedures, and international harmonisation:** procedures were compared, reinforced and harmonised between the utilities, the carriers and reprocessors. Contamination monitoring is made using a double step procedure with a whole screening of the equipment and a detailed localised regulatory sampling on a definite number of specific points. The points of control both on the flasks and the wagons are increased but there is a need to optimise the process through experience feedback: great care must be given to ALARA principle to avoid unnecessary doses to workers with due regard to the dose accrued during decontamination or during controls performed directly on the surface of the flasks; on the other hand, with the current standard level now in use for procedures, measurement devices and instrumentation, the efficiency factor for controls should be now closer to 1 than to 0.1 on usual surfaces.
- **Enhancement of the site management and shipper responsibilities,** in providing written declaration or commitment that the package is in proper condition, through its own QA system.
- **Good practices and experience sharing, transportation QA** in order to achieve a better demonstration of the transportation safety.
- **Declaration to regulatory body and public information**
Results above cleanness thresholds are fully recorded. Information process with regulatory authority should include adequate "risk informed" provisions according to the potential significance of events regarding experience feedback: in France, the importance of these events are balanced according to their risk significance and the accessibility of the contaminated surface. These measures should be harmonised at international level.
Experience feedback shows the importance of having clear explanation of the derived regulatory limits and pessimistic assumptions used, which should appear as additional comments in the regulatory textbook in order to enable better common understanding

and explanation to the public of the real nature of risks involved. This process should enable an open information of the public and transport companies.

- Information of railway companies and workers

Full information of the railway companies regarding the issue of spent fuel transport is necessary to clarify the risk involved. Dosimetry measurements at the workplace are important to bring an increase in the personnel feeling of security.

7/ Conclusion

It has been widely recognised that these events of contamination of spent fuel flasks and wagons had **no radiological consequence but it is fundamental to draw lessons from them and to restore confidence in nuclear transportation.**

More stringent procedures must apply to demonstrate compliance with regulatory limits and good practices must be exchanged between all entities involved. Enhancement of the shipper responsibility, experience feedback, QA in transportation activities, prevention, cleanness procedures and monitoring on nuclear sites and at reprocessor sites, while paying attention to ALARA principle, are necessary steps to progress.

It is necessary to **develop openness for the public and for railways companies** and in reporting to the regulatory authorities, within a harmonised risk informed framework to enable a prompt and risk graded information and treatment in case of discrepancy. It is also necessary to bring relevant information about the nature of risks and of preventive measures implemented in direction of all affiliated transport companies.

Utilities need a reliable and predictable transport system. Regulation and associated comments should be made clear as to the nature of the derived limits as cleanliness goal and more generally develop **more explanation concerning the pessimistic assumptions and risk assessment which are at the basis of the different limitations. There should be a clear nexus between regulations and health and safety goals** in order to have a better understanding for the public and to ensure reliability and stability of regulations. Regulation should also be written in order to present the basis which is needed to lend to better explanation to the public regarding transportation safety. Residual contamination which can occur is limited to non accessible surface and has never jeopardised public health and safety.

Whatever improvements are implemented, it is important to **maintain a view on good allocation of resources, unnecessary safety margins and overall reasonable balance between requirements level and health and safety benefits**.

The nuclear transports in European countries have proven to be safe and are the safest way of transporting material to produce the same quantity of energy if compared to fossil fuel. **Utilities and carriers, together with national regulatory bodies and international bodies like IAEA, must develop exchange of experience, co-operation and international harmonisation to further guarantee safe transport of nuclear material and to develop a clear explanation and understanding for the public.**

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A COST-BENEFIT ANALYSIS OF SPENT FUEL MANAGEMENT

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1. INTRODUCTION

Our industry is facing a changing world. Markets are pushing power companies (utilities) to be more reactive. Pay-out times are becoming shorter while customers are likely to quickly swap from one power company to another depending on their price appreciation.
Despite this apparent volatility, investors are also evaluating power companies for their ability to resist external aggressions such as energy prices changes, regulation reinforcements or public acceptance evolution. Nuclear is specially concerned. Actually, even if profitability has to be good, the value of a utility is appreciated for its policy versus industrial, environmental, economic and politic risks. The back-end of the fuel cycle is one of the most sensitive issues in this matter.
A cost-benefit analysis approach can utilities to balance advantages and drawbacks of committing themselves to a long term spent fuel management policy.
The present paper analyzes how spent fuel management can influence the risks and the costs incurred by a utility over the life time of its power plant(s), and what kind of benefit it can gain from a recycling strategy. It is a view from a fuel cycle company and thus has the limited purpose to offer a contribution to the solution of this tricky question.

2 BALANCING COST AND BENEFITS

The paper considers the situation of a utility in position to engage in a long term spent fuel management plan.
The lifetime of a nuclear power plant is 40 years and more. Many utilities are asking their safety authorities for lifetime extension licenses. The operational feed back of nuclear plants demonstrates the possibility for such long-term operation, which will improve shareholders' return.
So nuclear power plants operators need a long-term schedule. This must be taken into account through a long-term spent fuel management policy. This justifies the technical and economical needs for committing to a fuel purchasing and a back-end policy over the lifetime of the nuclear power plants.

2.1 DECISION MAKING

Utilities have different options for the management of their spent fuel: the position of "Wait and See" (W&S) i. e . postponing any decision until interim storage space is no longer available in the spent fuel pool, or a back-end strategy either Direct Disposal (DD) or Reprocessing/Conditioning/Recycling (RCR). W&S, DD and RCR have very different potentialities and consequences. When the legal and political environment allows a choice, any solution must be assessed in a cost-benefit analysis approach leading to a contribution to the kWh cost, which should be reasonable, that means acceptable for the customer, in the above-mentioned ever-increasing competition.

2.2 BENEFITS FROM MOVING SPENT FUEL AWAY

Moving spent fuel away from the reactor site is the first operational step towards a back-end strategy. It is useful to examine what the benefits can be for the utility:

- **Avoiding early reactor shut down**. After the minimum cooling period at the reactor pool, the spent fuel can be shipped out of the reactor site. For a typical PWR, each shipment of one fuel element generates storage space in the pool corresponding to more than one week of reactor operation. Doing that regularly reduces the risk of spent fuel accumulation and gives the operator planning flexibility.
  This benefit may be viewed as trivial but current events show that it is an actual one : The recent example of some German reactors is typical: should the reprocessing contracts be interrupted, some reactors would have to shut down due to the lack of storage capacity. Some US reactors are in the same threatening situation.
  Even when space is available, the perspective not to be in a position to propose a sustainable solution may convey a negative image to the public and the Authorities opinions. This may also adversely impact the plant life extension procedure and lead to early reactor shutdown. Furthermore, on-site interim storage might be requested if spent fuel is not moved in a timely fashion. This induces extra costs due to the lack of anticipation and to additional regulatory constraints.
  - ➔ The benefits of permanent space in pool can be calculated based on the likelihood of these schemes. This space is valuable to assess the ability of the reactor to produce low cost electricity during its remaining lifetime and consequently has a positive effect on the power company's assets.

- **Anticipating regulations strengthening** : International recommendations on environmental protection and safety standards aim to a smooth and rational implementation of nuclear energy. But locally and for less rational reasons, reinforcement of on-site and off-site regulations is always possible. Transportation and licensing conditions of today are very unlikely to be the same in the future. Many experiences of transports carried out in the past, albeit safely performed, should not be possible now at the same conditions and costs. Current interim storage licenses may in the meantime become obsolete. This continuous trend due to the changes in regulations may induce unexpected expenses.
  - ➔ Leaving the W&S attitude for adopting a real back-end strategy is a way to avoid future more severe regulations conditions as well as complying with the present authorizations. The evaluation of this benefit must take into account the evolution of nuclear regulations in a given country and forecast how these changes may be integrated into operating costs of nuclear power plants.

- **Reducing the decommissioning liabilities** : The guarantee that the spent fuel will not be stored at the reactor reduces the risks of delays and over-costs when the site has to be decommissioned. The spent fuel represents the major part of the radioactivity created over the life of the reactor. In many countries including France, when operators have to present their decommissioning plans to safety Authorities, the first stage examined is the complete spent fuel evacuation. After the reactor closure, this is a widely spread technical obligation before beginning any decommissioning activities. Maintaining large quantities of spent fuel on site require handling an important radioactivity level in the long-term precisely at a time when regulations may be more severe.
  - ➔ Any attitude consisting in disposing of spent fuel as soon as possible adds benefits: a sizable decrease of the overall decommissioning costs and the corresponding contingencies and provisions included in the kWh rates.

2.3 BENEFITS FROM A RECYCLING STRATEGY

The previous considerations strongly suggest that the sooner the decision to take care of the spent fuel the better, in a cost/benefit analysis view. But at this point, one can still select, if available, between the DD and the RCR solutions. Some additional benefits can be added-up in selecting the RCR strategy :

- **Cost predictability** : To face the upcoming deregulated market for electricity supply, power companies have to further enhance their competitiveness. Utilities bidding especially on the base-load electricity market have expressed their main criteria for competitiveness : low prices, flexibility and predictability. The latter applies particularly to back-end costs that are spread over a long period of time.

  There is a trend of an increasing request from both the public and the Authorities for the « polluter pays » principle. It is beyond doubt that this principle will be implemented and that utilities will have to take a stand regarding the fuel cycle back-end. Postponing actual expenses increases risk concerning retrieval or clean-up investments as well as associated operating costs even if money is saved in a dedicated fund over the operational life of the reactor. Were the costs under estimated, additional emergency payments would be requested. Adopting a strategy where economics are well controlled will eliminate unnecessary over-costs. The recycling industry has accumulated a large industrial experience by operating its reprocessing plants. At La Hague-France, COGEMA operates a 1,600-t/year nominal capacity plant for various customers. Wastes are conditioned in accordance with national and international safety standards. Valuable materials, reprocessed uranium and plutonium, are converted into new fuel elements. FRAGEMA, a joint venture of FRAMATOME and COGEMA, has supplied 1228 MOX assemblies into nuclear power plants located in France, Germany and Belgium. COGEMA is operating since 1995 the MELOX MOX fuel fabrication plant with a demonstrated capacity of 100 tHM/year.

  → By adopting the RCR strategy, utilities get an additional benefit: such option allows operators to better control their costs; there will be no need to include in the production cost any additional risk or large contingency provisions. So they will be more competitive on the market for base-load electricity supply, which often include clauses for insuring long-term supply together with low guaranteed prices. The evaluation of this benefit is deeply dependent on the situation of the utility on its own electricity market.

- **Hedging future uranium price variations** : There is no doubt that uranium resources are limited at least while the Fast Neutron Reactors are not implemented. Uranium demand will then become more active with time and although its availability is guaranteed for decades, prices may increase. Unpredictable fluctuations are a drawback for companies that are not in position to insure their supply beyond the short term. The RCR strategy includes the continuous property of reprocessed uranium and plutonium. Utilities can get 20 to 30% more energy from their uranium purchases by recycling these materials, thus becoming less dependent from the raw material market.

  ➔ It is possible for a utility to identify the benefit of long-term nuclear fuel availability provided by the RCR strategy by adding a new and reliable source of supply in its own long term fuel procurement plan. Investing in recycling today is a premium for the diversity of supply of tomorrow.

3. A MORE GLOBAL LOOK

There is another kind of benefits which is indirectly linked to utilities' activities. It may be more difficult to calculate them but a significant part of these benefits remains in the global credit of the utilities:

- **Preparing the next nuclear power generation.** Utilities are more and more asked for justifications and explanation whatever they propose to do. This is mainly due to the closer interactions between safety Authorities and the public opinion. The back-end issue is one the most examined one. Any action demonstrating that this issue is properly handled provides a global credit towards the company then towards the whole nuclear industry. This credit is gained facing safety or administrative Authorities while arguing in favor of a demonstrated back-end solution. The evidence of waste volume reduction and the control of well-known and reproducible specifications included the RCR strategy facilitates the demonstration. Quoting the several foreign safety authorities having approved the products delivered by existing plants should initiate discussions between national Authorities and reassure the public opinion.
  - ➔ The demonstration of a correct handling of the back-end through RCR gives the utilities a positive asset. It will be a very high value when the construction of a new generation of nuclear power plants is requested by future energy demand and environmental concerns.

- **Contributing to plutonium management.** Last but not least one should keep in mind that the operation of nuclear power plants naturally produce and burn plutonium. Such material contains a very high energy content. Disposing of that material as a waste is therefore an economical nonsense. Moreover, plutonium stockpiles in the form of spent fuel may cause serious concerns due for instance to its inherent high radiotoxicity. This can jeopardize the image of the nuclear industry and may induce irrational behaviour within the public. Utilities selecting the RCR strategy contribute to the global plutonium management: its use for energetic purposes.
  - ➔ So we add to the list of benefits a contribution to the improvement of the nuclear image which facilitates the development of nuclear energy. Such benefit is credited to the nuclear community and increase its global profitability.

4. CONCLUSION

The proposed cost/benefit analysis confirms the interest of a recycling option for power companies involved in nuclear electricity generation. For them, the list of benefits has to be compared with the commercial cost of a recycling contract, which can be firmly established. It would be interesting for each peculiar situation to calculate the impacts of the benefits previously mentioned: such estimation is not yet performed but one can anticipate at least several mills per kWh or 10% of the kWh rates.

It can also be considered from the perspective of independent investors when considering the possibility of buying shares of electricity generation companies. They will appreciate their value by performing such cost/benefit analysis. These present or potential investors may favor utilities who are able (1) to relieve their operation planning, (2) to mitigate the risks of production losses and (3) to reduce long term liabilities and improve utilization of assets, by using the recycling strategy.

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ACCEPTANCE AND TRACKING OF WASTE PACKAGES

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Andra (French National Agency for radioactive Waste Management) is responsible for long-term radioactive waste management and has to implement a waste management system which covers all stages, from waste generation to waste collection, conditioning, packaging and final disposal.

The nuclear power plants (57 PWR units in operation) of EDF (Electricité de France) generates about one third of the total volume of low and medium activity level waste packages disposed of at the disposal facility, the Centre de l'Aube, currently operated by Andra.

Moreover as several units are stopped, the partial dismantling already begun, generates waste which are also disposed of at the Centre de l'Aube.

The paper will describe :

⇒ the different types of waste packages manufactured by the NPP with regard to the following criteria :

- quantities,
- activity level,
- type of primary waste,
- conditioning process, packaging,
- activity measurements.

⇒ and the waste acceptance process.

Then, in order to verify the quality of waste packages manufactured by the NPP, Andra set up a quality assurance program that specifies the level of quality to be achieved by conditioning and packaging processes, defined quality control requirements and waste tracking requirements, from waste generation through final disposal.

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THE FRENCH UNDERGROUND RESEARCH LABORATORIES PROGRAM, CONTRIBUTION TO THE FEASIBILITY AND SAFETY STUDIES OF GEOLOGICAL DISPOSAL

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The paper will briefly describe *the status of the HLW disposal program in France* and then go on to present *the content of the research program* to be performed during the construction and the operation of the ANDRA's underground laboratories.

The HLW disposal program in France is ruled by the law of 30 December 1991 on research in radioactive waste management. This law sets three research directions, including the study of the possibilities of reversible disposal in deep geological formations, particularly through the construction of underground laboratories. The law sets a fifteen-year period for research, ending in 2006, after which the Government will submit an overall evaluation report of its research to the Parliament, accompanied by a bill which, subject to the necessary justification, will authorize the creation of a repository for high-level and long-lived radioactive waste. Half way through this 15 years period, and since the 9/12/98, the government has authorized ANDRA:

- to go on with the construction of an underground laboratory in the East of France (opalinus clay site) and to use it for research,
- to carry on researching a suitable granitic site.

The program to be undertaken in the opalinus clay site is organized around the following characterizations:

- Characterization of the Geomechanics (based on geomechanical & geotechnical measurements followed by mine by tests),
- Characterization of the radionuclide transfer process in the site (based on borewater sampling, permeability & pressure measure, tracer test),
- Characterization of the hydrogeology (based on measurements made during borehole & shaft sinking, geophysics).

The typical iterative process for pursuing the characterizations is the following:

- *Between now and 2001* : prepare site behavior models before starting each phase of the field works (bore holes drilling, shaft sinking, construction of underground galleries, specific experiments),
- *As from 2001* (*using results from the underground research program*): test and check each model through actual observations and measurements.
  Set the models to take the results of the former phase and predict the results expected during the following one. All these models, after validation, will be implemented to assess the safety related performance of the components of the potential repository as well as the whole facility.
- *As from 2004*: get necessary design data by using the results of the modelisation to design safety related components of the disposal facility (mechanical design, thermal design...).

Of course, this iterative process is adapted to the type of characterization undertaken.

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ANDRA'S CENTRE DE L'AUBE: DESIGN, CONSTRUCTION, OPERATION OF A STATE OF THE ART SURFACE DISPOSAL FACILITY FOR LOW AND INTERMEDIATE LEVEL WASTE

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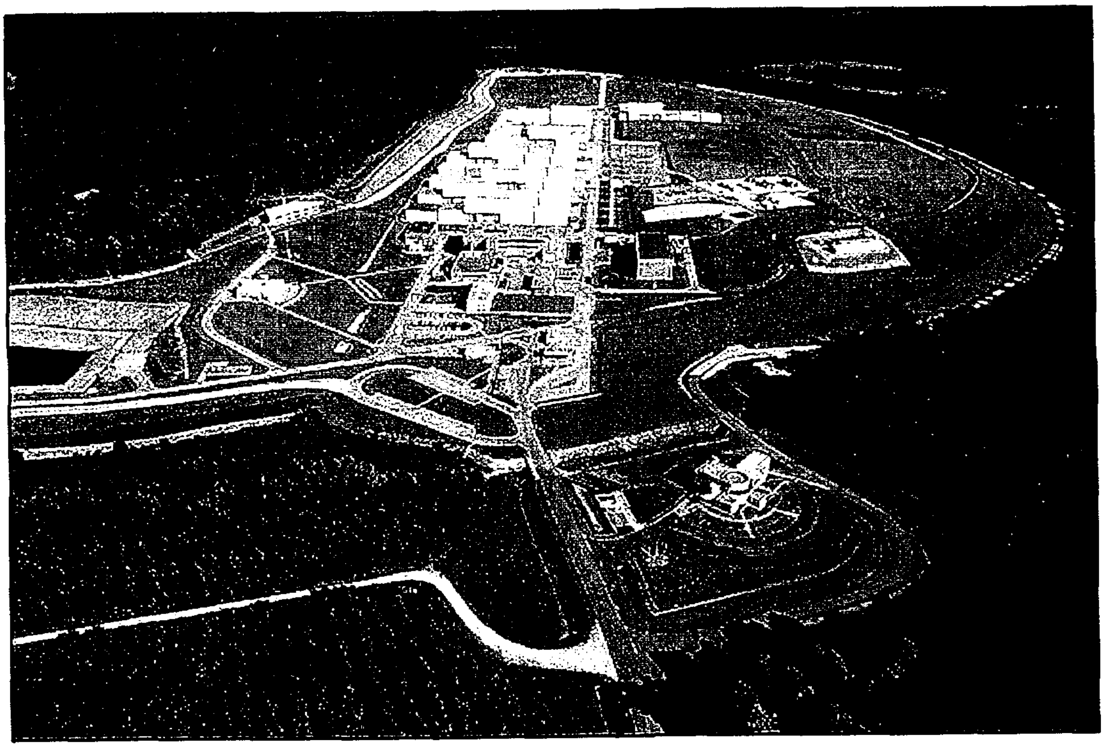

FIG.1. General view of the Aube Disposal Facility

Introduction

France's experience in the management of radioactive waste is supported by thirty years of operational activities in the field of surface disposal of low-and intermediate-level waste (LILW). The so-called Centre de l'Aube which started operation in 1992 is a new surface repository for LILW operated by the French Radioactive Waste Management Agency, ANDRA.

The Centre de l'Aube took over the so-called Centre de La Manche which was the first surface disposal facility opened in France and operated from 1969 to 1994. The total capacity of Centre de l'Aube is 1,000,000 m^3 of waste and its operating lifetime should exceed 50 years.

Regulations and management policy For LILW

In France, the final disposal of LILW is governed by Fundamental Safety Rules (FSR) which set performance objectives for radwaste repositories both on the short term during operation and on the long term, after closure.

According to these rules, the disposal system must be implemented to protect the general public and the environment and allow reuse of site after a monitoring period of about 300 years. The dose limit has been set at 0.25 mSv per year for the general public.

The management policy in France calls for surface disposal of short-lived LILW. The waste may contain a relatively small quantity of long-lived emitters. No decision has been taken yet on the long-term management of long-lived transuranic and high-level waste. Disposal in deep geologic formations is investigated as one possible option.

Safety - related design approach

Isolation of radioactivity contained in the waste is achieved through a multiple-barrier system consisting of

- waste packages ;
- engineered structures, including an earthen cap and a water collection system ; and
- site geological formation.

The waste isolation system must maintain its integrity throughout the operating period (a few decades long) and the institutional monitoring period (approximately three hundred years after operation).

Therefore, the engineered system must be designed and constructed, the waste packages be fabricated and the disposal site be selected to prevent or minimize radionuclides releases as long as the radioactivity remaining in the waste has not decayed down to background levels.

Siting

According to the French FSR, the site geology is required to contribute an additional guarantee of the adequate isolation of waste from water.

Therefore the site must possess hydrological and geochemical properties that would mitigate a potential failure of one of the barriers of the waste isolation system, by controlling the release of radionuclides into the ground.

Other selection criteria include

- site stability and low seismicity ;

- impervious substratum ;
- simple hydrogeology easy to model ;
- well identified outlet for surface runoff and ground water ;
- no valuable natural resources ;
- low density of population.

The Aube site was selected in 1985 after a two year program of geologic, hydrogeologic and geochemical characterization of a number of potential sites.

The Aube site fits the hydrogeologic model used for site screening, consisting of a semi-permeable formation over an impermeable layer.

Design and construction

The design developed by ANDRA for Centre de l'Aube provides a sound and durable engineered system consisting of waste containers stacked in concrete vaults ultimately covered after operation by a slab and lined with a waterproof coating. During operation, the waste packages inside the vaults are protected from the rain by movable steel-frame shelters.

Beneath the vaults, a water-collection system is used to collect and to detect the presence of infiltration water in the disposal units. Collected water is routed to an impoundment basin where it is monitored for radioactivity before release.

Operation

Remotely operated overhead cranes handle waste containers from the transport trucks to their final location in the vaults. The design capacity of Centre de l'Aube is 35,000 m^3 of waste per year.

Two types of concrete vaults are used for the disposal of waste. Concrete waste containers are placed in disposal structures with void spaces filled with gravel. Steel containers are placed in vaults where spaces are grouted with concrete.

For tracking and record keeping purposes, each waste package is identified by a bar code label. During transport and upon delivery, each package is tracked and its final location is recorded through a computerized system.

Closure and long-term monitoring

The long-term integrity of the waste isolation system is due in large measure to the effectiveness of the final cap placed over the disposal facility at the end of the operating period.

The multiple layer capping system implemented by ANDRA at Centre de la Manche consists of alternating layers of draining and impermeable materials. The hydraulic monitoring system integrated in the cap verifies that the water infiltration rate does not exceed a few liters per m^3 and per year.

After closure, the Centre de l'Aube site environment will be monitored during the 300 years of the institutional control period. As for the Centre de la Manche now, ANDRA will be responsible for site environmental monitoring and will report periodically to the licensing authorities.

Conclusion

The experience gained in France through 30 years of surface disposal activities demonstrates than radioactive waste can be managed in a safe and cost-efficient manner with good public acceptance.

ANDRA's Centre de l'Aube which has benefited of lessons learnt at Centre de la Manche in the seventies and eighties is being used as a reference by many countries for the surface disposal of their LILW.

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RADIOACTIVE WASTE FROM NUCLEAR POWER PLANTS AND BACK-END NUCLEAR FUEL CYCLE ACTIVITIES: THE FRENCH SAFETY APPROACH

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French disposal surface facility: The Centre de l'Aube is designed to receive very different types of waste generated by NPP, reprocessing, dismantling and also by Industrial application, research and medicine.

Such a diversity involves very different potential hazards which have to be apprehended in the safety analysis.

The paper will describe how are used the different safety analysis tools in order to manage the very large range of needs of the waste producers and to ensure a safe disposal.
These tools are :

- functional analysis;
- risk analysis;
- safety calculation:
- etc.

We will show that the most important acceptance criteria for the first barrier of containment, which is the waste package, are :

- the containment;
- the durability;
- the activity limitation;
- etc.

Then, we will give examples of derivation of some of these criteria from the safety scenarios (accident during operational stage, intrusion during post closure period.).

However, recently, waste producers request ANDRA to dispose of new types of waste initially not foreseen. Therefore, safety analysis has to imagine new scenario and derive new acceptance criteria. The paper will mention the example of seals sources, vessel closure heads of NPP WPR, racks of fuel elements, contaminated remote controlled robots, waste with high dose rate, pulverulent waste, etc. which lead to specific potential hazards. Indeed, some waste represent an unusual source of radiation, a risk of additional contamination in case of accidental situation, or increase the probability of occurrence of certain scenario such as the recovery of waste in the post-closure period.

The safety analysis must be adapted and must imagine some new scenarios to consider the acceptability of the waste described above and even, deduce acceptance criteria appropriate to the risks. The paper will give examples of studies which may be still under way at ANDRA offices.

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THE ACC FACILITY: THE R&D PROGRAMME

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COGEMA develops a policy of waste management optimization in the back-end fuel cycle which takes benefit from the La Hague plant's experience. The cornerstone of the waste management concept lies on the global philosophy of packaging and the Universal Canister Strategy (UCS) is one of the main contributing tool to reach this target.

The policy's rationale is focused on decreasing the overall volume of high activity waste stored in underground repositories to approximately 0.5 m^3. This target will be reached by extension of glass canister concept to waste other than fission products. This package standardization known as the Universal Canister Strategy is intended to be used for containing waste issued from the spent fuel: vitrified fission products, compacted hulls and end-pieces and technological wastes. The implementation of the unique conditioning will simplify handling and transportation operations.

The Universal Canister Strategy quick implementation requires the construction of a new facility for compaction. This workshop, known as the ACC (Atelier de Compactage des Coques), will compact high activity materials (hulls, end-pieces, and technological waste) in the shape of discs. Each canister contains between 5 and 7 discs depending on their thickness. According to COGEMA schedule, 2,000 canister will be produced per year at the ACC facility. That represents more than 12,000 boxes of compacted waste discs.

Therefore, an important campaign of R&D - which includes active and inactive tests - has been set up aimed at demonstrate the feasibility of the process in terms of conception, maintenance and investments. Three main programmes have been developed:

- on the process
- on the workshop safety
- on the canister characterization

The construction of ACC began in march 1995. The on site tests will be performed from the beginning of 1999 and the ACC commissioning is scheduled by beginning at the end of the year 2000.

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OPTIMIZED WASTE MANAGEMENT: LESS VOLUME AND LESS RADIOTOXICITY

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Reprocessing aims at separating valuable materials (uranium and plutonium to be recycled in NPP) and providing appropriate conditioning for non valuable nuclear waste according to their activity. As such it already constitutes an optimized management of nuclear materials. Moreover, it makes it possible to further decrease waste volume and radiotoxicity, since spent fuel radiotoxicity most significant contributors are :

- Fission Products in the first 300 years or so,
- Plutonium isotopes between 300 and 100,000 years,
- Uranium, Neptunium in the far long term - nearing 1,000,000 years.

In fact, with more than 30 years of industrial feedback in the field of reprocessing in its La Hague plants, COGEMA continues to optimize management of back-end fuel cycle solid waste, in order to reduce both volume and radiotoxicity of final residues.
The efforts have been first focussed on HLW and ILW. Lately, much efforts have also been made on TRU waste and LLW.

Less volume

HLW and ILW : This active waste is mainly constituted by fission products (HLW) and hulls and end-pieces (ILW).

⇨ volume reduction of fission products mixtures has been early done with the commissioning of the R7/T7 calcination, vitrification and storage of glass canisters facilities.

⇨ As for hulls and ends-pieces, the new ACC facility will provide one more forward step in volume reduction policy by compacting the hulls, the end-pieces and some technological waste instead of grouting them in cement. The new process will divide by about 4 corresponding final residue volume. The construction of the ACC began in march 1995. The on site tests started at the end of 1998 and the ACC commissioning is scheduled for the year 2000.

HLW and ILW will then account for less than 0.5 m^3/tonne of reprocessed fuel.

LLW : As for short-lived technological waste, a five years program aiming at reducing their volume to less than 0.8 m^3 per tonne of reprocessed spent fuel in year 2000. This is done by means of sorting and conditioning waste according to the actual activity contained. In this objective, the new implemented scheme is based on the following principles :

- Reduction of materials entering active zones, thus reducing potential waste volume,
- Sorting out at the workshops where waste are generated, including :
  - ⇨ Zoning of plant's workshops into two different zones,
  - ⇨ Precise measurement of α and $\beta\gamma$ activity contained in primary waste drums.
  - ⇨ Either fusion or incineration in the CENTRACO/SOCODEI facility, due to start operating in January 1999,
  - ⇨ Either La Hague on-site conditioning (including compaction whenever possible).

Less radiotoxicity

HLW : As for volume, efforts on radiotoxicity were first focussed on this highly active waste. In fact, La Hague process allows a (U,Pu) recovery rate of 99.88%. Thus Fission Products only contains a maximum of 0.12% of the plutonium present in spent fuel.

Nevertheless, R&D still goes on in order to further decrease HLW radiotoxicity (advanced reprocessing, incineration).

ILW and LLW : This less active waste is constituted by a variety of low and intermediate level short-lived or TRU residues. It comprises resins and sludge from effluents treatment as well as operating and maintenance waste. Once more, the aim is to reduce both radiotoxicity and volume.

- ⇨ As for resins, a cement stabilizing process has been demonstrated and is in commissioning stage
- ⇨ As for sludge, optimization has already been done with the implementation of extensive effluents recycling, as part of the innovative and comprehensive five years program which has been described elsewhere.
- ⇨ As for α contaminated waste, optimization is underway with two units which provide a solution to remove most of plutonium contained in waste and scraps produced on the COGEMA La Hague plants and MOX fabrication plants, by means of an electrochemical leaching process.

Waste management optimization was first focussed on lowering volume and radiotoxicity of the most active waste. These still go on. As for less active waste, the ongoing program will allow volume minimization using most appropriate treatments and conditioning thanks to innovative management options (zoning, sorting) as well as technical ones (CENTRACO facility).
The continuous commitment to waste volume and radiotoxicity minimization at COGEMA plants is fed by both experience feedback from more than 30 years operating experience and R&D activities.
It results in going-on optimized waste management, implying use of updated techniques and staff involvement. It illustrates COGEMA strong will for a sustainable development, with a safer and cleaner future.

IAEA-SM-357/32

DECOMMISSIONING OF NUCLEAR FACILITIES: COGEMA EXPERTISE DEVOTED TO UP1 REPROCESSING PLANT DISMANTLING PROGRAMME

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Over the last past decades, the french nuclear industry has acquired a great experience and know-how in the field of dismantling. Today this experience amounts to more than 200 000 hours. The fundamental aims within dismantling strategy are the same as for all nuclear facilities:

- minimising doses received by workers
- minimising waste volume and adapting waste management to radioactivity levels
- minimising costs

French experience is based on technologies which are currently used in nuclear maintenance facilities. Dismantling is a dynamic process especially in the field of decontamination (chemical and mechanical), cleaning, robotics and remote control operations.

The strategy for the dismantling of former UP1 reprocessing plant is based on the feedback of experience gained through the dismantling of other facilities such as AT1 workshop at La Hague. This workshop, a pilot for reprocessing of fast-breeder reactor fuels (Rapsodie and Phénix) has to be dismantled to IAEA level 3 (unrestricted site use), excluding civil works structures. Currently conducted by trained shifts, this dismantling project should end in 1999. The experience already acquired proves that chemical rinsings with the use of specific reagents is sufficient to decontaminate the hot cells and that the use of remote operations or robotics is not as important as previously envisaged.

The UP1 reprocessing plant of Marcoule runned from 1958 to 1997. End of operation was pronounced on the 31st of December, 1997. 20 000 tons of spent fuels were reprocessed at UP1. The cleaning and dismantling operations at the Marcoule site depend upon the CEA, EDF and COGEMA. The Defence and Industry ministries asked for a specific structure to be set up. An economic interest group called CODEM was created in May 1996. Therefore CODEM decides, finances and supervises dismantling operations, while respecting the constraints of nuclear safety, environmental protection and cost-effectiveness.

The cleaning operations of the Marcoule site are divided into 3 main programmes :

- the final plant shutdown (MAD) of the UP1 plant and its associated facilities,
- the dismantling of facilities (DEM) leading to a final status of Installation Classified for the Protection of the Environment (ICPE)
- the processing of wastes (RCD) temporarily stored on site.

These operations should stretch over about 30 years. Final shutdown operations for the UP1 reprocessing plant were initiated in January 1998. The shutdown procedures were prepared by analyzing the experience acquired during plant operation. The results of each cleaning steps are analysed to confirm the hypotheses assessed during the preliminary studies. In addition these results are used to further improve subsequent procedures in order to achieve the shutdown under optimum conditions.

IAEA-SM-357/33

THE R7/T7 VITRIFICATION IN LA HAGUE: TEN YEARS OF OPERATION

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The La Hague site, located in Normandy, hosts the world's largest reprocessing plants for spent LWR fuels. The reprocessing contracts signed in the end of the 1970s between COGEMA and several utilities around the world specify that waste resulting from the reprocessing operation should be returned to their owner in a form suitable for transportation and final storage, i.e. in glass canisters.

Today, French vitrification process clearly appears as one of the best solution both in terms of industrial maturity and efficiency. The French vitrification process is based on the AVM (Atelier de Vitrification Marcoule) process which operates continuously and is the first successful industrial experience. The R7 and T7 vitrification facilities at La Hague are similar facilities. R7 was started-up in 1989 and T7 in 1992.

Each facility is made up of 3 lines implemented in individual cells and designed to produce about 25 kg of glass per hour in the canisters, corresponding to 800 tU reprocessed. The process has proved to be highly flexible since fine particles from the dissolution step have been incorporated to the glass matrix.

In addition, the R7/T7 glass specifications were also approved by the regulatory authorities of COGEMA's customers in Germany, Belgium, Netherlands, Switzerland and Japan. Results accumulated so far at La Hague show that HLW vitrification in France fully meets criteria of a successful industrial process. Moreover, the resulting glass meets all required quality standards..

