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Storage of Spent Fuel from Power Reactors

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CONTENTS

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IAEA overview of global spent fuel storage

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Abstract. Spent fuel storage is a common issue in all Member States with nuclear reactors. Whatever strategy is selected for the back-end of the nuclear fuel cycle, the storage of spent fuel will contribute an imminent and significant part thereof. Notwithstanding considerable efforts to increase the efficient use of nuclear fuel and to optimise storage capacity, delays in realizing geological repositories in most countries or in implementing reprocessing in some countries entail in increased spent fuel storage capacity needs in combination with longer storage durations over the foreseeable future. An overview of global and regional spent fuel arisings and storage capacity is presented in this paper. Some trends are identified and recent Agency activities in the subject area discussed.

1. Introduction

Used or spent nuclear fuel is discharged from operating reactors and temporarily stored at the reactor pool. After a certain cooling time, the spent fuel will be moved from the at-reactor (AR) pool to away-from-reactor (AFR) storage facilities, either on or off reactor site, based on utility practice.

For the ultimate management of spent fuel discharged, the following options are being implemented or under consideration:

- the once-through cycle, i.e. the direct disposal of the spent fuel in a geologic repository;
- the closed cycle, i.e. the reprocessing of the spent fuel, recycling of the reprocessed plutonium and uranium, and disposal of the wastes from the reprocessing operations;
- the so-called “wait and see” policy, which means first storing the fuel and deciding at a later stage on reprocessing or disposal.

This wait and see policy, against the backdrop of delays in geologic repositories programmes in most Member States and in implementing reprocessing in some Member States, has entailed an increase of the amount of spent fuel to be stored and prolongation of storage duration. As a consequence, expansion of spent fuel storage capacity has been needed over the past decade. This trend will continue in the near future.

The situation is further complicated by today’s reliance on higher enrichment, higher burnup fuels as well as on mixed oxide (MOX) fuel, to generate electricity at a competitive cost. Given the much higher decay heat levels from these fuels, wet storage will remain the preferred approach for interim storage during the first decade after discharge. After sufficient decay and, especially when long term storage is foreseen (now storage up to and beyond 100 years is envisaged in some Member States), dry storage under inert conditions or in air becomes the preferred option, given the passive nature of dry storage systems.

2. Overview of spent fuel storage situation

2.1. Status of nuclear power

Today the growth in the number of nuclear power plants is at a standstill in Western Europe and North America, while expanding in parts of Asia and Eastern Europe. At the end of last year, 441 nuclear reactors were operating in 31 countries worldwide [1]. They provided about 2 780 TW·h, which is just over 16 per cent of the global electricity supply.

The total net installed capacity was 359 GW(e) and 33 nuclear power plants are under construction with a total net capacity of 27 GW(e). Table I shows the nuclear power status for four world regions, i.e. West Europe, East Europe, America and Asia & Africa.

Table I. Status of nuclear power in world regions

	Operating Reactors		**Under Construction**	
Regions	**No. of Units**	**Total Capacity GW(e)**	**No. of Units**	**Total Capacity GW(e)**
West Europe	146	125.7	0	0
East Europe	67	46.1	10	8.0
America	124	112.4	1	0.7
Asia & Africa	104	74.5	22	18.4
World	441	358.7	33	27.1

Status 1 January 2003 Source PRIS

2.2. Spent fuel arising

Worldwide the spent fuel generation rate, now at about 10 500 t HM/year, is expected to increase to about 11 500 t HM/year by 2010. As less than one third of the fuel inventory is reprocessed, about 8 000 t HM/year on average will need to be placed into interim storage facilities.

At the beginning of 2003, about 171 000 t HM of spent fuel were stored in storage facilities of various types (Table II). Most of this fuel is under water, but dry storage is becoming a commonly used technology with more than 12 000 t HM currently stored in dry storage facilities worldwide.

The total amount of spent fuel cumulatively generated worldwide by the beginning of 2003 was close to 255 000 t HM. Projections indicate that the cumulative amount generated by the year 2010 may be close to 340 000 t HM. By the year 2020, the time when most of the presently operated nuclear power reactors will be close to the end of their licensed operation life time, the total quantity of spent fuel generated will be approximately 445 000 t HM.

Table II. Status of spent fuel stored in world regions

t HM

Region	Amount
West Europe	36 100
East Europe	27 700
America	83 300
Asia & Africa	23 900
World	171 000

Status 1 January 2003

Assuming that current plans are maintained, one can observe the following regional trends (Fig. 1):

- West Europe will have slight decreasing quantities of spent fuel to be stored, due to reprocessing of spent fuel;
- East Europe will double the amount of spent fuel to be stored in the coming ten years;
- America will store all discharged fuel, thus the amount of spent fuel is constantly increasing;
- Asia & Africa like East Europe, will double the amount of spent fuel to be stored in the coming ten years.

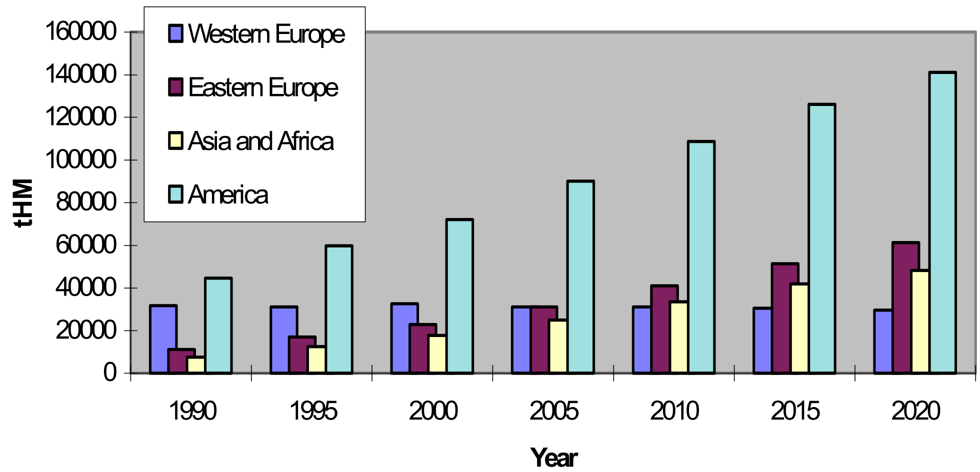

FIG. 1. Spent fuel stored by regions.

2.3. Spent fuel storage capacity

Various types of wet and dry storage facilities are operating in Member States with nuclear power plants (Table III). Early 2002, the global world storage capacity was about 243 000 t HM, with the bulk of storage capacity at reactor pools with 163 000 t HM. Member States operating nuclear power plants actually are or were increasing their existing storage capacity by re-racking the AR storage pools with high-density racks, by implementing burnup credit or by commissioning AFR storage facilities.

Table III. Capacities of operating spent fuel storage facilities

kt HM

Region	AR	AFR wet	AFR dry	Total
West Europe	28.3	32.3	11.3	71.8
East Europe	11.9	20. 8	1.5	34.2
America	94.7	1.7	8.5	104.8
Asia & Africa	27.9	3.3	1.7	33.0
Total	162.8	58.1	23.0	243.8

Status 1 January 2002

The storage capacity of new facilities, under construction in the various regions, is shown in Table IV. The total capacity is 24 000 t HM with 17 500 t HM as dry storage. This indicates that AFR dry storage is getting more and more preference.

Table IV. Capacities of spent fuel storage facilities under construction

kt HM

Region	AFR wet	AFR dry	Total
West Europe	3.0	1.0	4.0
East Europe	3.0	8.9	11.9
America		6.8	6.8
Asia & Africa	0.5	0.8	1.3
Total	6.5	17.5	24.0

Status 1 January 2002

2.4. Balance of spent fuel arising and spent fuel storage capacity

The global world storage capacity is about 244 000 t HM, and thus exceeded, by about 73 000 tonnes, the capacity needed by 1 January 2003. Globally all types of storage facilities have excess capacity available. On a worldwide basis, the spent fuel arising will fill the existing storage facilities and those under construction by around the year 2017, if no other new additional facilities will be built by that time. However, there is no reason to believe that no new construction projects for storage will be launched. Consequently, a storage shortage is not expected globally.

A worldwide or regional approach does not imply any problems. On a national level however, a shortage may occur if construction or expansion cannot be financed or licensed. Indeed, nationally the situation differs from country to country and sometimes even from utility to utility. In some cases, the storage pools are fully occupied by spent fuel allowing emergency core unloading only by special measures. Hence, additional storage capacity has to be installed in time, to avoid this problem. In other cases, additional storage capacity has to be installed timely to replace wet storage facilities. In particular in some Eastern European countries, plant operation might be jeopardized if additional local storage capacity cannot be installed in time.

In the past, most of the countries in this region heavily relied on the Soviet Union for their spent fuel management. Spent fuel return agreements signed in the past with the former Soviet Union were amended on a commercial basis. Due to economic constraints most countries did not opt for commercial contracts. As a result, many nuclear power plants in this region are or will be faced with a shortage of spent fuel storage capacity.

3. Trends

This section addresses a few selected trends in spent fuel management, in which the Agency has been active. It concerns the following topics:

- Regional spent fuel storage facilities;
- Long term storage; and
- Burnup credit.

3.1. Regional spent fuel storage facilities

Most Member States with power reactors are developing their own national strategy for spent fuel management, including interim storage. However, several Member States with a small nuclear power programme or only research reactors face the issues of extended interim storage of their spent nuclear fuel. The high cost for interim storage facilities for small amounts of spent fuel accumulated is obviously a handicap and therefore, from an economical point of view, access to an interim storage facility provided by a third Member State would be a solution, at least temporarily.

The safety and economic benefits from the implementation of regional spent fuel storage facilities could be attractive in terms of reduction of the number of spent fuel storage facilities world-wide, enhanced economy due to the scale of storage, and easier safeguarding to ensure non-proliferation.

On the other hand, such concepts involve political and public acceptance issues and therefore require a consensus among countries. The IAEA has assessed factors to be taken into account in the process of such a consensus during meetings in 2001 and 2002 on this task [2].

It appears that the concept of regional spent fuel storage facilities is technically feasible and potentially economically viable, without any obvious institutional deficiencies that would prevent completion of such a project. Storing spent fuel in a few safe, reliable, secure facilities will enhance safeguards, physical protection and non-proliferation benefits. However, political, social, and public acceptance issues are real and difficult to address. The added difficulty due to the regional nature of the facility could well be balanced by the benefits. The State considering hosting such a site and the States considering being customers for such a site will need to make their own decisions on the relative weights to place on these risks and benefits and the final decision on the establishment of a regional spent fuel storage facility.

3.2. Long term storage

The nuclear industry worldwide has accumulated significant fuel storage operating experience over the past 50 years. This experience, however, is largely based on safe and effective wet storage and the effect of time on structures and materials during this limited period of time.

The new challenges are to extend the life of existing and new wet and dry storage facilities and guarantee their safe performance for much longer periods of time.

Experts at Agency meetings [3] discussed various topics of relevance to long term storage for defining issues and questions to be addressed through future research and development:

- long term behaviour of spent fuel, fuel assemblies and packages;
- long term behaviour of dry storage systems;
- long term behaviour of wet spent fuel storage facilities; and
- regulatory concerns related to long term spent fuel storage.

The mechanisms that might have the potential to degrade the fuel and fuel structure need to be reviewed to identify possible gaps in knowledge, especially with respect to the long term behaviour of the materials during storage. Cask storage, in comparison to all other storage techniques, presents the greatest challenge (stress/strain) to long term fuel cladding performance, as a result of the high initial operating temperatures during the early years of storage. Stress and strain and the approach to the stress limit are the most important criteria in assessing cladding integrity.

Wet fuel storage is now considered to be a mature technology. In comparison, dry storage is an evolving technology, which has been developed over the past 20 years. Under present boundary conditions, dry storage can also be regarded as an established industrial technology. Unlike wet storage, dry storage can be more sensitive to fuel design changes and burnup increase, because of higher thermal output, which give rise to thermally activated processes.

In wet storage, there exist no urgent questions to be solved with regard to increasing operating life times. However, some recommendations, e.g. in the area of monitoring or technical optimisation were made.

In dry storage, there also exists a certain amount of supporting technical data, covering the burnup of the fuel loaded and the performance of the systems to date. For high burnup and MOX fuel, an extension of the knowledge on the creep behaviour of future cladding materials is needed. Additionally, a surveillance programme could demonstrate the long time behaviour of cask and fuel. For the development of advanced dry storage systems further R&D activities are needed, such as system performance for the perceived duty.

The regulatory objectives are very similar for all member states. Regulatory concerns include aspects of how technology changes are being handled and the extrapolation of material behaviour or performance for increasing storage duration.

3.3. Burnup credit

Experts explored the progress and status of international activities related to the burnup credit applications for spent nuclear fuel in 2002 [4]. Application of burnup credit to spent fuel management systems consists of implementation of a criticality safety assessment of the spent fuel management system of interest and of the application of the loading criterion.

Criticality safety is demonstrated with the aid of calculational methods verified by comparison to acceptable standards of known quality. Standards for comparison may be experiments, other accepted codes, or recognized standard problems.

The evaluation of the loading criterion is based on application of a criticality safety acceptance criterion to the results of the reactivity calculations. A criticality safety acceptance criterion is based on the safety margin required by the regulations for the application case, includes the biases of the applied calculation procedures as obtained from validation of these procedures and depends on the statistical confidence level chosen to express the impact of all uncertainties due to the applied calculation procedures and due to manufacturing tolerances of the system of interest.

Future issues of burnup credit are its application to long term storage/disposal of spent fuel, high burnup fuel, MOX spent fuel and spent fuel of advanced fuel designs.

4. IAEA activities on spent fuel management

Recent IAEA activities on spent fuel management consist of the following topics:

Dry Spent Fuel Storage Technology
An IAEA Technical Meeting/Workshop "Dry Spent Fuel Storage Technology" was held in June 2002 to give guidance to experts from Central and Eastern European Member States, operating WWER and RBMK nuclear power plants and to exchange information.

Spent Fuel Treatment
A consultancy meeting was held in October 2002, to prepare for a technical document [update to TECDOC-1103] on the subject of emerging technologies for spent fuel treatment. The TECDOC is expected to be published in 2003.

Operation and Maintenance of Spent Fuel Storage and Transport Casks and Containers
This is a new task intending to draw the pool of knowledge that has been accumulated from the industrial experience in the past several decades on the operation and maintenance of spent fuel casks. A Technical Meeting on this subject is planned for later this year.

Technical and Institutional Aspects of Regional Spent Fuel Storage
Meetings held in 2001 and 2002 determined that technical considerations and economic issues may be less significant than ethical and institutional issues for the development of a multinational project. A TECDOC is planned for 2004.

Optimization of Cask/Container Loading for Long Term Spent Fuel Storage
Meetings were held in 2002 and 2003 in preparation for a subsequent technical document on this topic.