Today, more than 6,200 canisters have been produced at R7 (over 3,700 canisters) and T7 (over 2,400 canisters). Since the active start-up of R/T7 facilities, vitrified canister compliance with specifications relies upon a complete quality assurance/quality control program including process control. Last, by using well-developed process, the R7/T7 facilities can safely produce more than two-third of the civilian vitrified HLW in the world.

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RATIONALIZING TRANSPORT OPERATIONS: THE TN 24 TRANSPORT STORAGE CASK APPROACH

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Introduction

As last year events have clearly shown, the issue of transport of nuclear materials is all the more vulnerable to hasty judgments from the public as it takes place on the public domain. The constant concern of transport casks designers and operators is improving safety and keeping the roads open.

Many actions contribute to that goal, in the fields of choice of material, of transport equipment, of routes, of training and procedures drafting etc. The one very significant contributor to rationalizing the ground transport operations is finding ways to diminish the number of these operations, especially for spent fuel and high level vitrified waste.

The approach presented here should meet the favor of competent authorities as this diminishes impact on the environment and on public acceptance.

The present paper presents the application of this principle of cutting the number of transports of spent nuclear fuel interim storage casks. It shows how it has been implemented in the TN 24 family of dual purpose casks. It shows that, for that purpose, standardization is more effective in terms of technology used rather than in terms of products.

How can diminishing the number of transport operations be achieved?

We proceed here on the rational basis that civil use of nuclear energy is beneficial to mankind. We reject the contention that the only way to diminish transport operations is to put an end to nuclear generation altogether. In fact this would not change the rationale of what we propose here for whatever needs be transported.

- First comes the diminution of quantities of radioactive materials to be transported. By careful choice of conditioning and sorting of materials, a first step in transport rationalization can be made. One clear way to this is going towards higher fuel burnups so as to diminish the number of reloads. Connected to that is reprocessing that actually allows a diminution of fuel handling operations, as it separates high level radioactive wastes and concentrates their volume. Transporting one glass canister avoids a transport of approximately 80 spent fuel assemblies!

  Most radioactive materials producers and users are working steadily on these approaches.

- Second comes the increase of payloads of individual transport containers: doubling the payload of a container will divide in two the number of transport operations to perform, and will improve safety. This improvement stems from the fact that potential consequences from transport accidents or malevolent attacks are proportional to the number of kilometers on the road or railway *but do not increase with the payload of a given container*.

  Why do they not increase?

The transport regulations are such that consequences of an accident are measured and limited with reference to the toxicity of the content, not with reference to the size of the content. In other words, if a given content has a toxicity index A_2, under the most severe accident condition, it may not release more than A_2 in one week. A_2 is a function of the isotopic composition of the material transported and *not* of the quantity of material transported. It follows that if a container contains ten times more than another one, this larger container may still release no more than the smaller one in case of accident.

Regarding radiations, the dose rate limits are as low for a large container as for a small one: it is therefore better to circulate less frequently larger containers, so that the cumulated dose uptake en route is diminished.

The example of the TN 24 casks family

It is this aspect that Transnucléaire has emphasized in the development of the TN 24 cask family of storage/transport casks listed below:

CASK NAME	For transport and interim storage of
TN 24 P	24 PWR spent fuel assemblies
TN 24 B	52 BWR spent fuel assemblies
TN 24 D	28 PWR 900 spent fuel assemblies
TN 24 DH	28 PWR 900 spent fuel assemblies
TN 24 XL	24 PWR 1300 spent fuel assemblies
TN 24 XLH	24 PWR 1300 spent fuel assemblies
TN 24 SH	37 PWR spent fuel assemblies
TN 24 G	37 PWR spent fuel assemblies
TN 52 XL	52 BWR short cooled spent fuel assemblies
TN 97	97 BWR spent fuel assemblies

As one can see from the above table, the contents can vary in a proportion of more than 50% in case of PWR fuel and almost 50% in case of BWR fuel.

How is optimization achieved?

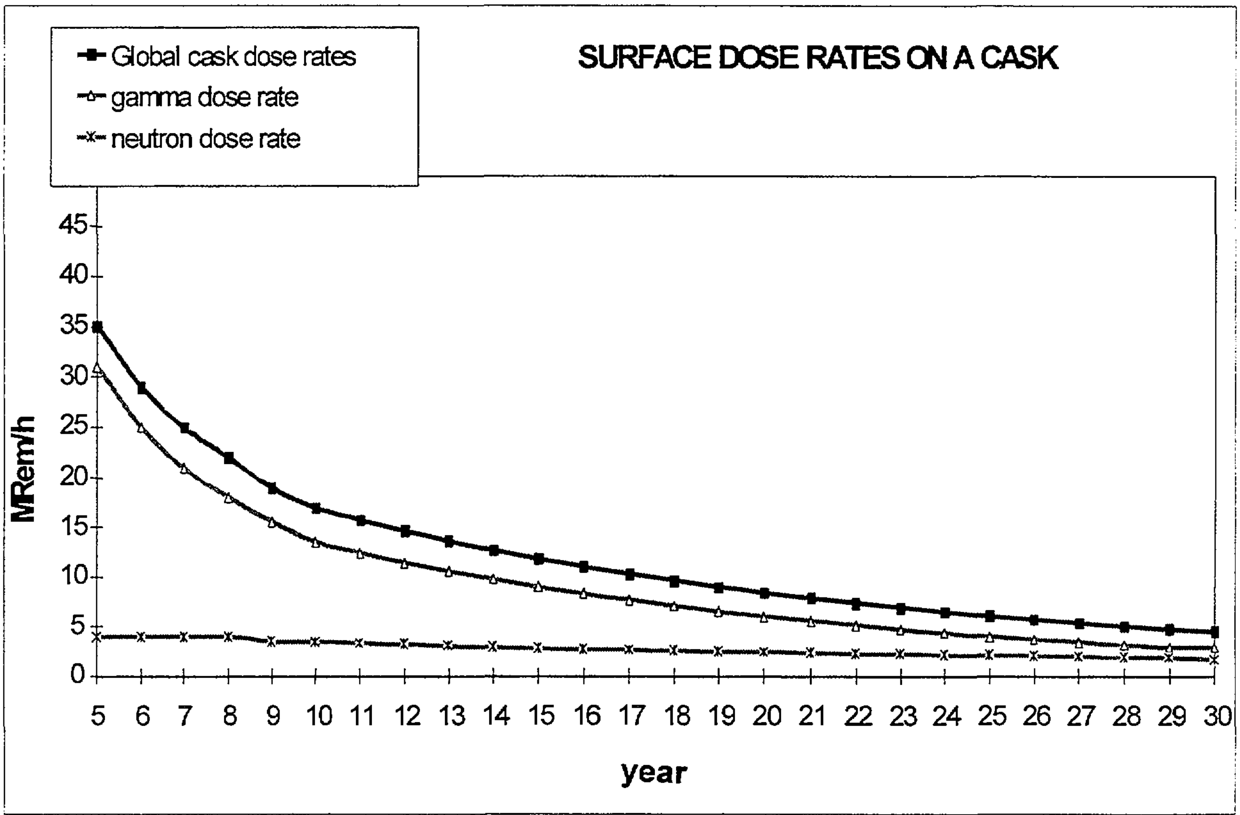
By carefully integrating fuel assembly specification differences in terms of burnup, decay time, initial enrichment and geometry, it has been possible to maximize efficiently the casks payloads.

This is based on two main technological choices:

Shielding

Shielding is designed so as to be able to uncouple neutron shielding issues from gamma shielding issues.

The neutron sources increase very quickly with burnup and have a very slow decay versus time, whereas gamma sources decrease exponentially with time, as shown by the curve below.



Uncoupling the shielding allows adapting readily to the actual need of the power plant and the set of fuel, and thus maximizes payload. In the TN 24 casks, the main gamma shielding is made of forged carbon steel, while neutron shielding is made from a neutron absorbing resin forming an outer layer on the gamma shielding. A steel outer shell protects the neutron shielding.

Because gamma radiation diminishes quickly with time, for a given initial global dose rate, a stronger neutron shielding will keep the global dose delivered lower than with a strong gamma shielding and a weak neutron shielding.

Criticality control and mechanical support of fuel assemblies

Baskets, that support the spent fuel and guarantee subcriticality, are basically boron aluminum structures.

This structure combines several advantages towards increasing the payload:

- Aluminum is a good heat conductor: less material is required to dissipate the decay heat while keeping low fuel cladding temperatures. The result is a smaller basket.

- Boron that captures moderated neutrons is distributed in the aluminum matrix at the best possible position for maximum efficiency. Furthermore, it cannot accidentally be separated from its aluminum matrix
- Aluminum alloys have a low specific density, hence the basket mass is minimal: this saves available weight for shielding purposes or for additional payload.

Conclusion

By aiming to improve the payload of transport/storage casks, a significant contribution is made to waste management.

This approach displayed here has been implemented steadfastly, and is being proven again by the current development of the TN 81 transport storage cask for high radioactive vitrified wastes that improves the capacity of such dual purposes casks by 40%.

These examples contribute in fact not only to limited handling and transport, but also to more compact interim storage facilities.

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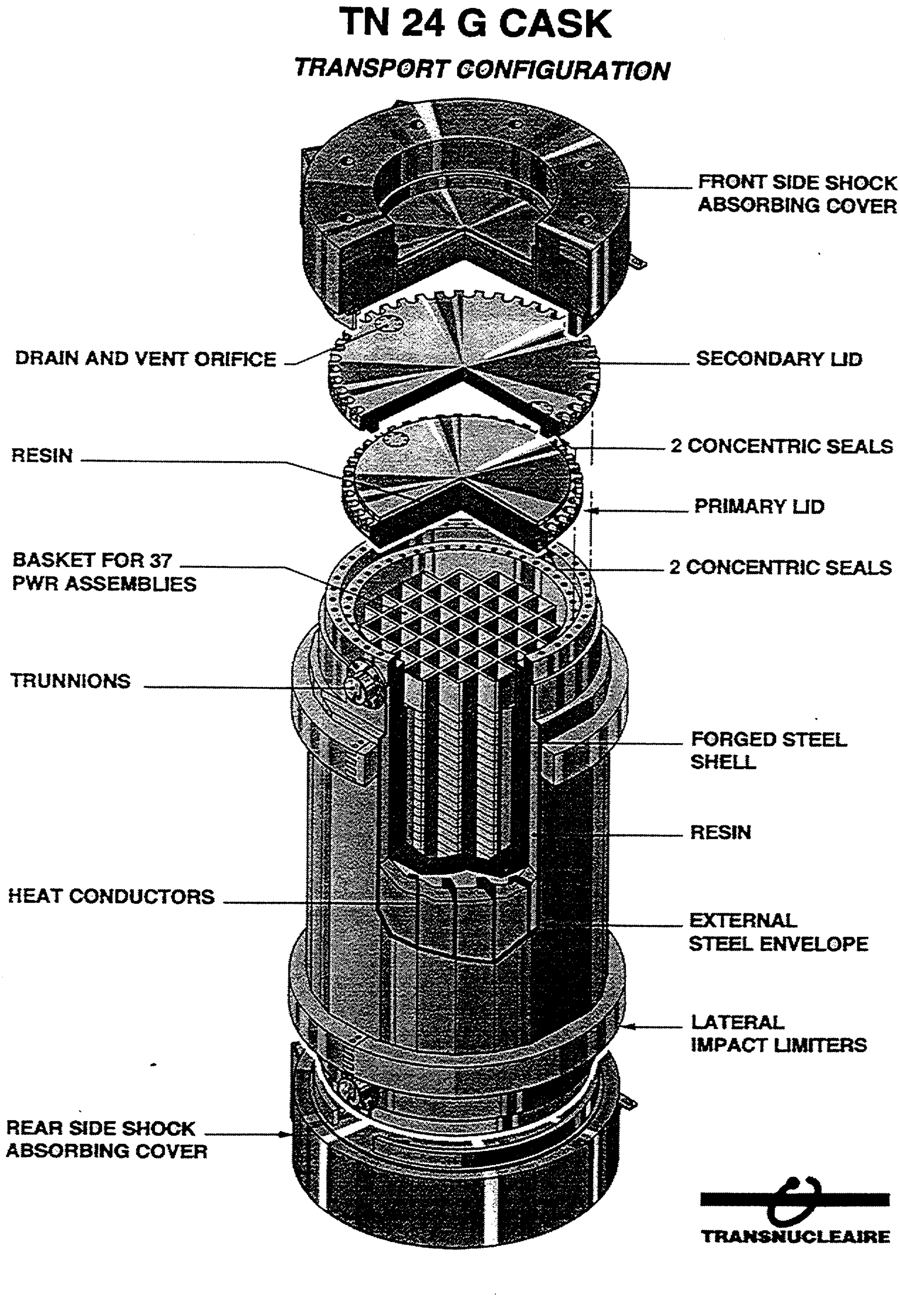

Figure 1:

The TN 24 G

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IAEA-SM-357/35

HUNGARIAN STRATEGY AND PRACTICE OF RADIOACTIVE WASTE MANAGEMENT

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In 1993 a National Programme was launched for the management of radioactive wastes of the NPP in Hungary. A complex strategy was elaborated, a site selection program for LLW/ILW started and research work began to prepare the disposal of HLW.

The new Act on Atomic Energy that entered into force on the 1 of June, 1997 regulated – among others – the performance and financing of tasks related to the radioactive waste management and decommissioning of nuclear facilities. As required by the Act the Hungarian Government set up a Central Nuclear Financial Fund and payments into this Fund started on January 1, 1998.

In 1998 - in line with the Act on Atomic Energy and its executive orders - the Director General of the Hungarian Atomic Energy Authority set up a non-profit company, the Public Agency for Radioactive Waste Management, which became responsible for the above mentioned tasks and continued the radioactive waste management projects started earlier. Recent results of these activities are the following:

- The re-evaluation of the safety of the repository used since 1976 for radioactive wastes from industrial, medical, scientific etc. applications is in preparation.

- Low level solid radioactive waste of the NPP Paks is -as far as feasible - compacted and stored on site.

- Liquid radioactive waste of the Paks NPP is stored on site and the possibility of distracting its boron content and conditioning it (to prepare concrete) is considered.

- Based on the investigations for a HLW disposal site in a claystone formation (Boda) construction and operation of an underground laboratory is under consideration.

- As no decision was taken yet on the back end of the fuel cycle, further 4 modules in the interim storage facility for NPP spent fuel are under construction.

- The siting process of a L/ILW final disposal facility will reach in 1999 a major decision making point, as a granitic formation in Üveghuta, supported by the local public could be selected for the establishment of a geological disposal facility. This decision is still pending on some further investigations, and it will require also the consent of the Parliament.

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OVERVIEW OF SYMPOSIUM ON STORAGE OF SPENT FUEL FROM POWER REACTORS

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1. INTRODUCTION

Continuous attention is being given by IAEA to the collection, analysis and exchange of information on spent fuel storage. Its role in this area is to provide a forum for exchanging information and to co-ordinate and encourage closer co-operation among Member States in certain research and development activities that are of common interest.

Symposia on this topic have been organised about once every four years since 1987. The purpose of the Symposium was to exchange information on the state-of-the-art and prospects of spent fuel storage, to discuss the world-wide situation and the major factors influencing the national policies in this field and to identify the most important directions that national efforts and international co-operation in this area should take.

The Symposium consisted of several oral sessions and one poster session. The oral sessions addressed four major topics: national programmes; technology; experience and licensing; and R&D and special aspects.

2. NATIONAL PROGRAMMES

It is noted that there continues to be world-wide growth in the generation of electric power using nuclear energy as its source. It is further noted that the rate of growth of nuclear energy generation has essentially levelled in Europe and North America while it has increased significantly in Asia. Although these trends have some impact on spent fuel management, including storage, the world-wide spent fuel production rate continues at about 10,800 t HM/y.

About 130,000 tHM spent nuclear fuel was stored around the world at 1 January 1998 (Table I). Over 70% (93,100 tHM) of this amount is stored in at-reactor (AR) pools in 32 countries, while the rest is in away-from-reactor (AFR) facilities, either wet or dry. Presentations from 20 countries in the session on National Programmes, and additional papers in the other sessions, covered 23 countries describing the technologies used to store more than 88% of the world total spent fuel to be stored.

TABLE I. STATUS OF SPENT FUEL STORED AT YEAR-END 1997

[kt HM]

Regions	AR	AFR Wet	AFR Dry	Total
West Europe	13.9	19.3	1.0	34.2
Asia & Africa	11.6	0.2	0.7	12.5
East Europe	7.8	9.9	0.3	18.0
North & South America	59.8	1.5	3.3	64.6
World	93.1	30.9	5.3	129.3

There are three major categories for classifying spent fuel management policies and practices. These include a closed fuel cycle which involves reprocessing of spent nuclear fuel, a once-through fuel cycle which , of course, ends with disposal of the spent nuclear fuel, and a "wait and see" approach.

3. TECHNOLOGY

The presentations on dry storage largely focused on the specific needs of different utilities and organizations whilst ensuring compliance with the stringent safety requirements applicable in the different countries. It was generally recognised that casks are needed to provide for both storage and transportation requirements. This flexibility is of great importance to meet requirements with regard to design work, licensing procedures and manufacturing work. Furthermore, cask designs have to accommodate different fuel types, including MOX fuel, higher burn-ups and specific needs of individual power plants. Another example is early consolidation and encapsulation of spent fuel in disposal canisters in Germany. Current cask designs are based on proven and cost effective technology.

4. EXPERIENCE AND LICENSING

In three sub-sessions, eight papers discussed regulatory and operational experiences with interim spent fuel storage. They described regulatory process and oversight, burnup credit analysis and measurements, and operational performance.

5. R&D AND SPECIAL ASPECTS

The information presented during the Session on "R&D and Special Aspects" can be grouped into 3 parts: Spent Nuclear Fuel (SNF) behaviour and properties; SNF treatment technologies; and International co-operation aspects.

6. CONCLUSIONS

There are three major categories for classifying spent fuel management policies and practices. These include a closed-fuel cycle which involves reprocessing of spent nuclear fuel, a once-through fuel cycle which ends with the disposal of the spent nuclear fuel, and a "wait and see" approach. One can view the decision to either reprocess or dispose as two ends of a spectrum of options. It should be noted, however, that countries, which choose originally the reprocessing option, envisage the final disposal of high burnup and MOX spent fuel. The "wait and see" strategy should not be viewed as avoiding a decision, but as a means of evaluating the possible options and maintain the retrievability of the spent fuel.

Messages retrieved from the Symposium are that the primary option for spent fuel will be interim storage for the next decades, the duration of interim storage becomes longer than earlier anticipated and the storage facilities will have to be capable for receiving also spent fuel from advanced fuel cycle practices (i.e. high burnup and MOX spent fuel).

It was noted that the handling and storage of spent fuel is a mature technology and meets the stringent safety requirements applicable in the different countries. However, it is performed in a flexible and dynamic way, continuously adapting to changes in nuclear policy and progress in technology, for example transportability of spent fuel, application of burnup credit and utilisation of advanced fuel types.

Wet storage remains dominant, even as the use of dry storage concepts increases. Wet storage is essential for cooling newly-discharged fuel, and will continue to be the method of storage used in connection with reprocessing. The industry has an extensive experience base in wet storage with an excellent performance record. Dry storage is being used increasingly, as more long-term storage of spent nuclear fuel is done. Dry storage may prove to be a cost-effective activity that can easily accommodate multipurpose systems (e.g., storage/transport, storage/transport/disposal).

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MANAGEMENT OF SPENT SOLVENTS OF REPROCESSING ORIGIN

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PUREX process, which forms the backbone of reprocessing activity, uses organic solvents (TBP in n-dodecane) for the extraction cycles. Severe conditions of extraction result in slow degradation of these solvents. Purification of this solvent by carbonate wash etc. helps in extending its repeated use in the extraction cycle. Eventually a stage is reached when the solvent starts affecting the extraction cycle adversely and hence is discarded as 'spent solvent'. The spent solvent being organic in nature, its management should not only be viewed from radioactivity consideration but also with regard to its chemical nature. Any acceptable process must address both these concerns.

Various options were tried out in our laboratory and were evaluated based on secondary waste volume generation, safety of the process and ease of operation. Although incineration could be a good option for organic wastes, its adoption gets limited in the present case due to the presence of phosphates in the solvent. TBP when incinerated results in generation of corrosive P_2O_5 which pose difficulty during off-gas handling.

TBP is susceptible to hydrolysis both by acids and bases. Hydrolysis of TBP using NaOH leads to formation of sodium salt of di-butyl phosphoric acid (NaDBP) and butanol. Both these products are soluble in water and hence are amenable to separation from n-dodecane. A treatment process based on 'Alkaline Hydrolysis' was developed for this spent solvent wherein the diluent is rendered free of activity and TBP. The diluent thus obtained can either be recycled after further purification steps, or if not acceptable can be easily incinerated. The products of alkaline hydrolysis along with the activity associated with them form a separate stream. This stream is compatible with cement matrices.

The alkaline hydrolysis treatment process adopted in this research center is described in this paper. The process leads to complete conversion of TBP to aqueous soluble reaction products, viz. sodium salt of di-butyl phosphate (NaDBP) and butanol. During the process, separation of the diluent (n-dodecane) virtually free of TBP and activity is also achieved . The overall reaction for the Alkaline Hydrolysis of TBP is:

$$(C_4H_9O)_3P{=}O + NaOH \longrightarrow (C_4H_9O)_2POONa + C_4H_9OH$$

The aqueous bottoms generated after hydrolysis are immobilised in cementitious matrices.

Studies were carried out to optimize process conditions for complete conversion of TBP and near total recovery of diluent (n-dodecane). Experiments included use of actual waste samples on laboratory scale (up to 7L) followed by demonstration of the process on plant scale (up to 200L) using inactive solvents. Studies were also carried out to immobilise the aqueous bottoms obtained after hydrolysis in cement matrices.

Results of experiments showed that about 5 –7 hrs. of reaction time at temperature of 100-115°C was required to achieve complete destruction of TBP. Table 1 gives results of a typical experiment showing that nearly 90% of TBP gets hydrolyzed in about 70 minutes followed by a reduction in the rate of hydrolysis. This indicates that the kinetics of hydrolysis is concentration dependent. It can also be seen that the radioactivity content in the top phase decreases with time. This showed that the activity reduction was mostly on account of destruction of TBP. Addition of water finally, helps in removing practically all the activity in the diluent phase. Traces of activity still remaining with the diluent phase can be attributed to the degradation products of the diluent, which are not water soluble.

Table 1

Feed Solution : 30% TBP in n-dodecane
Volume of feed : 7000 ml

Sr. No.	Time (minutes)	TBP hydrolysed (%)	Activity in diluent phase (Bq/ml)	
			Gross β	Gross α
1.	0	0.0	3330	370
2.	30	1.58	2775	12
3.	70	89.422	34	2
4.	120	98.693	8	2
5.	200	99.422	10	1.9
6.	300	100.00	2	1.8

The process parameters chosen for the operation of the demonstration plant were based on the observations during laboratory studies. Experiences during the operation of the demonstration plant have shown that all the set parameters could be maintained throughout the duration of reaction. The demonstration runs helped in establishing the process on a 200 lt inactive scale with respect to total conversion of TBP and recovery of the diluent . These runs have shown that greater than 99 % conversion of TBP and more than 98% recovery of n-dodecane is achievable.

Results of studies on immobilization of aqueous bottoms have proved their compatibility with cement matrices. Experiments carried out on 200L scale have established process parameters with regard to good homogenization of mix at 0.4 water to cement ratio. Leaching studies on active blocks have given acceptable leach rate values.

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INTEGRATED RADIOACTIVE WASTE MANAGEMENT FROM NPP, RESEARCH REACTOR AND BACK END OF NUCLEAR FUEL CYCLE- AN INDIAN EXPERIENCE

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India is one of the developing countries operating waste management facilities for entire nuclear fuel cycle for the last three decades. Radioactive liquid , solid and gaseous wastes of various categories are generated during operation of these facilities. The overall philosophy of waste management in India is to contain the activity to the extent possible and to release minimum possible radioactivity to the environment in accordance with ALARA principle to protect the worker, public and environment. This paper covers the waste management practices being adopted in India for treatment, conditioning, interim storage and disposal of low and intermediate level waste arising from the operation of nuclear power plant, research reactor and fuel reprocessing facilities.

Management of Liquid Waste

Radioactive liquid waste having different chemical composition and activity contents is generated during operation of nuclear facilities. Over the years, the low and intermediate level (LIL) liquid waste streams arising from reactors and fuel reprocessing facilities have been well characterised and different processes for treatment, conditioning and disposal are being practised. LIL waste generated in nuclear facilities is treated by chemical treatment processes where majority of the activity is retained in the form of sludge. The radioactive sludge is further concentrated using filters/centrifuge. The supernatant is subjected to ion exchange treatment before being discharged to the environment after monitoring. Decontamination factors ranging from 10 to 1000 are achieved depending upon the process employed and characteristics of the waste. At an inland PHWR site at Rajasthan, the LIL waste is concentrated by solar evaporation because of favourable meteorological conditions. The facility handles about half of liquid waste generated at that site. It has effectively restricted the release of tritium activity to the environment. To augment the treatment capability , a plant is being set up at Trombay to treat LIL waste based on reverse osmosis process.

Ion exchange resin based on resorcinol formaldehyde has been developed which is very specific for uptake of radiocaesium from the waste. Alkaline waste of intermediate level activity and spent fuel storage bay pool water are being treated by such resin on regular basis at Tarapur.

Management of Solid Waste

Solid radioactive waste is assayed , segregated and packaged suitably at the source itself. Depending upon their physico-chemical properties, these wastes are volume reduced by compacting, baling and incineration. Cement matrix is employed for immobilisation of process concentrate such as chemical sludge, ash from incinerators etc A facility has been operating for conditioning of spent resin in polymer matrix. The solid waste depending on the activity contents is disposed in underground engineered trenches in near surface disposal facility. The waste upto 2 mGy/h is disposed in brick/stone lined earth trench whereas, in RCC trench , waste upto radiation dose of 500 mGy/h is disposed. Tile hole is employed for disposal of waste of higher radiation field. Alpha bearing waste (>4000 Bq/gm) is presently being stored whereas waste with lower levels of alpha contamination is disposed along with beta gamma waste. The trench after the utilisation of its capacity is suitably sealed to avoid the seepage of ground/surface water. Bore well samples around the trench are drawn periodically to ascertain the effectiveness of the disposal system.

Management of Gaseous Waste

Unlike liquid and solid wastes which are managed by a central facility, the gaseous waste is treated at the source itself. High efficiency particulate air (HEPA) filter using micro glass fibre filter media is the last barrier between the plant and the environment to restrict the release of airborne particulate activity. Coconut shell based activated carbon impregnated with KI+KOH is widely used in India to minimise the release of airborne radioiodine to the environment during normal as well as accidental conditions.

The overall radioactive waste management programme in India is being persued encompassing research, development , design, construction and operation of facilities at all sites under nuclear fuel cycle. Radioactive waste discharges are kept well below the authorised limits prescribed by the regulatory authorities

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RADIOACTIVE WASTES MANAGEMENT AT NARORA ATOMIC POWER STATION IN INDIA

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1.0. INTRODUCTION

Narora Atomic Power Station is the fourth Nuclear Power Station having twin units each of 220 MWe capacity. It is situated on the bank of river Ganga at Narora in District, Bulandshahr (U.P.) & is 140 km. away from New Delhi, the Capital city of India. It is the first standardized and totally indigenous PHWR type Nuclear Power Station.

2.0. RADIOACTIVE WASTE MANAGEMENT

The waste management facility at Narora Atomic Power Station provides facility for segregation, collection, treatment, storage & safe disposal of liquid & solid radioactive waste. The philosophy of waste management is based on the principle of ALARA(As low as reasonably achievable) taking the economic and social factor into account. Three principles governing the management of radioactive wastes are (i) dilution and dispersal of low level wastes (ii) delay, decay and dispersal of waste containing short lived radio-nuclides and (iii) concentration and containment of high active wastes containing long lived radio nuclides after conditioning. The waste management facility is functionally divided into following four systems.

2.1. LIQUID EFFLUENT SEGERATION SYSTEM

This system is located in service building & provides collection & segregation at source of all the liquid wastes generated in the station based on level of activity & chemical nature so as to

- Minimize cross contamination
- To facilitate judicious decision in respect of management of each category of waste.

In this system, control over waste volume & activity generation is imphacised at source itself. Liquid waste received is sampled, monitored and pumped for treatment or dilution & discharge depending on activity content. All liquid waste carrying lines are of Stainless Steel & hydrotest at regular intervals are carried out to ensure it's integrity.

2.2 TREATMENT & DISPOSAL SYSTEM

The basic philosophy of various techniques such as chemical co-precipitaiton, flocculation & sedimentation is to concrete and contain as much activity as possible, prior to their discharge in an environmentally acceptable manner. Decontamination by chemical treatment involves co-precipitaiton using phosphates, ferro-cyanides, hydroxides etc. for effective removal of radio nuclides like Cs^{137}, Cs^{134}, Co^{60} & Sr^{90}. For dissolve activity decontamination factor of 10-20 & for suspended activity, decontamination factor of 90-100 is achieved. Low active organic waste is soaked in vermiculite & disposed off in drums as solid waste.

Treated waste is stored in post-treatment tank. Finally, it is re-circulated for homogenization, sampled & after filtration is injected into the condenser cooling water blow down line for dilution. Diluted waste is released to the flowing canals. On line proportional sampler is provided at the final outlet to check the specific activity level in diluted waste water being released. Sludge generated from the chemical treatment forms a part of solid waste. Activity released through liquid route is maintained well below Tech. Spec. limit and accounts for 5% & 50% of Tech. Spec. limit of gross Beta -Gamma and Tritium respectively.

2.3. SOLID WASTE MANAGEMENT SYSTEM

Radioactive solid waste generated consists of contaminated process equipment parts, protective clothing, used particulate filters, concentrated sludge, spent resin, cotton, papers & packaging materials. Control over intermixing of inactive waste with active waste is exercised to restrict the active waste generation. All radioactive solid waste as received is categorized on the basis of dose rates and physical characteristics of the waste depending upon the treatment and handling processes. These are as follows:

1. Waste with contact surface dose rate $\leq$ 200 mR/h.
2. Waste with contact surface dose rate > 200 mR/h but $\leq$ 2 R/h.
3. Waste with contact surface dose rate > 2R/h.

The waste collection at the source is being done to achieve proper segregation at the origin itself. For this purpose, administrative procedure and control is being exercised. Category-I waste is collected in unshielded standard drums with polythene bags inside. These bags are sealed after it is filled with wastes and drums are capped before despatch to solid waste management facility. category-II waste is collected in drums provided with local shielding, whereas category-III waste mainly consisting of spent resins is collected in adequately shielded casks & transported to solid waste management facility in a dedicated truck provided for this purpose. Conditioning /treatment of solid waste depends upon it's nature as mentioned below:

1. Non-combustible, compactible are baled to reduced its volume by factor of 5.
2. Combustible waste is incinerated to reduce it's volume by a factor of 35-40.
3. Spent filters and metallic parts are embedded in cement.
4. Sludge is immobilized in cement & vermiculite.
5. Spent resins are immobilized in polymer matrix.

As far as disposal of the waste is concerned, these are disposed in engineered container near surface disposal facility at plant site. Very low level waste package (2.5 mR/h.) are disposed in the vermiculite lined earth trenches. The waste package/containers having dose rate from 2.5 mR/h. to 50 R/h. are disposed in reinforced cement and concrete vaults whereas those having dose rate > 50 R/h. are disposed in retrievable steel lined high integrity containers.

2.4. GASSIOUS WASTE MANAGEMENT

An extensive ventilation system consisting of pre-filter and HEPA filters collects potentially active exhaust air from such areas as Reactor Building, spent fuel handling and storage area, the decontamination center and the heavy water management area. The active and potentially active exhaust air is routed to a gaseous effluent exhaust duct. This exhaust flow is monitored for noble gases, tritium, iodine and active particulates before being released to the atmosphere through 145 Meter high stack. Signals from the iodine, wide range gamma and particulate monitors are recorded in control room. Tritium monitoring is carried out by laboratory analysis of bubbler samples. Activity releases are maintained well below technical specification limit.

3.0 ENVIRONMENTAL MONITORING

Around solid waste disposal area, regular monitoring of underground water is being done by a series of monitoring bore wells to check the integrity of waste management facility. For assessment of environmental impact, a fully equipped environmental surveillance and micro meteorological laboratory is established which is functioning under directorate of health and safety, Bhabha Atomic Research Center. It carries survey upto 30 km. radius from Plant site & analysis of more than 1000 samples of water, soil, vegetation & food material are carried out. So far NAPS has not shown any environmental impact in its surroundings. The maximum dose to the public is 1-2% of the limit of 100 mrem/year recommended by International Commission on Radiation Protection (ICRP).

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XA9952262

IAEA-SM-357/41P

CHEMICAL ANALYSIS OF IRRADIATED ACTINIDES FOR TRANSMUTATION OF HIGH-LEVEL WASTE

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One of the options for high-level radioactive waste management is to transmute the actinide wastes into short-lived fission fragments by neutron-induced fission. In order to quantitatively understand the transmutation of minor actinides in neutron irradiation location, it is essential to obtain precise nuclear data on their neutron reactions.

In this study, the neutron capture cross sections and fission yields of americium nuclides have been measured radiochemically by alpha- and gamma-ray spectrometries. Several samples of uranium, neptunium and plutonium nuclides irradiated in a fast neutron reactor have been also analyzed to determine the contents of actinides and fission products in the samples.

From the viewpoint of nuclear waste management, the transmutation process of the actinides in reactor is discussed quantitatively by calculating the yields of minor actinides with the data measured in this study.

Highly-purified targets of Am-241 and Am-243 were irradiated in research reactors. The neutron capture cross sections of the nuclides have been radiochemically measured by alpha-ray spectrometry. Analyzing the gamma-ray spectra of the irradiated actinide targets and the chemically-separated fractions measured with a HPGe detector, the yields of the fission products are obtained. Table 1 shows the neutron capture cross sections of Am-241 measured in this study.

Table 1. Neutron capture cross sections of Am-241

Reaction	Cross section, b σ_0	I_0
$^{241}Am(n,\gamma)^{242m}Am$	85.7 ± 6.3	114 ± 7
$^{241}Am(n,\gamma)^{242g}Am$	768 ± 58	$1,694 \pm 146$

Enriched nuclides of uranium, neptunium and plutonium were irradiated in prototype fast reactor (PFR) [1]. Isotopic concentration measurements of the actinides and fission products formed by the irradiations were performed by chemical separation, alpha- and gamma-ray spectrometry and mass analysis. Table 2 gives the isotopic composition of Pu-244 sample irradiated during 2.7 years.

During the neutron irradiation, both the actinide target and the minor actinides produced change the atom numbers because of nuclear reactions such as capture and fission reactions and the decays as seen in Table 2. From the nuclear data of minor actinides, it is possible to understand more quantitatively the nuclear transmutation process due to the neutron capture reaction and fission on the actinide samples in several kinds of the neutron fields.

Table 2. Nuclear transmutation of Pu-244 sample irradiated during 2.7 years in PFR

Nuclide	Composition, % before irradiation	after irradiation
Pu-239	0.04	0.03
Pu-240	2.8	2.3
Pu-241	0.4	0.3
Pu-242	9.0	7.7
Pu-244	87.1	77.0
Am-241	0.6	0.4
Am-242		0.01
Am-243		0.7
Cm-244		0.1
Cm-245		4.2
Cm-246		0.2
Cm-247		0.01
fission		7.1

By mathematical procedure to calculate the time-derivative of the atom number on the nuclear transmutation of the actinides [2], the formation and decay processes of the minor actinides can be calculated using the nuclear data measured in this study. On the basis of the calculated values, the nuclear transmutation of Am-241 by the neutron-induced reaction can be understood quantitatively: 93.3% of the atom number of Am-241 (target) is transmuted to other nuclides after one year irradiation of thermal neutrons with flux of 10^{14} n/cm^{2}s in a reactor. The transmuted amount goes to Pu-238 (20.1%), Pu-239 (0.0001%), Pu-240 (0.001%), Am-242m (0.08%), Am-242g (0.04%), Am-243 (4.3%), Cm-242 (22.8%), Cm-243 (0.52%) and Cm-244 (0.13%), while the remaining percentage (45.5%) can be attributed mainly to the fission products. In order to quantitatively consider the transmutation of the minor actinides, it is essential to obtain the precise nuclear data, because the uncertainties of the calculated values of the transmuted amounts strongly depend on the nuclear data used.

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RADIOACTIVE WASTE MANAGEMENT ON THE ATOMIC ENERGY ENTERPRISES OF THE REPUBLIC OF KAZAKHSTAN

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There are many enterprises of atomic branch operated at the territory of Kazakhstan during more than 30 years. They are: nuclear power plant BN-350 in Aktau city, Ulba metallurgical plant producing nuclear fuel for NPP, four research reactors of National Nuclear Centre of the Republic of Kazakhstan. One of them is located in Almaty, and three of them are located in Kurchatov city (on the former Semipalatinsk Test Site).

Since 1991, after getting of the sovereignty, Kazakhstan started developing of its own legislative and regulatory system in the field of atomic energy use. In accordance with the Decrees of the President appropriate structures in Kazakhstan were created. They are: Atomic Energy Agency, as a main supervising governmental body, National Nuclear Centre combining all nuclear-related scientific institutes, and Corporation of Atomic Energy and Industry Enterprises KATEP. On 14 February 1994 Kazakhstan joined the International Atomic Energy Agency.

According to existing infrastructure the Republic of Kazakhstan needs the effective system for the assurance and guarantees for protection of population and environment against the possible negative influence of atomic energy usage. Conception of radioactive waste storage in Kazakhstan was elaborated as a first step [1]. The data connected with the total volumes of radioactive waste, classification of waste and the appropriate sites for the waste disposal had been determined, conducted and indicated. The quantitative criteria and the principle approaches for radioactive waste storage had been formulated based on the international experience, recommendations of IAEA, and existing safety rules and norms. The Conception contains main directions for the radioactive waste management: preliminary selection of sites for storage facilities, determination of the technology of collection, processing, transport of the radioactive waste, management and control of radioactive waste facilities.

In the legislative field of the safely use of nuclear energy several Laws of the Republic of Kazakhstan have been elaborated and adopted, they are: Law on Atomic Energy Use (14 April 1997 [2]), Law on Radiation Protection of Population (23 April 1998 [3]), Laws on the Environment Protection, on Licensing, on the Bowels (of the earth) and Bowel Use, on Social Protection of the Citizens Damaged from Nuclear Tests at the Semipalatinsk Test Site, on the Sanitary-Epidemiology Happiness of Population, on the Export Control of Arms, Military Engineering and Production of Double Purposes. The drafts of the Law on Radioactive Waste Management and Main Rules of Transport of Radioactive Materials are prepared for enforcement procedure now. The Norms of Radiation Safety NRB-96 [4] elaborated by specialists of Russian Federation and Republic of Belarus taking into account the international safety requirements [5] had been adopted in Kazakhstan in 1997.

Republic of Kazakhstan have signed and joined such international treaties and conventions as 1968 Treaty on the Non-Proliferation of Nuclear Weapons, Convention on Nuclear Safety, Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Thus, Republic of Kazakhstan took obligations to execute the International requirements for assurance of safe atomic energy use.

There are several practical works that should be noted. First regional storage for sealed radioactive sources has been entered into operation in Kurchatov city. Project on long term dry-storage of BN-350 spent fuel is under practical implementation now.

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THE PRESENT STATE AND FUTURE PROSPECTS OF THE RADIOACTIVE WASTE AND SPENT FUEL MANAGEMENT IN LITHUANIA

XA9952264

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The Ignalina NPP is the main source of the radioactive waste (RW) in Lithuania. Only a small part of the RW occurs in the medical, science and industrial institutions. Almost all RW is stored now at the Ignalina NPP site, and only the small part of them (mainly Low and Intermediate Level Waste) have been collected (until 1989 year) at Maishagala village nearby Vilnius.

Ignalina nuclear power plant has two RBMK-1500 graphite moderated channel type reactors, operating at reduced power (from 4800 MWth to 4200 MWth). The first reactor was put into operation at the end of 1983 year, the second one - in 1987 year. Ignalina NPP uses uranium dioxide fuel with 2,0% and 2,4% enrichment. Initially it was planned to return the spent fuel to Russia, from which the fresh fuel was supplied. Unfortunately after the collapse of former Soviet Union this way was closed, and now Lithuania is fully responsible for the management and disposal of its RW and spent fuel itself.

Ignalina NPP RW consists of solid and liquid waste, ion exchange resins and used lubricants (Table 1).

Table 1. Radioactive waste of Ignalina NPP

Waste type	Waste production, m^3/y
Solid	2 000
Contaminated water	240 000
Laundry waste	15 000
Evaporator concentrate	815
Spent ion-exchange resins	22
Perlite	14
Bitumenized waste	850
Spent nuclear fuel	50-80, MTU/unit

Solid waste without conditioning is placed into the special concrete vaults, which are located near the plant. Removable roof covers the upper part of the vault. Such method can be treated only as a temporary solution for the solid waste disposal.

Liquid waste is collected in the special tanks, and from them is directed to the evaporation station, which consists of two identical parts (Fig.1). The concentrate of liquid waste is mixed with bitumen and delivered to the storage place located nearby plant. The present scheme of liquid waste treatment is very complicated and incomplete. It has to be modernized in the near future. For such purpose we have proposed the original idea - cellular foam apparatus (Fig.2).

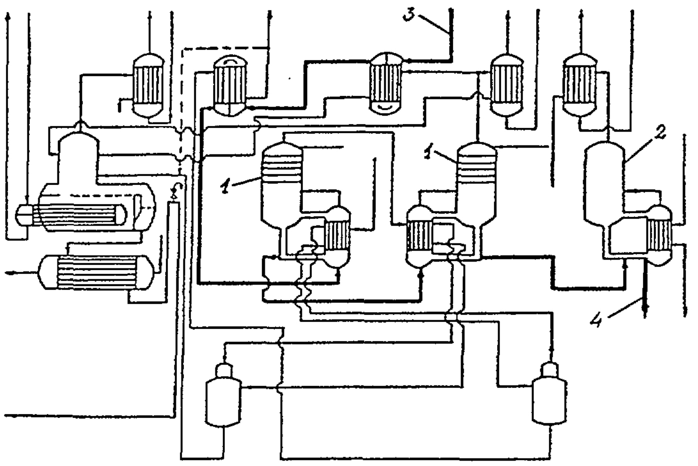

Fig. 1. Liquid waste evaporation station.
1 - main evaporators; 2 - final evaporator; 3 - initial liquid waste; 4 - concentrate to bitumen apparatus.

Lithuania, similar like some of other countries, decided not to reprocess the spent fuel from its nuclear power plant, and the spent fuel is stored now in the special water pounds nearby the reactor site. There are two such pounds for each reactor with a common volume equal to 1380 m^3.

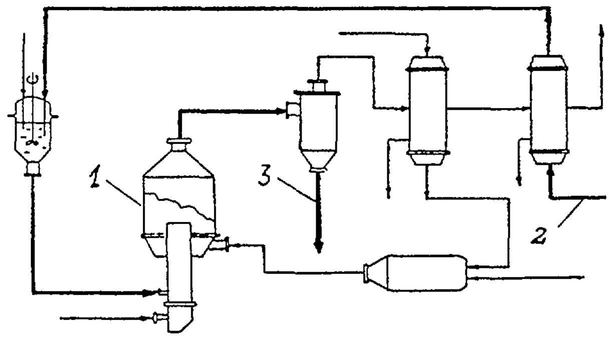

Fig. 2. Proposed evaporation station.
1 - foam apparatus; 2 - initial liquid waste; 3 - dry waste to bitumen apparatus.

Ignalina NPP operating age is fifteen years now. From 50 up to 80t uranium of spent fuel per unit every year is placed under the water in the pounds. Spent fuel assembles after discharging them from the reactor core are placed into the water pools for one year. After that they are taken out and cut in two parts inside the "hot chamber". Parts of assemblies are placed into the special baskets (51 assemblies in each basket) and sink into the water pounds for the temporary storage. The space of pounds is almost fully filled now by spent fuel baskets.

Future management policy of the spent fuel is to use dry storage cask. In December 1994 Ignalina NPP signed a contract with Germany Company GNB on the supply of the first part of the iron cask CASTOR for the dry storage of the spent nuclear fuel.

This type of the cask is supplied now to Ignalina NPP (20 CASTOR casks have been delivered already). It is very likely that the next supplier will be the Canadian Company AECL, which have proposed MACSTOR system for Ignalina NPP.

XA9952265

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RADIOACTIVE WASTE MANAGEMENT REGULATORY FRAMEWORK IN MEXICO

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The Mexican Federal Nuclear Law, assigned to Mines and Energy Ministry, at present Energy Ministry, the responsibility of radioactive waste management, and the regulatory activities related with the radioactive waste management is in charge of the National Commission of Nuclear Safety and Safeguards (CNSNS).

In 1988, CNSNS issued a General Radiation Protection Regulation that establishes basic radiation protection rules, dose limits, licensing process for radioactive materials and specifically for radioactive wastes establishes only generic points about their classification and the licensing process for temporal and definitive storage. This regulation does not establish specific technical criteria for the different stages of radioactive waste management such as criteria for their classification and characteristics for temporal and definitive storage, requirements related with their treatment and conditioning, and requirements for definitive storage facilities of radioactive wastes, however the regulation underlines that specific requirements shall be established in national standards.

The radioactive wastes produced in Mexico from nuclear powers plants arising from the two reactors at Laguna Verde Nuclear Power Station (LVNPS), at present do not exist another facilities related with the nuclear fuel cycle. The type of radioactive waste arising from LVNPS are low level radioactive waste and the spent fuel. The operation of LVNPS generates, average per year, 200 m^3 of low level radioactive wastes and they are stored temporarily in on-site facilities; while the spent fuel production is about 92 assemblies per reload and they are stored, also temporarily, in the reactors pools, whose capacity is for all useful life of the reactors.

Taking into account the capacity of on-site temporal facilities and the generation of low-level radioactive wastes, is probably that approximately within six years its will reach their capacity, and thus there is the possibility that a short term it will begin the construction of definitive storage facility for these type of radioactive waste.

As pointed above our regulation does not have specific criteria for regulating the activities related with the radioactive waste management, so it was necessary to begin establishing national standards and guides to accomplish the legal regulatory framework concerning with the final storage for low-level radioactive waste, with the objective of licensing, controlling and surveillance this type of activities. Basically, these standards considering:

a) Criteria for the classification of radioactive waste produced by nuclear industry regarding with its handling, treatment, conditioning, temporal and definitive storage

b) The methods for determining the activity, activity concentration, and identification of the radionuclides contained in a radioactive waste package, in order to get information on the material within radioactive waste package. This information will be useful for its treatment, conditioning process and definitive storage.

c) Requirements that the waste package must comply in order to be accepted in a near surface definitive storage facility.

d) The requirements under which a water leaching test must be performed in radioactive waste species. Such requirements include the establishment of a leaching index for those radionuclides released from solidified radioactive waste packages under controlled conditions and with a defined leaching agent.

e) The requirements for the site selection process, design, construction, operation, closure and institutional control for a near surface definitive storage of low-level radioactive waste facility.

The program of implementing these standards is according to characteristics of the wastes produced in México.

The criteria and requirements mentioned for the above standards are based on the recommendations established by the IAEA in the Safety Series No. 111 and the NRC regulations established in 10CFR part 61.

At present it is working in the elaboration of rules related with the criteria and requirements for clearance, from regulatory control of radioactive waste and the requirements for radioactive waste treatment, conditioning facilities. With these last rules, we think that legal regulatory framework will be full up for low-level radioactive waste management.

Concerning with the spent fuel, at present does not define the situation about which will be the final way.

XA9952266

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STATUS OF THE NATIONAL LOW LEVEL RADIOACTIVE WASTE REPOSITORY SITING STUDY

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The process of site selection for the repository of low level wastes was initiated in 1976 when the Philippine Government decided to go nuclear and constructed the 1st Philippine Nuclear Power Plant in the Bataan Peninsula. With the mothballing of the nuclear plant and the eventual final decision to convert the plant into a combined cycle power plant, the siting activities were suspended until May 1995 with the creation of a Nuclear Power Steering Committee. The Inter-Agency Committee is tasked to provide policies, directions, monitoring, evaluation and other functions necessary and appropriate to attain the objectives of the Philippines' nuclear power program. The Philippine Nuclear Research Institute is the lead agency tasked to study and identify a short list of potential candidate sites all over the country.

The paper describes the national effort to revise and update the established site screening criteria based on the important technical, environmental and social conditions currently prevailing in the country as well as developments in international standards and criteria for siting activities. Initial results of the site characterization study for a near surface disposal of radioactive wastes will be presented. A description of the site assessment methodology including, among others, site screening, scoring and ranking of candidate sites will also be described.

Safety assessments and analysis of the impact to the critical population using available computer models will also be presented. Preliminary results involving ground water pathway for a generic site with characteristics and proposed facility design for the Philippines will be briefly described including current work activities to determine the effect of the variation of distribution coefficient values and groundwater characteristics, among others, to the maximum dose at some point in time. These activities are intended to generate useful information for the regulatory authority to determine the critical site parameters and to establish an appropriate acceptance criteria for a disposal site from the point of view of radiation safety.

Activities concerning public acceptance/information program for selected areas and target audiences are currently pursued and will also be described. Lessons learned on national surveys conducted will be discussed and recommendations for additional work activities will be highlighted. The paper will also present an overview of the future work programs, directions and thrusts in the national nuclear power program.

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AN OVERVIEW ON THE NUCLEAR SPENT FUEL MANAGEMENT IN ROMANIA

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The sources of radioactive waste in Romania are users of radiation and radioactive materials in industry (including nuclear electricity generation), medicine, agriculture and research and also the processing of materials that are naturally radioactive, such as uranium ores, thorium associated ores and phosphate fertilizers. The different types of radioactive waste are classified into four categories of waste: excepted waste, low level waste, medium level waste and high level waste.

The Romanian Atomic Act, Low No.111/1996 on the safe deployment of nuclear activities has been in to force and Law provides the legal requirements for radioactive waste management. According to this Law, the waste producer bear the responsibility for the management of his radioactive waste and also for the financial and material arrangement for covering the collection, transport, treatment, conditioning and disposal of waste arise from deploying his activities and also for decommissioning of his facilities. The licensee shall pay a legal mandatory tax for financial contribution to the Radioactive Waste Management and Decommissioning Fund.

In Romania, for the moment, it is not the intention to reprocess the spent fuel that is considered as a waste, high level waste.

In Romania the spent fuels are produced both in research area and in NPP.

a) Research facilities

- *Institute for Nuclear Research(ICN) from Pitesti has in operation a* TRIGA type reactor, an American one, is in use here, since 1978.

- The Institute of Physics and Nuclear Engineering "Horia Hulubei" (IFIN-HH) is responsible for the waste management from its VVRS research reactor.

b) Cernavoda NPP

Starting 1996 Romania becomes the 30th country operating nuclear power plant. The Cernavoda Unit 1 equipped with CANDU type reactor satisfies about 9% of the Romania's needs of electricity. There are other 4 units already erected in Cernavoda NPP site. Only Unit 2 is preparing to start the work for to be finished. The spent fuel which retains 99% of total radioactivity will be stored for 7 - 10 years in a special concrete epoxy lined bay, called Spent Fuel Storage Bay. There is not an intermediate storage facility for the spent fuel.

The Spent Fuel Management Subprogramme as a part of Radioactive Waste Management Programme was initiated by the former RENEL (the Romanian Electricity Company in 1992. This Subprogramme had been included studying of the main methods and the existing technologies in the area of the design, operation and safety for an Interim Storage Facility (including transport aspects); also analysis upon the site selection for this facility and for spent fuel Final Disposal Facility.

The first objective that must be achieved in the back-end of the fuel cycle program is the performed of the Spent Fuel Intermediate Storage Facility. The analysis developed related to the spent fuel quantity stored in the S.F.I.S.F. have proposed the modular concept for building the facility, because the schedule for commissioning the N.P.P's Units 2 ÷ 5 is uncertain yet, and, also, for distributing the investments according to the real storage requirements.

First of all we have studied the possibility to place this Interim Storage Facility on the site of Cernavoda N.P.P. or in their neighborhood. As a conclusion of these studies (including geological aspects), the optimum site resulted to be inside N.P.P. boundary. This selection has as a direct advantage the avoidance of the transport on public roads and the simplification of the transport system.

The position close to Unit 5 of Cernavoda NPP on natural strata cannot make a significant perturbation of other activities into this area and have a benefit of existing road, utilities and other services, etc.

We have studied some solutions being in operation in the world and we have selected 3 variants that we have taken into consideration for choosing the best solution. The comparation analyze from economical, technical and safety point of view is on way.

Some studies concerning conceptual design for spent fuel disposal, material performances and geological assessment of some geological formations, especially saline formations have been included. After the preliminary studies six potential salt formations suitable for the site selection were determinate.

We are expecting that the experience gained during the design, construction and commissioning of Cernavoda NPP's first Unit could be an important factor for the cost reduction in the area of the design and project management for an Interim Storage Facility.

Within the frame of the subprogram 4, some studies concerning conceptual design, natural performances and geological assessment of some dupe geological formations, have been included.

The Within the frame of the subprogram 4, some studies concerning conceptual design for spent fuel disposal, material performances and geological assessment of some geological formations, especially saline formations have been included. After the preliminary studies six potential salt formations suitable for the site selection were determinate.

The milestones of our spent fuel management activity is now represented by a first phase for a medium period for spent fuel storage under safe conditions, so that the producers may provide protection of personnel, population and environment according to the responsibilities stated by Law No. 111/96.

The deadline for commissioning the Spent Fuel Interim Storage Facility is 2005. The achievement of this objective is on the way. The results from the studies performed in the last years will permit us to prepare the feasibility study next year and the documentation asked by our Regulatory Body for starting the process for obtain a site license of SFISF at Cernavoda.

A second phase is represented by the assessment of a strategy for long-period to use and adopt a proved disposal technology for the spent fuel, in direct correlation with a selected site.

Now, we are in progress with the preparation of the Feasibility Study for Cernavoda SFISF. Some different modes in which the problems are going to be solved and the results obtained until the Symposium time, will be presented in the paper.

IAEA-SM-357/48P

FUEL CYCLE OF BREST-1200 WITH NON-PROLIFERATION OF PLUTONIUM AND EQUIVALENT DISPOSAL OF RADIOACTIVE WASTE

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Serious expansion of nuclear sources - by an order of magnitude against the current level - can be achieved only around fast reactors in a closed fuel cycle. Large plutonium stockpiles accumulated in the first stage of nuclear power development, dictate the use of fast reactors with uranium-plutonium fuel, which have serious advantages over other reactor types and the thorium-uranium cycle.

The geography and scale of energy supply anticipated in the next century, impose new requirements on nuclear reactors and closed fuel cycle technology, in particular:
- full Pu reproduction in the core with BR~1. The slowdown in the expected rate of capacity growth and large amounts of accumulated plutonium (from water cooling reactor), eliminate the need for quick doubling of plutonium, which allows the use of reactors with BR~1 and moderate power density in the core;
- natural safety of reactors with deterministic exclusion of the most dangerous accidents such as prompt runaway, loss of coolant, fire, steam and hydrogen explosions, which lead to fuel failure and catastrophic release of radioactivity;
- lower radiation risk from radwaste (RW) owing to the transmutation of the most hazardous long-lived actinides and fission products (FP) in reactors and thorough treatment of RW to remove these elements, with provision of a balance between the activity of RW put to final disposal and that of uranium extracted from earth;
- facilities of a closed fuel cycle should not be suitable for Pu extraction from spent fuel for the purpose of its further use for weapons production; fuel should be physically protected against thefts (nonproliferation);
- fast reactors should be cheaper than existing LWRs, to make them competitive with fossils and gas in most countries and regions.

RDIPE has been working in the last decade on a concept of a fast lead-cooled reactor with UN-PuN fuel (BREST series). The studies carried out so far show that these reactors can satisfy all of the above requirements. The reactor survives any credible accident without fuel failure, has full internal Pu reproduction in the core (CBR~1), does not use uranium blankets and transmutes minor actinides (MA) as a part of the main fuel. These features make it possible to simplify reprocessing technology to a not too deep fuel purification from fission products, with Pu extraction from spent fuel neither required nor possible. Fuel reprocessing should preferably be set up on NPP sites in order to avoid large shipments of highly radioactive and fissionable materials.

The physical traits of fast reactors allow reprocessing in which 1% to 10% of fission products remain in the fuel. Also left in the fuel for transmutation are Am and some Np and Cm. Altogether, these impurities (1% of fission products + Am) account for the high radiation level of the fuel (approximately 50 Ci/kg), hence providing its inherent protection against thievery.

The radiation balance between natural uranium used for energy production in a closed system and resultant long-lived high-level waste (LLHLW) can be attained based on the transmutation of actinides and long-lived fission products in BREST reactors, extraction and utilization (or monitored storing) of Sr and Cs, with HLW put in monitored storage for about 200 years before final disposal in order to lower their activity thousand-fold, approximately. It is assumed in the fuel cycle concept suggested that go to waste are 0.1% of uranium, americium and curium, 0.01% plutonium, 100% of the other actinides, 2% of cesium, technetium and iodine, and 100% of all other fission products. Additional (desirable) requirements: extraction of Np (which may be sent to waste) and Cm (to be stored out-of-pile for 50-70 years, with Pu resulting from Cm decay sent back to the reactor). Np, Cm should be extracted so that 1-10% of them remain in fuel.

The existing commercial technology of spent fuel reprocessing based on aqueous extraction and other radiochemical techniques studied now are tailored to Pu extraction and hence cannot satisfy the nonproliferation requirements. The new techniques should take advantage of the possibilities opened by reactors of the new generation, and should be proliferation-resistant. In this context, the main feature required of a reprocessing technology is that it leaves no room for Pu separation from uranium wherever in the process, which means that the two should always go together in a certain ratio. Inseparability of U and Pu should take its root in the chemical processes and equipment used in reprocessing.

Several refining technologies have been appraised against the above requirements: aqueous, molten fluorides, gas fluorides (low and high temperature modification), electrochemical refining in molten salts. Investigations were also made into the use of unconventional refining techniques such as metallurgical processes or direct annealing of fuel composition. The researches of these and some additional technologies are continuing now. All these techniques were found to be basically capable of assuring inseparability of U and Pu during reprocessing and providing the requisite level of fuel cleaning from actinides and fission products.

XA9952269

THERMOCHEMICAL TREATMENT OF SPENT ION EXCHANGE RESINS

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Spent ion exchange resins (IER) saturated with radionuclides and water is a principal type of radioactive waste constantly generated by nuclear plants of various functions. The reduction of volume of this waste and its treatment to the forms suitable for long-term disposal is an urgent problem facing the present-day atomic energetics. Nowadays the technological process THOR (Studsvik, Sweden) based on the thermodestruction of IER is the best developed and realized on the industrial scale [1]. However, this process requires expensive equipment and great energy consumption it takes for the moisture to be evaporated and thereafter IER to be destroyed by heat. This might be a disadvantage in case of treatment of IER of low and intermediate level of radioactivity.

Meanwhile the capability of some elements has long been known and found practical use of active interaction with water in combustion regime with the generation of great amount of heat (see Table).

TABLE. CHARACTERISTICS OF STOICHIOMETRIC BURNING OF SOME HYDROREACTING ELEMENTS WITH WATER

Element	Heat of combustion (kJ/kg)	Equilibrium temperature (K)
Mg	13000	2747
B	19550	1898
Al	15050	2872
Si	10170	2278
Ti	7720	2326

This property of the elements (metals) has been used in the development of new technology of treatment of IERs in SIA "Radon" [2, 3].

Wet IER is mixed with powder metal fuel (PMF) which represents a mixture of metal fuel, a quantity of burning activator and some technological additives. On initiating, the mixture of IER with PMF burns without extra energy supply to generate enough heat for the moisture to be evaporated and products of IER decomposition to be destroyed and evaporated. To burn out the products of IER evaporation and hydrogen resulting from the reaction of the metal with water, the air is used.

Theoretical data obtained from the thermodynamic simulation (e.g., see Fig.) and the results of experiments using a pilot plant show that radionuclides contained in IER are chemically bound in ash residue consisting (depending on the type of PMF) of metal oxides, silicates, sulphates, etc. According to the experimental data radionuclides in amounts of 90 % or more of Cs-137 and up to 95 % of Sr-90 and Co-60 are fixed in the ash residue. The residue volume is more than ten times less than the initial volume of IER.

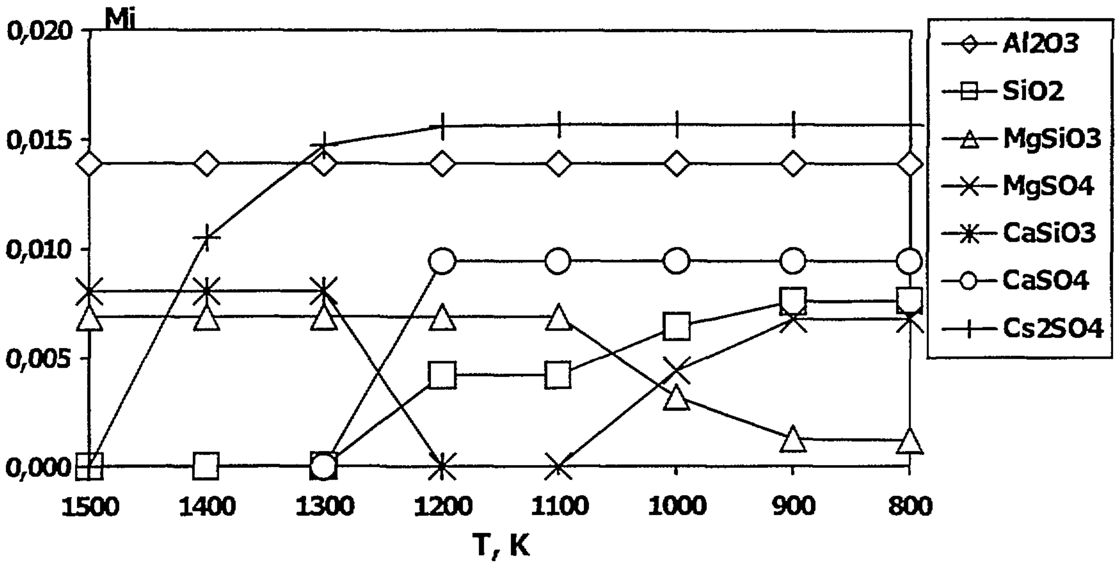

FIG. Variation of equilibrium composition of the ash residue of the combustion products (with Cs) on cooling ($\alpha = 2$).

Concentrations of hazard gases in off-gases do not exceed maximum permissible ones accepted in different countries.

The technological process is easy to perform; it does not require sophisticated equipment and great energy consumption, which allows its realization on a mobile facility.

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XA9952270

IAEA-SM-357/50

The technologies for treatment, conditioning and disposal of radioactive waste in the Slovak Republic

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Original soviet design of NPP operated in Slovakia (HWGCR KS 150,WWER 440) was based on storage strategy of non-treated solid waste and evaporated liquid waste until decommissioning of the plant.

New Slovak approach to the waste management at the end of 80s resulted to strategy to install technologies able to transform in principle all radwaste into form suitable for disposal. Following steps supports this strategy :

a) conditioning of radioactive waste into a form suitable for disposal or long term storage
b) disposal of radioactive waste to near surface repository
c) storage of conditioned radioactive waste non-acceptable for near surface repository
d) research and development of deep geological repository

Because of relatively large quantity of arising RAW in Slovakia (average year production at the Bohunice site with four WWER units is 400-500 m^3 of concentrates, 100-200 m^3 of solid waste and about 25 m^3 spent sorbents) great effort is paid to their minimisation.

The incineration, bituminization, vitrification, fragmentation and decontamination facilities are available.

Bohunice radioactive waste conditioning centre designed by NUKEM is now under commissioning. In this facility, the solid and liquid burnable waste will be incinerated; supercompactor will be used for treatment of other compactible solid waste. Liquid RAW will be cemented into concrete containers. Drums with bituminized waste, cemented/dried resins, sludges and ash as well as packages of compacted solid waste will be grouted with radioactive concrete mixture in special fibre reinforced concrete containers (license of French company Sogefibre).

The near surface repository was built near the NPP Mochovce and improved on the basis of IAEA's Waste Management and Technical Review Programme (WATRP) recommendations focused mainly on repository stability and more detailed safety analyses. Modifications of drainage system and backfilling were based on national review and assessment of safety analysis report. The repository is now under licensing process. It is designed for a disposal of solidified low and intermediate level radioactive waste from NPP operation and partly decommissioning, which comply with acceptance criteria for disposal.

Basic limitations on the acceptability of waste for near surface disposal include the specific activities and total quantities of radionuclides in the waste as they were determined on a basis of site-specific long term safety assessment for following 19 nuclides: ^{14}C, ^{41}Ca, ^{59}Ni, ^{63}Ni, ^{79}Se, ^{90}Sr, ^{93}Mo, ^{93}Zr, ^{94}Nb, ^{99}Tc, ^{107}Pb, ^{126}Sn, ^{129}I, ^{135}Cs, ^{137}Cs, ^{151}Sm, ^{238}Pu, ^{239}Pu and ^{241}Am. Short term safety assessment (handling, transport) was performed for next three nuclides: ^{3}H, ^{55}Fe and ^{60}Co.

The deep geological repository is in the early stage of research and development.

IAEA-SM-357/51

A STRATEGY FOR UPGRADING MANAGEMENT OF SPENT NUCLEAR FUEL AND RADIOACTIVE WASTE AT THE IGNALINA NUCLEAR POWER PLANT

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The Lithuania electricity production depends to more than 80 % on the Ignalina Nuclear Power Plant, INPP consisting of two RBMK reactors. The first unit was taken into operation in 1983 and the second in 1987. With the location in the north-east part of Lithuania, close to the borders of Belarus and Latvia the reactors were planned to provide electricity to the whole region. Also today a large part of the electricity produced is exported to neighbouring countries.

When designed, the waste management strategy of the former Soviet Union, based on its Norms and Rules, was implemented. This means in brief that the spent nuclear fuel (SNF) should be reprocessed the Soviet Union and the management of radioactive waste should be done in connection with the eventual decommissioning of the reactors. The major facilities for management of radioactive waste were evaporation of liquid waste and subsequent bitumenisation of the sludge, treatment of liquid with ion-exchange techniques and subsequent storage of the resins in tanks, sorting of solid waste according to its activity content and storage on site.

Solid radioactive waste generated during plant operation is collected and segregated into different groups depending on dose rate and composition. It is loaded into containers and transported to special stores. There is a complex of four stores at the INPP, with auxiliary systems and equipment for their operation. The auxiliary systems and equipment include loading devices, transport containers, special cars, facility for washing of cars and transport containers and gas fire extinguishing station. Waste composition and proposed processing methods are presented in the report.

Following the independence of Lithuania in 1991 Lithuania is successively replacing the former Norms and Rules with Lithuanian laws of which the Law on Nuclear Energy is in force and the Law on Radioactive Waste Management exist in draft. Lithuania has signed both the Convention of Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The INPP has the intention to meet all national and international requirements on safe management of its SNF and radioactive waste and to demonstrate its capability to achieve this. Recognising, however, the financial constraints of the INPP and the lack of experience of western SNF and radioactive waste management, the disruption of the support from Russian waste management organisations, international co-operation for the upgrading is necessary during a transition period.

Since the return of the spent fuel to the Russian Federation no longer is a possible option, a new spent fuel storage facility, based on dry storage of the spent fuel in casks, is established at the site. For the realisation of an improved waste management system at the INPP, Sweden is since several years assisting INPP in this very important task

The paper will give an overview of the work done, in progress and planned, primarily from an operational point of view. It will give example of practical problems which has to be overcome and how very tuff prioritisation has to be made because of lack of resources. Many of the problems are associated with the fact that the waste management strategy successively has to be changed at the same time as the generation of SNF and radioactive waste which require proper management, is continued. A lot has already been achieved, especially regarding the SNF. However, the main work is still to be done until the INPP can declare that its waste is properly managed in full accordance with a national waste management strategy which eventually will end with the disposal in licensed repositories

XA9952272

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INTEGRATED PLANNING OF LABORATORY, IN-SITU, MODELLING AND NATURAL ANALOGUE STUDIES IN THE SWISS RADIOACTIVE WASTE MANAGEMENT PROGRAMME

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After more than 25 years of development, the Nagra radioactive waste management programme can be considered relatively mature. The selection of a proposed L/ILW repository site at Wellenberg is scientifically accepted by all relevant technical bodies; however, it requires a positive decision at the cantonal level to allow final stages of site confirmation to be initiated. The basic feasibility of HLW disposal in Switzerland was demonstrated already in 1985 and, since then, effort has concentrated on identification of a potential siting region in North Switzerland - in either the crystalline basement or an overlying sedimentary rock. A project confirming siting feasibility will be finalised around 2001. Although HLW disposal in Switzerland remains the reference case, the option of participation in a multinational repository project remains open and is being studied, e.g. within the Pangea project. Most of the essential technical components required to site, implement and assess the safety of a repository are now well established, as is the infrastructure (e.g. underground test facilities, specialist laboratories) and experienced personnel needed to carry out any additionally required R&D work.