Long term Storage of Spent Nuclear Fuel
To address new trends on long term storage of spent nuclear fuel, several meetings were held until 2000 with the results published [3].

Spent Fuel Performance Assessment and Research
Spent fuel storage technology (particularly dry storage) is undergoing evolution, new fuel and material design changes are coming on stream and target burnups are increasing. The report of the Co-ordinated Research Programme on spent fuel performance assessment and research programme (SPAR) has been published [5].

Selection Criteria for AFR Storage Facilities
Based on meetings held in the period 2001-2003, a TECDOC is planned for 2003 to provide guidance on selection criteria and methodology for AFR facilities, together with updated information on technical development and changing circumstance in the relevant area.

Economics of Spent Fuel Storage
Economics is a major factor of consideration in spent fuel storage projects and its significance will be amplified in the future with the increasing amount of spent fuel to be stored and the associated costs for implementation. A meeting held in 2002 served as a key step toward providing a TECDOC on this subject .

Implementation of Burnup Credit in Spent Fuel Management
A TECDOC [4], exploring the progress and status of international activities related to the burnup credit applications for spent nuclear fuel, will be published in 2003, based on a meeting held in 2002.

Data Requirements and Maintenance of Records for Spent Fuel Management
Guidelines on information management are required for long term management of spent fuel. A meeting, planned for next month is expected to lead to a subsequent TECDOC on records management.

5. Conclusions

The following conclusions can be drawn:

- At present, there is sufficient spent fuel storage capacity on a worldwide basis. However, nationally or on a specific site basis, the situation is different and needs urgent attention;
- Wet fuel storage is presently a mature technology and plays a major role in spent fuel storage;
- Under present boundary conditions, dry storage can also be regarded as an established industrial technology;
- The first geological repositories for the final disposal of spent fuel are not expected to be in operation before the year 2010. Many Member States have not yet started specific site investigations. As a consequence, the use of interim storage will be the primary spent fuel management solution for the next decades in many countries;
- Even more spent fuel storage capacity is required if countries defer their decision to open geological repositories;
- The storage duration becomes longer than earlier anticipated, due to the selection of the "wait-and-see" policy chosen by many nuclear power countries. The use of higher enriched fuel with higher burnup results in higher decay heat and longer storage periods;
- With longer storage periods dry storage becomes more and more important.

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Spent fuel storage, a long term engagement
OECD/NEA overview

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Abstract. Storage of the spent fuel, as opposed to disposal, will be the dominating feature of spent fuel management for the next century. Proven technologies exist for long term storage and it is inexpensive to enlarge the storage capacity. Sustainability of the long term storage depends on assumptions on the future development. In order to improve the sustainability of nuclear energy, one target should be the safe and cost-effective reduction of radioactive waste volumes and toxicity. NEA have several activities (in policy level as well as technology and scientific projects) related to storage of spent fuel.

1. Introduction

After forty years of commercial nuclear energy production 150 000 tU of spent fuel is currently in interim storage awaiting either deep geological disposal or a decision regarding its ultimate management. Part of it is likely to be reprocessed in accordance with the policy of the owner of the fuel. Currently, only about 30% of the irradiated fuel arising yearly from the nuclear power programmes of OECD member countries are committed to reprocessing.
Political decisions on all major industrial, infrastructure projects take a considerable time in our modern world and those concerning nuclear energy are no exception. The siting and development of final disposal facilities for irradiated fuel have been a particularly time consuming task and they will probably continue to be so. The longer the delays will be, increasing amounts of fuel have to be stored for longer.

In USA and in Finland, the lead countries, the first disposal facilities will not start to be operated until around 2015 - 2020. This means that storage, as opposed to disposal, will be the dominating feature of spent fuel management for the next century. The current storage capacity is more than 200 000 tU. It is inexpensive to enlarge this capacity.

A number of approaches have been developed for interim storage of spent fuel, each adapted to national policy and choices on fuel cycle, i.e., once-through or recycling option. Extensive experience has been accumulated on wet storage in cooling pools and dry storage in casks. It indicates that the technologies available for both approaches are suitable for storage over extended periods, probably up to one century. Additionally several commercial concepts exist for both wet and dry storage types creating a natural competition between manufacturers. New developments in the front-end of the fuel cycle, including trends to higher burn-up and use of mixed oxide fuel (MOX), will affect spent fuel management but are not expected to pose problems for long term interim storage, provided adequate technology adaptation will be implemented.

The evolution of the nuclear fuel cycle industries so far has been driven mainly by economic optimisation at each step while complying with increasingly stringent regulations on safety and radiation protection. In the future, new emphasis will be placed integrating aspects of sustainable development in the optimisation of nuclear systems, i.e., reactors and fuel cycles.

2. Sustainability

The concept of sustainable development evolved in the late 1980s and defined by the Brundtland Report [1] as “a development that meets the needs of the present generation without compromising the ability of future generations to meet their own needs”. In a broad sense, sustainable development incorporates equity within and across countries as well as across generations, and integrates economic growth, environmental protection and social welfare. A key challenge of sustainable development policies is to address those three dimensions in a balanced way, taking advantage of their interactions and making relevant trade-offs whenever needed. The central goal of sustainable development is to maintain or increase the overall assets (natural, man-made and human and social assets) available to future generations [2].

A very long storage period of large volumes of irradiated fuel is not generally regarded as desirable. For example the Radioactive Waste Management Committee (RWMC) of NEA has stated that from an ethical standpoint the responsibilities to future generations are better discharged by a strategy of final disposal than by reliance on stores [3]. In order to evaluate the sustainability of an activity like prolonged storage of spent fuel, certain basic assumptions should be made for the storage period, such as what is the stability of society, the speed and the direction of economical and technological development.

Long term storage requires surveillance bequeathing the next generations with responsibilities of care of the spent fuel. Final disposal has been presented as an alternative to reduce these responsibilities. Comparing these two alternatives and assuming that the structural stability of a society could not be presumed during the storage period, long term storage could be considered as less sustainable. Similarly an assumption of slow economical and technological development may increase the burden transferred to future generations making long term storage could be less “sustainable” compared to final disposal.

On the other hand, if a stable society is assumed and a similar speed of economical and technological development as during last 10 - 20 years in OECD countries could be expected, a long term interim storage might turn out to be more sustainable than final disposal. In this situation it could be argued that the society would benefit from using the economic and the human resources needed to develop and to construct a geological disposal facility for something else, such as education or health care.

In order to improve the sustainability of nuclear energy, one target should be the safe and cost-effective reduction of radioactive waste volumes and toxicity. The toxicity reduction which takes place naturally could be supplemented by introducing advanced nuclear fuel cycles that include reprocessing and consequently recycling. Additionally, recycling enables more efficient use of natural resources that is in-line with the objectives of the sustainable development. However, even if utilisation of these advanced fuel cycles were to be started today, it would take many decades before they are able to significantly change the world’s needs for long term interim storage of irradiated fuel. Though, the commitment now to develop and utilise an advanced fuel cycle increases utility of the interim spent fuel storage, since the spent fuel will then be ready for reprocessing and reuse as a fuel.

Utilisation of advanced fuel cycles does not obviate the need for an interim storage period after the fuel is removed from the reactor. The spent fuel needs to be cooled before it is reprocessed. After removed fron the reactor, new fuels containing recycled actinides and perhaps fission products produce more decay-heat than the currently used fuels. As a consequence some may argue that recycled fuel need longer cooling period before reprocessing and this way increasing the storage capasity. Though, new reprocessing technologies, such as pyro-reprocessing, tolerate higher heat production, and shortens the cooling period.

3. Role of Governments

The long storage period of spent fuel i.e. beyond one generation increases the responsibility of government as the ultimate legatee of the wastes. It invokes organisational, financial and economic issues as well as having safety and technological consequences.

In general, the government has the responsibility to define national policy on nuclear waste. It carries out policy oversight and regulation, defines the processes for funding, siting and environmental assessment of the facilities and possibly implements them. These responsibilities are similar as in the case of other industrial activities, however, the long time scale related to nuclear waste adds some unique features.

In most countries, the funds needed for spent fuel management are collected during the time of electricity production, well in advance of the long term liabilities related to spent fuel management. Thus, the challenge for the government is first to ensure collection of sufficient funds and then to ensure preservation them until they are needed. These funds could be transferred to future generations directly for example via a fund or indirectly for example by investing the resources in improving parts of the civil society infrastructure. Additionally, government may need to ensure that some entity is available to carry out the waste management itself, since one cannot have confidence that the power companies that produced the spent fuel will still exist several decades after power production has been ceased.

The back-end of the fuel cycle, which covers all steps after unloading of the spent fuel from the reactor, has a number of strategic implications related to radiation protection, safety, resource management and safeguards. In most countries, technology and policy choices for the back-end of the fuel cycle are overseen, and in some countries decided, by governments rather than being left to the initiative of utilities and industry.

From the safety point of view, the long period, for which the spent fuel should be isolated, demands a certain stability and capability of the society and a consistent perception of the requirements. One important aspect is to ensure the availability of the competence necessary for the safe and adequate treatment and disposal of radioactive materials.

4. Current NEA activities related to spent fuel storage

4.1. Overall waste management approaches and system analysis

Under the NEA waste management programme the progress in national radioactive waste management programmes are monitored and NEA Member countries are assisted in defining and implementing waste disposal strategies and policies. The potential of specific variations to geologic disposal, including those of extended storage and partitioning and transmutation are assessed. Waste management principles and concepts are discussed, including the role of risk concepts for safety evaluation of waste management approaches; they are put into the context of sustainable development and environmental safety practices. Approaches for a stepwise repository development process have been defined and a common understanding is sought how it is applied and and the usefulness of the approach. The regulatory systems overseeing the implementation of geologic disposal of long-lived waste are analysed in order to better understand differences in national criteria and programmes to improve the overall effectivness of the regulation. International peer reviews are organised to review national waste management programmes if requested by NEA Member countries.

4.2. Impact of advanced nuclear fuel cycle options on waste management policies

A study is under way to analyse a range of advanced fuel cycle options from the perspective of their impact on waste repository demand and specification. The focus is first to assess the characteristics of radioactive waste arising from several advanced nuclear fuel cycle options. Then repository performance analyses are performed using source terms for different wastes arising from each advanced nuclear fuel cycle. New options for waste management and disposal of such wastes may also be identyfied. A report is expected to be published early in 2005. More information on this study can be found on the NEA web-page at http://www.nea.fr/html/ndd/eg-fuel-cycle.html.

4.3. Working Party on Nuclear Criticality Safety

This Working Party on Nuclear Criticality Safety (WPNCS) deals with reactor criticality safety issues relevant to the fabrication, transportation, storage and other operations related to the nuclear fuel cycle. More information on the work of WPNSC can be found on the NEA web-page at http://www.nea.fr/html/science/wpncs/index.html.

As a part of the Working Party an Expert Group on Burn-up Credit has organised several benchmark problems. It has also been a place for technical discussions on burnup credit among NEA member countries. Burn-up credit is a term that applies to the reduction in reactivity of burned nuclear fuel due to the change in composition during irradiation. Burn-up credit is applied to criticality safety in the transportation, storage, and treatment of spent fuel for a wide range of fuel types, including UOX and MOX fuels for PWR, BWR, and VVER. Activities of the expert group are:

- to carry out international comparison exercises and benchmarks and to assess the ability of computer code systems to predict the reactivity of spent nuclear fuel systems, including comparison with experimental data as available;
- to investigate the physics and predictability of burn-up credit based on the specification and comparison of calculational benchmark problems; and
- to publish the results for the benefit of criticality safety community, so that the work may be used to help establish suitable safety margins.

More information of the work of the Expert Group can be found on the NEA web-page at http://home.nea.fr/html/science/wpncs/buc/index.html. A separate NEA paper (SFCOMPO: A Database for Isotopic Composition of Nuclear Spent Fuel; Current Status and Future Development) describes in more detail one part of the work of the Expert Group.

4.4. Nuclear energy and civil society

Several of the NEA's standing technical committees have launched activities that aim to analyse national and local experience with local society and to communicate lessons learned. Two of these activities are related to waste management, the Expert Group on Society and Nuclear Energy and the Forum on Stakeholder Confidence (FSC).

The Expert Group is carrying out a study that would provide policy makers with findings, guidance and recommendations on communication and consultation with civil society in connection with nuclear energy policy decisions, including siting of spent fuel storage facilities. The processes used or intended to be used in member countries would be mapped and experiences on consultation and communication aspects reported and analysed. The combined programmes of industry and government would be addressed, and if the examples were too numerous, case studies would be selected. The results of the study would be discussed at a workshop before being published.

The FSC facilitates the sharing of international experience in addressing the societal dimension of radioactive waste management. It explores means of ensuring an effective dialogue with the public, and considers ways to strengthen public confidence in governmental decision-making processes.

4.5. Nuclear energy data ("The Brown Book")

NEA publishes, annually, data of NEA member countries on electricity generation, on nuclear power and on nuclear fuel cycle. Spent fuel storage capacities and amount of spent fuel arising yearly are included in this data. The electricity generation and production data for fuel cycle services refer to these facilities located within the country, and thus exclude imports. The fuel requirements, however, refer to amounts of fuel cycle materials and services necessary for national nuclear programmes.

4.6. Role of Government in nuclear energy policy making

NEA has initiated a project to study to review and to analyze the role of governments in the nuclear energy field in the context of the policy-making framework of the beginning of the 21st century. The aim is to draw conclusions and to make recommendations on developing the NEA activities in order to maximize support to Member countries on national nuclear energy-policy issues. The spent fuel storage is discussed as a part of the waste management.

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SESSION 1

NATIONAL PROGRAMMES

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Management of spent fuel from power reactors in Argentina

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Abstract. A brief description of the two operative nuclear power plant and their fuel assemblies is given, as well as the fuel consumption and the expected quantities of spent fuel to be accumulated at the end of their lives. It is also described the legal framework, the organization and the current practice for the storage of the spent nuclear fuel in both nuclear power plants, as well as the management strategy for the future.

1. Introduction

In Argentina the radioactive waste and spent nuclear fuel management activities are carried out according to the Act Nº 25018 ("Radioactive Waste Management Regime") passed by the Parliament in the year 1998. This Act designs the Comisión Nacional de Energía Atómica (CNEA) as the governmental organization responsible for the strategic planning for the management of the spent fuel and the radioactive waste generated in the country and their final disposal. The generators of radioactive waste and spent nuclear fuel must pay for their complete management and are responsible for their conditioning and storage until they are transferred to the CNEA, in accordance with the acceptance conditions established by that organization.

Argentina is a Contracting Party of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, adopted in Vienna on September 1997 and entered into force on June 2001. Moreover, all the activities related to radiological and nuclear safety are regulated and controlled by the Autoridad Regulatoria Nuclear (ARN).

Particularly, in the case of the spent fuel from power reactors, utilities are responsible for its management during the operation of the nuclear power plants (NPP), including the interim storage. After the final shutdown of the NPP the spent fuel will be transferred to the CNEA who will be also responsible for the NPP decommissioning.