A priority for the Nagra R&D programme is filling remaining gaps in system understanding. In particular, this involves more detailed analysis of specific waste forms which have been somewhat neglected in the past, including high burn-up spent fuel (conventional UO_2 and MOX) and long-lived ILW ("TRU"). Also important is clearly establishing the significance of previously identified "open questions" on potential perturbing factors - including colloids, organics, microbes and the high pH plume from cementitious materials. A more general aim is the validation of models and databases and maintaining system understanding at a state-of-the-art level as a contribution to technical confidence building. As repository projects come closer to realisation, such confidence building becomes increasingly necessary for a range of audiences. The requirement of winning public acceptance may also justify projects which are not strictly necessary from a scientific level, such as studies of monitoring and retrievability or demonstration of operational technology. Finally, from the viewpoint of the repository implementer, continual analysis of the potential to reduce costs (without sacrificing safety) through optimisation of design or operational procedures is expected.

In general, most of the straightforward R&D work has already been carried out and hence the topics which are identified for future work tend to be particularly challenging technically and/or very time-consuming and costly to carry out. Characteristics of the resulting projects are thus that they require integration of laboratory, in-situ, modelling and natural analogue studies and that they are focuses for international collaboration. A good example is the ongoing work on quantification of the influences of a high pH plume from TRU waste which includes:

- Joint Nagra/SKB/Nirex funded experiments at BGS, UK and supporting Swiss studies at the Paul Scherrer Institute (PSI) and University of Berne
- Model development (SANTA-CHEM) in collaboration with JNC, Japan associated with their GEOFRONT experimental studies and complementary development of the model MCOTAC by PSI
- In-situ experiments with a wide range of international partners at the underground test facilities at Grimsel and Mont Terri
- The international natural analogue project at Maqarin, Jordan
- Focused studies on the rôle of colloids, organics (natural and anthropogenic) and microbes in such an environment.

Confidence-building is challenging because of the long timescales involved in many of the key processes, which require the use of mathematical models to evaluate system evolution over periods up to hundreds of thousands of years or even more. Active participation in a range of studies like those listed above, involving integration of input from many technical disciplines and temporal extrapolations far beyond normal engineering practice, is aimed at providing convincing demonstration of a fundamental understanding of the key processes involved and that long-term extrapolations can be justified by observational data. It is, of course, impossible to predict future system behaviour precisely and hence formal procedures need to be established to describe a range of future evolutions of the repository system (scenarios) which should bracket extremes of the entire range of possibilities. To enhance confidence that the range of scenarios is complete, Nagra is active in an international project, coordinated by the NEA, to develop and maitain a comprehensive catalogue of all relevant features, events and processes to be considered during scenario development.

Monitoring, retrievability and technology demonstration are all components of major international projects such as FEBEX (under the leadership of ENRESA, Spain) for HLW and GMT (run by RWMC, Japan) for L/ILW and TRU, which are ongoing at the Grimsel test site. In collaboration with various organisations in Japan, the output of such projects is being utilised to develop concepts for optimised disposal systems which can improve the transparency of the safety case, ease operational quality assurance procedures and also reduce costs. Such work may, in turn, lead on to further international demonstration projects.

Emphasis is very much on the Swiss national programme but, increasingly, Nagra's resources of data, technology and experienced staff have been made available to support repository projects elsewhere. Such consulting can be attractive to both sides, allowing Nagra to develop a wider range of experience in the application of accumulated expertise while allowing clients access to a knowledge base accumulated at a cost of more than 700 M SFr. over the last 25 years.

The full paper will provide a more detailed overview of Swiss repository concepts, the associated Nagra R&D programme and a discussion of procedures used to set priorities.

IAEA-SM-357/53P

INITIAL DESIGN PROCESS OF THE REPOSITORY

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Initial design process can be used for taking preliminary information for conceptual design options. Initial design process layout is shown in *Fig. 1*.

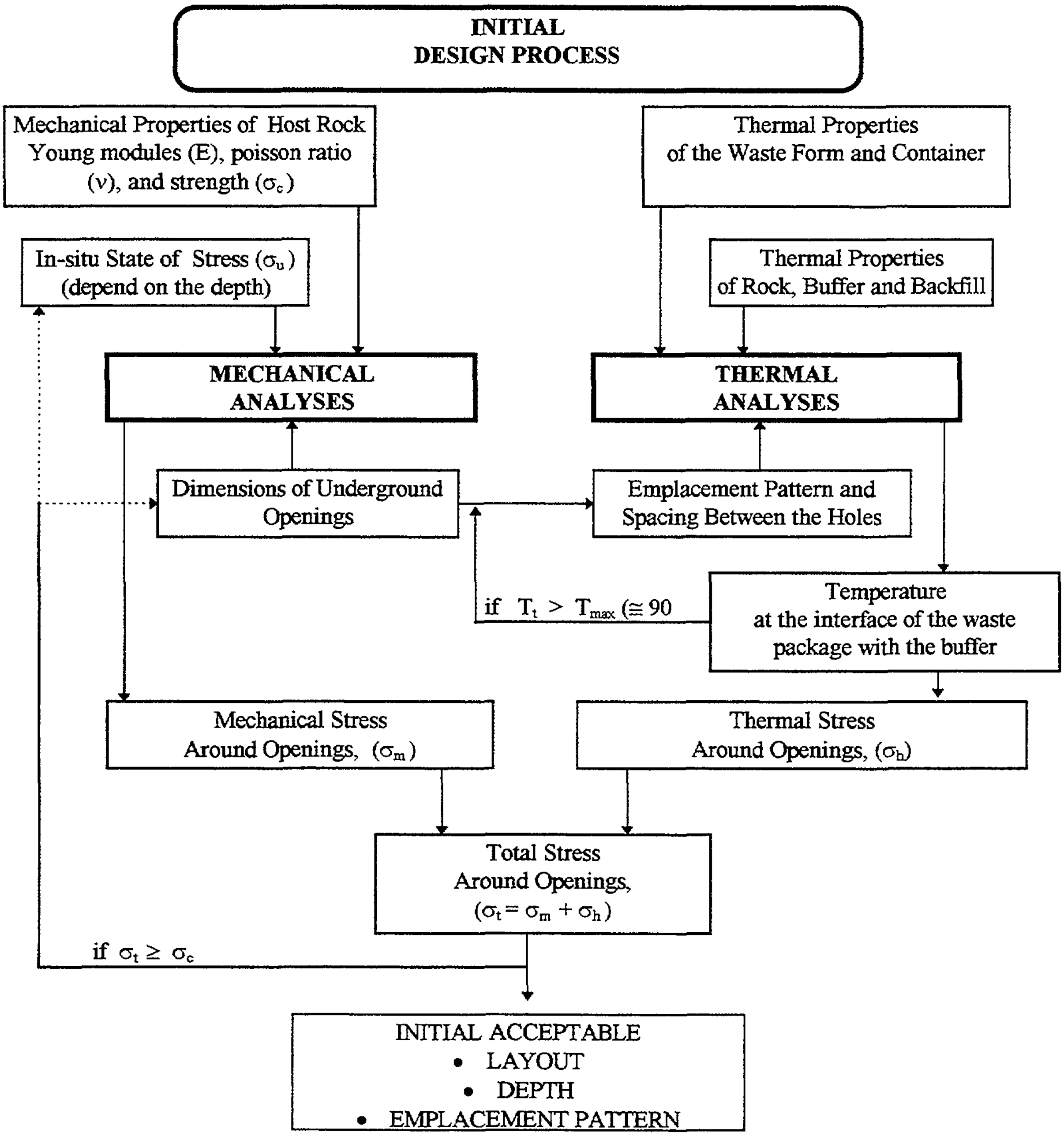

FIG. 1. Initial design process layout.

For making a proper design, much information is needed about factors related to mechanical properties of the host rock, thermomechanical properties of the waste form and the container. Initial design process includes two main analyses; mechanical and thermal analyses. If the required data are taken into these main analyze process and an interaction between these two analyze processes can be satisfied, in this case initial and optional depth of the repository, repository layout and the emplacement pattern of the repository can be determined.

Young modules (E), Poisson's ratio (ν) and the strength of the host rock (σ_c), are the main parameters which will be used for analyzing of the initial stability of underground openings in the repository. Thermal conductivity of the buffer material and the host rock are some of the main effective factors for thermal analyses. And there is a limitation for the temperature at the interface of the waste package with the buffer, because of the boiling temperature of water and/or the negative effects on the thermomechanical properties of the buffer material, interface temperature must not exceed 100 °C (90 °C is recommended).

Although hydraulic conductivity of the potential host rocks can be known generally (between 10E-6 to 10E-12 m/s according to the type of the host rocks). Hydraulic conductivity requirements will be specific to the site during the excavation period.

For a conceptual design, much information is required. After the site selection period, several investigations in the underground research laboratory and especially in-situ tests by boreholes should be done for getting realistic and site specific data.

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XA9952274

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UK SAFETY STANDARDS FOR RADIOACTIVE WASTE MANAGEMENT AND DECOMMISSIONING ON NUCLEAR LICENSED SITES

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HM Nuclear Installations Inspectorate regulates radioactive waste management and decommissioning on the UK's licensed sites. Over 99% by volume of the UK's intermediate level and high level wastes (ILW, HLW) are created and stored on licensed sites. The NII have developed guidance which takes into account current waste and decommissioning activities in the UK, including the prospect of long term storage of wastes in the absence of a disposal route for ILW and HLW. It includes the following:

Decommissioning:

- involves the systematic and progressive reduction of hazard until there is no danger
- undertaken as soon as reasonably practicable
- only deferred if there are substantial safety benefits in doing so
- costs discounted once decommissioning methods chosen and timing substantiated
- if same techniques and technology for dismantling are involved after deferral then dismantling should be undertaken sooner rather than later
- full use made of existing knowledge for safety and to minimise information transfer to future generations
- full use made of available disposal routes
- a passively safe state achieved as soon as reasonably practicable
- plans kept under continuous review to ensure best options selected
- plans included in safety cases, which address the provision of staff, adequacy of plant knowledge and site infrastructure and services: safety cases reviewed at least every 10 years until the site is delicensed.

Radioactive waste management:

- waste managed safely
- leakage and dispersal prevented
- leakage and spills contained and cleared promptly
- full use, within authorised limits, made of existing disposal routes
- waste which cannot be disposed processed promptly for long term passive safe storage
- waste avoided and volumes and radioactivity minimised
- radioactivity immobile
- waste form physically stable and chemically inert
- potential energy acting on stored waste and its storage environment minimised
- a multibarrier approach taken to containment
- waste form and package resistant to degradation
- waste package and storage system resistant to foreseeable hazards
- active safety systems, monitoring and maintenance minimised
- monitoring and maintenance to ensure safety should be minimised
- wastes be inspectable and retrievable
- wastes accessible to enable a response to foreseeable accidents
- storage arrangements should facilitate retrieval for final disposal
- lifetime of storage arrangements and waste package appropriate for planned storage period and method of disposal
- no requirement for prompt remedial actionwaste stores designed to allow periodic and continual refurbishment.

Successful application, by regulation, of the above principles and standards will ensure that radioactive waste and decommissioning in the UK are managed safely both now and for the future pending final disposal.

XA9952275

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URANIUM OXIDE RECYCLING TO GIVE MORE SUSTAINABLE POWER GENERATION

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Introduction.

The background of nuclear reprocessing in the United Kingdom is given. Open versus closed fuel cycles will be considered, including the value of recycling spent nuclear fuel to produce products which represent a sustainable way of utilising nuclear energy resources effectively.

Description of The Nuclear Fuel Cycle.

The concept of the nuclear fuel cycle is presented, looking at the cycle in the context of Life Cycle Assessment (LCA). The cycle begins at the mining and milling stage, progressing through to fuel manufacture, nuclear power production and either, recycling or direct disposal. This concept is then extended to include further fuel cycles, fabrication and use of Mixed Oxide (MOx) fuel, and the potential use of future fuel cycle technology. Waste volumes for different cycles are considered in particular open versus closed cycles, including single MOx cycles and second generation MOx cycles. The associated energy value of the recovered product produced by open cycles is assessed and presented for the Uranium Oxide, Mixed Oxide and Fast Breeder cycles. The value of this energy is offset against direct disposal and the Global Warming Potential averted compared to other forms of energy production.

Life Cycle Assessment and The Performance of BNFL's Thorp Plant.

Consideration is given to BNFL's Thorp plant and its performance compared to the design flowsheet, demonstrating the LCA performed on the design flowsheet is valid. Comparisons are then made between open and closed cycles (Uranium Oxide only) in terms of the impact of radioactive environmental burden [1]. Distinctions are then drawn between the impact from mining and milling and that from recycling.

Viability of Nuclear Waste Forms.

Direct disposal of fuel assemblies is compared with that of engineered waste forms for Intermediate Level Waste (ILW) and vitrified High Level Waste (HLW). An explanation of the reasons why these engineered options were chosen is given in terms of their storage and disposal, relating to their volume, radioactivity, and their physical and chemical state.

Comparison of Open Fuel Cycles in Terms of Toxic Potential.

Comparisons based on the radioactivity (TBq) of waste forms, from closed and open cycles, provides a simplistic method for comparing fuel cycles. However, this gives insufficient weight to the longer lived nuclides of relatively low specific activity, notably the actinides. Adequate weighting can be given to important toxic radionuclides, by the use of a methodology, which calculates radiological toxic potential [2].

Toxic Potential (TP) has been developed as a methodology for assessing the impact of different waste forms. The Toxic Potential of a given quantity of any radionuclide is defined as the volume of water into which it would have to be completely dispersed so that the water is considered safe to drink. More specifically it is the volume of water into which it would have to be dispersed, such that an "average" man would not exceed his annual recommended dose limit of 1 mSv if all his drinking water came from a source contaminated with that radionuclide.

$$\text{Toxic Potential } (m^3 \text{ water}) = \frac{\text{Radionuclide Activity (Bq)} \times 0.712 m^3 \text{ water/year.}}{\text{Annual Limit of Intake (Bq/year)}}$$

This concept will be presented for different fuel cycle scenarios to indicate the benefits in terms of long term storage/ disposal, of extending the fuel cycle to include MOx (first generation and second generation) and ultimately the use of future fuel cycle technology.

The value of Integrated Toxic Potential (ITP) is discussed as a criterion for judging waste management options, indicating that volume alone is a poor indicator.

Sustainable Benefits of Open fuel Cycles Compared to Direct Disposal.

In conclusion the sustainable benefits of open cycles are compared to those of direct disposal. This information is presented to support the concept that cycles involving nuclear fuel recycling are a sustainable option for power generation and have specific benefits over and above that of direct disposal.

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XA9952276

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CURRENT EVOLUTIONARY STAGE OF L/ILW MANAGEMENT AT OPERATING NPP IN THE USA

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The volume of Low and Intermediate Level Waste (L/ILW) from commercial nuclear power plants(NPP) buried annually in the USA has decreased dramatically over the last two decades. The reduction in disposal volume has occurred for both dry active waste(DAW) and wet wastes (e.g., resins, filters and sludges). Decreased disposal volume has occurred at both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). Figure 1 below shows the reduction in disposal volume for PWRs.

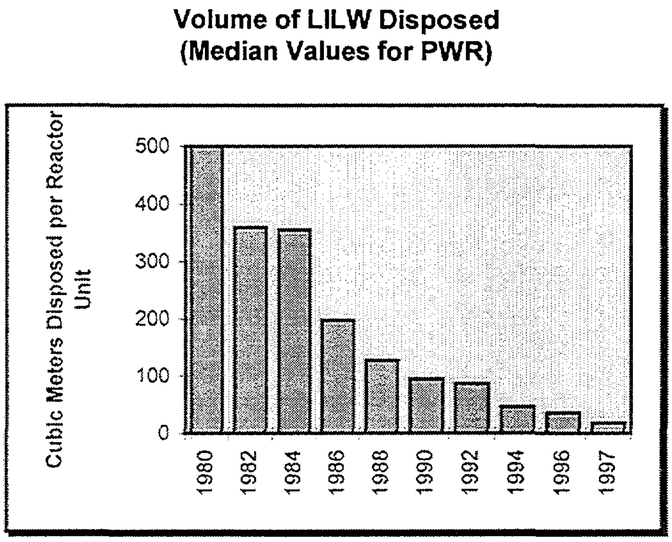

Figure 1

The reduction in disposal volume occurred in stages. The major driver to reduce disposal volume in the commercial sector of the USA was increasing burial costs. Although the USA is a large country with many excellent sites for L/ILW disposal, no political consensus for commercial disposal siting has been achieved. This lack of political will has resulted in increased burial costs.

The initial means to reduce disposal volume for DAW was to separate clean material from radioactive and to decontaminate metallic waste. The next major stage for DAW was the use of off-site processors to incinerate or compact waste prior to disposal. The lower fees for incineration versus compaction drove plants in the USA to limit the use of disposable plastics that could not be incinerated (e.g., PVC). Ironically, this substitution of disposable contamination control supplies reduced costs since many incinerable plastics are lower in cost than PVC[1].

Although off-site processing services greatly reduced waste disposal volume, they did not decrease costs substantially. Waste processors only needed to charge slightly below the cost of commercial disposal to insure waste would be sent to them. Further pressure to reduce plant costs forced operators to examine waste generation sources versus obtaining processing services for waste that had already been generated. This examination lead to the replacement of disposable contamination control supplies with either launderable items or elimination of the item[2]. Thus, waste minimization or pollution prevention of DAW in the commercial sector of the USA came late in the process. Figure 2 depicts the evolutionary steps in DAW reduction.

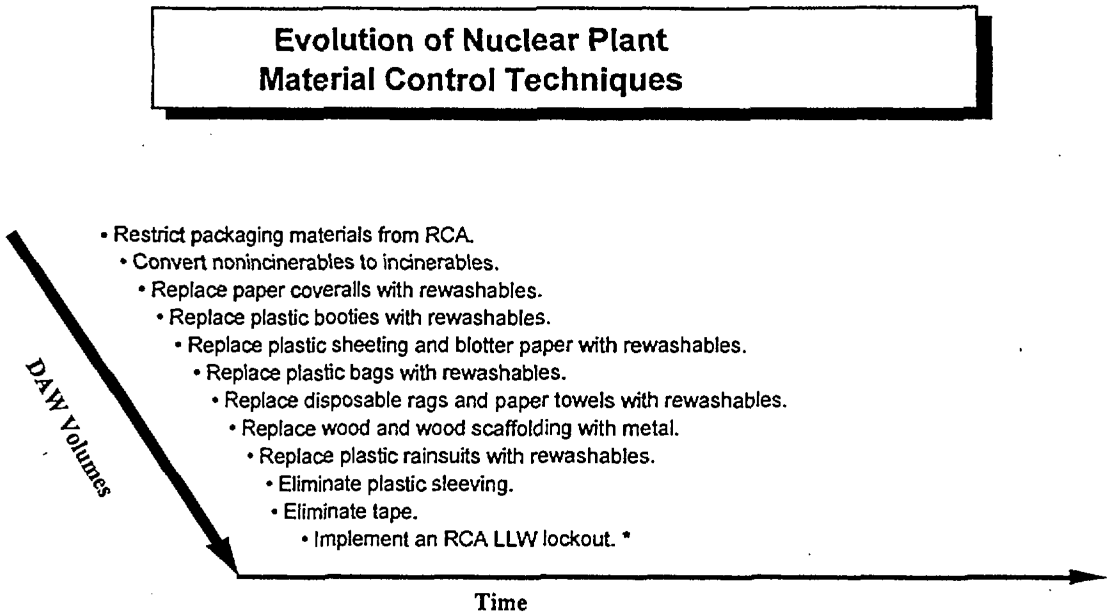

Figure 2

The benefit to others from reviewing the history of DAW reduction in the USA is that the intermediate step of incinerable product substitution can be skipped. Moving directly to extensive use of launderable items or eliminating the use of needless disposable contamination control items can save a substantial amount labor and cost.

Wet waste disposal volume was also reduced over the same time period. The stages of wet waste reduction in the USA have been the reverse of DAW. Initially wet waste disposal volume was reduced by ceasing cement solidification of spent resin and instead dewatering this waste. Evaporation of liquid waste was replaced by filtration and ion exchange at many plants.

The use of selective ion exchange media, polymer addition and segregated beds greatly increased liquid throughputs and generally decreased discharge activity [1]. Reducing liquid inputs to radwaste treatment systems was also pursued. These waste minimization processes to reduce wet waste were routinely implemented prior to the advent of off site processing services for wet waste.

Recently, thermal destruction of wet waste at off site processors has become available in the USA. Much of this is focused on lower activity spent resin from condensate polishing systems. On site volume reduction of high activity cartridge filters using remotely operated shielded shears has also been adopted at several plants within the last few years.

The current drive to further reduce operating costs to meet a deregulated electric market has unexpectedly resulted in further waste reductions. Shorter refueling outages yield increased revenue but, also result in less waste. Increased heat rate improves electrical generation but, also reduces liquid input to radwaste. Thus, the invisible hand of self interest in a mature market strives to eliminate waste of all sorts, including L/ILW, providing a public good.

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COMPARATIVE ANALYSIS OF RADIOACTIVE WASTE MANAGEMENT TECHNOLOGIES BY RISK ESTIMATION

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One of approaches under the development of radioactive waste treatment conception is based on the principle of natural and artificial radioactive series radioactivity equivalency.

The basic natural radioactive elements are included into four radioactive series (Tab.1). These are: thorium series, neptunium series, uranium series and uranium-actinium series.

All of radioactive series articles are bond by irreversible reciprocal transformations. Therefore closed systems reach the equilibrium during the definite period.

Natural series contains the radioactive elements accumulated during millions of years. Great quantities of the radioactive daughter isotopes are in equilibrium with uranium and thorium (Tab.1). The series elements are take out thanks to geochemical processes, difference of physical and chemical properties and miming [1]. Thus the system may be opened. It causes the breach of radioactive equilibrium. Therefore it is necessary to know the system open state degree and time to estimate the series radioactivity.

The reactor actinides are predecessors for defined radioactive series. The actinide production depends on correlation between fission and capture cross-section and neutron spectrum. Consequently it depends on fuel kind and reactor type. Thus the thermal reactor produces more neptunium series isotopes than the fast reactor [2]. But there are some general regularities: the most quantity of produced isotopes is plutonium; neptunium and americium are produced less by one order than plutonium; curium isotopes are approximately produced less by two orders than plutonium. Approximate actinide quantity submits to following correlation: Pu : (Np, Am) : Cm $\approx$ 1 : (0,1) : 0,01. Basing on the closed nuclear fuel cycle (it foresees uranium, plutonium and neptunium inclusion), it is possible to estimate the radioactivity of radioactive waste reactor series, taking into consideration their basic radioactive elements. For example, 1g-samples of corresponding curium isotope of 1 and 10^3 years standing each were taken as the parent isotope base (Tab.2) [2].

The calculations showed (taking into consideration the relation between uranium and curium ($1 : 10^{-5}$)) natural and reactor uranium series ($4n + 2$, $4n + 1$) radioactivities are commensurable and the execution of radioactivity equivalency principle is possible for these series (compare Tab.1 and 2). It is possible to realize this principle in tens of thousands of storage years.

The principle of radioactivity equivalency is impracticable for neptunium series since natural series has practically the back-ground radioactivity and reactor neptunium series is increasing its radioactivity during hundreds of thousands of years.

TABLE 1. Radioactivity (A) of natural series ($t \approx 1 \times 10^9$ y)

Series	Parent isotope	Equilibrium establishment period, y	A, Bq/kg
4n	^{232}Th	$\sim 5 \times 10^3$	$4{,}1 \times 10^7$
4n + 1	^{237}Np	$\sim 1 \times 10^6$	0
4n + 2	^{238}U	$\sim 1 \times 10^7$	$1{,}6 \times 10^8$
4n + 3	^{235}U	$\sim 5 \times 10^6$	$3{,}7 \times 10^8$

TABLE 2. Radioactivity (A) of reactor series.

Series	Parent isotope	Time of standing	
		1 year	10^3 years
		A, Bq/g	A, Bq/g
4n	^{244}Cm	$2{,}8 \times 10^{12}$	$7{,}58 \times 10^{10}$
4n + 1	^{245}Cm	$6{,}6 \times 10^9$	$1{,}66 \times 10^{10}$
4n + 2	^{242,}	$2{,}6 \times 10^{13}$	$4{,}73 \times 10^8$
	^{246}Cm	$1{,}14 \times 10^{10}$	$9{,}85 \times 10^9$
4n + 3	^{243}Cm	$1{,}86 \times 10^{12}$	$4{,}5 \times 10^9$

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XA9952278

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SPECIALIZED HOMOGENEOUS REACTOR FOR SPENT FUEL TREATMENT

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Specialized homogeneous reactors on fast neutrons with circulating metal fuel and liquid metal coolant, in which new method for circulation of metal fuel - with phase transition from liquid state to granulated one and vice versa, can be used effectively in nuclear power system, consisting of the reactors both on fast and on thermal neutrons, with the aim to annihilate highly active and long-lived products of nuclear fuel irradiation - minor actinides, to reduce the mass of produced plutonium and radioactive wastes to be disposed. The method of fuel circulation is examined at demonstration facility with using simulating media. The results of experimental investigations are given.

IAEA-SM-357/59

ADVANTAGEOUS TECHNOLOGY TREATMENT OF LAUNDRY WATERS

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A special laundry there is on majority atomic power stations and other major enterprises, which connected with using of radioactive materials. Cleaning of sewage waters on such laundry presents a hard task to crack because of following reasons:

- a laundry solutions have a high salt content (before 5 g/l),
- surfactants cause a significant formation a spume in evaporate apparatus, which using for clean-up of radioactive waters at atomic power station,
- cleaning of the laundry solution by removing from it all admixtures brings about the loss of majority of valuable components of laundry solution.

In this paper on the base of preliminary experimental studies is offered a principle technological scheme of installation for the cleaning of laundry water (Fig.1), which allows again to use water and components of laundry solution and is characterized by the low amount of secondary nuclear waste.

Main particularity of the offered technological scheme is that sewage after the rinsing (60-80% from the common amount of sewage) process in hyperfiltration apparatus, but sewage after the laundry (20-40% from the common amount of sewage) - in the ultrafiltration apparatus.

Concentrate after reverse osmosis desalination of sewage (after rinsing) is kept a majority of laundry solution components, since a hyperfiltration membrane nearly does not miss salts and surfactant molecules. For the reason their recuperation the concentrate directs to processing in the ultrafiltration apparatus. Desalinated water (permeate) after hyperfiltration apparatus reapply.

For cleaning of water from radionuclides it is use a method KOUF "coagulation and complexing with next ultrafiltration" [1]. As coagulants it is use reagents, transforming radionuclides in form, which cannot pass through ultrafiltration membrane (in colloid and other forms). For example, hexacyanoferrat Ni or Co is use for precipitation of $^{137,134}Cs$, polyacril acid (PAC) – for complexing ^{60}Co and etc.

The stage of oxidation of some organic substances, which are kept in permeate after ultrafiltration, and following filtration of cleaned solution required for discoloration laundry solution after the laundry of color linens. Conducted by us experiments have shown that in this case method of electrooxidation in two-chamber electrolyser with porous diaphragm is much efficient.

The testing of pilot ultrafiltration installation with spiral filtrate elements on the real laundry solutions are shown that when using a ultrafiltration membranes with pore size $5 \cdot 10^{-8}$ m the membrane delay the submicron and colloid particles, but dissolved components of laundry solution get through the ultrafiltration membrane to permeate.

The changing for chemical composition of laundry solution after ultrafiltration is shown below (laundry solution/permeate after ultrafiltration), mg/l: common salt content – 4235/3847; pH – 7,38/8,30; permanganate oxidation – 410,8/67,6; Na^+ - 1817/1792; K^+ - 12/14,1; NH_4^+ - 6,0/7,6; Ca^{2+} - 24,1/8,0; Mg^{2+} - 14,6/10,9; Fe_{sum} – 0,56/0,26; Cl^- - 62,4/57,0; NO_3^- - 1,34/0,97; SO_4^{2-} - 2,4/<0,1, HCO_3^- - 1927,6/2372,5; CO_3^{2-} - <10/120; PO_4^{3-} - 1556,3/1375. After ultrafiltration the contents of surfactants in laundry solution fall on 40-60%.

Now the technological scheme of installation for the radioactive water of special laundry reprocessing, described in given paper, will be realized at the special laundry of MosNPO "Radon". Entering an installation for the cleaning of water in the usage scheduled for 2001 years.

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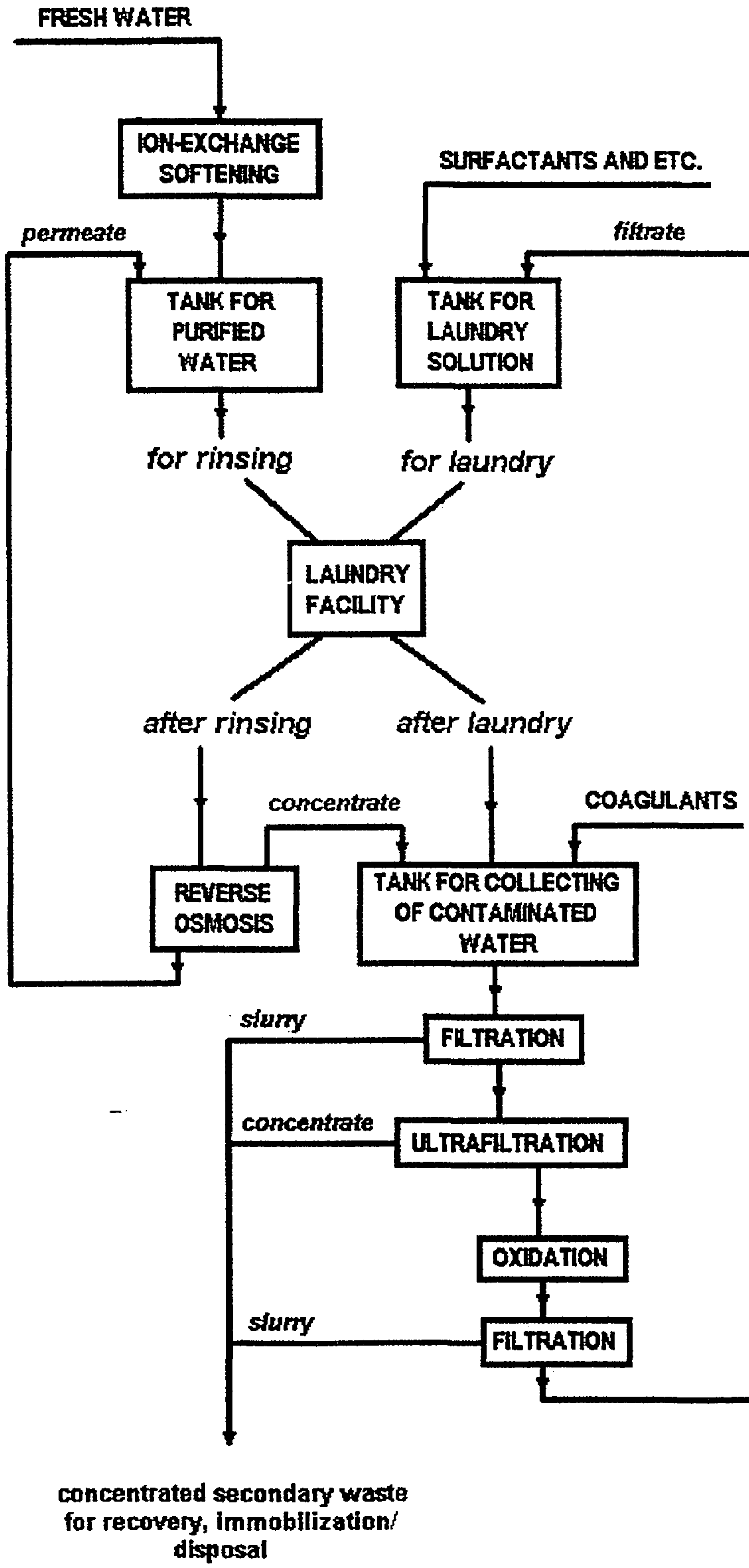

FIG. 1. The principle technological scheme of installation for the radioactive water of special laundry reprocessing.

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IAEA-SM-357/61

INFRASTRUCTURE NEEDS FOR WASTE MANAGEMENT

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Significant technical progress has been achieved and considerable experience has been accumulated in many countries for the safe management of radioactive waste. The system for management of short-lived waste is sufficiently developed, and internationally accepted practice is coded. Proper infrastructure to manage this type of waste has been established and is operating.

For long-lived waste, deep underground disposal is internationally the preferred option. In this area, too, significant technical progress has been achieved in recent years. Technical capabilities have been developed to address technical issues associated with the design, evaluation and regulation of those facilities. Repository engineering and design have been improved and the capabilities of technical concepts demonstrated. Site characterisation has been carried out to establish potential deep disposal sites. Several underground research laboratories have been operating and generating valuable technical inputs, while clarifying technical and geological features crucial to disposal concepts. Significant progress is also noted in the evaluation of the long-term safety of deep repository systems, including treatment of uncertainty.

On the basis of these technical achievements and experience in management of short-lived radioactive waste as well as non-radioactive waste, the community working on implementing geological disposal programmes has been developing technical infrastructure and clarifying requirements for implementing geological disposal of long-lived waste.

Recent experience in siting of geological disposal repository clearly indicates that the confidence in the long-term safety of the geological disposal gained by the experts is not necessarily shared by non-expert groups. There is widespread recognition, within the technical community, that the critical path towards implementation of disposal facilities is increasingly determined by a broader community. This means a need for additional institutional mechanism for integrating environmental, societal, economical and other factors in the decision-making process.

IAEA-SM-357/62

THE STATUS OF THE RADIOACTIVE WASTE MANAGEMENT IN KOREA

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In Korea, fourteen nuclear reactors are in operation and by 2015, a total of twenty-eight nuclear reactors will be in operation. The current nuclear share occupies about 34.2 % of the total generating capacity of electricity and 46.3 % of the total production of electricity. The active nuclear program causes an inevitable increase in the build-up of radioactive waste, including spent fuel. Therefore, the reliable and effective management of radioactive waste and spent fuel has become a key for the continuous growth of the nuclear power program.