2. Nuclear power plants and their fuel assemblies

Argentina has two nuclear power plants in operation, supplying 12% of the national electricity production. The Atucha-1 NPP started commercial operation in 1974, it is a 340 MW(e) Heavy Water Reactor with pressure vessel, of Siemens design. The Embalse NPP started commercial operation in 1984, it is a 600 MWe PHWR with pressure tubes of AECL design (CANDU). The nuclear power plants are owned and operated by the state company Nucleoeléctrica Argentina S.A.

The fuel assembly (FA) of Atucha-1 has an active length of is 5.3 m and has a circular cross-section of 0.10 m diameter, with 36 fuel rods plus one structural rod. Each FA is loaded with approximately 176 kg of UO_2. Atucha-1 was fuelled with natural uranium during the first 27 years of operation, the average burnup of the spent fuel was approximately 6 000 MW·d/tU. In January 1995, the utility started a programme to gradually convert the fuelling to slightly enriched uranium (SEU), using an enrichment of 0.85% U-235. The programme was completed in August 2001, since then the whole core is fuelled with SEU and the average burnup of the spent fuel (SF) is approximately 11 300 MW·d/tU. This change produced an important saving in fuel consumption: from approximately 395 FA/full power years (fpy) to approximatelly 210 FA/fpy.

The CANDU fuel assembly of the Embalse NPP has a length of 0.5 m and a circular cross section of 0.10 m diameter, with 37 fuel pins. This NPP is fuelled with natural uranium, achieving an average burnup of approximately 7 500 MW·d/tU. The weight of UO_2 per fuel assembly is approximately 22 kg, the fuel consumption is approximately 4 800 FA/fpy.

3. Spent fuel management

3.1. Current practice and storage strategy for the future

In Atucha-1 the SF is stored in water pools located at the reactor site. The storage pools are made of concrete with stainless steel lining. The monitoring programme of the whole installation has not detected any failure or degradation of the components neither of the SF. The original management strategy considered the transfer of the SF to dry interim storage after the final shutdown of the NPP. Nevertheless, it is foreseen the necessity of operating the wet storage installation during at least 10 years after the final shut down, to allow for thermal cooling and radioactive decay of the SF belonging to the last core.

At present Atucha-1 is planning a re-racking of the SF in order to enlarge the capacity of the wet storage pools. However, it is foreseen the necessity of additional storage capacity to comply with the 12.5 full power years of remnant life of the reactor. Therefore, it is necessary to anticipate the original planning in order to put in operation a dry storage installation before the final shutdown of the NPP. So, there is under development a conceptual design of a modular facility for dry storage of the SF to be located at the reactor site, using the concept of concrete silos[1].

In Embalse NPP the SF is stored in water pools during 6 to 8 years for thermal cooling and radioactive decay and7 then are transferred to dry storage in silos, both interim storage facilities are located at the reactor site. The pools for wet storage are made of concrete with epoxy lining. The silos for dry storage are made of concrete, each silo contains 9 sealed stainless steel baskets with 60 spent fuel elements each one. This installation is of modular type, 120 silos were already built with a total capacity of 64 800 spent fuel assemblies, when needed, new silos will be added to the existing ones. It is planned that 6 to 8 years after the final shutdown of the NPP all the SF will be in dry interim storage. The stockpile of SF stored at both nuclear power plants is shown in Table I.

3.2. Final disposal

In the management of spent fuel from the nuclear power plants, Argentina has not defined yet its fuel cycle back-end strategy, although in principle the fissile material contained in the spent fuel is considered as a potential resource. The spent fuel will be kept in interim storage until a decision is taken whether reprocessing it or not. According to the current planning, a

[1] See paper IAEA-CN-102/2 by D.O. Brasnarof and J.E. Bergallo, pp 135-140 (Session 2).

decision about this issue should be taken before the year 2030. In any case, a deep geological repository is considered necessary for the disposal of the SF or the high-level waste that would be generated in the case of their reprocessing.

Table I. Stockpile of spent fuel from the nuclear power plants

NPP	STORED FAs[a]		EXPECTED AT EOL	
	WET	DRY	Quantity	t HM
Atucha-1 340 MW(e)	8 546	--------	11 600	2 040
Embalse 600 MW(e)	40 900	46 877	144 300	3 175

[a] Status at 31 December 2002

The schedule for the construction of the geological repository is mainly linked to the decommissioning of the nuclear power plants. Considering its design life, the Atucha-1 NPP would cease operation in the year 2015; a three stages deferred strategy will most probably be adopted for its decommissioning, ending in 2058. Therefore, it is planned that the deep geological repository should be in operation in the year 2050, at present are being carried out geological studies for repository siting. This schedule is tentative and depends strongly on economical, political and social issues.

Spent fuel storage facilities in the Czech Republic

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Abstract. There are six nuclear power reactors type PWR (VVER) operating in the Czech Republic. In the course of their operation they will generate in total approximately 3 300 t of spent nuclear fuel. Spent fuel storage concept has undergone many changes since 80s, the last one was approved by the Czech government in May 2002. This concept sets down basic principles of the nuclear fuel cycle back-end strategy which could be characterized as "wait and see" approach. Among other points this concept determined storage spent fuel technology, i.e. storage metal casks placed in dry storage facilities, appointed localities for spent fuel storage facilities at NPP sites and determined the underground storage facility in location Skalka as the back up solution. The interim spent fuel storage facility was built at the Dukovany NPP with capacity of 600 t of spent fuel and has been in operation since 1995. The construction of further spent fuel storage facility for the Dukovany NPP with capacity 1340 t of spent fuel has been contracted including storage dual purpose casks. The construction of the facility should start in 2003 and the storage should be put into operation in 2006. Next spent fuel storage facility in Temelin NPP with capacity of 1 370 t of spent fuel is in the planning phase at present, the site permit is expected in 2006 and the construction approval in 2008. The operation should start in 2014. As the back-up solution for the Temelin spent fuel storage facility the underground storage facility in locality Skalka has been kept. This locality used to be an underground option for the central spent fuel storage facility and has already obtained site permit.

1. Introduction

There are six nuclear power reactors operating in the Czech Republic. Four of them type VVER-440 in the Dukovany NPP were put into the commercial operation in 1985-1987, first unit type VVER-1000 in the Temelin NPP was put in the eighteen-month trial operation in June 2002 and the second one of the same type in April 2003. The operation of the Dukovany NPP will generate approximately 1 940 t of spent fuel during 40 years of operation and the Temelin NPP 1 370 t in 30 years of operation. This amount represents the capacity of the spent fuel we have to deposit in the spent fuel storage facilities.

2. Storage concept

The concept of the preparation and construction of the spent nuclear fuel storages has been changed several times since the 80s of the last century. The last one was agreed by the government in May 2002 in the document called "Concept of the disposal of radioactive waste and nuclear spent fuel" [1]. This concept sets down basic principles of the nuclear fuel cycle back-end strategy which could be characterized as "wait and see" approach. Among other points this concept determines:

- Spent fuel will be stored in dry storage facilities, in storage casks or in dual-purpose casks (transport and storage). This concept enables, in the case of urgency, to carry out the transport immediately without any transhipment operation;
- Priority is given to place the storage facilities at the nuclear power plants. The locality Skalka will serve as the back up site (underground storage);
- Concurrently, in connection with the preparation of the deep repository, possibilities of the nuclear spent fuel recycling and adoption of new technologies, aimed at decreasing the nuclear spent fuel volume and toxicity, will be pursued;
- The deep repository will be put into the operation in 2065.

3. Dukovany NPP - Interim Spent Fuel Storage Facility (ISFSF)

Spent fuel is stored in dual-purpose (transport and storage) casks of the CASTOR 440/84 type. The ISFSF has been in the operation since 1995. Its storage capacity of 600 t will be exhausted by the end of 2005.

4. Dukovany NPP - Spent Fuel Storage Facility (SFSF)

The preparation of the 1 340 t SFSF started in 1997. The EIA process took 2 years. ČEZ, a. s., got the site permit in 2000. The State Office for Nuclear Safety issued the permission for starting the construction procedures in November 2002, the construction approval is expected in 2003. Duration of the construction work is scheduled to 23 months in order to start the trial operation at the beginning of 2006.

In the course of the bidding process the cask type CASTOR 440/84M for 84 fuel assemblies was chosen. The cask CASTOR 440/84M is a modification of the type 440/84 used at present. The body is made of ductile cast iron, the basket of boron-aluminium mixture, the neutron shielding by means of polyethylene placed in boreholes. The inner room is filled with helium at a lower pressure than the atmospheric one, the pressure of the helium in the space between lids is of 0.7 MPa, each lid is provided by 2 kinds of sealing (metallic and elastomer sealing). The modification of this new type of cask in comparison with the type CASTOR 440/84 consists in:

- new design and material of the basket which guarantees a subcritical stability for fuel cassettes with higher enrichment and an improved exhaust of the heat into the body of the cask;
- reduction of inner diameter of the body of the cask (the gamma emission shielding thickness is sustained);
- modification of the geometry of the surface cooling gills;
- adding of a further row of boreholes for the moderator in the cylindrical wall of the cask and of a moderator plate both at the bottom and at the lid of the cask in order to improve the neutron shielding in consequence to the increased burn-up.

The parameters of the cask CASTOR 440/84M meet the requirements of the new nuclear fuel with higher average burn-up (44 000 MW·d/tU in comparison with the original 40 000 MW·d/tU). The weight of the empty cask is 93.7 t, the weight of the filled one is 112 t. The temperature of the covering coating of the fuel element does not exceed 350˚C.

The contract for delivery of 25 casks with the option for other 4 casks was signed in May 2001. Start of construction of the spent fuel storage building is expected in 2003 after the construction approval enters in force.

5. Temelin NPP - Spent Fuel Storage Facility (SFSF)

The capacity of the pools situated near the reactor enables the operation of the Temelin NPP until 2014. For this reason the preparation of the new SFSF with the capacity of 1 370 t began in 2002 by carrying out the feasibility study for various storage locations in the nuclear power plant site. The selected place and the proposed technical solution of the spent fuel storage facility enable to increase the capacity of the storage halls, if necessary.

According to the project plan, the site permit is supposed to be obtained by July 2006, the construction approval by September 2008 and the SFSF should be put into the trial operation in February 2014.

The storage concept in the Temelin SFSF is very similar to the Dukovany SFSF. It means a dry storage facility with dual-purpose storage transport casks. The difference is in the disposal solution of the storage building. We suppose double ship arrangement of the storage halls, where each ship of the storage hall has its own crane. Receiving and service part of the building, that is situated vertically to the storage halls, will have also its own crane.

6. Skalka Site - back up location for the Temelin SFSF

The Skalka site used to be an underground variant of the central SFSF. Nowadays it is kept as a back up site for the spent fuel coming from Temelin NPP. The State Office for Nuclear Safety allowed the SFSF to be located in Skalka in January 2000. In March 2001 the authority issued the site permit. The validity of the site permit expires in 2011.

Based on an extensive siting studies the Skalka site was chosen as the most suitable location of all, which were screened outside of the NPP sites in 1993-1997. The Skalka project considers placing the spent fuel dual-purpose casks to the underground horizontal shafts. For the purpose of a detailed geological exploration the 450 m long access shaft and 300 m long exploration shaft were set up and fixed laboratory tests were carried out. The said storage project assumes to locate about 2 900 t of spent nuclear fuel in transport-storage casks transported to the storage facility by rail. Geological conditions enable to double the storage capacity.

7. Planned lifetime of the spent fuel storage facilities

The usual time period for the spent nuclear fuel storage prior to the disposal is several decades. With regard to possible further prolongation of that time period due to constant research on spent fuel recycling and on new transmutation technologies, the required lifetime of the dual-purpose casks is 60 years. The state of spent fuel storage casks is continuously checked and assessed. The behaviour of spent nuclear fuel during storage has been verified by long term experiments. The spent nuclear fuel is expected to be, for the purpose of the final disposal, replaced from the dual-purpose casks to the special, so-called, repository casks. According to the state spent fuel management concept the deep geological repository should be put into the operation in 2065.

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Spent fuel dry storage in Hungary

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Abstract. Paks Nuclear Power Plant is the only NPP in Hungary. It has four VVER-440 type reactor units, which provide approximately 40% of the total domestic electricity generation. The fresh fuel is imported from Russia. According to the original fuel strategy the Soviet Union (later Russia) undertook not only to supply new fuel but also accepted of the spent fuel for reprocessing. This arrangement included also that all products of the reprocessing process (all radioactive wastes, plutonium, uranium) were supposed to stay in the Soviet Union. Conditions, laid down in the original concept have changed. To provide assurance of the continued operation, Paks NPP's management decided to implement an independent spent fuel storage facility and chose GEC-Althom's MVDS design. The facility now has 11 vaults, thus providing storage space for 4 950 assemblies. After 6 years of operation, in May 2003, there were 3 047 assemblies in the spent fuel store, and in its present form it is expected to provide space required for the storage of spent fuel generated until the end of 2007. New capacity needs to be constructed until that time. The paper gives an overview of the situation, describing implementation of the dry storage facility at Paks and its operating experience. Finally, some information is given about the future plans, including the process of selecting the type of next dry storage.

1. Introduction

Hungary is a typically small country, with a population of 10.3 million inhabitants and a land area of 93 000 km^2. Paks Nuclear Power Plant (NPP) plays an important role in energy supply of the country. The plant has four Pressurised Water Reactors of the Soviet VVER-440/213 type, the first of which was commissioned in 1982, while the fourth in 1987. The rated electric performance of each unit is 460 MW. During the last years, approximately 40% of the total domestic electricity consumption has been supplied by Paks NPP.

Fresh nuclear required for the operation of the plant is so far imported from Russia. Spent fuel has been returned to the Soviet Union, later to Russia, until 1998. Owing to political changes in the beginning of 1990s made inevitable to review the concepts of fresh fuel procurement and the strategy of spent fuel management. In the present Paper, the aspects of closing the fuel cycle in Hungary are reviewed in more detail.

2. General questions of the fuel cycle strategy

2.1. Fresh fuel supply

Fresh fuel is being imported form Russia (previously from the Soviet Union) since the beginning of the operation of the Plant. Even when slightly used fuel assemblies of the Greifswald NPP (Germany) were shipped to Hungary, that fuel was also of Soviet origin. To diversify the fresh fuel procurement, Paks NPP – together with the Finnish utility IVO – signed a contract in 1995 with BNFL (UK), to develop an alternative VVER-440 fuel. The task was successfully accomplished and this fuel is operated satisfactorily at the Loviisa NPP in Finland. Import permit for the alternative fuel has been issued, loading and operation of this fuel is possible any time.

Procuring fuel from the UK is hindered by the Russian-Hungarian Inter-Governmental Agreement of 1994, which is still in force. According to this Agreement the Hungarian Party will buy fresh fuel, during the whole lifetime of the Plant, from Russia, and in exchange the Russian side will receive the spent fuel for reprocessing without the return of the wastes from the process. Contrary to the conditions of the Inter-Governmental Agreement, – not surprisingly, and in line with international practice – the Russian side requested the Hungarian side to accept the wastes and by-products from the reprocessing of the spent fuel. Hungary declared that the country is not in the position to accept the proposal. After a review of the above Inter-Governmental Agreement, introduction of an alternative fresh fuel supplier may become possible.

2.2. Spent fuel generation

Operation of fuel in the nuclear reactor is a safety related subject, as well as a question of economy. The primary goal of the Operator is to increase the burnup of the fuel, while observing the safety limits of operation. To achieve this goal, it is important to know the parameters of the reactor core, and to continuously develop programmes for the design of different loadings and, besides these, to improve such fuel parameters as form of enrichment and power shaping.

Using the above mentioned possibilities, the generation of spent fuel in the Paks NPP is continuously decreasing. At the beginning of the Plant's lifetime, the 4 reactors generated 460 - 480 spent fuel assemblies at a relatively low burnup rate (28-33 GW·d/t HM). Nowadays, the yearly discharge rate from the 4 reactors is 372 assemblies, with a higher burnup (40 - 46 GW·d/t HM). This corresponds to 44.6 tU. During the whole lifetime of the NPP, according to the calculations, the whole amount of discharged spent fuel is 13 400 assemblies, that is equivalent to 1610 tU. From this amount 2 331 assemblies have been shipped back to Russia, until 1998. If a 20-year lifetime extension will be accomplished, this will result in production of 7 440 additional assemblies, corresponding to 890 t U heavy metal weight.

2.3. Spent fuel management

2.3.1. Original ideas

According to the original fuel cycle strategy, the Soviet Union agreed to accept the spent fuel from Paks NPP for reprocessing after three years cooling, without the return of the reprocessing wastes and fissile materials. This arrangement has been modified several times. The first change was the extension of the At-Reactor (AR) decay cooling time for spent fuel to five years. In order to fulfil this requirement, the AR storage capacity of the Plant was doubled by reconstruction of the pool racks. Reprocessing services by Russia originally were free of charge, later prices increased gradually. Return of spent fuel to (Soviet Union) Russia under ever changing conditions were carried out between 1989 and 1998.

2.3.2. Spent fuel shipments to Russia

A total of 2331 spent fuel assemblies were shipped to Russia between 1989 and 1998. There were no further transports after 1998. The shipments, especially those between 1993 and 1996 were on an exceptional basis only, owing to internal changes of the regulations taking place in Russia. Recognizing this fact, and taking into account the importance of the Paks NPP in the

Hungarian electricity system, the management of Paks NPP started to get prepared for long term interim storage of the spent fuel within the country.

2.3.3. Implementation of the dry storage facility

Paks NPP started to review the spent fuel strategy in the early 1990s, looking for new alternatives. Between 1990-92, the NPP staff reviewed the existing storage technologies. Representatives of Hungarian scientific and technical organizations, regulatory agencies, local administration and an international consultant participated also in the process. Advice from the International Atomic Energy Agency was also sought during the selection.

After a pre-selection process, 7 companies were invited to submit Feasibility Studies for different storage technologies. As a result of the evaluation, GEC-ALSTHOM's (UK) Modular Vault Dry Storage (MVDS) system was selected, and a design contract was signed in September, 1992 between the 2 companies. Licensing process for the Independent Spent Fuel Storage Facility (ISFSF) took place between 1993 and 1995. As a basis for licensing, the relevant US regulations were taken as a basis.

Construction of the facility started with the earthworks in 1995. Construction of the first 3 modules of the MVDS was finished early 1997, and the store was commissioned by the end of 1997. Since 1998, extension works of the ISFSF are performed parallel with its operation.

3. Dry storage of spent fuel

3.1. Description of technology

The MVDS provides for 50 years of interim storage for VVER-440 fuel assemblies and followers in a contained and shielded system. The fuel assemblies are stored vertically in individual Fuel Storage Tubes, the Storage Tubes are housed within a concrete vault module that provides shielding. To prevent the development of eventual corrosion processes, the fuel assemblies are in an inert nitrogen environment inside the Storage Tubes. Decay heat is removed by a once-through, buoyancy driven, ambient air flow across the exterior of the Fuel Storage Tubes, through the vault and the outlet stack. There is no direct contact between the fuel assemblies and the cooling airflow.

The storage facility functionally can be divided into three major structural units. The first one is the storage vault, where the spent fuel assemblies are stored in vertical tubes. Each vault is capable to accommodate 450 spent fuel assemblies. The second major structural unit is known as charge hall, where the Fuel Handling Machine travels during the fuel handling operations. The third major unit is the so-called transfer cask reception building in which the reception, preparation, unloading and loading of the transfer cask takes place. The fuel handling system and other auxiliary systems are installed in this building.

The fuel assemblies are transported to the MVDS from the AR pool using Paks' C-30 transfer cask and its railway wagon. The transfer cask is received in the transfer cask reception building where it is removed from the railway wagon and prepared for fuel assembly unloading. The fuel is raised into a drying tube directly above the cask where it is dried prior to being lifted into the Fuel Handling Machine. The fuel assemblies are transferred, within the Fuel Handling Machine, to the vertical Fuel Storage Tubes located in the vaults.

Once the Fuel Handling Machine has moved away from the Storage Tube the air is evacuated from the tube and replaced with nitrogen. After that operation the tube is connected to the built-in nitrogen system monitoring the storage environment of the spent fuel assemblies. In the first phase of construction, the transfer cask reception building and a vault module including three vaults was erected.

3.2. Experience with operation

Originally, the organization in charge of constructing, licensing and operating the ISFSF, was Paks NPP. A new organization, the Public Agency for Radioactive Waste Management (PURAM), independent from Paks NPP, was created and took over first the responsibilities of the extension, later became the licensee and operator of the facility.

Certain operational tasks at the ISFSF are performed by Paks NPP within the frame of a contract with PURAM, such as operation of the Fuel Handling Machine, radiation protection tasks, etc. This solution is more cost effective, because it helps to avoid duplication of operating staff and provides full-time work for the operators of the ISFSF elsewhere.

After 6 years of operation, on 1 May 2003, there were 3 047 assemblies in the spent fuel store, which corresponds to 356.5 t U heavy metal weight. The present total capacity of the ISFSF allows storage of 4 950 assemblies, which, taking into account the NPP discharges, provides enough storage space until the end of 2007.

One of the responsibilities of the Licensee is to collect the Annual Report dealing with operation and safety aspects, and getting its approval. Based on these previous Reports, it can be said, that radiation doses to personnel and releases to the environment are within the regulatory limits. The total collective dose from operations in 2002 was 8 582 person×mSv, which is sufficiently higher than those in 2001 and before. The reason for the increase is in connection with filter replacements at the Fuel Handling Machine. Filter contamination and replacement is an activity defined as normal operation. Replacement of the filters in 1999 resulted in a 0.283 person×mSv collective dose, while the same process in January 2002 alone caused a 3.726 person×mSv collective dose as a result of handling assemblies with crud on their outer surface.

In order to avoid similar events in the future, receipt of spent fuel from the Plant will be more strict, all filters of the Fuel Handling Machine will be fitted with gamma dose meters, and in the future, all filters will be replaced together with their housing. The level of environmental releases is so low, that they can only be calculated. The total annual dose from all effluents and airborne releases to an individual during any of the last years was calculated to be in the range of $1\text{-}3\times10^{-5}$ μSv, while the regulatory limit for the critical group of the population is 10 μSv. TLD dosimeters are mounted on the fence of the ISFSF, measuring the dose rate from direct and reflected radiation as well, with account of the distance factor. Data from these dose meters consistently showed dose levels, which are in the range of background radiation at the site and its vicinity. Measured average for 2001 was 99 nSv/h, for 2002 106 nSv/h, while the background varied between 70 and 100 nSv/h.

3.3. Future plans

During the decision process about the type of the storage facility – although dry storages existed already – there was no design, which would have reference for the storage of spent

VVER-440 fuel. This fact and the lack of information about the behaviour of irradiated Zr1Nb clad fuel during long term dry storage were main factors affecting the choice. This situation changed for the last 10 years, because new technologies came to the market, and experience exists with the dry storage of this type of fuel in different systems, e.g. in casks.

Based on unit costs offered by the existing spent fuel storage technologies, it is without doubt that cheaper solutions could be found, but taking into account the existing facility and the specifics of Paks NPP, the overall situation is more complex. Therefore a complex review is necessary to support the decision about the future extension of the dry store.

The decision about continuing construction of similar MVDS modules, or selecting a new technology is being prepared through a multi-level process. As a first step, based on a request for bids, the available storage technologies were selected, and after their evaluation, the real alternatives to the existing technology were short-listed.

The final offers for the MVDS and for a concrete cask system (CONSTOR, GNB Germany) were evaluated by PURAM and its experts. The technical content was found to be technically feasible. The implementation costs showed no significant difference. It was concluded by the review that licensing of the systems may raise additional issues, therefore the selection process was terminated pending further investigations.

The accident with the spent fuel cleaning system at Unit 2, on 10 April 2003, puts the subject of Paks dry storage technologies in a new perspective. PURAM management decided to initiate actions in support of speeding up preparations for licensing of the 2 above mentioned possible dry storage alternatives. A decision about the actual type of the store may be made at a later stage.

Spent fuel storage in India

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Abstract. India has gone for closed nuclear fuel cycle option to reprocess the spent fuel for recovery of uranium and plutonium to meet energy demand. Wet storage of spent fuel has been the predominant mode of storage in India pending reprocessing. Three regional spent fuel storage facilities are being constructed at different reactor sites to meet the storage requirement. The paper describes important issues related to layout, design and licensing in addition to operating experience.

1. Introduction

The Indian nuclear power programme has grown from twin BWR reactors at Tarapur to 12 PHWRs of 220 MW(e), each working at various locations. Additionally, 6 reactors having a total installed capacity of 1 960 MW(e), 4 × 220 MW(e) and 2 × 540 MW(e) reactors, are in an advanced stage of construction. It is necessary to augment the existing facilities/construct new facilities for storage of spent fuel from these reactors, since there is a time gap between generation of spent fuel from the reactors and their reprocessing. To meet this requirement, three regional spent fuel storage facilities (SFSFs) are being constructed. This is being done on an EPC (engineering, procurement and construction) mode as the technology is well established and participation of Indian industry will help in speedy completion of the project.

2. Spent fuel storage practices followed

Wet storage of spent fuel has been the predominant mode of storage in India at various nuclear reactors and reprocessing plants.

2.1. Research reactors

The fuel for CIRUS AND DHRUVA research reactors is aluminium clad, uranium metal rod. The fuel storage facility is adjacent to the reactor. The fuel pool of CIRUS is epoxy painted whereas the fuel pool of DHRUVA is stainless steel (SS) lined. The spent fuel rods are stored vertically in SS racks. The pool water chemistry is maintained well within the specified limits. The behaviour of fuel material and pool components during the storage period of 10-15 years has been satisfactory. There have been few instances of pitting corrosion of aluminium cladding in the storage environment for long term periods. The fuel from Fast Breeder Test Reactor (SS clad, uranium carbide and plutonium carbide) at Kalpakkam is stored in air at a dry storage vault.

2.2. BWRs

The fuel for TAPS-1 & TAPS-2 at Tarapur, is zircaloy clad uranium oxide, having an enrichment of 2 to 2.4% in 6×6 array. The spent fuel is stored vertically at-reactor (AR) in a wet storage facility in SS racks. This AR storage facility was originally designed for 528 spent fuel assemblies (SFAs). The capacity was augmented to 1 500 SFAs by re-racking using high-density racks. A separate wet storage facility away-from-reactor (AFR) has been designed, constructed and made operational for storage of 2 000 fuel assemblies, extendable to 3 200 fuel assemblies.

2.3. PHWRs

The fuel for PHWRs is zircaloy clad, natural uranium oxide in 19-pin bundle of size 82 mm diameter × 495 mm length. The spent fuel is stored in AR fuel pools. An additional facility has been constructed at RAPS, Kota for dry storage of spent fuel in concrete casks with minimum 10 years cooling at AR. The latest PHWRs are being designed to have fuel pools with storage capacity of 10 reactor-years.

2.3. Reprocessing plants

The spent fuel is stored in underground fuel pools lined with SS plates. Their engineering has been on the line of PHWR fuel pools. Their storage capacity is much less as compared to fuel pool of PHWRs since they are meant to meet operational requirement of reprocessing plant.

3. Design aspects of new independent SFSF

3.1. Capacity

The capacity of new SFSFs was decided based on generation of spent fuel from reactors vis-à-vis take off for reprocessing and storage capacity at reactor.

3.2. Siting

The SFSFs are located at existing nuclear plant site to take maximum advantage of infrastructure, nearness to operating reactor and the approved site for nuclear facility.

3.3. Plant layout

Various issues related to the fuel pool being underground or above ground, single wall or double wall structure were considered before taking up a decision with respect to ease of operation, time and cost of construction. The decision was made in favour of underground fuel pool with single wall construction. See Fig. 1 for the layout of the new facility. The plant layout is designed to take care of following aspects:

- Separate vehicle air locks for trailer entry and exit;
- Cask decontamination and cask storage facility;
- Cask handling in the pool;
- Separate zoning of active & inactive areas.

3.4. Seismic design criteria & civil structure design

3.4.1. Seismic design

Tarapur and Kalpakkam sites lie in the seismic zone III as per the latest Indian Standard (IS-1893: 2002) for earthquake resistant design of structures, which is used for the design of conventional civil structures. Entire India is divided into four seismic zones starting from zone II (lowest seismicity) to zone V (highest seismicity) depending on the seismicity of various areas. The independent SFSF are designed based on the guidelines given in IAEA

TECDOC-1250 [1]. These guidelines are followed for various safety classifications of system & components for Nuclear Fuel Cycle Facilities (NFCF). These facilities are designed for OBE (Operating Basis Earthquake) level of earthquake. The design of pool building and other associated building is performed by using the local soil/rock data obtained through a geotechnical investigation. A weighted average of shear wave velocity up to a depth of 25 m below the founding level has been considered for the purpose of seismic design and qualification of civil structures. The soil-structure interaction has been considered as per ASCE 4-98 standard. Two horizontal and one vertical component of the site-specific OBE response spectra have been used for the design of civil structure.