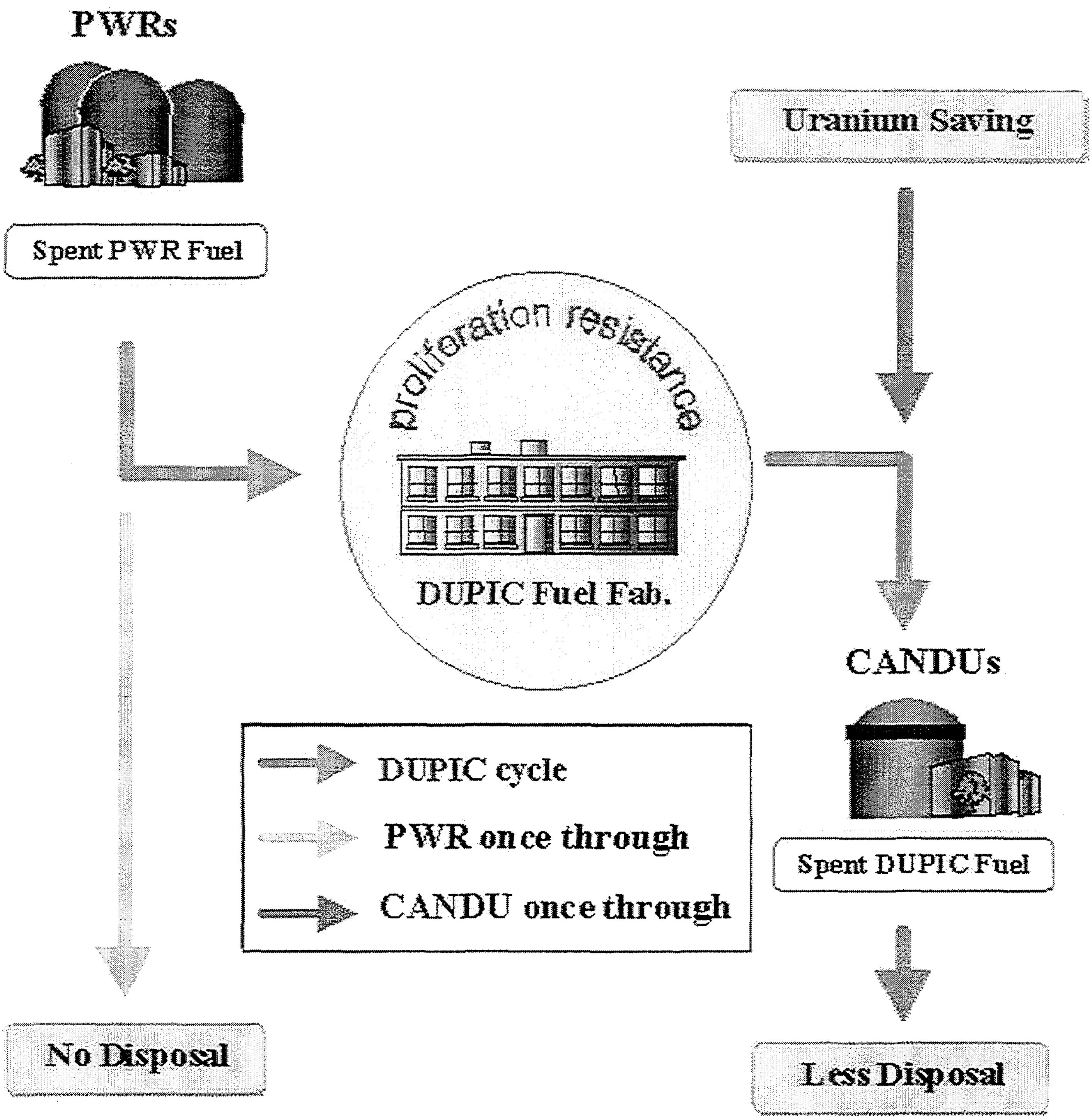
By 2000, a total of 84,413 drums of LILW shall be generated and it shall drastically increase to 256,520 drums by 2020. The cumulative amount of spent fuel from existing nuclear power plants has reached 3,233 MTU by the end of 1997. It is expected that approximately 11,000 MTU and 19,000 MTU will be accumulated by the year 2010 and 2020, respectively. At present, the PWR spent fuels generated from nuclear power plants have been stored temporarily in storage pools at plant sites, while the CANDU spent fuels have been stored in storage pools and dry storage (concrete canister) at plant sites. By the new national planning, AFR storage facilities for spent fuels shall be built by 2016 and a repository for LILW radioactive disposal shall be in operation by 2008.

Even though Korea has a "wait and see policy" for spent fuel management, several alternative studies on spent fuel management such as DUPIC have been carried out. The DUPIC program, as shown in the figure, is integrated with a number of associated scopes of work, such as compatibility with a CANDU reactor system, safeguards systems development, waste management, etc. A preliminary experiment for characterization of spent PWR fuel materials is now ready to look at key technical parameters that would be essential for the DUPIC fuel fabrication experiments starting next year.

In parallel, R&D activities to develop the needed technologies for the permanent disposal of spent fuel and HLW have been implemented. A site-generic concept is being developed under assumptions that an underground repository would be located in a type of crystalline rock in Korea and an appropriate multi-barrier system would be provided for the isolation of the HLW from the biosphere. To reach the target for the development of a reference deep geological repository concept suitable for Korean geological circumstances by the year 2006, the basic R&D program on four fields have been set up; performance assessment and disposal system development, geo-environmental science research, engineered barrier development, and radionuclide migration study.

A few concepts for the transmutation of long-lived radionuclides are being considered at present in KAERI. One of prospective technologies for the transmutation is to utilize an accelerator-driven system, namely, a hybrid system composed of a proton accelerator and a sub-critical reactor. The current research on the hybrid system is carried out focusing on the sub-critical reactor. Since the nuclear fission in the sub-critical reactor does not occur by chain reaction but by the neutron supplied from the outside of the fuel system, the control of the reactor would be easier. Also, it would ease the conditions of nuclear fuels compared with those of conventional power reactors. If fuel conditions are eased, then more impurities may be allowed in the fuel material of trans-uranium, resulting in great advantages for fuel processing and preparation. The concept of the relevant fuel cycle is also being studied in order to choose the optimum process. The pyro-processing option might be preferable in the viewpoint of process simplicity, nuclear nonproliferation, and economy. However, more intensive studies would be required to develop an applicable technology.

In addition, active R&D on the treatment of radioactive waste from the various nuclear fuel cycle as well as the decontamination and the decommissioning of nuclear facilities are in progress.


PWRs
Spent PWR Fuel
Uranium Saving
proliferation resistance
DUPIC Fuel Fab.
CANDUs
DUPIC cycle
PWR once through
CANDU once through
Spent DUPIC Fuel
No Disposal
Less Disposal

IAEA-SM-357/63

APPROACHES FOR DEVELOPING THE REFERENCE CONCEPT OF A GEOLOGICAL DISPOSAL SYSTEM AND THE RELEVANT R&D' STATUS IN KOREA

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In Korea, eleven PWR units and three CANDU units are in operation and another 6 units are presently under construction. According to the Nuclear Energy Plan of the Long-term National Power Development Plan announced in 1998, seven more units would be added by the year 2015.

The Korea Atomic Energy Research Institute (KAERI) has undergone a R&D program for HLW disposal since 1997. The main purpose of the program is to establish a reference HLW repository concept by the year 2006 that, as a prototype, could be the first repository model to accommodate any social, nuclear industrial and environmental conditions anticipated in Korea.

This paper represents the basic directions and approaches with specific essential activities needed to develop a reference deep geological repository system. The relevant R&D status and some results carried out during the past years are also included in this paepr.

The primary function of a HLW repository is to isolate radioactive waste from the accessible environment for a sufficiently long time. Therefore, it should be well defined with reasonable propriety of engineering aspects (technology, economics and long-term safety).

As shown in Fig.1, activities to develop a reference disposal concept, as a step-by-step processes, are as the followings :

- Selection of reference spent fuel,
- Establishment of ground rules and assumptions,
- Establishment of general criteria for system design and safety requirements,
- Most promising options regarding spent fuel packages and emplacement methods of packages (analysis, comparison and ranking of feasible alternatives),
- Pre-conceptual design of surface facilities and underground repository, and
- Sensitivity analyses.

According to these step-by-step processes, relevant R&D activities have been performed during the past two years.

Two types of spent fuels arise from the Korean nuclear power program – CANDU-type and PWR-type fuel. For this study, disposal capacity was estimated to be about

36,000tHM (20,000tHM of PWR fuel and 16,000tHM of CANDU fuel) based on the Nuclear Energy Plan (23 PWRs and 4 CANDUs). The reference spent fuel of the two types have been defined based on screening the representative characteristics of all spent fuels from the existing and planned nuclear power plants. The reference spent PWR fuel is giving an average of 45MWd/kgU burnup and is being cooled for 40 years after discharge from a reactor prior to its disposal. A variance level of burnup, as an alternative for spent PWR fuel, is 55 MWd/kgU.

One of the subsystems in the disposal concept conceived for this study is to encapsulate the spent fuel in a corrosion resistant container. High-Ni Alloy and copper are considered as the corrosion resistant material of the disposal container and carbon steel is the insert material. Three types of the containers for the two separate packages of 4 spent PWR-fuel assemblies and 333 spent CANDU-fuel bundles, and for one package, in which both of 4 spent PWR-fuel assemblies and 72 spent CANDU-fuel bundles can be accommodated, have been preliminarily designed.

The containers are then to be deposited in a deep underground facility located at several hundred meters deep in crystalline rock. No site has been specified but a generic site is considered for this study. The containers should be placed in vertical boreholes from the floor in a disposal tunnel or in horizontal boreholes from an access tunnel. Seven different alternatives concerning the emplacement patterns of the containers have been prepared, and also, based on their thermal loads, the distances between the deposition hole centers and between the tunnel centers have been estimated.

The layouts of waste packages in the underground repository with respect to each alternative are sketched, and then the specific thermal loads, the required disposal area, and the excavation rates are estimated. Based upon this information, the alternatives are being narrowed down to one or two most promising option(s) by a typical pair-wise comparison method.

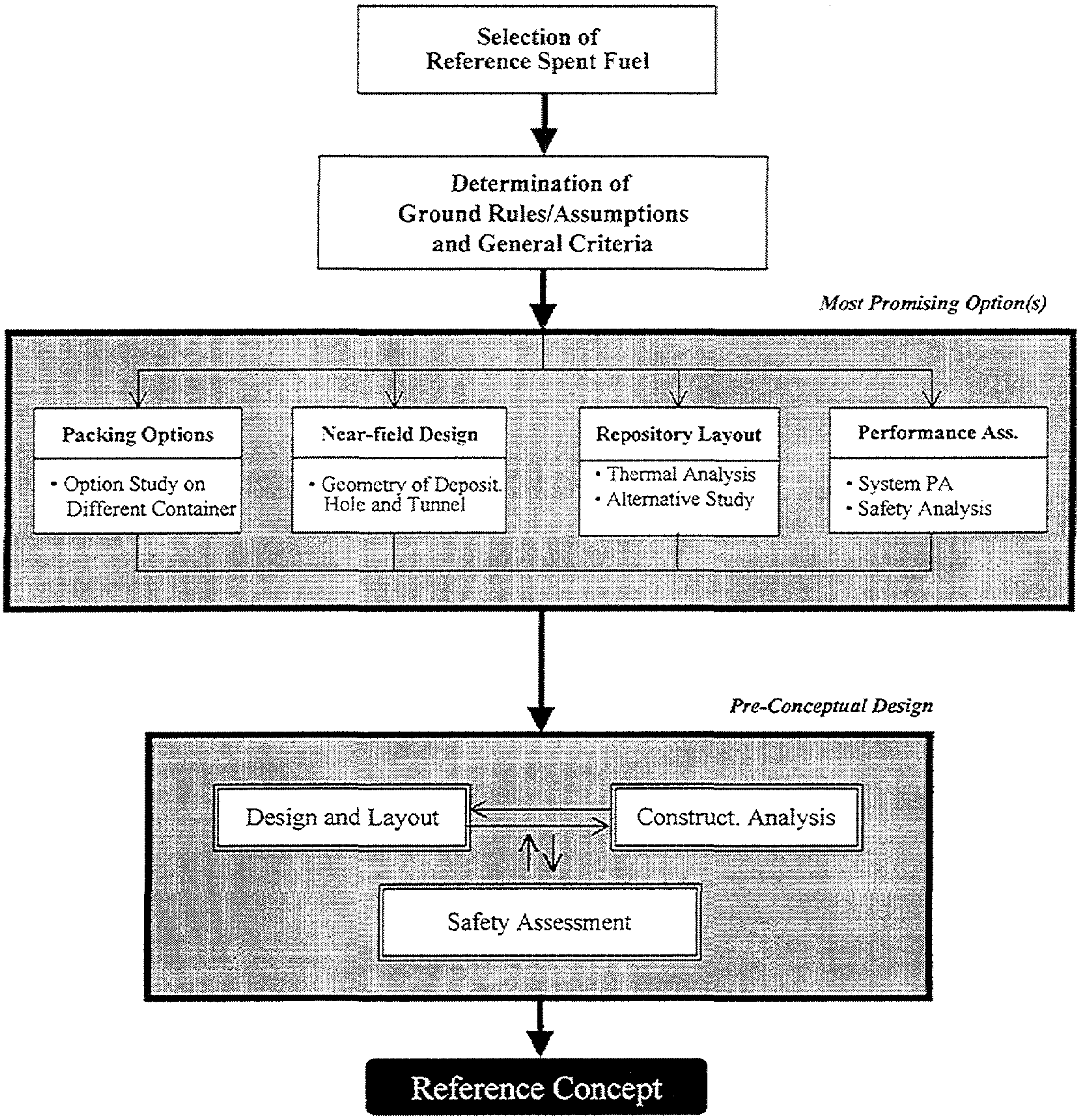

Fig. 1 Approaches for Developing of a Reference Disposal System

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KOREAN WORKING TOWARDS LILW VOLUME REDUCTION

XA9952283

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Environmentally sound design and construction of LILW management facility are very important. However, new treatment technology development for LILW, which can remarkably not only reduce the disposal volume but also enhance the characteristics of waste forms such as leach resistance, gas generation, etc, is more important because it can fundamentally improve the disposal safety. In addition, the new treatment technology enables the design and management of LILW management facility much easier.

Considering the above mentioned matters and delay of disposal site acquisition, the KEPCO/NETEC has made lots of efforts to reduce the disposal volume and to develop a new treatment technology. The KEPCO established the short-term goal to reduce the number to 250 drums per one reactor-year and the long-term goal to 35 drums per one reactor-year. To reach the short-term goal, the KEPCO has encouraged the service inventions and adopted new treatment equipment such as super compactor, spent resin drying system, concentrate waste drying system, etc. As a result the KEPCO could reduce the waste drum per one reactor-year from 550 drums in the early 1990s to 350 drums in 1998(see Fig. 1). The short-term goal for waste volume reduction is expected to reach within 2 or 3 years [1].

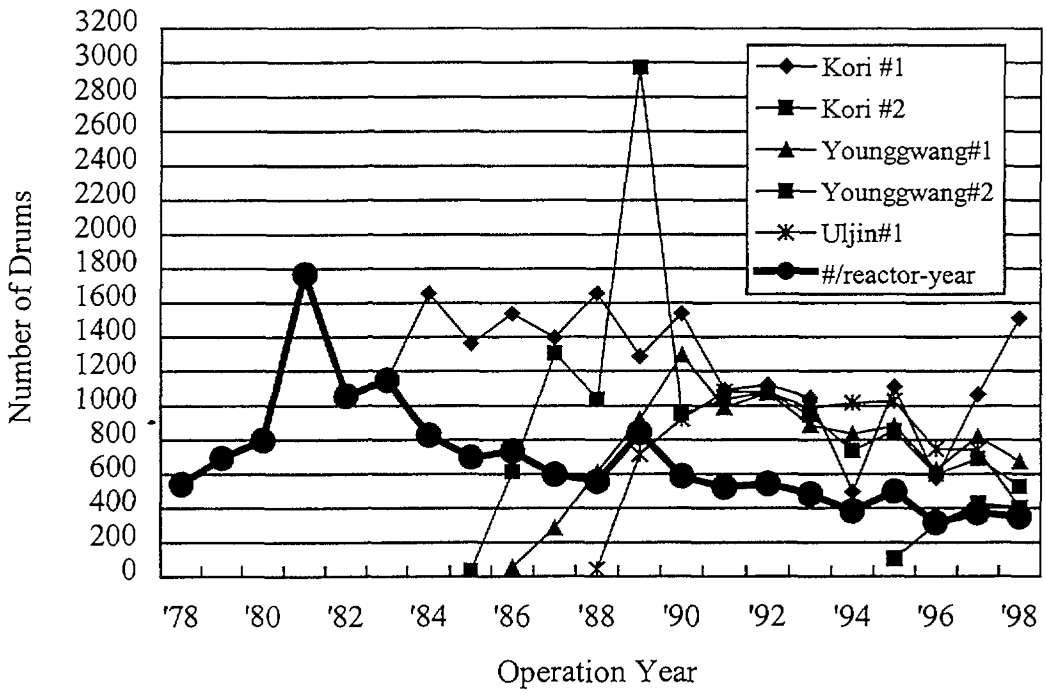

Figure 1. Generation Trends of Waste Drums(200 L size)

Currently, the KEPCO/NETEC has developed the new treatment technology, LILW vitrification technology, in order to both reach the long-term goal and increase the inherent safety for LILW disposal. The vitrification technology shall be able to reduce the number of waste drum generated per reactor-year 10 to 25 drums and to realize the production of environmentally clean waste forms. At the same time, the technology will contribute to change the NIMBY(Not In My Back Yard) to the PIMFY(Please In My Front Yard) attitude of local residents around disposal facilities.

We had already completed the feasibility study on LILW vitrification from 1994 to 1995[2]. Since 1996, we have been carrying out the second vitrification project to develop the pilot plant including cold crucible melter heated by induction current(CCM), Plasma torch melter(PTM), and off-gas treatment system(OGTS).

The pilot scale vitrification facility will be constructed at NETEC by June 1999. The facility consists of two melters such as CCM of 50 kg/hr throughput and PTM of 10 kg/hr capacity, and OGTS composed of high temperature filter(HTP), post combustion chamber(PCC), scrubber, selective catalytic reduction(SCR), etc. This facility might enable all liquid and solid radioactive waste from nuclear power plants to make more stable glassy/slag waste forms with remarkable volume reduction while minimizing the entrainment of volatile chemicals[3].

Shredded combustibles and spent resin are directly fed onto the molten glass in CCM through feeding system with about 20% excess oxygen. The CCM is cooled with cooling water of 110 °C so that its surface contacting molten glass could be kept low temperature of around 200 °C. With cooling, CCM can exclude refractory materials which are inevitably necessary for any others high temperature melter and which make several problems such as increase secondary waste, frequent maintenance for refractory replacement, etc.

The plasma torch of PMT is reverse polarity type and uses nitrogen as plasma gas in order to prevent the generation of NO_x. The melter was designed to have an external refractory and an internal graphite crucible. The crucible has unique structure to easily drain the molten slag and metal. It is favorable that incombustible such as filters, metals, sand, etc are handled with PMT.

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THE APPROACH FOR THE PERFORMANCE ASSESSMENT FOR A HLW REPOSITORY IN KOREA

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In the middle of the '90s, Korea Atomic Energy Research Institute(KAERI) started to work on the fundamental R&D on the permanent disposal of high-level radioactive wastes in Korea. One of the key tasks in this project is the study of performance assessment. In parallel, the Korean Reference Disposal Concept is studied to develop the optimum concept for deep geologic disposal in crystalline rock. To check the safety of the reference concepts as well as alternative options, it is necessary to develop all needed technologies for performance assessment and then apply them to the Total System Performance Assessment (TSPA) of disposal options suitable for the geologic and social conditions of Korea.

The national regulatory frame on the issues of the post closure safety assessment of the HLW repository does not exist yet. However, the guidelines given in the regulations for the disposal of ILLW can be applied to judge the safety of potential disposal concepts in HLW. The tentative post closure safety target at this point is to check 1) whether the annual dose to the individual in the critical group is lower than 2 mRem/yr and 2) whether the probability of fatal accidents such as cancers caused by the repository is lower than 10^{-6} /yr.

The major R&D goals for the study of performance assessment to see whether the Korean disposal system can satisfy the regulatory guidelines are :

1) to develop the tools and database for scenario development
2) to develop needed computational tools and database for TSPA, and
3) to apply the TSPA tools and database to assess the safety of the disposal concepts.

1) Scenario Development

KAERI currently studies the world-wide databases for FEPs to find out the relevant FEP lists for general geologic conditions in Korea. From the list of FEPs, after working group discussions, it is planned to extract important FEPs and then construct feasible scenarios by properly combining these FEPs. In this process, the Rock Engineering System(RES) as well as Process Influence Diagram(PID) methods are applied for the systematic development of scenarios on sub-systems such as natural barriers as well as the total disposal system. Then the scenarios developed in this process shall be reviewed by an expert group and classified as the reference scenario, feasible scenarios, etc.

Once the reference and feasible scenarios are identified, the appropriate TSPA tools for the sequence of events in concerned scenarios shall be identified. This approach shall be valuable for the quality assurance, potential review by outside expert groups including the regulatory body, and continuous update of approaches to TSPA.

Along with these tasks, the parallel R&D to find out the frequencies of natural disruptive events and other external events in Korea shall be studied and recorded into the database.

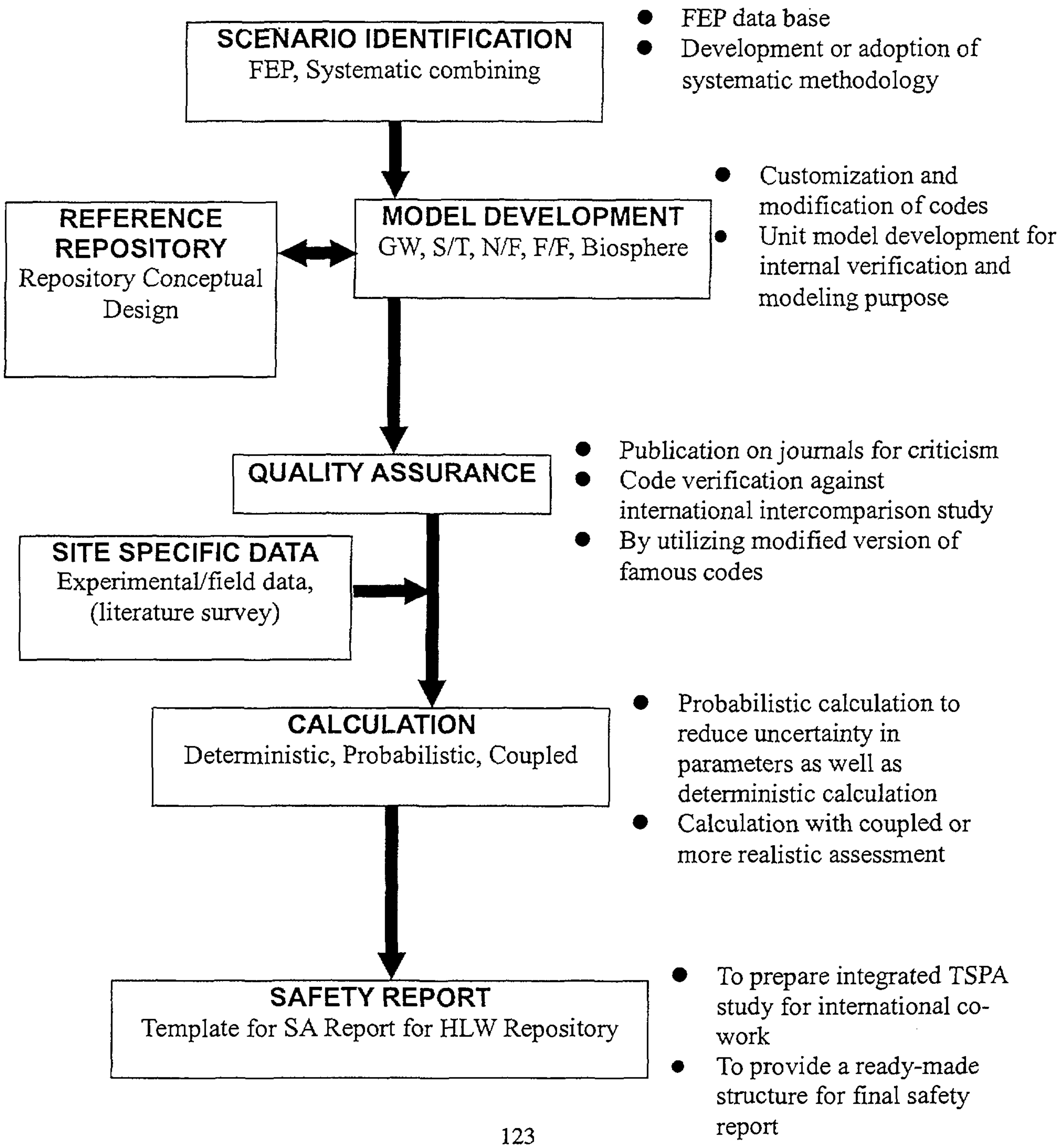
2) Development of TSPA tools

Currently KAERI is developing the overall safety assessment code, MASCOT-K, originally developed for the probabilistic safety assessment of the ILLW repository. In this program the response of the sub-models, i.e. buffer and fractured rock, in the system is transferred to the adjacent sub-system(s) by the Laplace transformed functions which shall eventually be transmitted to the final sub-system, biosphere. New sub-models suitable for the disposal conditions in Korea were and shall be developed. For example, the GAP model was developed to see the importance of highly soluble nuclides such as Cs-135, Cs-137, and I-129 dissolved in the void gap inside canisters which shall be filled with intruding groundwater. New sub-models to express the congruent release, colloid transport, multiple fracture, etc., are under development. In parallel to this approach, a new tool for Time Dependent PSA(TDPSA) shall be developed in the future to see the effect of a potential climate change during the time frame of post closure assessment.

In addition a study to see the coupling effects of thermo-hydraulic-mechanical(THM) processes on the stability and safety of the repository is underway. The near term R&D target in this study is to find out whether the decay heats from the embedded spent fuels creates new significant groundwater flow channels in the vicinity of the deposition holes. The appropriate constitutive laws to govern couplings are reviewed and the evaluations on thermal effects are underway.

3) Generic TSPA for the Idealized HLW Repository

During the fiscal year of 1999, the preliminary TSPA shall be pursued for the idealized repository in the coastal area. Based on the geological study geologic strata to the deep basement shall be assumed. The geologic medium at the location of the disposal tunnels shall be appointed as crystalline rocks such as granite or gneiss. Then the potential pathways as well as travelling times for groundwater from various locations of the disposal tunnels and deposition holes shall be calculated considering the effect of salt intrusion which shall influence the pressure profiles in the coastal area.. In addition, the retardation coefficients and solubility limits of major radionuclides shall be evaluated for the Korean reference deep groundwater. Also the dose conversion factors for Korean critical group shall be evaluated. All of these will be expressed in terms of probabilistic density functions and used as input for the MASCOT-K. In 1999, the preliminary reference concepts shall be scrutinized for safety study. And by 2006 the Korean reference concept and major alternative ones will be assessed to find out whether these proposed systems can satisfy the safety targets.



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IAEA-SM-357/66P

GROUNDWATER EVOLUTION OF THE GRANITE AREA, KOREA

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The groundwater. chemistry is very important in the performance assessment of geological disposal for radioactive waste. Crystalline rocks such as granite and gneiss have been considered as suitable host rocks of radioactive waste repository in Korea. KAERI is conducting hydrogeological and hydrogeochemical investigations of granitic rocks as a part of Radioactive Waste Management Research Program. The granite area of the diversion tunnel for water supply to Yeongcheong dam was chosen as a research site for investigation of bedrock groundwater chemistry.

The hydrochemistry of groundwater belongs to the Ca-HCO_3 type, and is controlled by groundwater systems and water-rock interaction in the flow conduits(fractures) (Fig. 1). The oxygen-18 and deuterium data are clustered along the meteoric water line, indicating that the groundwater in the site is commonly of meteoric water origin and is not affected by secondary isotope effects such as evaporation and isotope exchange. Tritium data show that the groundwater was mostly recharged before the pre-thermonuclear period and have been mixed with younger surface water flowing down rapidly into the tunnel along fractured zones. Water chemistry indicates the groundwater is evolved under partially open system in which CO_2 is supplied continuously.

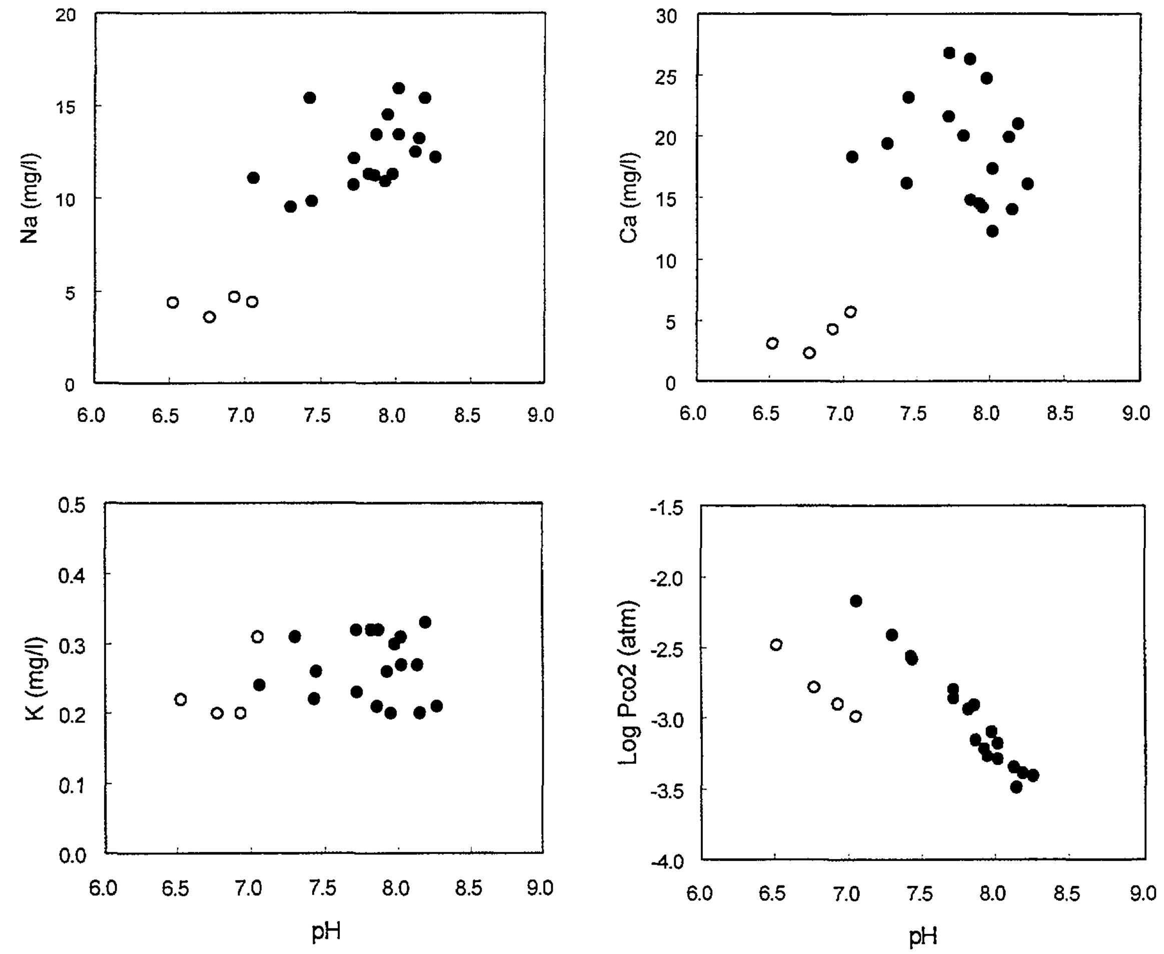

Fig. 1. pH versus Na, Ca, K and Pco_2 diagrams for water samples in the granite area. Open circles: surface water; solid circles: groundwater.

Carbon isotope of groundwater (-17.1 to -17.9‰) reflects the important role of organic carbon in the groundwater in this area. SO_4 in waters is ascribed to oxidation of pyrite formed during hydrothermal alteration in the granite. Sulfur isotope data of groundwater ($\delta^{34}S_{CDT}$ = +2.6-+4.5‰) in the granite area indicate that sulfates are originated from hydrothermal fracture-filling pyrites.

Based on the mass balance and reaction simulation approaches, using both the hydrochemistry of groundwater and the secondary mineralogy of fracture-filling materials, we have modeled the low temperature hydrogeochemical evolution of groundwater in the area (Fig. 2). The results of geochemical simulation show that the concentrations of Ca, Na and HCO_3 and pH of waters increase progressively owing to the dissolution of reactive minerals in flow paths. The concentrations of Mg and K increase in the early stage with the dissolution of reactants, but later decrease when montmorillonite and muscovite are precipitated, respectively. The continuous adding of reactive minerals, namely the progressively larger degrees of water/rock interaction, causes the formation of secondary minerals with the following sequence: hematite, gibbsite, kaolinite, montmorillonite, muscovite and microcline. The reaction simulation results agree well with the observed water chemistry and secondary mineralogy.