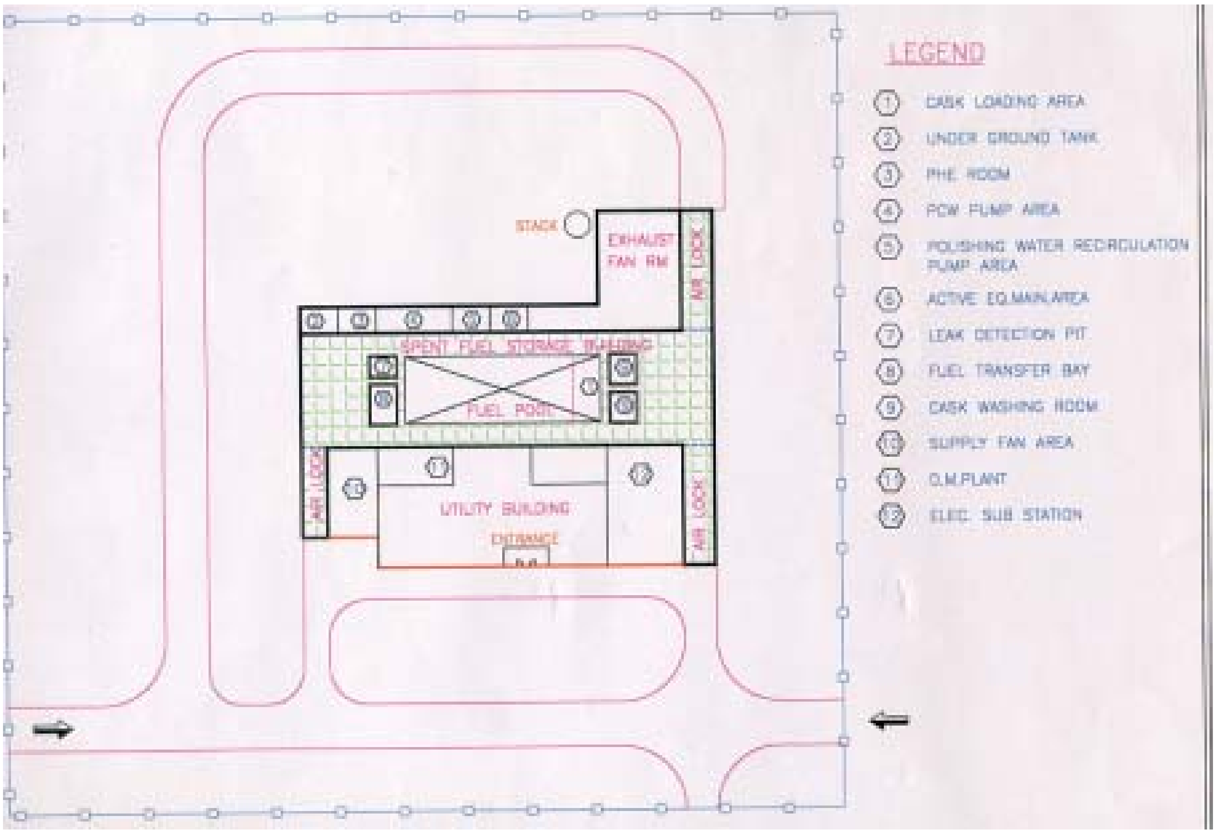

FIG. 1. Layout of spent fuel storage facility

Single time history compatible to the ground response spectra is being used for the generation of floor response spectra (FRS) that are used for the design of various systems and components after peak broadening and flattening. The design of various mechanical system and components is carried out as per the respective design codes and standards based on their safety classification and seismic categorisation.

The pool structure has been designed for hydrodynamic response during seismic event to check for the stability of the structure. Increase in the pool water pressure due to both convective and impulsive modes has also been considered in the design of the fuel pool. Sufficient free board height has been provided in the fuel pool so that the pool water does not spill outside the pool due to sloshing effect during a seismic event.

3.4.2. Civil structure design

The design of civil structure is carried out for a life span of 50 years. The design of the civil structures would ensure that:

(i) The applied bearing pressure under all load combination shall not exceed the safe bearing pressure of the sub grade;
(ii) The total & differential settlements are of zero order;
(iii) No liquefaction for the sub grade below should be there under seismic condition;
(iv) Adequate factor of safety against overturning, sliding & flotation;
(v) Adequate structural integrity.

The design analysis is carried out by using thermal loading of pool in normal operating condition (40°C) as well as in accidental condition (60°C) along with other loading like live loads, dead loads, equipment loads, environmental loads such as thermal loads for building, seismic load (OBE), fire load etc. A safety analysis is carried out for the buildings structure for (i) natural events, like earthquake, wind, flood, solar radiation (ii) man induced events, like fire, explosion, internal flooding, temperature rise of pool water for non-availability of cooling system, etc.

3.5. Fuel pool

The fuel pool is designed as under ground structure on a foundation raft sitting on a hard rock strata. The cask handling and seating zone of pool is located close to trailer receiving area so that the cask is not moved over the stored fuel bundles. The pool walls and raft are made using High Performance Concrete (HPC), added with micro silica for improved leak tightness of the concrete. Additionally, waterproofing compound is provided on both sides of the wall before lining.

3.5.1. Shielding

The depth of pool is based on minimum biological shielding of 3 meter above top most tray of the stack and handling of cage during cask unloading. The radiation levels, estimated at water surface and at working level are less than 1 µSv/hr. The shielding analysis is carried out by using the ORIGIN–2 computer code for source estimation and using 2-D transport theory code DOT–3 for dose rate calculation. The fuel bundles and pool water up to the tray height are homogenized and 18-group gamma source obtainable from ORIGIN–2 is condensed in 3-group source.

3.5.2. Leak collection and detection system

The pool walls are lined with 3 mm thick SS plate and floor is lined with 6 mm thick SS plate to avoid ingress/egress of pool water. The fuel pool is provided with an elaborate leak collection/detection system. The leak collection channels on the backside of pool liners play the dual role of supporting the liners and collection of any water leakage through the welding. These channels are connected to pipe headers terminating at a deep leak detection and collection pit. In addition to the above, there are infiltration bore wells/infiltration galleries provided around the SFSF building to maintain the ground water below the pool raft.

3.5.3. Pool water-cooling and polishing system

The maximum pool water temperature is limited to 40°C in normal condition and 60°C in accidental conditions. Suitable plate type heat exchangers have been provided to remove the heat generated for the spent fuel bundles. In the event of power failures, these cooling systems shall run on Class-3 power supply.

The pumps, valves, heat exchangers, piping are made of SS 304 / 316 material to avoid corrosion and ease of decontamination. There is 100% redundancy in the system. The inlet for primary pool cooling system and polishing system is taken from top surface of the fuel pool through a slotted pipe, which ensures that during accidental condition, pool water will not get siphoned off below predetermined level. Other piping embedments are also taken above the maximum water level to ensure that water does not leak from the fuel pool.

A polishing system for pool water clean up has been provided to remove the fission product impurities like Cs^{137}, Sr^{92} etc. for a turnover time of less than 72 hr. This is based on cat ion unit followed up by mixed bed ion exchange unit. The cat ion unit has a disposable cartridge and mixed bed is regenerative type. This system has been finalized taking into account long experience with only regenerative type of cat ion and anion system. The liquid waste generation has been considerably reduced.

A DM water plant is also provided to meet make up water requirement of pool due to evaporation losses and for cask washings.

3.5.4. Fuel handling

Single failure proof EOT crane of 75 t capacity has been provided to handle 70 t shipping casks. The crane has sufficient safety features like double wire rope system, two rope drums, two independent brakes, VVF drive etc. The reach of the crane has been limited to cask handling area of the pool by the proper layout. A pool bridge carries out the handling of fuel storage trays and fuel bundle within the fuel pool. The pool bridge is equipped with suitable electrical hoist and tong for handling of tray and fuel bundle. These cranes and bridge are also designed to withstand OBE level of earthquake.

3.5.5. Impact absorber for pool

In addition to having single failure proof crane, it is planned to provide an impact absorber inside the pool to take care of accidental fall of shipping cask in the pool while handling. The impact absorber would absorb the bulk of the impact energy. The stresses developed in the raft because of the residual energy of the fall of the cask would be well within the safe limits. The impact absorber is made of pipe in pipe construction and shall get damaged during the impact, but save the fuel pool structure.

3.5.6. Muck cleaning system

There is a compact filtration system, which operates under water for cleaning of pool floor for muck/debris collected during cask handling and fuel storage. The filters are remotely handled and disposed off through a lead shielded cask. The filters have 20-micron fine particles capacity and made of cellulose fibre and layers of wire mesh and silicon paper.

4. Other features

In addition to the above specified features, the SFSFs are also provided with following safety systems:

i) Ventilation System;
ii) Fire detection, fire alarms and fire mitigation system;
iii) Access Control System and CCTV monitors;
iv) Radiation monitoring/Radiation protection system;
v) Class-III and Class-II power supply;
vi) Status and alarms of main systems in control room;
vii) Air locks for tractor- trailer entry/exit.

5. Licensing of the facility

A systematic safety review and design review is carried out by the independent expert groups and safety committees at local and national regulatory body levels. The licenses are given in phases for carrying out construction, commissioning and operation of the facility.

6. Operation of spent fuel storage facilities

With the present experience of spent fuel storage of BWR & PHWR fuel over a period of three decades, it has been found that wet storage is a safe method for storing zircaloy clad fuel from BWRs and PHWRs as the integrity of zircaloy clad is intact in pool environment. Failed fuel bundles account for the increase in pool water activity. The pool water chemistry is maintained with pH of 6-8 and specific conductivity of less than 1 µS/cm. The specific activity of pool water is maintained in the range of 1 000 - 5 000 µµCi/ml. There is a temporary rise in activity levels during fuel loading and receipt operation due to crud movement. There have been a few cases of drop of fuel bundle inside the fuel pool wherein few fuel pins have been separated/damaged. However, this does not increase pool water activity appreciably.

7. Research and development in the field

India has actively participated in two IAEA sponsored CRPs on "Irradiation Enhanced Degradation of materials in Spent Fuel Storage Facilities" [2] and "Corrosion of Research Reactor Al-clad Spent Fuel in Water" and has contributed significantly. The conclusion of the study are quite encouraging and are confident of wet storage of zircaloy clad fuel for storage period of even more than 100 years.

8. Conclusion

India has mastered the technology for design, construction and operation of spent fuel storage facility (wet type) meeting all the international safety standards.

ACKNOWLEDGeMENTs

The authors thank Mr. V.P. Balakrishnan, Associate Director (Projects) and Mr. N.K. Bansal, Associate Director (Operation) Nuclear Recycle Group for their keen interest and guidance.

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New interim spent fuel storage facility at INPP

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Abstract. The Radwaste Management Strategy was issued and approved by the Lithuanian government in February 2002. The Strategy defines a long term strategy for management of spent nuclear fuel. It is intended to use storage facilities for spent nuclear fuel with dual purpose, which would be suitable both for long term storage and transportation. At the moment, INPP does not have a disposal route for its spent nuclear fuel. A potential alternative to the interim storage of the spent fuel is reprocessing. At present, there are no facilities for the reprocessing of RBMK spent fuel. The existing interim spent fuel storage is about 75% filled. The storage will be full before July 2004. The INPP wet storage pools will be able to accumulate spent fuel assemblies for about 2 years after the existing dry storage facility is full.

1. Introduction

The Ignalina nuclear power plant (INPP) is located in the northeast of Lithuania, closer to the borders with Belarus and Latvia. There are 2 units, each of which is equipped with an RBMK-1500. The RBMK-1500 is a graphite moderated, channel-type, boiling water reactor. Its design thermal power is 4 800 MW. The maximum number of loaded fuel assemblies (FA) into the nuclear core is 1 661. The operation lifetime of a nuclear fuel assembly in reactor core is 4 – 6 years (it depends on the load place in reactor). For all reactor operation periods 4 different types of fuel enrichment have been used, i.e. 2.0%, 2.1% (from reprocessed U), 2.4% and 2.6% U^{235}. In 2004, it is planned to use nuclear fuel with an enrichment of 2.8% U^{235} in reactor of Unit 2. Technical characteristics of the RBMK-1500 fuel assembly are presented in Table I. The RBMK-1500 reactor is the most advanced version of the RBMK designs and, as an example, the reactor hall and wet storage of spent fuel (cooling pools hall) are in the same building. At present, about 12 500 fuel assemblies are stored in the cooling pools. Each fuel assembly consist of 0.126 t of nuclear material. The on-site interim storage facility has been built in 1997. It is a dry storage facility in which spent fuel assemblies (SFAs) are stored in containers. The initial design was for 60 containers. Practically, it is possible to increase the quantity of stored containers up to 80. To date, 56 containers accommodate 2 856 fuel assemblies or ~360 t of nuclear material.

2. Brief description of nuclear fuel management at INPP

2.1. General description

Nuclear fuel management at INPP can be represented as several independent subsystems, each performing its own function in a certain sequence. The system includes the following subsystems:

- transporting fresh and spent fuel within the INPP;
- holding & storing spent fuel discharged from the reactor (prior to cutting process, cooling period is at least for 1 year);

- cutting spent fuel assemblies into fuel bundles and placing them into 102-piece baskets;
- holding 102-piece transport baskets with cut fuel assemblies in storage pools;
- transporting spent fuel (after cooling in pools for at least 5 years) to the dry storage facility.

Table I. Technical characteristics of RMBK-1500 fuel assemblies at INPP

Characteristics	Initial enrichment fuel value				
	2.0%	**2.1%[a]**	**2.4%**	**2.6%**	**2.8%**
Nominal mass fraction of U^{235} in uranium, %	2.0	2.1	2.4	2.6	2.8
Mass fraction of mixture $U^{234} + U^{236}$ isotopes, %	No	0.5	No	No	No
Nominal mass fraction of U^{235} in screen pellet, %	0.4	0.4	0.7	0.7	0.7[b]
Mean mass fraction of burnt erbium absorber (E_2O_3), %	No	No	0.41	0.50	0.60[b]
Mass U^{235}, kg	~2.20	~2.28	~2.65	~2.80	~3.00[b]
Uranium mass (isotope composition), kg (±1,60 kg)	111.20	111.20	111.20	111.08	111.08
Screen pellets mass U, kg	~1.02	~1.02	~1.02	~1.02	~1.02
Mass UO_2, kg	~126.00	~126.00	~126.00	~126.00	~126.00
Average density of fuel pellet, g/m^3	$1.04\times10^{7} \div 1.07\times10^{7}$				

[a] reprocessed
[b] insignificant changes are possible

2.2. Detailed description

2.2.1. The first step – fresh fuel supplied

Fresh fuel units have been supplied to the INPP fresh fuel storage from Russia. Fresh fuel has been transportated by 10-piece shipping casks in a specially designed railway wagon. The annual quantity of supply is approximately 900 units. Fig. 1 shows a view of the RBMK-1500 fuel assembly. An RBMK-1500 fuel assembly consists of two half-assemblies or bundles, lower and upper caps, central rod with an extension tube, fasteners and retainers which guarantee connection of bundles and correct positioning of fuel rods in the assembly. Each of the bundles contains 18 fuel rods rigidly fixed by a framework.

2.2.2. The second step - fresh fuel assembling

For reasons of the radiation safety, the fresh fuel units are assembled with a hanger by welding. Welding is performed in the vertical position using special equipment. The total length of fuel assembly is ~ 17 m, the nuclear part is 3.6 m long.

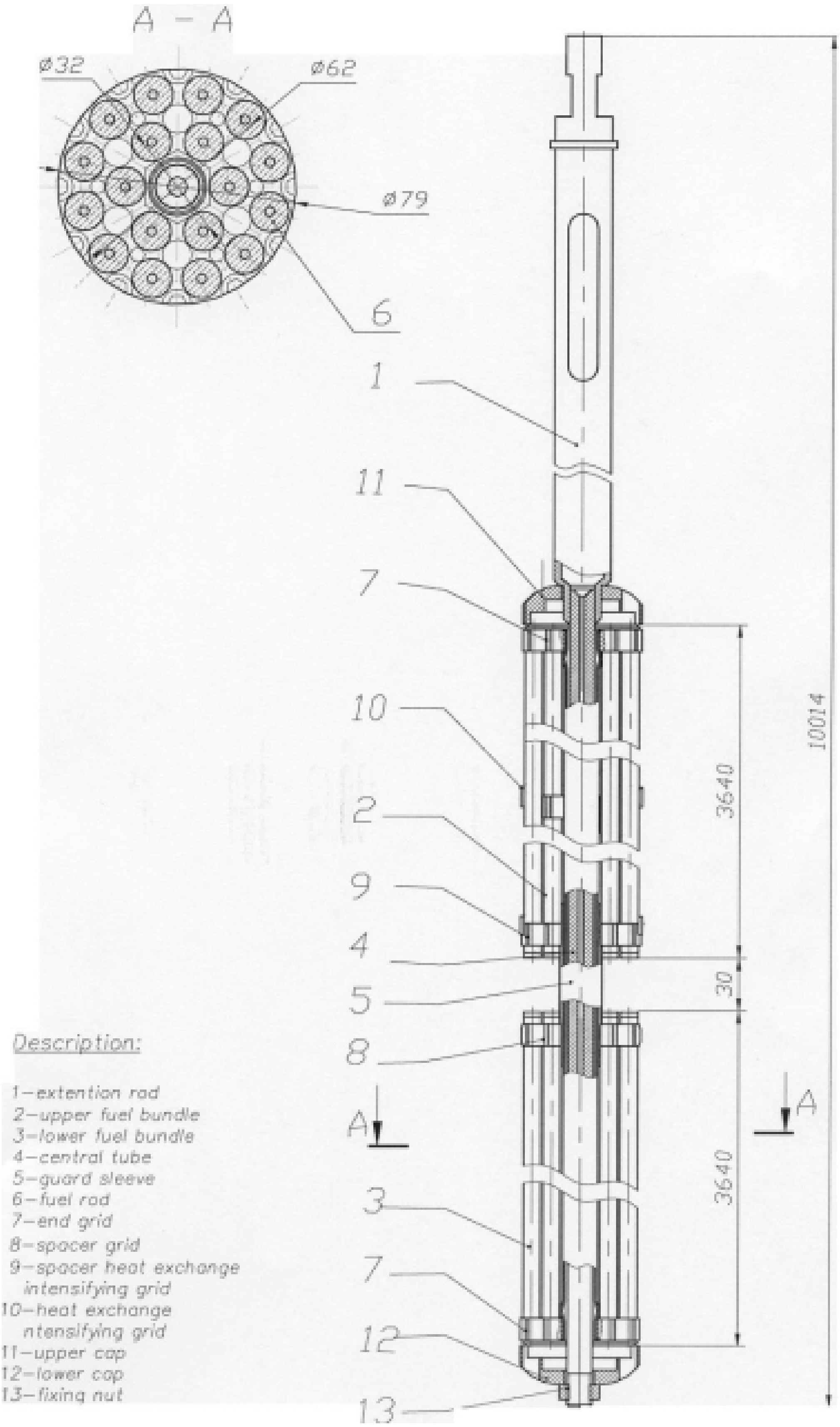


FIG. 1. RBMK-1500 fuel assembly.

2.2.3. The third step - continuous refuelling

Now the fuel assembly can be loaded into the nuclear core by the refuelling machine. The refuelling machine is positioned over the socket into which a fresh fuel assembly has previously been loaded and retracts one inside. Then the refuelling machine is positioned over the reactor channel, joins and seals it, unloads a spent fuel assembly and loads the fresh one. Refuelling is usually performed during a steady-state reactor operation. The monthly reloading rate is ~ 40 fuel assemblies.

2.2.4. The fourth step - 1-year cooling in the pools (or waiting for cutting)

After that the spent fuel is transported from the reactor hall to the pool hall (wet storage) in a transport trolley. Fig. 2 shows a layout of the wet storage hall at INPP. The spent fuel storage and handling system consists of 12 pools:

- 2 pools are used for cooling uncut spent fuel extracted from the reactor (they can accumulate about 1 700 spent fuel assemblies);
- 5 pools are used for storing spent fuel in 102-piece baskets after cutting (they can accumulate about 5 700 spent fuel assemblies in 111 baskets);
- 1 pool is used for collecting spent fuel assemblies prepared for cutting, cutting hanger from fuel assembly; transporting spent fuel assembly to the hot cell and 102-piece baskets in the hot cell and back to the pools;
- 2 pools and 4 transport corridors are used for handling 102- piece baskets and casks.

2.2.5. The fifth step - cutting in the hot cell

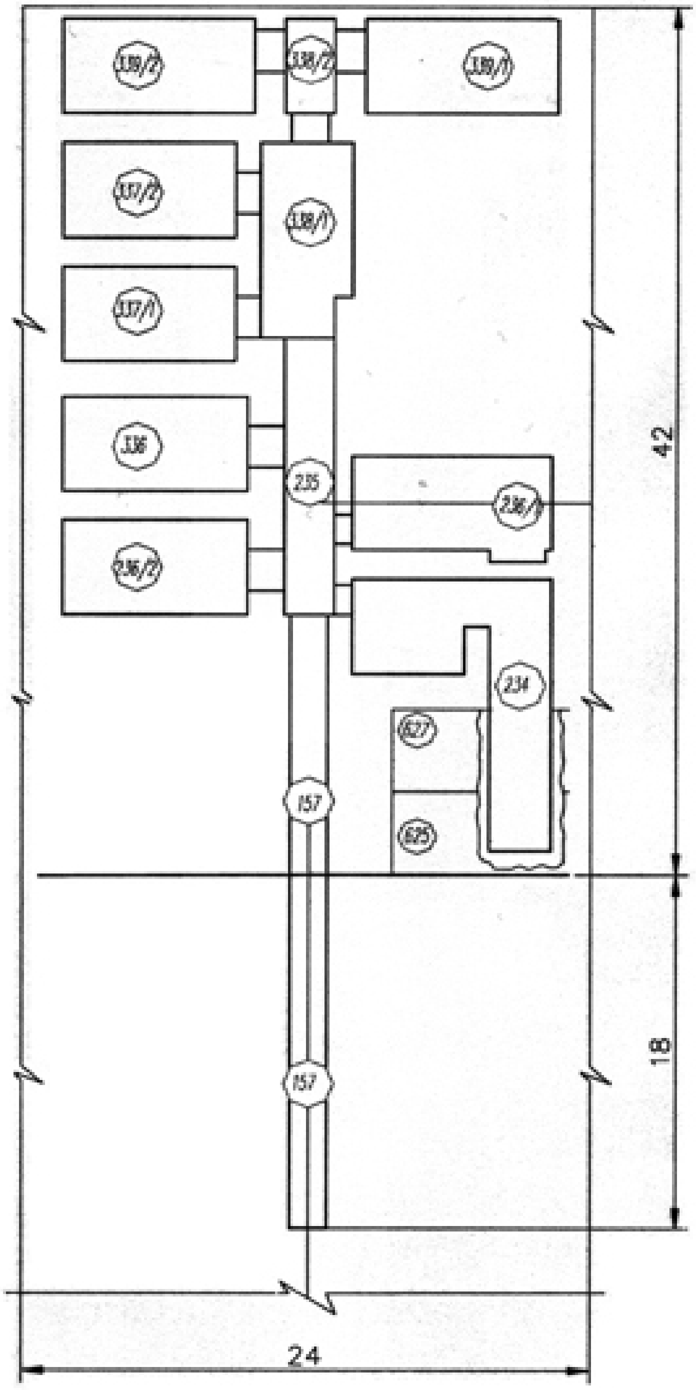
After 1-year cooling, the spent fuel assembly can be cut and loaded into the 102- piece transport basket -32M (see Fig. 3) for further wet storage. Before that the hanger is removed. The spent fuel assembly is transferred in a channel to the hot cell for cutting. The hot cell is equipped with special facilities and tools, which perform such activities as cutting spent fuel assemblies into the extension tube and two fuel bundles. This process is accomplished by an abrasive milling and diamond-cutting machine. Then, the cut bundles are loaded into the transport basket by the hot cell crane. Parts of the fuel assembly, such as central rod, extension tube, lower and upper caps, are cut into smaller pieces, placed in containers and taken away for disposal.

2.2.6. The fifth step - wet storage of filled baskets

The filled basket has been transferred into one of 5 pools for further storing. The filled baskets can be stored in two tiers. The first tier is located at the bottom of the pool, while the baskets of the second tier are placed on special metal beams equipped with stops to prevent the casks from sliding down and falling (see Fig. 4).

2.2.7. The sixth step - dry storage in CASTOR/CONSTOR casks

After at least 5 years, the spent fuel bundles housed in baskets may be moved to the on-site interim storage facility. Spent fuel assemblies are transported and stored in CASTOR-RBMK (Fig. 5) and CONSTOR-RBMK (Fig. 6) containers supplied by GNB. The CASTOR-RBMK cask body is made of ductile cast iron and casted in one piece. The cask has lids made of corrosion protected carbon steel.



rooms 157, 235– transport canals

room 234– transport canal in the cutting bay

rooms 236/1,2– spent fuel assembly storage pools

rooms 336,337/1,2, 338/2, 339/1,2– storage pools for 102 pcs. baskets with cut SF

room 627– control room of hot cell

rooms 513, 0101, 046, 047 – Cutting Fucility

room 338/1 – pool for loading transport/storage container

room 625– hot cell

FIG. 2. Layout of spent fuel pools hall.

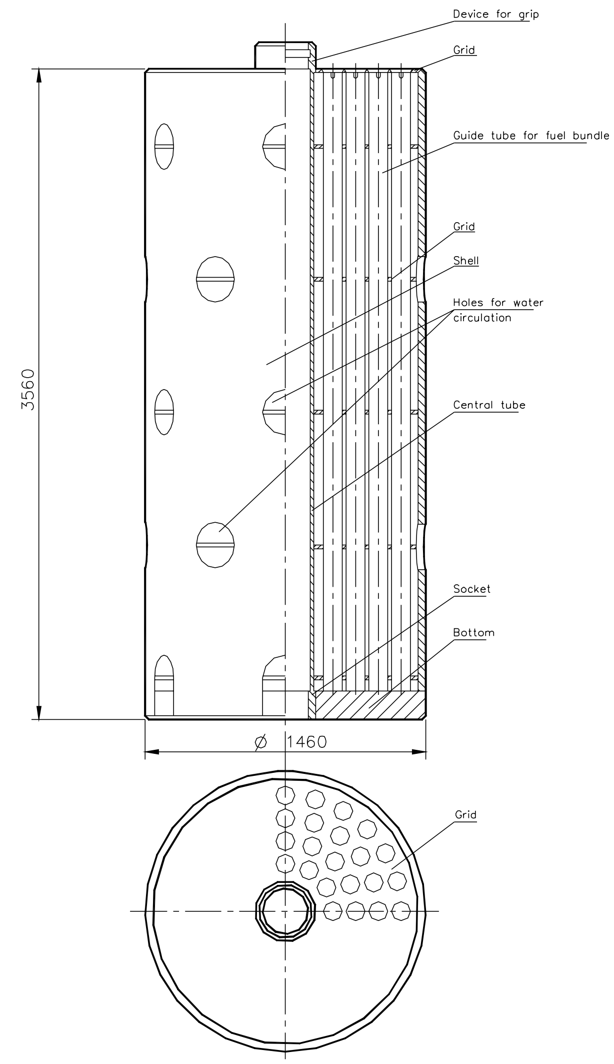


FIG. 3. Arrangement of 102-piece basket for fuel bundles.

The first lid is provided with a double-barrier sealing system to secure leak tightness. The second lid (guard plate) minimizes radiation exposure and ensures weather and additional corrosion protection.

The CONSTOR-RBMK cask body is a welded double shell construction made from grained low temperature structural steel with an integrated reinforced heavy concrete shielding. The cask has three lids. The first lid (primary) is bolted to the head ring of the cask body and provided with an elastomer seal to ensure leak tightness after loading of the spent fuel. The intermediate (seal) lid and the outer (secondary) lid are welded to the steel ring of the cask body.

All actions with spent fuel have been performed at under water (excluding the actions in hot cell) to prevent radiation exposure on operational personnel. The casks are filled with inert gas (Helium), which ensures corrosion protection and improves the passive heat removal. At the moment, the process of nuclear fuel management at the INPP and in Lithuania ends at the existing ISFS site.

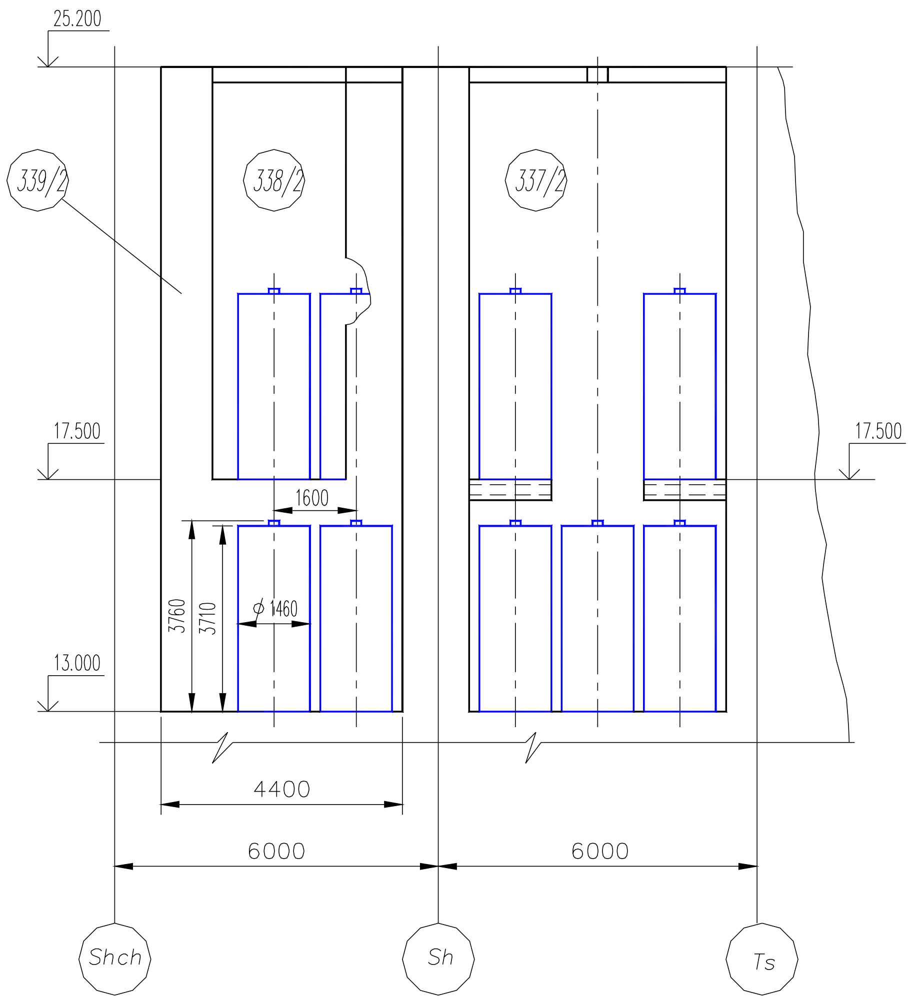


FIG. 4. Arrangement of 102-piece baskets in storage pools.