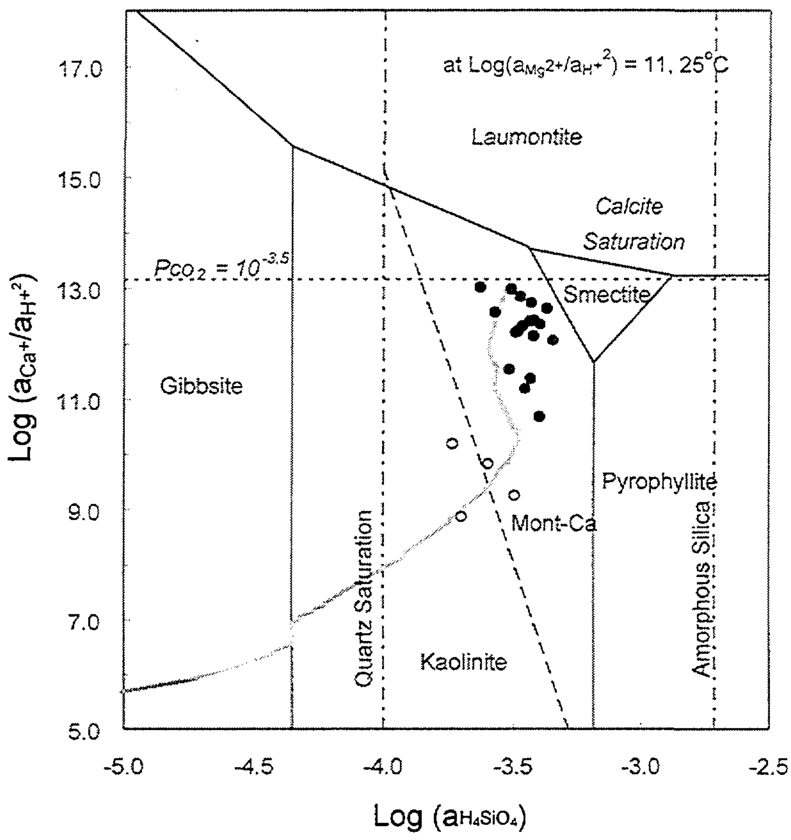

Fig. 2. Stability diagrams for some minerals in the system CaO-Al_2O_3-SiO_2-H_2O at 25°C, showing the probable path (thick solid line) of groundwater evolution. Symbols are same as in Fig. 1.

IAEA-SM-357/67P

LITHOLOGICAL SUITABILITY FOR HLW REPOSITORY IN KOREA

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Korea is located in the area where the Eurasian content is continent with the west Pacific mobile belt. Whereas the Japanese archipelago is characterized by active mobile belt, the Korean peninsula has a close affinity with the Asian continent in geology and tectonic setting. The eastern margin of the Korea-China platform belongs to a part of the shield area regarded as stable cratonic nature, but the peninsula has some difference from the stable platform and is considered as the marginal geosyncline phenomena of platform with superimposed tectonic elements during the Meso-Cenozoic era. For the geotectonic units of the peninsula, the Mesozoic orogeny is most important as a transitional development of platform into geosyncline stage. The Mesozoic tectonic activity is the most vigorous crustal movement in the entire Korean peninsula followed by igneous intrusion. Such tectonic movements intensified gradually from the north to the south and from the west to the east, resulted from the eastward subduction zone of the Pacific plate.

The lithology of the Korean peninsula consists of a complex structure of 29 rock types from Archean to Quaternary. Among these rock types, the preliminary screening is considering based on the geological properties which are important for the barrier function of the far field such as geohydrological properties, geochemical characteristics and regional lithological uniformity. In the present stage, the main criterion is mainly concerned on a large areal extent of rock types allowing flexibility of siting. The plutonic intrusion in Korea occurred from the Proterozoic-Cretaceous era and are occupied by nearly one-half of the peninsula. The majority of igneous rocks are granites and their varieties. Intermediate plutonic rocks of mainly diorite are exposed as small stocks in the southern part and along the east coast of the northern part. The distribution of mafic and ultramafic plutonics is almost limited to the tectonic regions. During the Jurassic-Cretaceous orogeny(180-130 Ma BP), igneous activity resulted in forming a large batholith of granitic rock (Daebo granite). The plutonic activity ceased at the end of Paleogene (60 Ma BP), but transformed into volcanic activity building offshore volcanic islands and rift valleys.

The Daebo granitic batholiths in southern part of Korea, which is the primary host rock, lie parallel to the Okchon fold belt (NE-NNE) and between the precambrian basements (Kyonggi and Sobaeksan massif) (Fig. 1). All samples of plutonics rocks contain microcline and most of them grouped into granite to granodiorite in composition.

Large scale-regional fractures are chiefly developed in the limited area, in contrast small scale fractures are evenly distributed throughout the southern peninsula (Fig. 2). Three dominant sets of NNE (N10W-N50E), NW (N30W~N50W) and WNW (N60W~EW) are defined through the fracture orientation. Regional fractures could be classified into four orders on the basis of their trace lengths. First order having a trace length of over 40km is tectonically most important in the peninsula. Third order fractures of 1-20 km length commonly formed local fracture zones and abundantly throughout southern Korea regardless of geologic settings and tectonic province.

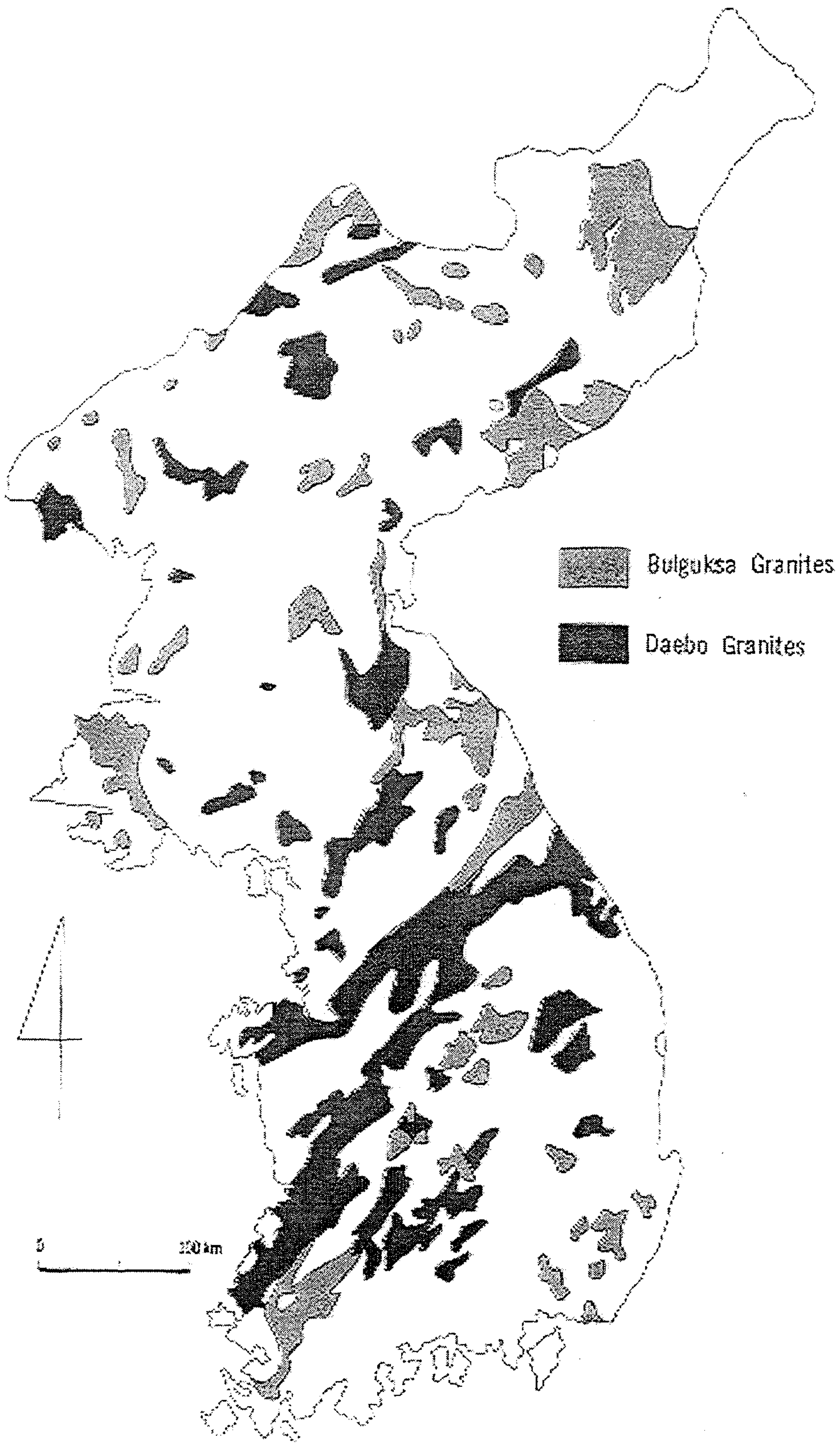

Fig. 1. Distribution of the Daebo and Bulguksa Granites in Korea

Through the geologic history, the peninsula was experienced for a significant tectonic movement in the period between 180 and 100 million years ago. And then subsequent tectonic activities have been diminished and limited to in the local areas. In the present stage, the disruptive natural phenomena such as seismicity and volcanism are characterized as sudden activities in the localized particular areas, which can be avoided by excluding certain areas in the early stage of siting.

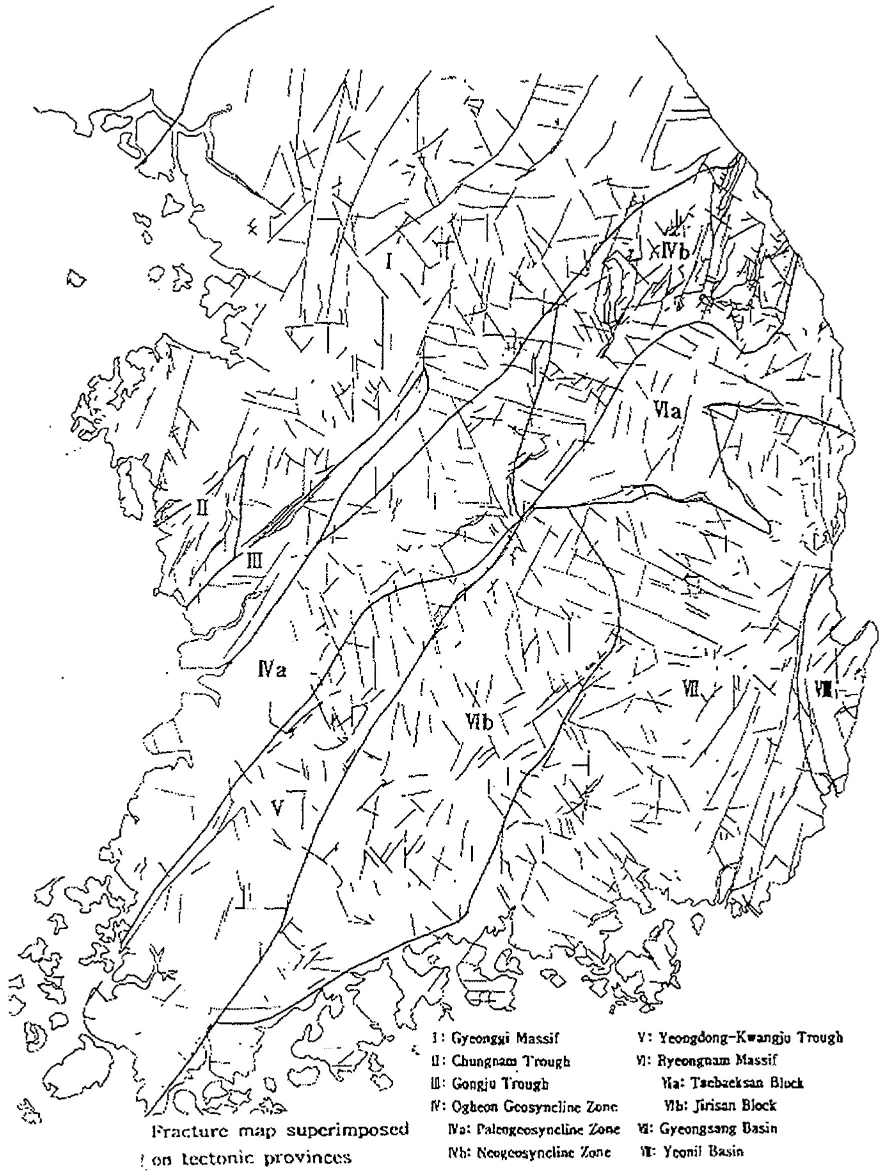

Fig. 2. Fracture map superimposed on tectonic provinces in Korea.

IAEA-SM-357/68P

SWELLING AND HYDRAULIC PROPERTIES OF CA-BENTONITE FOR THE BUFFER OF A WASTE REPOSITORY

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A repository for high-level radioactive wastes would be constructed in the bedrock deep below ground surface, and the present design concepts [1,2] of an underground repository in granite formation include the use of compacted clay-based materials as both buffer and backfill. The buffer material is required to have a good sealing property and thus have a low hydraulic conductivity in order to minimize the penetration of ground water from the host environment.

The bentonite has been considered as a candidate buffer material, and in Korea, is mainly produced from tertiary sediments in eastern Kyungsangbuk-do. This study intends to present the swelling and hydraulic properties of the domestic bentonite. The bentonite was identified to be a Ca-bentonite, and the weight percentage of montmorillonite was obtained to be about 70 %.

Swelling tests, which were designed according to Box-Behnken's experimental scheme, were carried out to investigate the effect of dry density, bentonite content and initial water content on the swelling pressure. Measured swelling pressures were in the wide range of 0.7 Kg/cm^2 to 190.2 Kg/cm^2 under given experimental conditions. Based upon the experimental data, a 3-factor polynomial swelling model was suggested to analyze the effect of dry density, bentonite content and initial water content on the swelling pressure. The swelling pressure increased with an increase in the dry density and bentonite content, while it decreased with increasing the initial water content and, beyond about 12 wt.% of the initial water content, leveled off to nearly constant value.

The hydraulic conductivities of the compacted bentonites with the dry densities of 1.4 to 1.8 Mg/m^3 at 20°C are in the range of $2x10^{-14}$ to $3x10^{-12}$ m/s. The hydraulic conductivities decrease with increasing dry density of bentonite and the relation between the logarithm of the hydraulic conductivity and the dry density of bentonite can be fitted to a straight line. The hydraulic conductivities decrease faster with increasing dry density at higher porosities. The hydraulic conductivities increase with increasing temperature. The hydraulic conductivities at the temperature of 150 °C for bentonites with the dry densities of 1.4 Mg/m^3 to 1.8 Mg/m^3 increase up to about one order of magnitude higher than those at 20°C.

The swelling and hydraulic properties of Kyungju bentonite are comparable with those of bentonites suggested as a candidate buffer material for high-level waste repository in several other countries [3]. The experimental results obtained here will be useful to the selection of a candidate buffer material for a high-level waste repository in Korea.

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XA9952288

IAEA-SM-357/69

ENVIRONMENTAL FRIENDLINESS OF BACK-END FUEL CYCLE OPTIONS

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The environmental friendliness of back-end fuel cycle options was studied with one option for spent fuel direct disposal and two options for thermal recycle, by reprocessing and DUPIC (Direct Use of spent PWR fuel In CANDU reactors), respectively.

The amounts of deep geological disposal wastes, i.e. intermediate- and high- level wastes and the thermal and radiological toxicity of those wastes were investigated for the three different options in evaluation of environmental friendliness during the long-term environment management of the wastes.

BASIC ASSUMPTIONS

Korea now has 14 PWRs and 4 CANDUs in operation or under construction. The basis of the study was to compare the wastes from the three fuel cycle options with the same amount of the electricity generation, i.e. 9.6 GWY and 4.9 GWY from PWR and CANDU, respectively. The electricity generation ratio of 2 to 1 by PWR and CANDU makes the best use of the DUPIC option with PWR spent fuel in Korea by fabricating DUPIC fuel with uniform fissile contents. The fissile contents of DUPIC fuel are 1.0 % and 0.45 % for uranium and plutonium, respectively, which resulted from the optimization study by using 3600 PWR spent fuel assemblies in Korea the fissile contents of which were very different: ± 71.4% and ±20.5% for U-235 and Pu, respectively with 95 % confidence level.

The fuel compositions and the burn-up used in the studies are listed in Table 1.

Table 1. Fuel Burn-up of PWR, DUPIC and CANDU

Fuel Type	PWR	DUPIC	CANDU
Burn-Up,[MWD/MTU]	35,000	14,800	7,000
Composition of DUPIC Fuel	PWR SF(82.5%) + SEU(7.8%)+ DU(9.7%)		

WASTE GENERATION

The specific amount of radioactive waste generation from the three fuel cycle options, i.e. in terms of m^3 per MTU spent fuel were estimated by using the following sources:

1) DUPIC conceptual design study report by Scientech and Gamma engineering on a DUPIC plant with a scale of 400 MTU/year,
2) La Hague and Thorp reprocessing plant data published and privately communicated and
3) the OECD/NEA report.

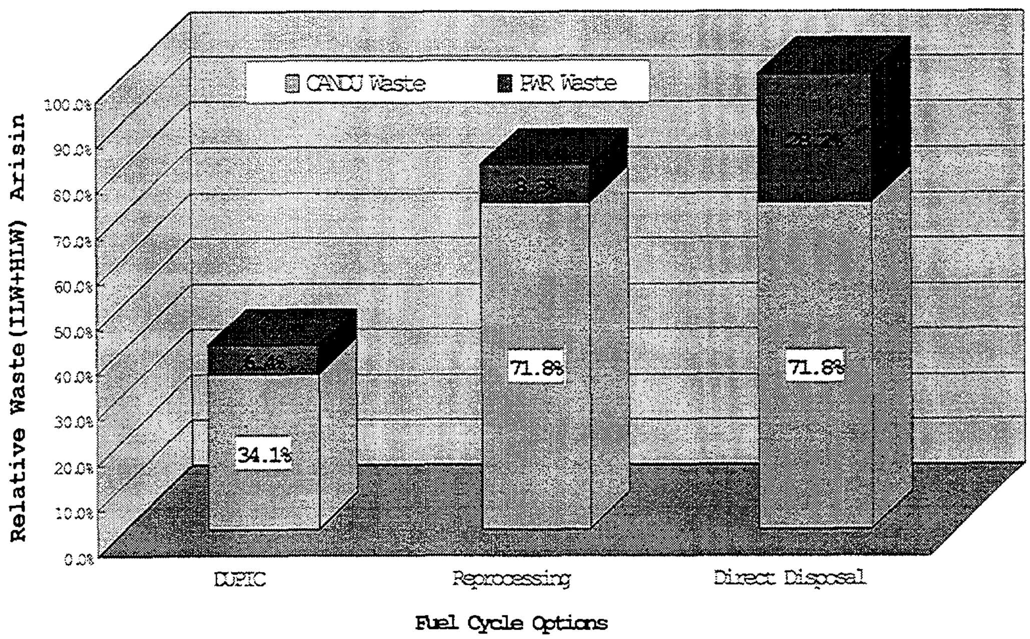
The amounts of spent fuel arising from the three fuel cycle options based on the same electricity generation are shown in Table 2.

Table 2. Fuel Cycle Material Balance, [MTU/yr]

Fuel Cycle Options	`Fuel Composi-tion	PWR			CANDU		
		Fresh Fuel	Spent Fuel	**Electricity**	Fresh Fuel	Spent Fuel	**electricity**
DUPIC	PWR	100	-	**3500 GWD** (9.6 GWY)	100	121	**1790 GWD** (4.9 GWY)
	SEU				9.5		
	DEU				11.5		
Reproces -sing	PWR	100	-	**3500 GWD** (9.6 GWY)		255	**1790 GWD** (4.9 GWY)
	SEU						
	NU				255		
Direct Disposal	PWR	100	100	**3500 GWD** (9.6 GWY)		255	**1790 GWD** (4.9 GWY)
	SEU						
	NU				255		

Figure I show that DUPIC generates small amounts of ILW and HLW: 80% of reprocessing option and 40 % of direct disposal option. The DUPIC option uses two times higher burn-up fuel than the other options so that DUPIC produces two times less spent fuel wastes. The ILW and HLW from the DUPIC process is also less than that of reprocessing.

Figure 1. Deep Geological Disposal Waste Generations from the nuclear fuel cycle options.

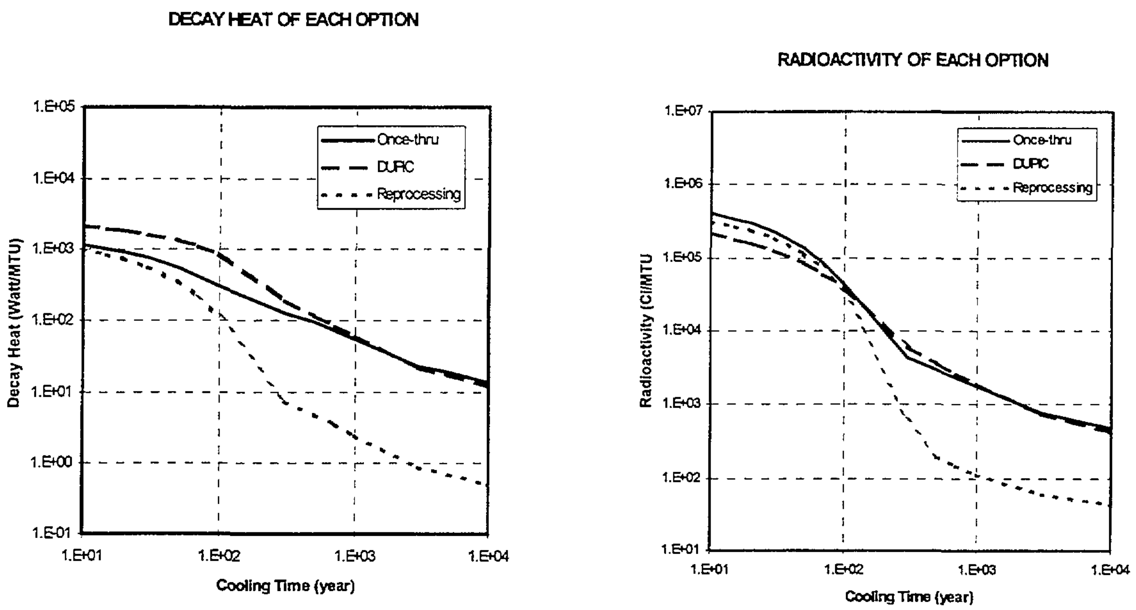


RADIOLOGICAL TOXICITIES

The long-term radiological characteristics of the high level waste from three fuel cycle options were estimated from the spent fuel compositions of PWR, CANDU and DUPIC by decay calculation using ORIGINE 2 where the fuel composition of DUPIC fuel was transferred to WIMS-AECL to perform a DUPIC lattice calculation.

The long-term radiological, thermal and ingestion hazards of the high level wastes of the three fuel cycle options were represented in Figure 2. We found that in a short period of time up to about 100 years the radiological hazards of the three options are comparable to each other but in the long term that of the reprocessing option is the least due to the recovery of plutonium which has a dominant impact in the long time. The radiological hazards of DUPIC and direct disposal options are comparable. This result indicates that the DUPIC option does not add much hazard to PWR spent fuel by one more burning in CANDU reactor.

Figure 2. Radiological hazards of the three nuclear fuel cycle options.



CONCLUSIONS

By comparing the amount of wastes and their long term radiological impacts we concluded that DUPIC is the best choice among the three options from the view point of environment friendliness, when we use two types of reactors; PWR and CANDU.

IAEA-SM-357/70P

CATALYTIC PROCESS FOR TRITIUM EXCHANGE REACTION

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Tritium concentrations in the moderator and the heat transport systems of a CANDU reactor have been increased due to the neutron irradiation of heavy water. This leads to additional public exposure, which subsequently requires the tritium removal facility(TRF) at the PHWR site to reduce the tritium concentration to the desired extent.

The liquid phase catalytic exchange(LPCE) process combined with cryogenic distillation(CD) is one of the available processes for practical application. A hydrophobic catalyst is essential in the LPCE process for preventing the dramatic reduction of the catalytic activity during a reaction, and it has been improved by AECL since the 1970s[1]. Recently, KAERI/KEPRI have also developed a hydrophobic catalyst consisting of polystyrene support and platinum. The catalyst has enough surface area and water expelling property for forwarding the gas-vapor-water reaction adequately.

Activities for the catalyst have been measured in the gas-vapor phase exchange reaction, since the catalyst concerns only the gas-vapor exchange reaction. The results of the catalytic activity have showed fair performance, comparing with the commercial catalyst[2]. The overall reaction for transferring a hydrogen isotope from the liquid phase to the gas phase, however, consists of two steps: a catalytic reaction at the catalytic bed in gas-vapor phases, and a mass transfer reaction at the hydrophilic bed in vapor and liquid.

This study described the performance of a column in which each stage comprised a catalyst bed for the gas-vapor reaction and a hydrophilic bed for the mass transfer between vapor and liquid. The experimental apparatus is shown in Fig. 1. The column included a humidifier to saturate the inlet gas and two stages in which both the catalyst bed and the hydrophilic bed were configured. The deuterium in heavy water streamed in from the top of the column and the hydrogen gas flowed up from the bottom of the column, resulting in the transfer of deuterium from the heavy water to the hydrogen gas. The outlet concentration of deuterium in the hydrogen gas was measured by a gas chromatograph to analyze the column performance.

The model calculation for the outlet concentration of deuterium in hydrogen providing the efficiencies of the catalytic bed, η_c and the hydrophilic bed, η_p are available elsewhere[3,4]. Fig. 2 shows the experimental results of the outlet concentration of deuterium, and compares them with the computed results carried out with assumed η_c and η_p values. The column efficiency increased with the number of stages, as shown in the figure. The increase of the molar ratio of gas to liquid also raised the column efficiency. The experimental results lay between the computed results with η_c=1.0 and η_p=1.0 and the results with η_c=0.9 and η_p=0.9. This implied that the column performance with respect to the internal design configuration was excellent.

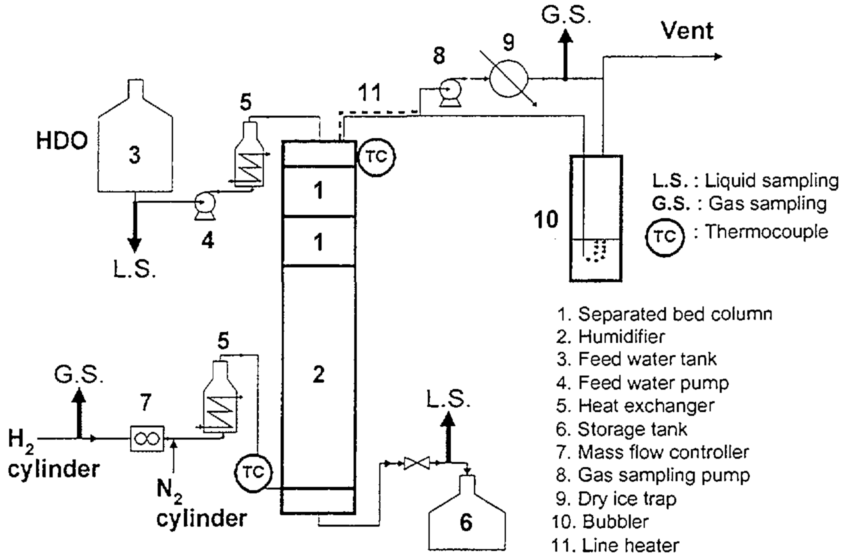

Fig. 1. Schematic view of experimental apparatus for hydrogen isotope exchange reaction.

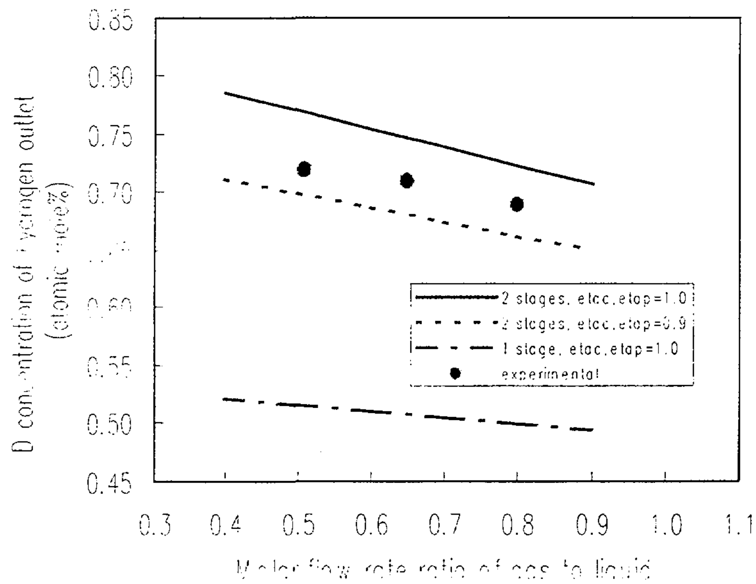

Fig. 2. Experimental results for hydrogen isotope exchange reaction

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XA9952290

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TRAPPING CHARACTERISTICS FOR GASEOUS CESIUM GENERATED FROM DIFFERENT CESIUM COMPOUNDS BY FLY ASH FILTER

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The semi-volatile gaseous radioactive wastes such as cesium and ruthenium can be released from the OREOX (Oxidation and Reduction of Oxide fuel) process for manufacturing the DUPIC(Direct Use of PWR Spent Fuel in CANDU) nuclear fuels[1]. Among semi-volaitile gaseous radioactive wastes, cesium is one of the most hazardous and leachable radioactive fission products. Cesium compounds such as Cs_2O, CsI, Cs and CsOH are expected to generate during OREOX and sintering processes of DUPIC fuel fabrication. Therefore these cesium compounds must be trapped as a stable form in the off-gas treatment system of DUPIC fuel fabrication process. Experiments are performed to evaluate trapping characteristics of gaseous cesium with fly ash filter.

Trapping experiments for gaseous cesium generated from cesium silicate, CsI and CsOH by fly ash filter were performed in a two-zone tube furnace under the air and hydrogen conditions at 800□ for 12 hours. As shown in Fig. 1, cesium compounds in the first hot zone were used to generate a controlled source of gaseous cesium, which was scheduled to pass through fly ash filters mounted in the second hot zone. To manufacture fly ash filter, fly ash powder and binding material were mixed together to make a uniform slip solution. This slip solution was impregnated with a polyurethane sponge. Fly ash filter was made by removing extra slip solution followed by drying and sintering processes. The fly ash filter has an inner diameter of 45 mm, thickness of 10mm, average weight of 16.1g. The weight ranges from 14.5 g to 16.7g.

The color of the fly ash filter was brown. As shown in Fig. 2, when cesium silicate glass was used as the source of gaseous cesium, the color of the fly ash filter changed from brown to dark brown under the air atmosphere. It changed from gray to black under the hydrogen atmosphere. When cesium iodide and cesium hydroxide were used as the source material, the same color change was observed except that it changed from gray to dark blue under hydrogen atmosphere in cesium hydroxide case.

The trapping results of fly ash filter for gaseous cesium generated from cesium silicate, CsI and CsOH by fly ash filters indicated that pollucite ($CsAlSi_2O_6$) and Cs-nepheline ($CsAlSiO_4$) were mainly formed.

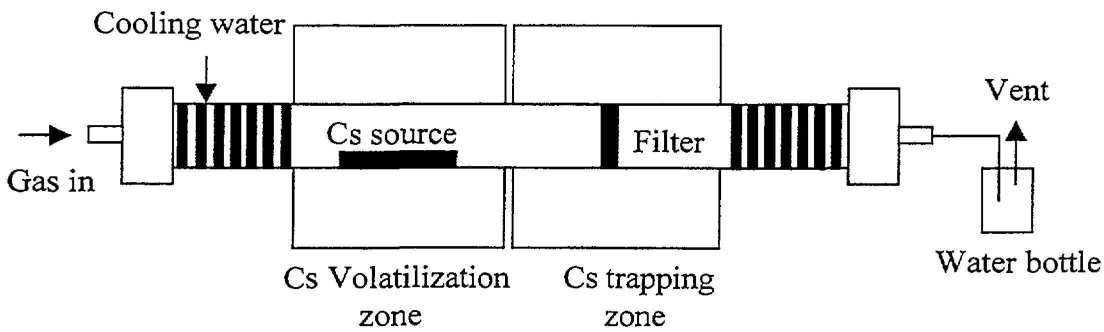

Fig.1. Schematic diagram of the experimental apparatus for trapping Cs

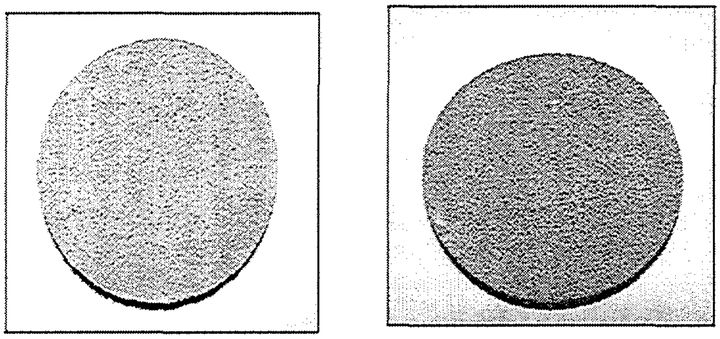

Fig. 2. Photographs of fly ash filters before and after trapping gaseous cesium.

Fly ash filters will be used for trapping gaseous cesium generated during OREOX and sintering processes of DUPIC green pellets. In order to show an example of fly ash filter to be applied for DUPIC manufacturing process, off-gas treatment system for sintering process to be used in a hot cell of IMEF (Irradiated Material Examination Facility) is shown in Fig. 3. As shown in Fig. 3, it consists of Cs trapping unit, iodine trapping unit, TGT (Thermal Gradient Tube) and HEPA filter, etc. More than of 90% of cesium is expected to be volatilized in sintering process, which will be implemented at 1650□ under hydrogen(Ar+4% H_2) condition during 8 hours. Cs trapping unit will be performed to trap gaseous cesium at about 800□. Fly ash filter will be mounted into the Cs trapping unit as a form of disk.