3. Inventory of nuclear materials at INPP at final shut down date (on 31.12.2009)

At the moment, the inventory at the INPP consists of:

- 2 856 fuel assemblies stored in casks (dry storage);
- 9 698 fuel cut assemblies stored in baskets (wet storage);
- 2 712 uncut fuel assemblies stored in pools (wet storage);
- 3 307 fuel assemblies in the nuclear cores (in operation).

Approximately 4 000 fuel assemblies will be produced by 2010. The assumed total quantity of spent fuel assemblies accumulated at INPP will be – 22 500 fuel assemblies or ~2 800 t of high level spent nuclear material.The calculated average burnup of the spent fuel is 2 000 MW·d/FA. The burnup band of the SFAs stored at the INPP pools ranges from 1 500 to 2 000 MW·d/FA.

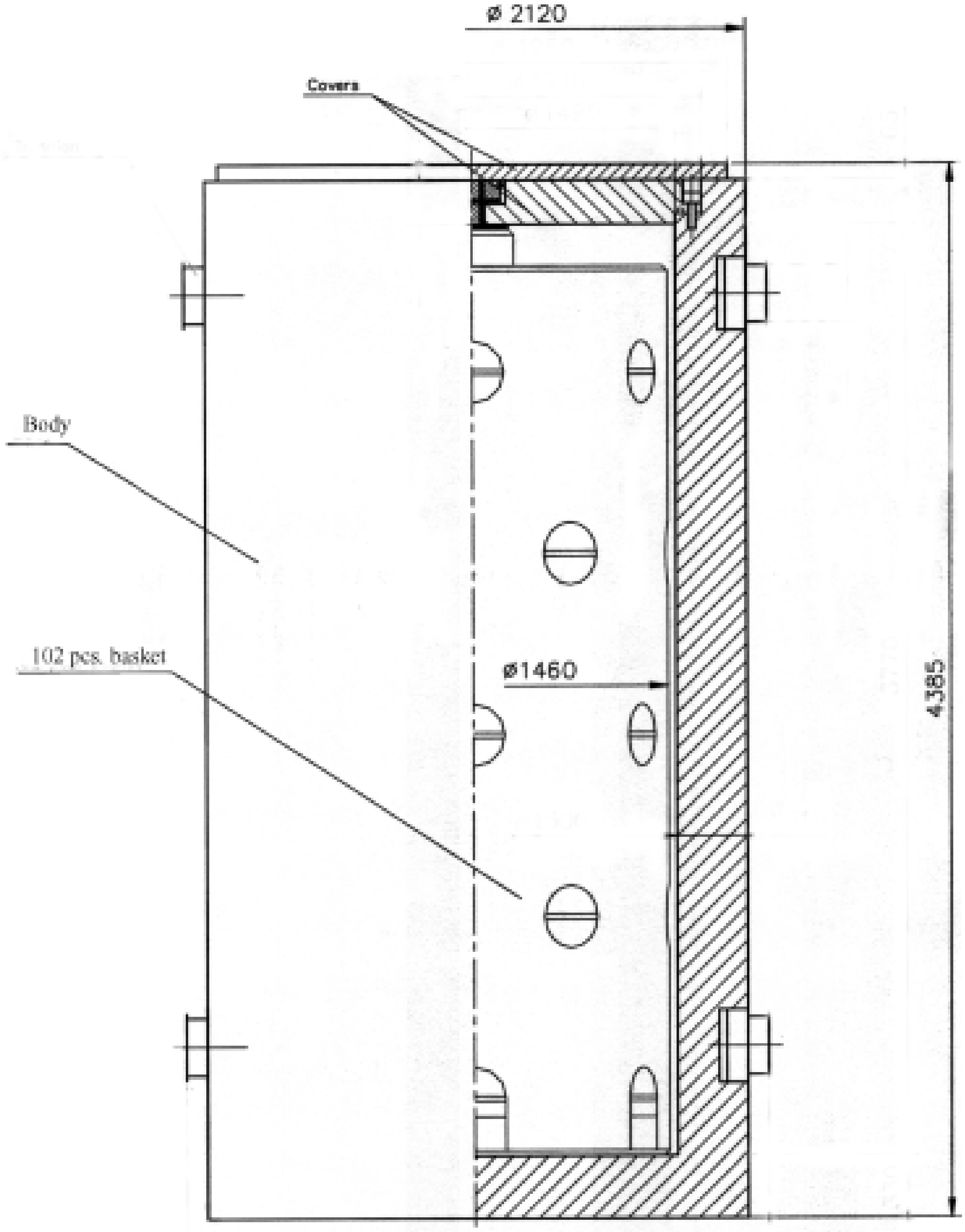


FIG. 5. CASTOR RBMK-1500 cask.

4. Practical dry storage experience

The existing ISFS was put into operation in spring 1999 and on 12th of April, the first cask was filled. 20 CASTOR casks and 34 CONSTOR casks were filled and put in the storage facility. Along the whole perimeter of the storage there is a system providing the continuous dose-rate control with the signal's output to the INPP radiation control board. The casks design provides the following storage criteria:

- Spent fuel residual heat per container - 1,6 kW (design maximum - 6,1 kW);
- Cooling time of spent fuel – no less 10 years;
- Storage period – at least 50 years;
- Maximum cask surface temperature - 36°C;

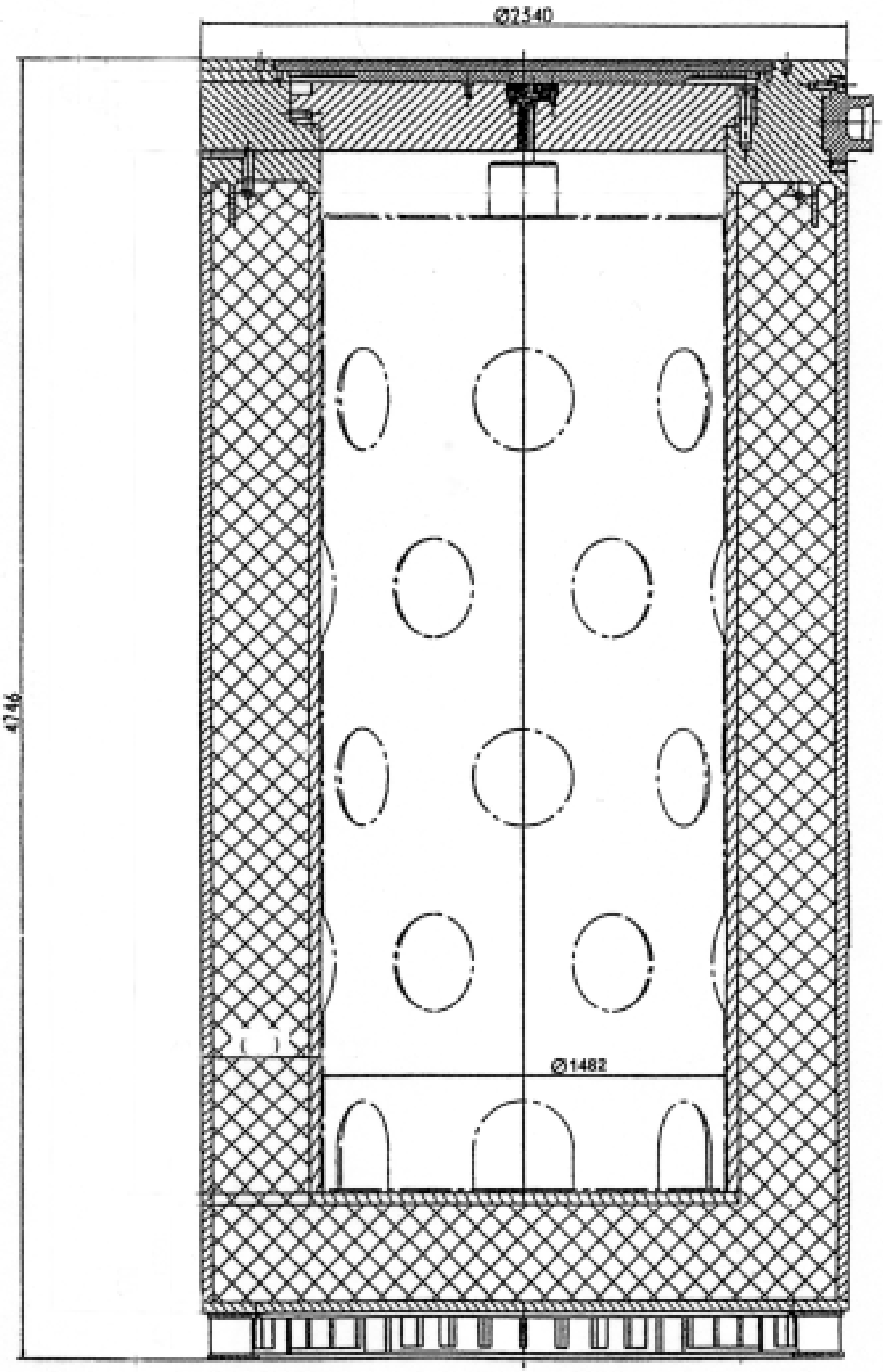


FIG. 6. CONSTOR RBMK-1500 cask.

- Effective multiplication factor (k_{eff}) is below 0.95;
- Removable surface contamination – maximum 300 Bq/m²;
- Surface dose rate:
    - of the CASTOR cask:
        - maximum 90 μSv/h (γ) – (design maximum 727 μSv/h);
        - maximum 200 μSv/h (n^1) – (design maximum 273 μSv/h);
    - of the CONSTOR cask:
        - maximum 110 μSv/h (γ);
        - maximum 100 μSv/h (n^1);
        - total design (γ + n^1) maximum 1 000 μSv/h.

Also there is implemented a checking of sufficient Helium-pressure in the inner cavity between CASTOR cask sealed lids each year. A checking of sufficient Helium-pressure in the inner CONSTOR cavity is implemented before welding of lids. At the moment there were not any failures.

5. National spent fuel storage strategy

The national spent fuel storage strategy [1] was developed to implement the provisions of the Law of the Republic of Lithuania on Radioactive Waste Management and approved by the Lithuanian government in February 2002.

5.1. Strategy objectives

The objectives of this strategy were to:

- strive for implementation of proper radioactive waste management policy;
- develop the radioactive waste management infrastructure based on modern technologies;
- create the effective financing system for radioactive waste management;
- provide for the set of practical actions that shall bring management of radioactive waste in the Republic of Lithuania in compliance with the radioactive waste management principles of IAEA and with the good practices in force in EU Member States.

5.2. Strategy tasks

The tasks of this strategy were to:

- improve the legal basis for radioactive waste management;
- modernize a system of radioactive waste management at the INPP and to implement the new radioactive waste classification system according to the "Requirements on the Pre-disposal Management of Radioactive Waste at Nuclear Power Plant" approved by the order of the head of the State Nuclear Power Safety Inspectorate;
- be ready for the management of radioactive waste, which will result from the INPP decommissioning providing the plant with necessary radioactive waste management facilities;
- modernize the management infrastructure for radioactive waste generated by small producers;
- construct new repositories for radioactive waste in compliance with requirements of legal documents.

5.3. Strategy concerning the management of spent nuclear fuel

The best case would be to store the spent nuclear fuel in the dual-purpose storage systems applicable for both the long term storage and transport. Until 2011, it is foreseen to expand the existing spent nuclear fuel dry long term storage facility at Ignalina NPP or to construct a new facility in the vicinity. Ready for storage spent nuclear fuel shall be removed to the dry storage facility in order that the Ignalina NPP decommissioning activities would be performed effectively. Striving for the safe disposal of the spent nuclear fuel it is essential to:

- draft and implement the long term research programme "Possibilities to dispose of the spent nuclear fuel and long-lived radioactive waste in Lithuania";

- analyze the possibilities to have in Lithuania a deep geological repository for spent nuclear fuel and long-lived radioactive waste;
- analyze the possibilities to create a regional repository taking joint efforts of few counties;
- analyze the possibilities to dispose of the spent nuclear fuel in other countries, and estimate the justification for a price of such disposal;
- analyze the possibilities to prolong the storage period in interim storage facilities for up to 100 and more years.

6. Technological requirements for new ISFS facility

After the above mentioned aspects had been studied and analyzed in accordance with the indicated demands and taking into account references [2 to 5], the requirements for the new ISFS facility at the INPP were developed. The new ISFSF at the INPP shall include the following technologies and equipment:

- to provide interim storage facilities for (approximately) 18 000 spent fuel assemblies;
- to remove spent fuel half-assemblies stored in storage/transfer baskets from the cooling ponds;
- to remove damaged fuel assemblies (non-tight and mechanically) including fuel debris from the full-assembly cooling ponds, arrange cutting of the damaged assemblies and insertion of the cut assemblies and debris into suitable storage/transfer baskets;
- to remove experimental fuel assemblies from the full-assembly cooling ponds, arrange cutting and insertion of the cut assemblies into suitable storage/transfer baskets;
- to provide a safe means of transport of the spent fuel and fuel debris to the interim storage facility;
- to provide effluent treatment equipment and systems for the treatment of contaminated solid and liquid waste arising at the processing facility and to remove external contamination from the surface of the equipment prior to storage;
- to inspect and identify/record the assemblies prior to storage;
- to monitor the storage conditions (including temperature, radiation, gas leakage from the casks, drain water, environment conditions);
- to provide security arrangements at the fuel store and connections to the existing utilities, railways and roads including fences where necessary;
- to place the spent fuel and fuel debris into safe and secure interim storage for a period of at least 50 years.

7. Current status of project

The successful execution of contracts for large buildings, civil engineering, supply and installation projects, and major custom-made equipment requires that contracts are awarded to competent contractors, usually on the basis of competitive tendering procedures. Large contracts financed under loans from the European Bank for Reconstruction and Development are awarded, as a role, through open tendering.