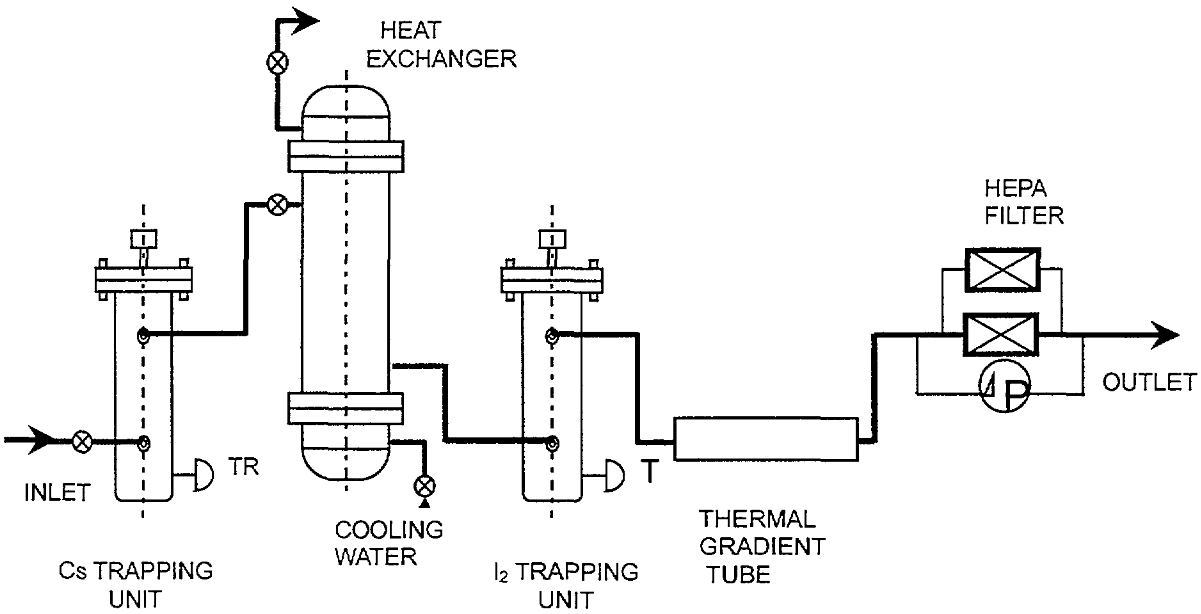

Fig. 3. Flow diagram of off gas treatment system for sintering process in IMEF

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IAEA-SM-357/72P

EVALUATION OF OPTIMAL SILVER AMOUNT FOR THE REMOVAL OF METHYL IODIDE ON SILVER-IMPREGNATED ADSORBENTS

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The adsorption characteristics of methyl iodide (CH_3I) generated from the simulated off-gas stream on various adsorbents such as silver-impregnated zeolite (AgX), zeocarbon and activated carbon were investigated. An extensive evaluation was made on the optimal silver impregnation amount for the removal of methyl iodide at temperature up to 300°C. The degree of adsorption efficiency of methyl iodide on silver-impregnated adsorbent is strongly dependent of impregnation amount and process temperature.

Preliminary adsorption test for unimpregnated adsorbent was conducted to screen an efficient base adsorbent. The adsorption capacity of activated carbon is markedly decreased as the temperature increases. On the other hand, 13X showed the higher adsorption capacity of methyl iodide as the temperature increases. Zeocarbon shows a similar adsorption pattern compared to that of activated carbon at higher temperature range. It is implied that synergy effect due to the mixing of two materials does not appear in adsorption capacity. Low adsorption capacities on both activated carbon and zeocarbon at high temperature would be caused by higher desorption rate than adsorption rate. As shown in previous many investigations, these results would be another evidence that activated carbon has a poor methyl iodide retention at high temperatures compared to zeolite 13X. The increase in the adsorption capacity of methyl iodide on 13X at higher temperature is due to both physisorption and chemisorption occurred in the substrate of 13X.

Adsorption capacities of methyl iodide on the various AgX were obtained as a function of adsorption temperature and the amounts of silver impregnation. Tests were conducted to determine the maximum adsorption capacities of 13X and three types of AgX, such as 10 and 20, 30 wt%. Variations in net adsorbed amounts and the desorbed amounts of methyl iodide with adsorption temperature are shown in Fig. 1 and 2. As shown in Fig. 1, the adsorption capacity of methyl iodide on each AgX showed maximum value in the range of 150°C to 200°C. It would be considered that the optimal temperature for removal of methyl iodide by AgX is about 200°C. The decrease in adsorption amounts at 300°C may be due to higher desorption rate than adsorption rate including chemical reaction between silver and methyl iodide. Effect of silver impregnation amount on the adsorption capacity is observed that the higher silver impregnation amount, the greater adsorption capacity of methyl iodide. The adsorption amount of methyl iodide, however, is not proportional to their silver amounts. This result would be inferred that the all of the silver impregnated in the matrix would not react with the methyl iodide as silver amount increases. The reason that the higher silver impregnated adsorbent has the much portion of the unreacted silver in the matrix was considered that the partial blocking of the adsorption sites was occurred.

Based on the adsorbed amounts of methyl iodide on various AgX, the optimal silver utilization for the removal of methyl iodide was obtained under the conditions that impregnated silver amount is about 10 wt% at the temperature of 150°C.

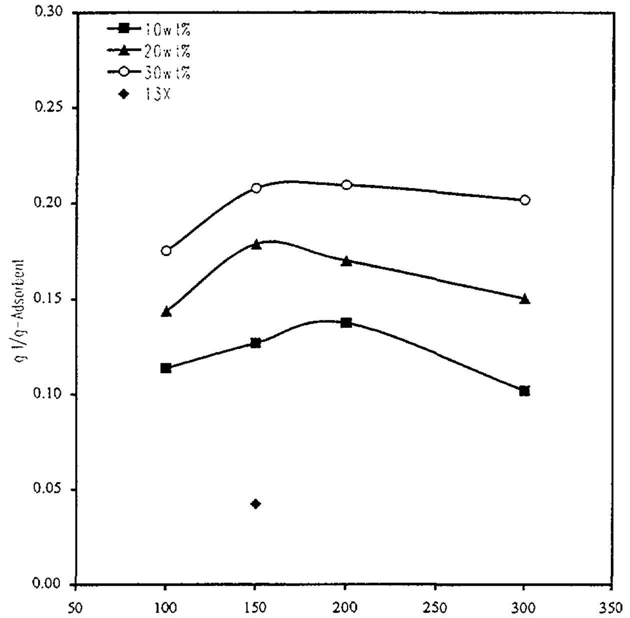

FIG. 1. Adsorption amount of methyl iodide on 13X and various AgX as a function of adsorption temperature.

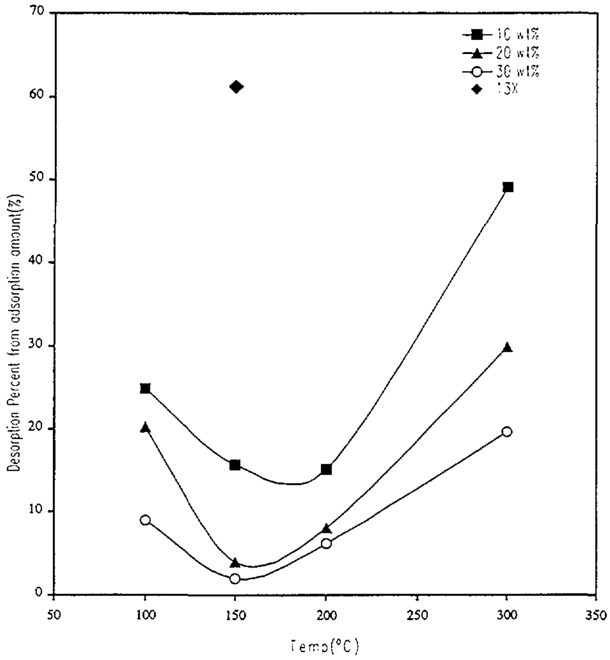

FIG. 2. Desorption percent of methyl iodide adsorbed on 13X and various AgX as a function of desorption temperature.

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XA9952292

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DECONTAMINATION OF SOIL CONTAMINATED WITH CS^{+} AND CO^{2+} IONS

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The two kinds of soil decontamination techniques were studied, one by electrokinetic remediation and the other by soil washing.

In electrokinetic remediation, electrodes are implanted in the soil, and a direct current is imposed between the electrodes[1]. In this study, the feasibility of using electrokinetics to remove Cs^{+} ion from the clayey soil is discussed. Preliminary results of the variation of the electroosmotic velocity, along with the decontamination efficiency on the Cs^{+} ion, are presented.

Saturated clay samples were prepared from the dry powder kaolin clay of Showa Chemicals Inc.. The dry powder was mixed with an aqueous solution of 0.01M Cs^{+} ion to a liquid fraction of 40 % by weight. The homogenized mixture was loaded into the test cell gradually. Tapping the test cell was repeated until nearly all of the air bubbles were removed. A schematic diagram of laboratory scale experimental apparatus is shown in Fig. 1.

The electroosmotic velocity was calculated from the relationship between the cumulative volume of effluent and time. When the applied electric field was changed from 1.5 to 2.0 V/cm, the electroosmotic velocity changed from 1.04×10^{-4} to 1.38×10^{-4} cm/sec. The electroosmotic permeability coefficient of the system is found to be 4.27×10^{-6} cm^2/V.s.

When the applied electric field was 1.5 V/cm, motion of acid front from the anode and base front from the cathode was also investigated. In an early stage of the test, the speed of acid front was faster than that of base front. After the initial 24-hr period, it appeared that the both fronts gradually speeded down.

After the 72 hrs' experiment was terminated, soil samples were removed at various positions and the relative concentration of Cs^{+} ion was measured by XRF. When the applied electric field was 1.0 V/cm, Cs^{+} ion was not detected at the normalized distance of 0.75 from the anode.

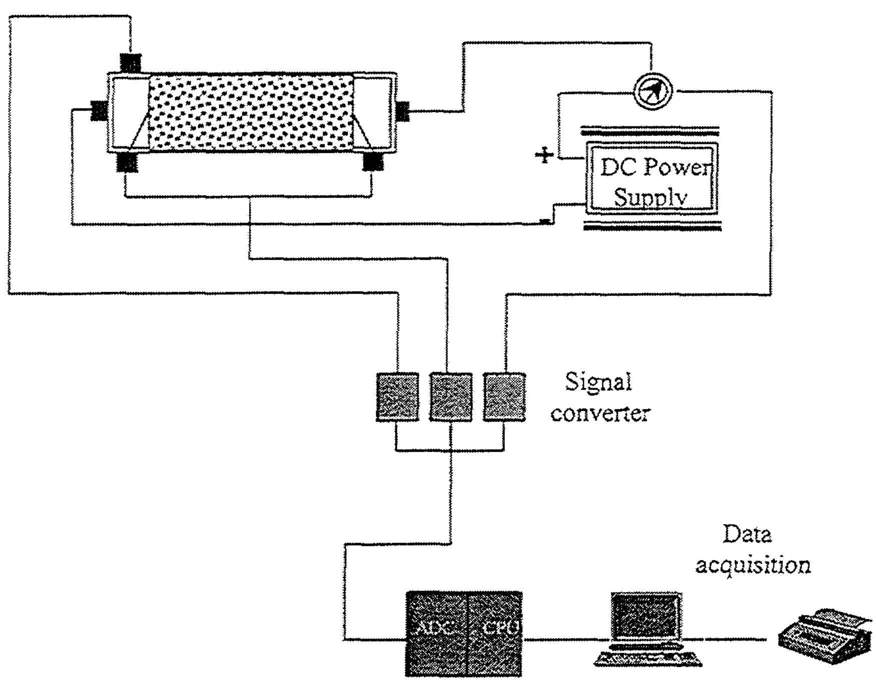

FIG. 1. A schematic diagram of experimental apparatus for electrokinetic remediation of Soil.

In soil washing, complexing agent such as EDTA, EDA, DTPA etc. can be added to the washing solution[2]. A series of batch scale tests was conducted to investigate the decontamination efficiency of EDTA on soil artificially contaminated with Co^{2+} ion in the temperature range from 25 to 55℃, under the pH solution of 4.0 to 9.0. The results showed that the decontamination efficiency was decreased with an increase of the solution pH. As shown in Fig. 2, this tendency is well in accord with the increase of surface potential of quartz type SiO_2 which is the main component of soil. Ferric ion was dissolved out from the soil. This can be explained by the dissolution of FeO(OH) by H^+ ion and then by the $CoEDTA^{2-}$ ion. At a given temperature, it was found that the amount of desorbed Co^{2+} ion was directly proportional to that of the dissolved Fe^{3+} ion. At a given time, the amount of dissolved Fe^{3+} ion steeply increased with the solution temperature.

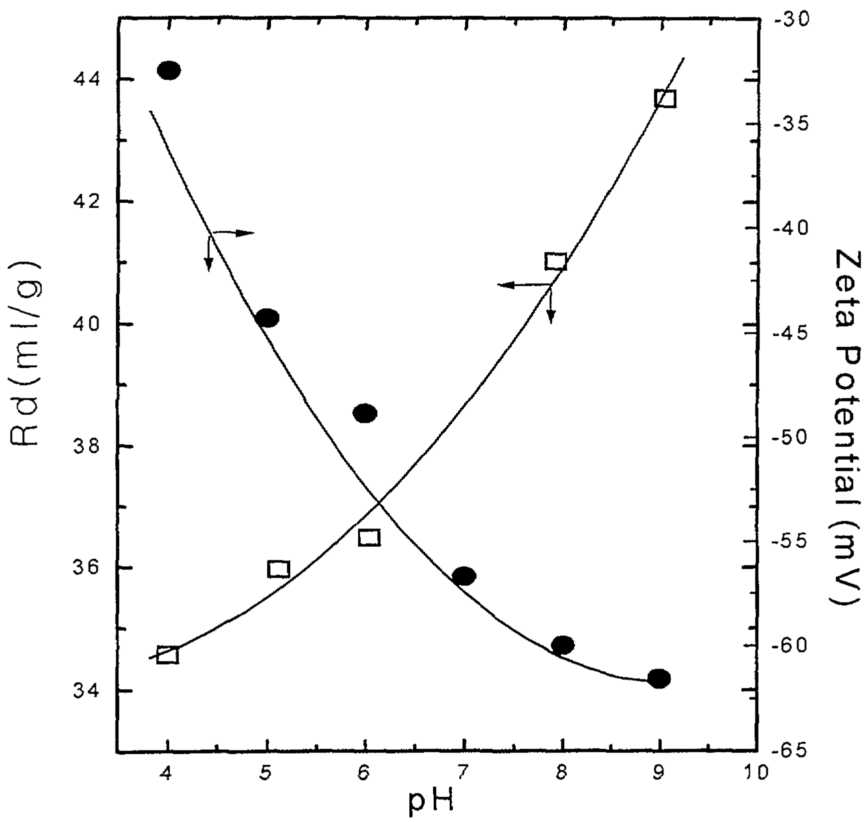

FIG. 2. Double plot of Rd value of Co-EDTA and zeta potential of SiO_2 against pH.

Acknowledgement
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IAEA-SM-357/74P

SORPTION DATA BASE FOR PERFORMANCE ASSESSMENT OF RADWASTE REPOSITORY

XA9952293

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Sorption data base (SDB) provides readily available data for the performance assessment of radwaste repository when site specific data are not available and/or more reference data are needed. The software developed in KAERI (SDB-21C) is a graphic user interface (GUI) program that provides efficient and user friendly tools for evaluating the large amount of sorption data. The data base compiled in the program includes about 11,000 NEA data and 1,100 KAERI data up to now while the addition of new data is under progress. The functions of adding, searching, graphic representing and analyzing sorption data are all integrated in the SDB-21C.

1. INTRODUCTION

SDB-21C was developed not only to provide a user-friendly tool for sorption data base but also to compile the sorption data produced in KAERI during the last ten years. Unlike the NEA sorption data base, SDB-21C provides the functions of adding, searching, graphic representing and analyzing sorption data in an integrated manner, and also contains the parametric model. The sorption data produced in KAERI were added in the format used in the NEA sorption data base.

The program can be used to:

- Determine a single K_d value in the specific conditions of interest,
- Determine the range of K_d value when the specific data are not available,
- Estimate overall trend of sorption between the specific geological material and radionuclide of interest,
- Predict sorption behavior with three-dimensional K_d graph produced by the parametric model.

2. COMPILATION OF KAERI SORPTION DATA

Around 1,100 distribution coefficients were collected up to now in a consistent manner, and the addition of new data is under progress. The format for the sorption data and related experimental information was same as that of the NEA sorption data base in order to develop a general sorption data base. The compiled sorption data in SDB-21C were shown in Figure 1.

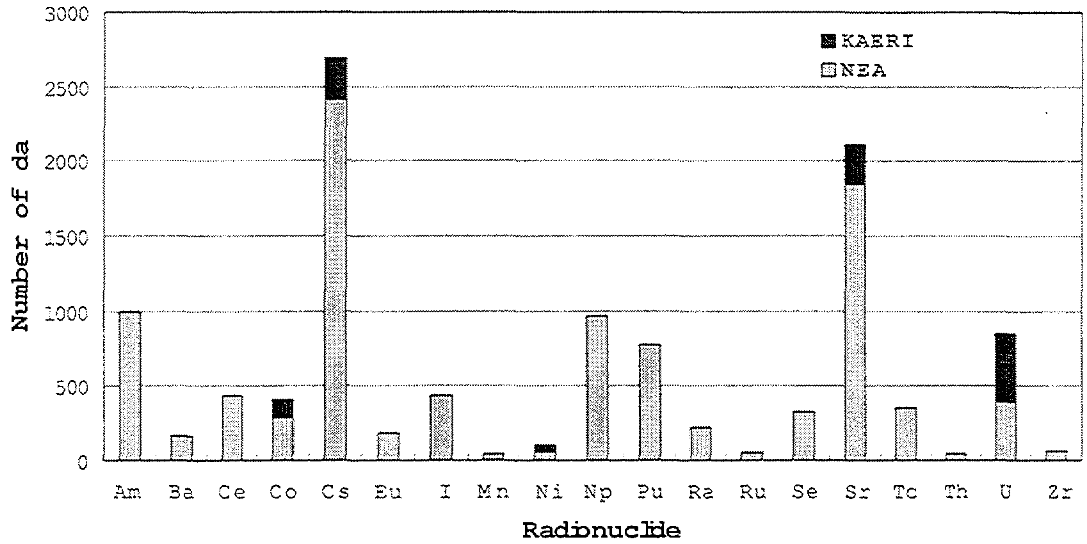

Figure 1. The compiled sorption data for radionuclides in SDB-21C.

3. FUNCTIONS OF SDB-21C

The important functions included in SDB-21 are as follows.

(1) Search system

- The operation is faster compared with dBASE III and MS Access since less memory is used to execute it.
- It is a user-friendly graphic user interface (GUI) tool.
- A variety of customized menu is provided thus it is efficient to search sorption data.

(2) Tabular and graphic representations

- It tabulates search result and the table can be saved as text format, thus the result can be exported to other spreadsheet programs.
- It can analyzes search result by three different graphic presentations named K_d, % and isotherm distributions.
- The Kd values specially obtained for the parametric model can be evaluated by a dynamic three-dimensional K_d graph.

(3) Other functions

- New sorption data can directly incorporated into data base.
- Search results such as table and graph can be printed out.
- Sorption data and the related information for reference and solid type are directly interconnected in the program.

4. CONCLUSIONS AND FURTHER WORKS

The computer program, SDB-21C, is versatile software for the evaluation of sorption data and contains large amounts of sorption data produced by NEA and KAERI. By adopting a graphic user interface (GUI), it is easy to operate and it provides various representation tools for sorption data.

Up to now, only KAERI sorption data have been added, but the addition of new data from literature is under progress. And the development of statistic analysis of sorption data will be included in the program.

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A STUDY ON THE GENERATION OF RADIOACTIVE CORROSION PRODUCT AT PWR FOR EXTENDED FUEL CYCLE

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Nowadays nuclear power plant operation practice shifts to extended fuel cycle such as from 12 month operating cycle to 18~24 month operating one. It is important to emphasize that the current trend to longer fuel cycles has complicated the dilemma of finding optimum pH range for the primary coolant chemistry. Typically, current 12-month fuel cycles start with no more than 1200ppm boron concentration at the start of a cycle, so a maximum value of 2.2 ppm lithium is supplied to satisfy the pH=6.9 requirement. The recent ICRP recommendation (ICRP publication No. 60) for the radiological protection requires more strict reduction of ORE in the nuclear power plant. Although CRUD is not high level waste, it is very important products because CRUD is the major source of ORE and its transport mechanism is not specified yet. The major sources of the radiation are produced by the neutron activation of the non-radioactive corrosion products at the reactor core, and then the radioactive corrosion products are transported back to the outside of the core, and usually accumulated near the steam generator side at PWR.

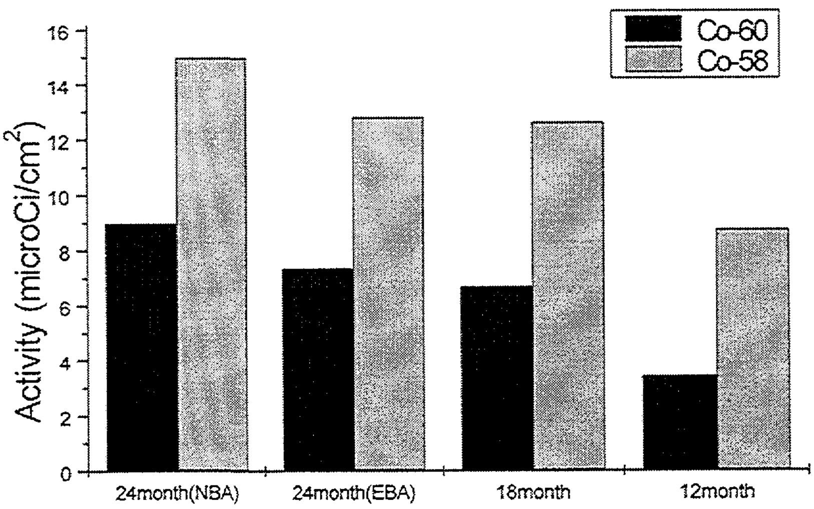
To simulate the generation of CRUD at the extended fuel cycle, the COTRAN code used, which was developed to simulate the behavior of the radioactive corrosion product (CRUD) for Korean Nuclear Power Plant and was developed with double layer concept model and are differentiated as soluble and particulate CRUD. The original COTRAN code was utilized to predict the behavior of the CRUD during only one fuel cycle. In this paper, however, the behavior of the CRUD for multi fuel cycle is simulated and predicted. In order to describe the corrosion products behavior for multi fuel cycle, COTRAN code is modified to consider the effect of decontamination, refueling and also applied to KNGR (Korea Next Generation Reactor). It is known that a certain amounts of the CRUD are decontaminated utilizing coolant shutdown chemistry technique. It is assumed that one-third of the fuel is refueled every fuel cycle and consequently, one-third of the generated CRUD is removed at core. The simulated results show that not only selecting the optimum pH value but also cycle duration are important parameters reducing the generation of the CRUD. It turned out that the activities of CRUD decreases as the pH of the coolant increases, and for the same period of different fuel cycle, as the operating fuel cycle duration increased, the generation of the CRUD increases.(Figure.1) As the operating cycle duration is extended, the ratio of Co^{58}/Co^{60} becomes smaller. Especially, activities of Co^{60} and Co^{58} shows similar trend compared to the measured activities of those at steam generator tube surface of Millstone Point 3 PWR.

In this paper, enriched boric acid (40% enriched B^{10}concentration) for the reactivity control is adopted as the required chemical shim rather than natural boric acid. The effect of the enriched boric acid (EBA) is that the neutron absorption capability of the chemical shim is maintained while decreasing the required boron and lithium concentration in the reactor coolant system. By applying enriched boric acid, the amounts of generated CRUD are reduced because the high pH operating period is extended. In the point of waste view, more filters or ion exchangers to remove CRUD are required and amounts of waste are increased at the extended fuel cycle. To solving this problem, the study on new water chemistry regime or advanced material for primary loop has to be continued.

It is recommended to acquire empirical formula in order to predict the amounts of CRUD with several key parameters (cycle duration, pH, decay constant, Time). Using multiple regression analysis method, the empirical formula was developed and it is well agreed with the simulated computer code results.

Ultimately, the purpose of this study is eventually to reduce ORE of PWR workers through the more careful and precise estimation of CRUD generation.

Fig 1. Activity of Steam Generator tube Surface After 1800EFPD



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RADIOACTIVE WASTE MANAGEMENT AT EDF'S PLANTS: GENERAL OVERVIEW AND PERSPECTIVES

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INTRODUCTION

This paper presents the experience and lessons of general interest drawn by EDF regarding Low and Intermediate Level Waste and forward perspective for the future.
A significant decrease of solid radwastes generated at EDF plants has been recorded. However, the remaining issues are two : first, some particular wastes can't get an optimal issue with processes presently in use so that EDF has developed new ones such as fusion and incineration ; second, minimizing packages volumes makes unavoidable increase of activity contained, so that B-Type containers are now required for transport of many filter cartridges from the primary circuits .

RW - INTEGRATED PRE-DISPOSAL SYSTEM

EDF applies the same QUALITY approach to RWM as that used for other activities with national instructions regarding the collection, conditioning, transport and storage of ILLW. Each plant define responsabilities and relations between the different services involved in RWM.

The various tasks are carried out so that the final product meets ANDRA specifications. On the one side, ANDRA teams regularly make sure the packages are in accordance with the requirements as to the kind of waste contained, the conditioning method used, the mechanical characteristics and the evaluation of the radioactivity. On the other side, Quality controls are performed by EDF on-site teams and each supplier does set up a QA system. To meet this objectives, management decisions aim to keep dosimetry and costs ALARA.

This dispositions have been completed since January of 1991 by a full computerized data tool providing the characteristics of each package to be disposed of, which are sent for acceptance to the ANDRA Centre before shipping.

CURRENT TECHNOLOGIES AVAILABLE for each waste stream

Technological Waste issuing from maintenance activities are currently packed in 200 liter-metallic drums after compaction when useable. Some exceptions to be noticed are :

- metallic pieces, oils, solvents (to be sent to the CENTRACO Plant : see further)
- VLLW (mainly generated during decommissioning) to be sorted waiting for a dedicated disposal
- technological waste with dose rate exceeding 2 mSv/h (to be packed in concrete containers).

In the next future burnable technological waste are to be sent to the CENTRACO incinerator, such as Ion-exchange spent resins issuing from the SG blowdown system (temporary stored at the plants).

ILW spent resins generated by the treatment of primary circuits are kept in tanks at the plants to be embedded in a polymer matrix using a mobile unit (about every two years). The container is made of « high performance concrete » and contains metallic shields ensuring dose rate at the surface is far below the acceptable one (2 mSv/h). Shielding thickness (shields will later be made of recycled scrap metals) depends on the massic activity of the resins measured before packaging. New mobile unit allows optimizations of dosimetry, cadences of conditioning and waste volume per container.

ILW filter cartridges are to be packed either in metallic drums (after drying only if the contact dose rate doesn't exceed 2 mSv/h), or in concrete container for higher dose rates ; in this case, lead shields can be added before cementation of the cartridges.

Although their activity is quite low compared to those of spent resins and filters, evaporator bottoms are cemented in particular concrete containers provided with a stirrer.

For packages to be disposed of, ANDRA Specifications set up an activity level (per nuclide) below which a simple immobilization of waste ensures required mechanical resistance ; on the opposite for higher activities, an extra enclosure (usually concrete) is needed to delay as long as possible the water influx leaching the package. When the container itself assures nuclides containment during its planned lifetime, it is called a « high performances » one.

As to the radioactivity evaluation two methods are currently in use :

- gamma spectrum on samples (resins and concentrates)
- standard average spectrum connected to a dose rate measurement after packaging (technological and filter cartridges).

Activity of beta/gamma long-lived emitters is calculated using correlations issuing from analysis on filters and resins samples.

PERFORMANCES ASSESSMENT

The production of RW from EDF's plants expressed in volume of waste packages ready for final disposal was reduced from 360 m3/plant/year in 1985 to about 100 m3 at the present time. Main progresses concern process wastes and at the same time a significant decrease has also been recorded for liquid and gaseous effluent releases and, although less significant, for technological wastes volumes.

Du to more stingent requirements, a few waste streams needed pre-treatment and some of them were strictly limited or prohibited. To face these particular problems, EDF decided to set up centralized treatment methods such as fusion and incineration (both presently to be in use at CENTRACO plant). Technical and environmental considerations (volume reduction and full-stabilized issuing waste) have leaded EDF to promote the incineration project. The fusion project planned in the 1990's aims at the same, including partial re-use (within the nuclear industry) of metallic pieces such as shields. The CENTRACO plant (see below) is supposed to make sure a suitable issue for each kind of waste that needed pretreatment before disposal.

FUTURE PERSPECTIVES

The CENTRACO Plant includes a melting unit for contaminated scrap metals (just starting ; 1,500 ton/year capacity) and an incineration one for low activity level technological wastes (3,500 ton/year solids + 1,500 ton/year liquids ; to start in april of 99).
The melting unit will generate two main products (depending on the activity of the entering scraps) :

- iron shields to be re-used in concrete containers (low activity scraps)
- ingots to be disposed of (scraps activity exceeding recycling criteria)

pointing out clearly EDF's liability to recycle what can be re-used and to replace lead shields by iron ones (quantities of toxic materials as lead are now limited on the disposal centre).

Concerning the incineration of LLW the main advantages are two and apply both for the future and the present :

- reducing the volumes of packages will allow to extend the lifetime of the disposal center;
- transforming raw wastes containing a low proportion of organic materials into stable and homogeneous secondary waste will consequently increase safety of the disposal centre and safety of the plants.

Concerning VLLW, discussions are launched with french authorities to define a dedicated disposal.

CONCLUSIONS

Significant progresses recorded as to volume reduction of RW at EDF NPPs will be achieved in the next future with the CENTRACO plant commissioning, so that the treatment of ILW and LLW will be considered as completely solved. The next challenge for EDF is now to make available a suitable repository for VLLW (to issue mainly from decommissioning) in order to keep disposal both safety and costs within acceptable limits.

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AN OPTION FOR THE MANAGEMENT OF RADIOACTIVE WASTE IN ARGENTINA

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The Nuclear Program started in Argentina in the 50's in the field of scientific applications for peaceful uses of Atomic Energy. In 1960, due to the world oil crisis, a decision was taken to start a program on nuclear power generation to have an autonomous alternative. In 1974 the first Nuclear Power Plant, Atucha 1 (360 MWe, PHWR) was connected to the grid and ten years afterwards Embalse NPP (600 MWe, Candu) start operating; together they produce 12% of the country's electricity. A third Nuclear Power Plant Atucha 2 (700 Mwe, PHWR) is under construction.
In order to be independent from foreign supplies the complete front end of the fuel cycle was developed and is running successfully.

It is expected that approximately 60.000 m^3 of waste, without considering recycling, will be generated when the present nuclear facilities will be decommissioned. Around 90 % of this amount will come from the dismantling of the NPPs, approximately 6.000 m^3 will be Medium Level Waste and the rest can be disposed off as Low Level Waste.
Considering its design life, the Atucha 1 NPP, the oldest in Argentina, will cease operation in the year 2015. As a three stages deferred strategy, ending in 2058, will most probably be adopted for the decommissioning, the necessity to route wastes from this plant indicate the timing for the MLW and HLW repositories.

Medium and Low level Wastes from NPPs operation, nuclear applications in medicine, industry and R&D must also be considered, although the amounts are negligible if compared with those from decommissioning.

At present, treatment conditioning and disposal of LLW is managed at the trench type repository, and auxiliary facilities, located at Ezeiza Atomic Center. MLW is temporarily storage at the same Center and in the Nuclear Power Plants.

There are plans to build, in the near future, a MLW near-surface monolithic concrete type repository, similar to those erected at L'Aube in France or El Cabril in Spain, with an estimated capacity of 60.000 drums.

For the spent fuel elements the national policy at present is the "wait and see" (deferral) option. Although the reprocessing technology was developed in Argentina and a pilot plant is 80% built, a decision to stop construction was taken in the 80's together with the postponement of the original Nuclear Program which planned 6 NPP for the country before the year 2.000.

Although a final decision, regarding reprocessing or direct disposal, will be taken in the future, a final repository for HLW will be needed. A decision was taken that the spent fuel elements, or the vitrified waste arising from the reprocessing process, will be finally disposed off in a deep geological repository.

Research and development in vitrification, employing borosilicate glass by fusion and hot pressing, and different materials for containers were conducted in the last 20 year at the National Atomic Energy Commission laboratories. Studies are also conducted in partition and transmutation of Actinides.

Recently a National Nuclear Law and a Nuclear Waste Disposal Act were approved by Parliament regulating responsibilities and obligations of Government and waste producers, to comply with Law requirements a new Strategic Plan for radwaste disposal is under elaboration at CNEA.

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RESEARCH AND DEVELOPMENT OF TREATMENT TECHNIQUES FOR LLW FROM DECOMMISSIONING
- DECONTAMINATION AND VOLUME REDUCTION TECHNIQUES -

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Reduction of amount and/or volume of low-level radioactive waste are very important for decommissioning of nuclear reactor. The present paper discusses research and development of decontamination and volume reduction techniques in the JAERI.

I. Decontamination

The Japan Power Demonstration Reactor (JPDR) decommissioning program was begun in 1986 by the JAERI under a contract with the Science Technology Agency (STA). The program contained development of efficient decontamination techniques for pipe system before dismantling and for component after dismantling. High decontamination factor (DF) could be attained by developed techniques such as sulfuric acid - cerium chemical and wet flowing abrasive methods [1] . It is, however, necessary to reduce drastically the amount of secondary waste that will be generated in decommissioning of commercial nuclear power plants.

Therefore, the JAERI has been developing advanced decontamination techniques generating a small amount of secondary waste since 1990 [2], [3] .

The techniques developed are as follows:

Decontamination for pipe system before dismantling

-- Gas carrying abrasive method

Inside of pipes is polished by abrasive on swirling flow of air. On the application for large pipes, this method has the advantages of higher flow velocity of air and easier separation of abrasives compared with the wet flowing abrasive method.

-- In-situ remote electropolishing

Small inside area of pipes closed by two balloons is electropolished using a small amount of electrolyte. This method is superior to general chemical decontamination for pipe system and applicable to partial decontamination of pipes.

Decontamination for component after dismantling

-- Thermal shock method

After surface contaminant of metallic waste is fused with a flux, the contaminant in the form of stable solid solution is efficiently removed by thermal quenching. The treatment of the secondary waste is relatively simple.

-- Laser induced chemical process

Contaminant is removed by laser ablation or laser induced chemical reactions. Laser light can be remotely controlled by using mirrors and fibers.

II. Volume Reduction

There are various volume reduction techniques such as cutting, crush, compaction, incineration and melting. Especially, melting is considered as a desirable one for reasons of high volume reduction ratio and stabilization. Accordingly, the JAERI has been conducting melting tests of metallic waste and miscellaneous waste.

The objects of melting tests are:

1. To collect melting characteristics of waste,
2. To collect operational data for the advanced volume reduction system under construction in JAERI,
3. To clarify the behavior of radionuclides during the melting process,

4. To examine the radionuclide distribution and characteristics of the homogenized product for future recycling or final disposal.□

Radioactive metal melting tests were carried out using radioactive metallic waste arising from the decommissioning of JPDR and completed in 1995. The radionuclides were distributed to ingot, slag, and off-gas by their thermodynamic and physical properties during melting. The homogenization in melting process was also confirmed [4] . □

Melting tests of miscellaneous solid waste, which consisted of metal and non-metal, have been conducted by using a plasma-induction hybrid melter with off-gas treatment system since 1998. The induction furnace is used for melting metallic waste and plasma torch is used for non-metallic waste. □

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RADIOACTIVE WASTE MANAGEMENT PLAN DURING THE TRIGA MARK-II & MARK-III DECOMMISSIONING

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Decontamination and decommissioning(D&D) project of the TRIGA Mark-II & Mark-III KRR 1&2) was started in January 1997 and will be completed by December 2002.

In the first year of the project, work was performed in preparation of the decommissioning plan, start of the environmental impact assessment and setup licensing procedure and documentation for the project with cooperation of the Korea Institute of Nuclear Safety(KINS).

In the second year, Hyundai Engineering Company(HEC) was designated as the main contractor to do design and licensing documentation for the D&D of both reactors. And British Nuclear Fuels plc(BNFL) is the technical assisting partner of HEC. After pre-design, a hazard and operability (HAZOP) study checked each step of the work.

Meanwhile, all the spent fuel from KRR 1&2 were transported to US in June 1998. At the end of 1998, the decommissioning plan documentation including environmental impact assessment report was finished and submitted to the Ministry of Science and Technology(MOST) for decommissioning license. It is expected to be issued the permission at the end of September 1999. Practical work will then be started at the end of the year 1999.

Safe treatment and management of the radioactive waste arising from the D&D activities is utmost importance for successful completion of the practical dismantling work. This paper summarizes general aspects of radioactive waste treatment and management plan for the TRIGA Mark-II&-III decommissioning work.

1. Transportation of the spent fuel

Enrichment of the nuclear fuel used at the KRR 1&2 are 20% and 70% respectively. After they stopped their operation, spent fuels were stored at the KRR 2 reactor pool in Seoul and the HANARO, new research reactor, spent fuel storage pool in Taejeon.

Meanwhile the US made a spent fuel management policy that the US origin spent fuel are returned to US. The Korean government agreed with the US policy and all the spent fuel from KRR 1&2 were safely transported to the US in the middle of 1998.

2. Radioactive waste treatment / management

One of the activities associated with dismantlement of the KRR 1&2 is the treatment and management of the resulting wastes. Planning a waste treatment/management programme requires knowledge of the types and origin of wastes that will be generated by the decommissioning activities.

Radioactive wastes produced during D&D work are classified into solid, liquid and gaseous types. But all of the radioactive waste will be controlled by radioactivity level, by the shape and size, combustibility and non-combustibility and possibility of recycling or reusing.

Because of the good condition of the ventilation system in the KRR 2 reactor hall, most of gaseous waste will be treated with this system. But a temporary containment system including ventilation is planned to install after removal of the non-activated concrete of the reactor's biological shield.

All of the liquid radioactive wastes are under the free discharge level, low or very low level(less than 10^{-6} μCi/cc) with a maximum quantity of 500 m^3. However, as there is no dilution sink to be discharged near the KRR 1&2 site, it will be concentrated by a natural evaporator nevertheless it is under the permissible release level. Such a natural evaporation facility, for low and very low level liquid waste, is operating at Taejeon KAERI site without any environmental risks, and with a good decontamination factor. The shower or laundry waste water, which maybe more than the detecting level will be treated by using a membrane separation facility. Then the concentrate will be solidified with cement or asphalt in a 200 liters drum. A simple and movable cementation equipment will be installed on site for the solidification of concentrated liquid waste from natural evaporation.

Solid wastes are generally low-level. Medium level radioactive wastes from reactor structures, mainly stainless steel components, will be dismantled and stored in a shielding container. The rest of low level solid wastes will be packed in a 4 m^3 ISO containers for non-compactable and 200 liters drum for compactable or combustible wastes and then stored in a temporary storage facility until a low- and medium-level radioactive waste disposal site is ready, probably year 2008. The general flow diagram for radioactive and non-radioactive wastes is shown in figure 1.

Simultaneously all of the packed radioactive waste will be correctly labeled and recorded by data base for further treatment or final disposal.

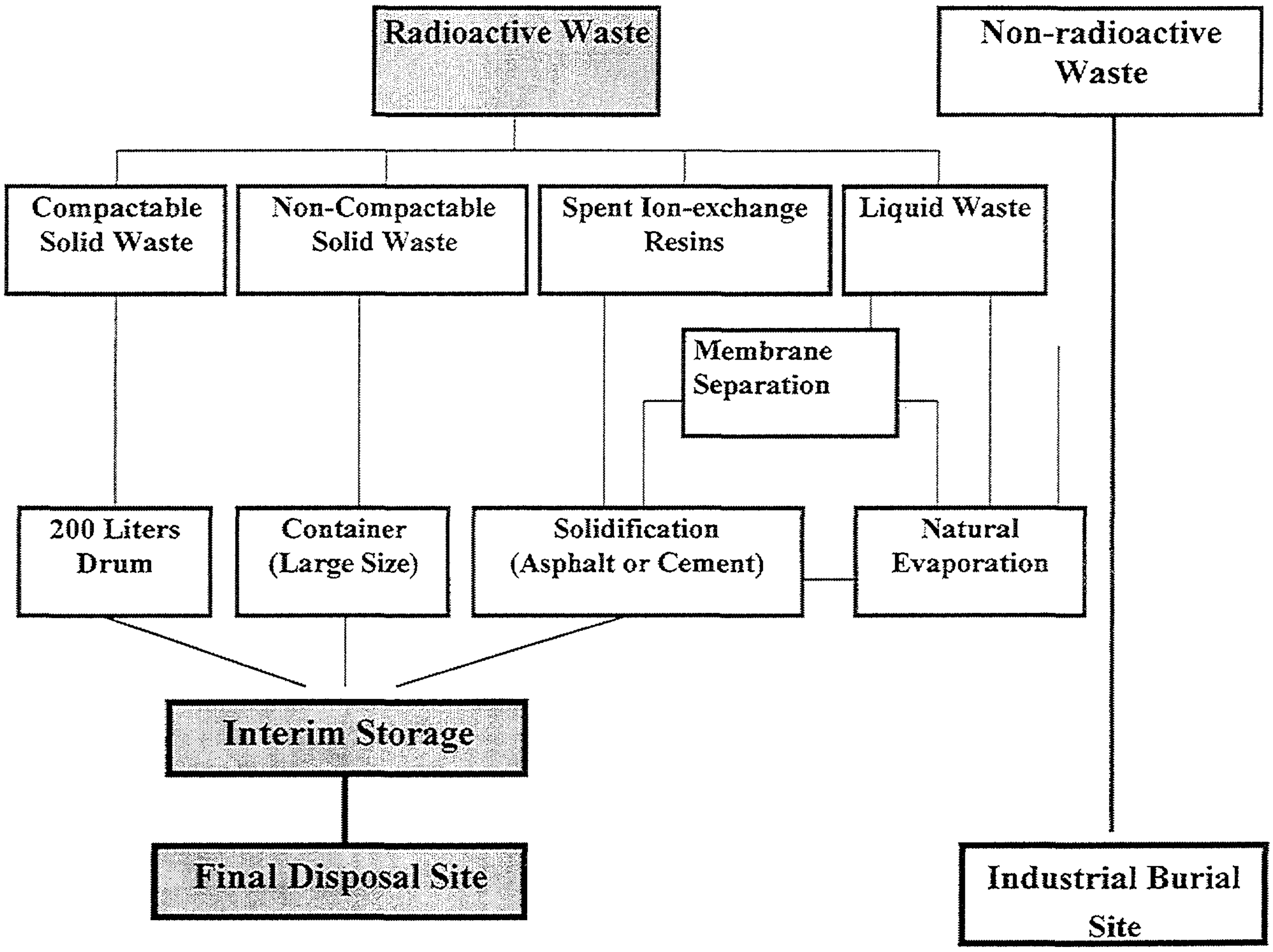

Figure 1. Flow diagram of radioactive wastes management

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DEVELOPMENT AND APPLICATION OF GROUNDWATER FLOW METER IN FRACTURED ROCKS: MEASUREMENT OF VELOCITY AND DIRECTION OF GROUNDWATER FLOW IN SINGLE WELL

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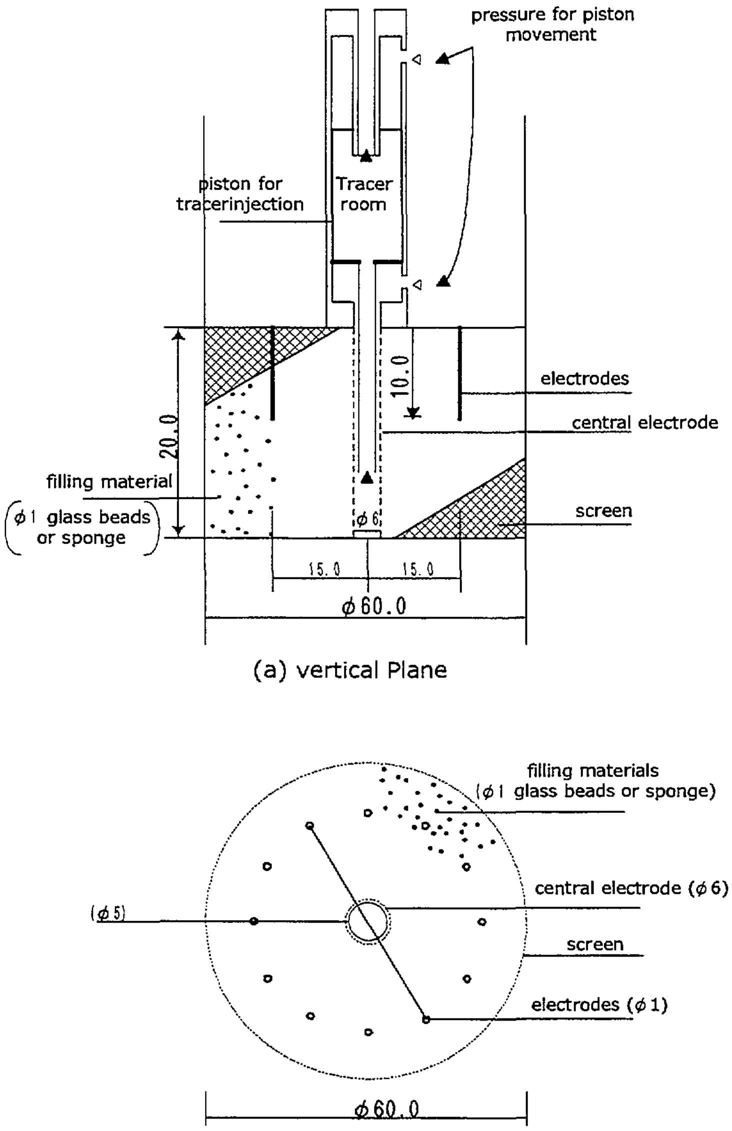
For the confirmation of safety for the geological disposal of high-level radioactive wastes, it is very important to demonstrate the groundwater flow by in-situ investigation in the deep underground.

We have developed a groundwater flow meter to measure simultaneously the velocity and direction of groundwater flow by means of detecting the electric potential difference between the groundwater to evaluate and the distilled water as a tracer in a single well.

In this paper, we describe the outline of the groundwater flow meter system developed by CRIEPI and Taisei-Kiso-Sekkei Co. Ltd., the evaluation methodology for observed data by using it in fractured rocks and a few applied results to in-situ tests [1][2].

The diagram for the outline of measurement apparatus of groundwater flow meter developed and modified by us is shown in Fig.1.



(a) vertical Plane

(b) horizontal Plane

Fig.1 Basic Structure of Measuring Room of Groundwater Flow Meter

This meter is capable of simultaneous measuring both the velocity and direction of the groundwater flow. This can be achieved by using the distilled water as a tracer and preparing only a single borehole.

We have developed the data analysis method to specify the velocity and direction of groundwater flow in fractured rocks and confirmed the basic applicability of this flow meter system by comparison between the laboratory tests and analyzed results.

Furthermore, we have applied this groundwater flow meter system to the in-situ tests that had been performed as co-operative research project in fractured rocks at JNC's Tono mine in Japan and SKB's Aspo HRL site in Sweden flow meter (See Fig.2 and Fig.3).

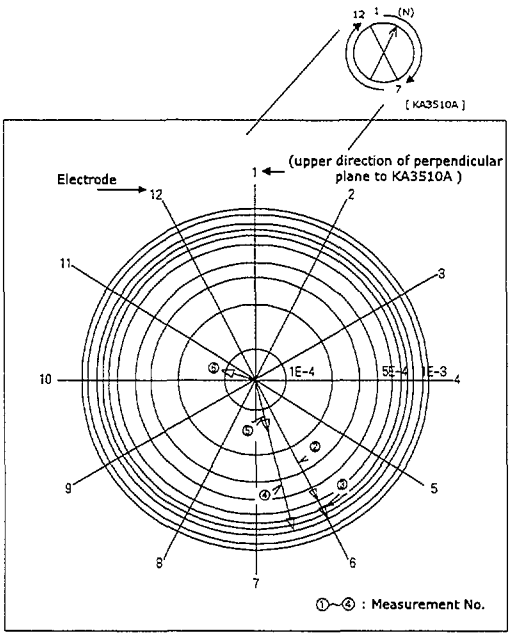


Fig.2 Observed Results of Groundwater Flow in a Fractured Zone (SKB's Aspo HRL)

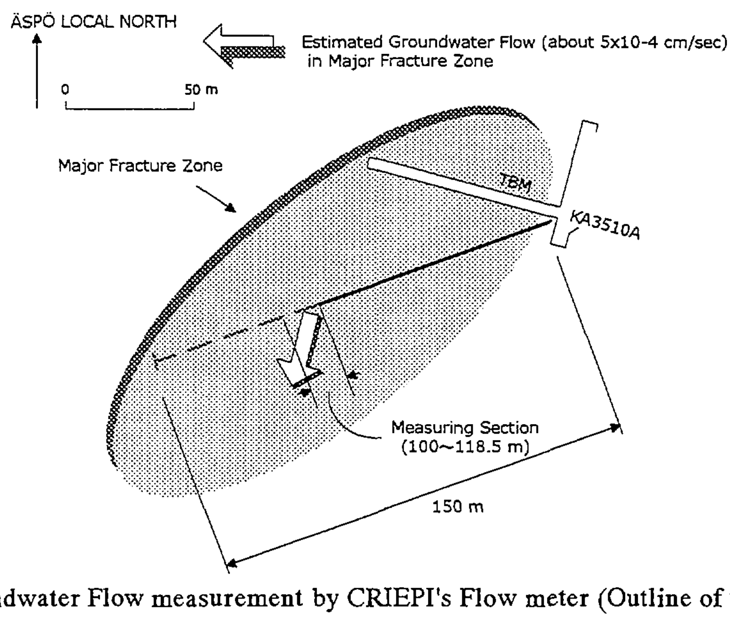


Fig.3 Groundwater Flow measurement by CRIEPI's Flow meter (Outline of the estimated Groundwater flow in a major fracture zone (Aspo HRL)

From these results, it is has been made clear that this flow meter system can be practically used to measure the groundwater flow direction and velocity as low as order of about $1\times10^{-2}\sim1\times10^{-6}$cm/sec.

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EXECUTION TECHNIQUES AND APPROACH FOR HIGH-LEVEL RADIOACTIVE WASTE DISPOSAL IN JAPAN : DEMONSTRATION OF GEOLOGICAL DISPOSAL TECHNIQUES AND IMPLEMENTING APPROACH OF HLW PROJECT

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In Japan, the high-level radioactive waste disposal project is expected to start fully after establishment of the implementing organization, which is planned around 2000 and to dispose by 2030s to at latest middle of 2040s. (See Fig.1)

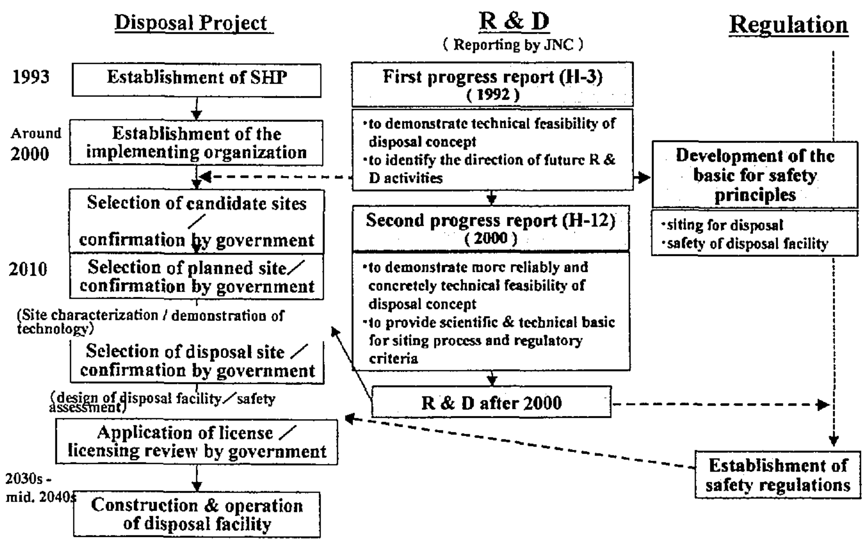

Fig.1 National Program for Disposal of HLW in Japan

Considering each step of implementing the HLW disposal project in Japan, in this paper, the execution procedure for a series of HLW disposal project, such as the selection of candidate/planned disposal sites, the construction and operation of the disposal facility, the closure and decommissioning of facilities, the institutional control and monitoring after the closure of disposal facility, are discussed in detail from a technical viewpoint for the rational execution of HLW project in Japan. (See Fig.2)

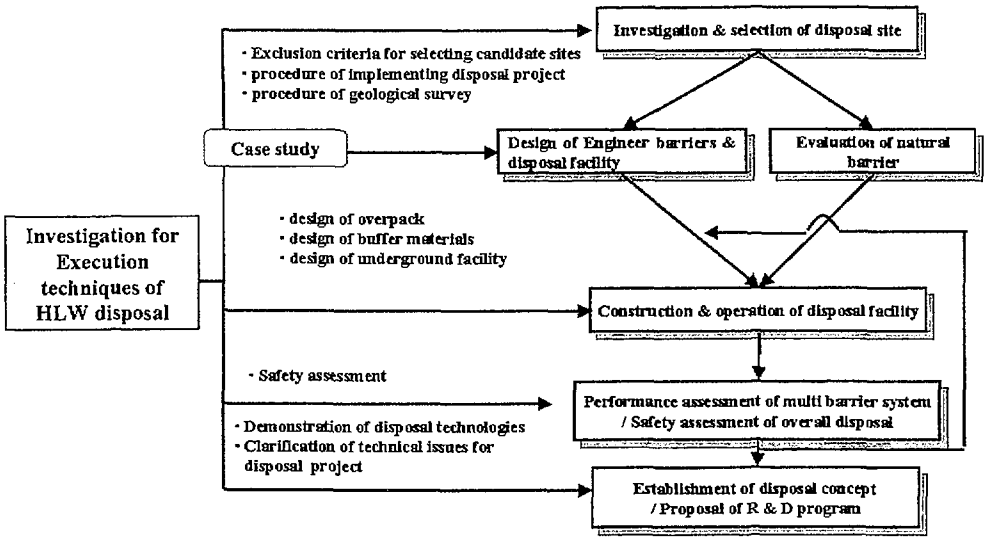

Fig.2 Schematic Diagram of Investigation for Execution of HLW disposal

Furthermore, we investigate and propose an idea for the concept of the design of geological disposal facility (See Fig. 3),

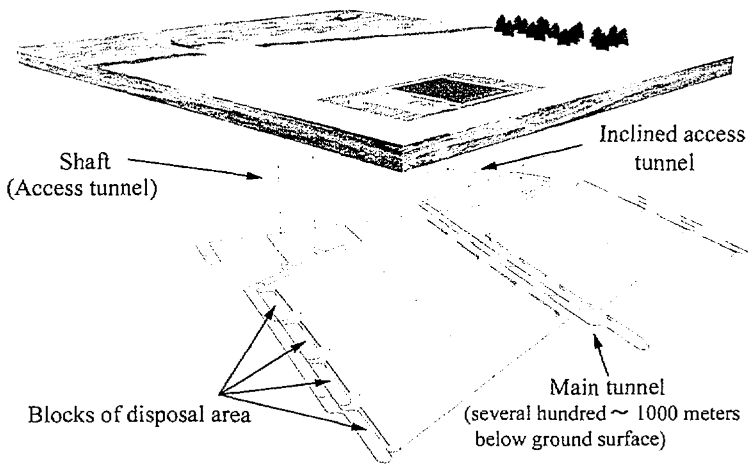

Fig.3 Basic Design Concept of HLW Disposal Facility

The validation and demonstration of the reliability on the disposal techniques and performance assessment methods at a candidate/planned site. (See Fig.4)

Based on these investigation results, we made clear a milestone for the execution of HLW disposal project in Japan.

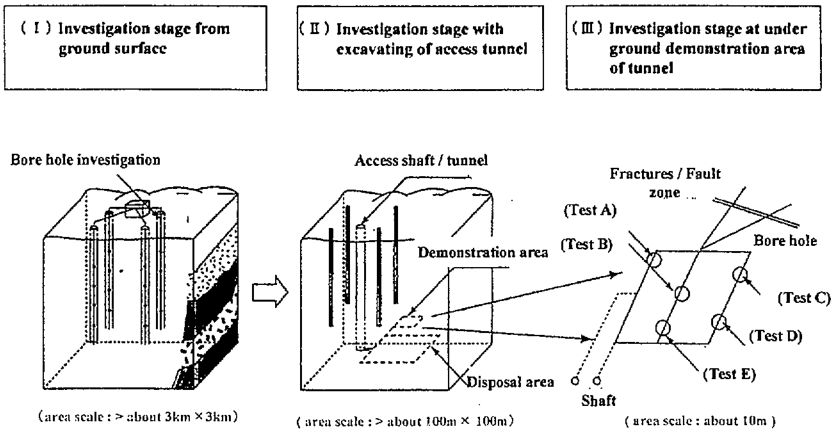

Fig.4 Basic Concept for Validation & Demonstration of HLW Disposal Techniques at a Planned Site

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INDUSTRY PERSPECTIVES ON THE US SPENT NUCLEAR FUEL MANAGEMENT PROGRAM

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The US Spent Nuclear Fuel Management Program has reached a critical juncture. Since the start of the US nuclear energy program, disposal of spent nuclear fuel has been the responsibility of the US government. It has been 17 years since the US Congress authorized the use of geologic disposal and 12 years since Congress made a decision to study geologic disposal at Yucca Mountain, Nevada as the Nation's singular long-term option. During this time there has been considerable scientific study of the capabilities of Yucca Mountain to isolate long lived radionuclides for thousands of years. There has also been considerable design effort on the engineered barriers that will work in conjunction with the proposed repository's natural features to ensure that health protection standards can be met. Much confidence has been gained concerning the soundness of the geologic disposal option.

However, while progress on the technical aspects of geologic disposal has been substantial, the policy level decision-making needed to move the program forward has been yet another matter. As spent fuel continues to accumulate at 103 operating US commercial nuclear power reactors at 65 sites and storage space at those locations is becoming increasingly scarce, a number of important waste management policy questions remain unanswered. This paper will outline the US's spent nuclear fuel storage needs, highlight what the Nation's scientists have learned about geologic disposal, discuss what is being done to meet near term storage requirements, describe the emerging regulatory framework associated with the program and summarize the ongoing policy debate.

The 65 operating power reactor sites are among approximately 130 locations in 40 of the 50 US States where spent nuclear fuel is currently stored (shutdown reactors, government storage facilities and research institutions make up the rest). Clearly this is an issue which touches the lives of all Americans. These sites are responsible for producing approximately 20% of the nation's electricity, enhancing the national well being through world class scientific research, and providing for the national security. Nuclear power plants met 40% of the nation's increase in electricity demand over the past 2 1/2 decades while displacing billions of barrels of oil, billions of tons of coal, trillions of cubic feet of natural gas, and thereby preventing what would have otherwise been a 25-30% increase in harmful emissions. All of these accomplishments have resulted in 39,000 MTU of spent fuel being discharged from US reactors. This number is expected to grow to 86,000 MTU by 2040 as America continues to reap the economic and environmental benefits of nuclear energy.

The US government has a long-standing policy of not reprocessing commercial spent nuclear fuel. Virtually all of the US industry's spent fuel continues to be stored at the reactor sites - approximately 95% of it in reactor pools that are rapidly filling up. A growing number of sites have begun to address this space crunch with at-reactor dry storage facilities (11 such facilities have been licensed thus far). However, if sufficient additional dry storage can not be implemented or other interim measures are not taken, more than half of the nation's power reactors will lose full core discharge capability within the next decade. With the federal government projecting that a permanent repository could not be ready until 2010, at the earliest, the industry has been advocating that a centralized interim storage facility be sited in the vicinity of Yucca mountain while the repository work continues to progress. So far, the US Congress has been unable to enact the legislation needed to allow central interim storage in Nevada. Private initiatives to locate interim storage facilities in Utah and Wyoming have also run into stiff opposition from state governments. Meanwhile, the additional costs of maintaining what are essentially 130 separate and scattered interim storage facilities stretches into the billions of dollars.

Although stalled in Congress, the argument for interim storage at Yucca Mountain continues to be bolstered by progress on the scientific front that provides ever increasing confidence that this site will eventually be found suitable as a permanent geologic repository. The United States is indeed fortunate to have within its borders a site which combines a dry climate, a stable geologic environment, a repository horizon that is both 1000 feet above the water table and 1000 feet below the surface, and isolation from populated areas. The current repository schedule calls for the US President to make a suitability determination in 2001. An important step towards this milestone was reached when the US Department of Energy completed its "Viability Assessment of Yucca Mountain". This report concluded,

"*Over 15 years, extensive research has validated many of the expectations of the scientists who first suggested that remote, desert regions of the Southwest are well-suited for a geologic repository. Engineered barriers can be designed to contain waste for thousands of years, and the natural barriers can delay and dilute any radioactive material that migrates from the waste packages*".

The Viability Assessment also outlined in extensive detail, the remaining tasks to be completed and their associated costs to characterize, design, construct, and license an operating repository in 2010.

The US Nuclear Regulatory Commission (NRC) is charged with regulating the proposed geologic repository. The NRC has demonstrated strong leadership by getting in front of the ongoing policy debate and publishing its proposed standard for a Yucca Mountain repository in February 1999. This proposed rule would be a risk-informed regulation built on the 1994 recommendations of the National Academy of Sciences. However, one important aspect of this standard has yet to be defined. Congress has granted the US Environmental Protection Agency (EPA) the authority to set the specific public radiation protection threshold for Yucca Mountain. The precise value at which the EPA will set this threshold and how it will be calculated continues to be the subject of much discussion.

In summary, the fate of the US Spent Nuclear Fuel Management Program, from interim storage in the near term to radiation protection over a period of thousands of years, continues to be tangled up in a far-reaching policy debate, while the costs of indecision continue to mount. But progress is being gradually made towards - in the words of Energy Secretary Richardson, "a decision based on science". This paper will attempt to shed light on both the scientific successes and policy challenges being met on the road to geologic disposal.

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INTERNATIONAL PERSPECTIVE ON REGULATION AND RADIOACTIVE WASTE MANAGEMENT

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In recent years, within the framework of national as well as international programmes, notable advances and considerable experience have been reached, in particular in the areas of minimization of radioactive waste arisings, conditioning and disposal of short-lived low - and intermediate - level waste, vitrification of fission product solution on an industrial scale and engineered storage of long-lived high level waste, i.e. vitrified waste and spent fuel. Based on such results near-surface repositories have successfully been operated in many countries. Furthermore, geological repository development programmes are now pursued, addressing the development and application of appropriate methods for site-specific safety assessments, too.

In addition, there is a need for the awareness of international developments and the considerations of their implications at a national level. One of the most important achievements in international agreements and guidance represents the establishment of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. This convention considerably contributes to a higher degree of transparency in national radioactive waste management programmes not only for countries developing such a programme but also for countries having an established system.

Despite the fact that confidence of most experts in the safety of a repository during its operational and post-operational phase has been confirmed, this is not necessarily matched by an equally favourable attitude of politicians, regulators or non-expert groups. Thus, several repository - development programmes are undergoing increased scrutiny and are expected to be adjusted according to new findings and evaluations. In the Federal Republic of Germany and in Great Britain the hitherto radioactive waste management and disposal programmes are presently re-examined and revised programmes are to be developed seeking national consensus in both countries, respectively. On the other hand, at the end of March 1999, the Waste Isolation Pilot Plant in the United States of America started operation as the first geological repository licensed in particular for long-lived low - and intermediate - level waste.

In addition to scientific-technical areas the consideration of issues regarding economical, environmental, ethical and political aspects has been increased during the last years. Hence, there is a need for the examination of such issues in more detail and, if appropriate, for introducing respective results in further radioactive waste management and disposal options and/or planning work. For example, the elaboration of criteria for choosing a candidate repository site from a number of potentially suited sites or the retrievability of waste packages from geological repositories should be faced and transferred to broadly accepted solutions. As far as economical aspects are concerned respective considerations, to some extent, still seem to be rather in their initial phase and need a better understanding. Thus, specific features of a nationally implemented waste management system may be favoured (or not) and, consequently, adjusted in an appropriate way.

Taking differences in national approaches, practices and constraints into account, it is to be recognized that future developments and decisions will have to be extended in order to include further important aspects and, finally, to enhance confidence in safety-related planning work and proposed waste management and disposal solutions.

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RADIOACTIVE WASTE MANANGEMENT AT KANUPP

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This paper describes the existing Radioactive Waste Management Scheme of Karachi Nuclear Power Plant (KANUPP). It also describes a modification carried out in the Spent-resin collection system in which a locally designed removable tank replaced the old permanent tanks. Presently the low level combustible solid waste is incinerated and stored. It is proposed to replace the present method by compacting this waste and storing it in steel drums underground.

KANUPP is a 137 MWe CANDU PHWR located at Karachi. The reactor has been generating power since October 1971. The plant continues to operate despite withdrawl of vendor support, through indegenous efforts. It has till now produced over 9.2 billion units of electricity and now uses fuel bundles manufactured in Pakistan. KANUPP faithfully adheres to its original safety and public risk targets. The average personnel radiation exposure has been well within the limits prescribed by the ICRP. The release of radioactive material through gaseous and liquid effluents has been less than 4% of the maximum permissible limit.

A wide variety of radioactive wastes, solids, gases and liquids of various activity levels and half-lives result from the operation of the nuclear reactor. These are handled and disposed of by a corresponding variety of methods such that they are not hazardous to the public.

Some of the gases and vapours produced are Argon, Krypton, Xenon, Iodine and Tritium. Of these Tritium and Iodine require special care as they constitute significant internal hazard. The solid wastes comprise a great variety of materials such as laboratory waste, secondary floor covering, disposable plastic suits, rubber gloves, (which are all combustible items) and active piping, components of valves and pumps, filters, ion-exchange columns etc. (which are non-combustible items). Liquid radioactive wastes arise from laundry and personnel shower effluents, evaporator distillates, effluents of the decontamination centre, spent fuel bay water clean up, accidental spills, ion-exchange column regeneration and evaporator concentrates etc. Three different classes are loosely defined, as low level, medium level and high-level wastes.

The largest amount of radioactive waste are the Spent Fuel bundles. These are stored underwater within the plant building in a specially designed Spent Fuel Bay. The water temperature and purity is carefully controlled. The Spent Fuel Bay is designed to contain and keep all the spent fuel for the life of the plant. For other solid wastes, a Waste Storage Area has been allocated near the plant. This area, measuring about 36m × 70m, about 1 km north of the plant and within plant's boundary has been provided for a long-term, retrievable storage of the plant's solid waste. The combustible waste with only trace amounts of radioactive contamination is first burnt in an incinerator to reduce its volume before the ash is collected in plastic bags, sealed and placed in concrete-lined trenches.

Liquid effluent from all the active areas of the plant which could generate radioactive effluent e.g. Decontamination Area etc is collected in 32,000L hold-up tank. When this tank fills up, its contents are transferred to one of the two 180,000L dispersal tanks. When this tank is nearly full (in about 2-3 weeks) its contents are analyzed in the Radiation Laboratory to determine its radioactivity content and identify its radio-nuclides. The effluent is then pumped out to sea after mixing it with the plant cooling water to such an extent that the discharge point radioactivity level of the effluent is below the permissible set-point. The discharge is continuously monitored by drainage monitor. If the pre-set safe discharge limit is exceeded the monitor blocks the flow of the active effluent and diverts it into the hold-up tank where its storage plus further dilution may render it safe for discharge.

The gaseous effluents are released at the top of a 45m high stack after passing though absolute filters. The effluents are continously monitored by a set of different radiation monitors for radio-iodine, noble gases, radioactive particulates and tritium. If the released radioactivity approaches 10% of the permissible limits, alarms are annunciated in the Control Room so that, if necessary, additional filteration or hold-up may be incorporated.

Ion-exchange resin columns installed in the Primary Heat Transport, Moderator and other systems carry out the removal of contaminants, the highly-radioactive fission and corrosion products from the systems. As a result, the resin becomes highly contaminated. When spent, the resin is flushed into storage tanks by remote handling. In the original KANUPP design, two large capacity tanks were provided for long-term storage of the spent resin. The tanks are buried underground, normally inaccessible and their capacity was originally expected to be adequate for storage of 30 years of plant's output of spent resin. However, both of these tanks were completely filled a long time ago.

Various alternatives were considered and it was decided that for further storage of spent resin, a new smaller tank be installed in the vacant space of Bay Water Circulating Equipment Room. The tank, 2.4m high and 1m in diameter, made of stainless steel is fixed on a 1.2m × 1.2m platform which is fixed on castors. The tank is shielded with 2.5cm thick lead bricks. The whole tank assembly, when installed, is further enclosed in thick concrete walls. This tank is sufficient to store the resin generated during plant's one-year operation. The total weight of the tank assembly, including its trolley, shield and resin is about 7,000 Kg. The shielding provided around the tank (lead and concrete) is adequate to reduce the dose-rate in immediate vicinity to permissible level. This tank, when filled with spent resin, is removed and transported to Waste Storage Area for long-term storage in a reinforced, concrete-lined trench. A new tank of similar description is installed in its place for further storage. Three such tanks have already been filled up and transported and buried in the Waste Storage Area.

Presently the low level radioactive combustible solid material is burned in an incinerator and then the ash is collected and stored in concrete lined underground trenches. It is now proposed that to eliminate the burning and to compress the waste using a heavy duty compactor, store it in steel drums and bury in the trenches in the Waste Storage Area.

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