Creation and construction of the new ISFS at the INPP is financed by the EBRD in the framework of the INPP decommissioning programme. A consortium led by the British company NNC Ltd. has won the tender and will be the main consultant for implementation of the project. The Project Management Unit (PMU) is made up of representatives from the INPP and from the consortium. In additional to the new ISFS at the INPP, the PMU, acting on behalf of the plant, will manage several other decommissioning projects. At the moment, the following three main project documents are developed and issued:

- **Technical specification** – a document produced in compliance with this procedure, which specifies the engineering, quality assurance and documentation requirements for the supply of equipment and services to INPP;
- **Pre-qualification process** is one of parts of the tender process, which enables tenderers who may be insufficiently qualified on their own to avoid the expense of tendering or to enter into a joint venture, which may have a better chance of success;
- The pre-qualification process should not be used to limit arbitrarily the number of tenderers and all pre-qualified applicants must be permitted to tender;
- **Invitation to Tender** should follow as soon as possible after pre-qualified tenders have been notified. The tender documents should be issued only to pre-qualified firms, and should refer to the need to provide specified updates information and pre-award verification requirements.

The contract award is expected in spring 2004.

8. Conclusion

The implementation of the project “Interim Dry Spent Fuel Storage Facility at the INPP” shows the following results:

- All technical requirements for the new storage facility are developed and reflected in the Technical Specification;
- The pre-qualification process showed the potential Contractors, their abilities and experience;
- INPP personnel has gained experience and skills in the contracting process.

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Implementation of Romanian NPP spent fuel management strategy *A regulatory approach*

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Abstract. The NPP Cernavoda Unit 1 started operation in 1996, producing around 100 t spent fuel per year. Unit 2 is under construction and restart of construction of Unit 3 is considered. After considering different options, it was found that the best strategy for the management of the NPP spent fuel is the dry storage for at least 50 years, with further possible extension of the dry storage period. This option allows for the time necessary for siting, construction and commissioning of the Romanian geological disposal of the spent fuel, if another more favourable option will not occur meantime. The paper describes the regulatory framework that governs the spent fuel management and details the regulatory process, presenting the main requirements formulated by CNCAN, by regulatory dispositions and authorization conditions, as well as the system of inspections at the site during the construction phase. The last part of the paper presents some considerations related to geological disposal of spent fuel. Romania is a relatively small country, and subject to earthquakes. Nevertheless, the Government considers that siting a geological repository within the country is necessary for secure implementation of the nuclear programme. However, considering the benefits that an international repository can present, Romania would like to keep such an option open, under the condition that international safety and security standards are met and regulatory control at the receiver country is established.

1. Introduction

NPP Cernavoda Unit 1, started to operate from 1996, producing around 100 t spent fuel/year. The plant is of CANDU-6 type using natural uranium as fuel, at a mean burnup of the spent fuel of around 7 800 MW·d/tU. The construction of Unit 2 was restarted, and commercial operation of this unit is expected for 2006. Romania intends to restart also the construction of Unit 3.

In establishing the NPP spent fuel strategy, the characteristics of the design of NPP, which allows a capacity of safe storage of the spent fuel in the wet pond for around 7 years and 6 month of operation (i.e. more than 6 years of wet cooling), were also taken into consideration. After considering different options, it was found that the best strategy for the management of the NPP spent fuel is the dry storage for at least 50 years, with further possible extension of the dry storage period. This option allows for the time necessary for siting, construction and commissioning of the Romanian geological disposal of the spent fuel, if another more favourable option will not occur meantime.

The selected solution for dry storage is the Canadian AECL “Monolithic Concrete Module” type MACSTOR. The design of the dry storage covers 30 years of operation of 2 CANDU-6 units. If Unit 3 will enter into operation further capacity will be needed, and a supplementary storage has to be added.

As Romania has not in place specific regulations for siting, construction and operation of spent fuel storages, National Commission for Nuclear Activities Control (CNCAN) has decided that the Initial Nuclear Safety Analysis required for the siting authorization, and the Preliminary Nuclear Safety Report required for the construction authorization, shall observe the structure and the requirements of U.S. Regulatory Guide 3.48 “Standard Format and Content for the Safety Analysis Report (Dry Storage)” , with some modifications related to the characteristics of the CANDU type spent fuel as well as of Romanian regulatory framework. The assessment of the above documents shall be done by CNCAN according to the applicable requirements of U.S. NUREG 1567 “Standard Review Plan for Spent Fuel Storage Facilities” and to the regulatory dispositions issued by CNCAN.

After a relatively complex process, the siting authorization was issued on 12.08.2001, followed by the construction authorization, issued on 05.06.2002.
The authorization for commissioning of the storage is expected in the first part of 2003. This authorization will be followed by the test operation authorization and by the operation authorization, to be renewed every 2 year.

2. Legislative and regulatory framework

According to Romanian policy, no reprocessing of spent fuel is foreseen, so spent fuel is considered as radioactive waste. The Romanian legislative framework that governs safety of spent fuel and radioactive waste management includes the following:

- Law no. 111/1996 on safe conduct of nuclear activities (as amended); the last amendment is in the final process of approval in the Parliament;
- Law no. 137/1995 on environmental protection (as amended);
- Law no. 98/1994 on public health;
- Governmental Ordinance no. 47/1994 on defense against disasters, endorsed by the Parliament by law no. 124/1995;
- Law no. 106/1996 on civil protection;
- Law no. 105/1999 on ratification of Joint Convention on the safety of spent fuel management and on the safety of radioactive waste management [1];
- Law no. 703/2001 on civil liability for nuclear damages;
- Governmental Ordinance no. 11/2003 on the management of spent nuclear fuel and radioactive waste, including final disposal;
- Governmental Ordinance no. 7/2003 on the peaceful use of nuclear energy.

Law no.111/1996 (as amended) establishes the regulatory framework for nuclear activities. According to this law the regulatory body, National Commission for Nuclear Activities Control (CNCAN), under the coordination of the Ministry of Waters and Environmental Protection is empowered with the regulation, authorization, and control of nuclear activities. According to the law, any (no excepted) nuclear activity (including only possession) and any (no excepted) radiation source within the nuclear activity shall be authorized.

Beside the general requirements for nuclear safety, radiation protection, quality assurance, safeguards, physical protection, emergency planning, preparedness and implementation, Law no.111/1996 (as amended) has also specific requirements regarding radioactive waste management (as spent fuel is considered radioactive waste, these requirements apply also to spent fuel):

- The holder of authorization is responsible for the management of radioactive waste generated by his own activity;
- The holder of authorization shall bear the expenses related to the collection, handling, transport, treatment, conditioning, temporary storage and disposal of the waste produced in its activity;
- The holder of authorization shall pay the legal contribution to the Fund for management of radioactive waste and decommissioning;

- On discontinuation of the activity or decommissioning of nuclear installation, as well as in case of transfer of sources or installations, the holder of authorization shall obtain an authorization to hold, decommission or transfer them, as applicable;
- An authorization for a nuclear activity shall be granted only if the applicant disposes of material and financial arrangements adequate and sufficient for the collection, treatment, conditioning, and storage of radioactive waste generated from his own activity, as well as for decommissioning the nuclear installation when it will cease its authorized activity, and has paid his contribution to the Fund for management of radioactive waste and decommissioning;
- The import of radioactive waste shall be prohibited, except situations in which import follows directly from processing outside Romanian territory of a previously authorized export of radioactive waste, including spent nuclear fuel.

According to the provisions of Law no.111/1996 (as amended), CNCAN issued a set of regulations and internal procedures regarding the regulation, authorization, control and enforcement process. Till now, the following new regulations were issued:

- Radiological Safety Fundamental Norms /2000 (transposing the Council Directive 96/29/EURATOM - the Romanian regulation has a supplementary chapter on the transfer in environment of the radioactive waste);
- Radiological Safety Norms on Operational Protection of Outside Workers /2001;
- Radiological Safety Norms – Procedures for Agreement of External Undertaking /2003
- Radiological Safety Norms –Authorization Procedures /2001;
- Norms for Designation of Notified Bodies in Nuclear Field /2000;
- Norms for Authorization of the Work with Radiation Sources Outside the Special Designated Precinct /2002;
- Individual Dosimetry Norms /2002;
- Norms for Issuing the Work Permits for Nuclear Activities and Designation of Radiological Protection Qualified Experts /2002;
- Norms for Decommissioning of Nuclear Objectives and Installations /2002 (the regulation does not refer to NPPs);
- Radiological Safety Norms for Operational Radiation Protection for Uranium and Thorium Mining and Milling /2002;
- Radiological Safety Norms for Radioactive Waste Management from Uranium Mining and Milling /2002;
- Fundamental Norms for Safe Transport of Radioactive Materials /2002;
- Norms for International Shipments of Radioactive Materials Involving Romanian Territory /2002;
- Norms for International Shipments of Radioactive Wastes Involving Romanian Territory /2002;
- Norms for Transport of Radioactive Material – Authorization Procedures /2003
- Safeguards Norms for Nuclear Field /2001;
- Detailed List of Materials, Devices, Equipment and Information Relevant for the Proliferation of Nuclear Weapons and Other Explosive Nuclear Devices /2002;
- Norms for Physical Protection in Nuclear Field /2001;
- Norms on Requirements for Qualification of the Personnel that Ensures the Guarding and the Protection of Protected Materials and Installations in Nuclear Field /2002;
- Norms on Radiation Protection of the Persons in Case of Medical Exposures /2002;
- Norms on Radioactively Contaminated Foodstuff and Feeding stuff after a Nuclear accident or other Radiological Emergency /2002 (issued together with the Ministry of Health and Family);

- Norms on Irradiated Foodstuff and Alimentary Additives /2002 (issued together with the Ministry of Health and Family).

From the old regulations, still in force till the new regulations will be issued, we mention, related to spent fuel and radioactive waste management:

- Republican Nuclear Safety Norms for Nuclear Reactors and Nuclear Power Plants / 1975: part I: Safety Criteria for Nuclear Reactors and Nuclear Power Plants and part II: Authorization of Operator Personnel for Nuclear Reactors and Nuclear Power Plants;
- Republican Nuclear Safety Norms – Working Rules with Nuclear Radiation Sources / 1975;
- Norms for Prevention and Extinguishing of Fire and for Providing Vehicles, Installations, Devices, Apparatus, Protection Equipment and Chemical Substances for Preventing and Extinguishing of Fires in Nuclear Field / 1978;
- Republican Nuclear Safety Norms for Planning, Preparedness and Intervention for Nuclear Accidents and Radiological Emergencies / 1993;
- Republican Quality Assurance Norms: QA Requirements for the Project Management of the Nuclear Objectives and Installations / 1991;
- Republican Quality Assurance Norms: QA Requirements for the Design of the Nuclear Objectives and Installations / 1991;
- Republican Quality Assurance Norms: QA Requirements for the Procurement of the Products and Services of the Nuclear Objectives and Installations / 1991;
- Republican Quality Assurance Norms: QA Requirements for Manufacturing Products and Providing Services for the Nuclear Objectives and Installations / 1984;
- Republican Quality Assurance Norms: QA Requirements for Construction of the Nuclear Objectives and Installations / 1991;
- Republican Quality Assurance Norms: QA Requirements for the Commissioning of the Nuclear Objectives and Installations / 1991;
- Republican Quality Assurance Norms: QA Requirements for the Operation of the Nuclear Objectives and Installations / 1991.

Other regulations are issued by the Ministry of Health and Family:

- Norms for Medical Examination for Hiring Workers and for Periodical Medical Examination / 2001;
- Norms for medical surveillance radiation workers / 2001.

It has to be mentioned that till now CNCAN has not issued specific regulations for siting, design, construction, operation, maintenance, inspection, and administration of spent fuel and radioactive waste management facilities nor for waste classification, treatment and disposal.

In order to fill the gap, CNCAN intends to issue in 2003-2004 a set of norms for radioactive waste management. For this purpose, in a PHARE project requested by Romania, was proposed a task for preparing this set of documents.

Till this moment, international regulations are used (e.g. IAEA regulations, Canadian Standards, and USNRC Regulatory Guides and NUREGs). In the recent case of licensing of the siting and of the construction of NPP Cernavoda Spent Fuel Dry Storage (2001 and 2002), the assessments and the review of the Initial Nuclear Safety Analysis (required for siting authorization) and of the Preliminary Nuclear Safety Report (required for construction authorization) were performed using as a reference the applicable requirements of the following documents:

- Canadian Standard N292.2-96 Dry Storage of Irradiated Fuel [2];
- 10 CFR 72 Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste [3];
- Regulatory Guide 3.48 Standard Format Content for the Safety Analysis Report (Dry Storage) [4];
- NUREG -1567 Standard Review Plan for Spent Fuel Dry Storage Facilities [5].

The Final Nuclear Safety Report submitted to CNCAN for the commissioning authorization was performed using the same requirements.

Taking into consideration the complexity of the problems, and the fact that there are only a few spent fuel and radioactive waste disposal and treatment facilities, it is expected that even after issuing the set of norms for radioactive waste management, for detailing of the requirements, there will be used regulations of US and of other developed countries, as well as IAEA safety requirements and guides. This will be particularly true for the spent fuel management. Thus, in reviewing the Final Nuclear Safety Report revised for operation authorization, CNCAN will carefully take into consideration also the following documents:

- IAEA SS no. 116 "Design of spent fuel storage facilities", 1994 [6];
- IAEA SS no. 117 "Operation of spent fuel storage facilities", 1994 [7];
- IAEA SS no. 118 "Safety assessment for spent fuel storage facilities", 1994 [8].

The Governmental Ordinance no. 11/2003 on the management of spent nuclear fuel and radioactive waste, including final disposal (to be endorsed by the Parliament) establishes the attributions of the National Agency for Radioactive Waste (ANDRAD).

The main tasks of ANDRAD are:

- to elaborate the National Strategy on medium and long term for the management of spent nuclear fuel and radioactive waste, including final disposal and decommissioning;
- to elaborate the Yearly Activity Plan and to establish the financial resources necessary for the coordination at national level of the management of spent fuel and radioactive waste;
- to create and maintain the national data base regarding the spent fuel and radioactive waste;
- to analyze the characteristics of spent fuel and radioactive waste in view of their management;
- to establish the spent fuel and radioactive waste inventory to be produced in each year, in view of elaboration of the Yearly Activity Plan;
- to elaborate technical standards and procedures for the management of the spent fuel and of the radioactive waste, including disposal and decommissioning
- to coordinate feasibility and siting studies, of design, construction, commissioning and operation of final repositories for spent fuel and radioactive waste;
- to coordinate the decommissioning process for the nuclear installations;
- to cooperate with similar foreign organization to assure the use of the best available technologies for spent fuel and radioactive waste disposal.

The same ordinance establishes the following responsibilities of the holders of authorization:

- to report every year to ANDRAD the quantities and types of spent nuclear fuel and radioactive waste generated during the current year and those estimated to be produced in the next year, in order to allow the actualization of the data base for coordination at national level of the process of management of the spent nuclear fuel and of the radioactive waste, including final disposal and decommissioning;
- to bear (during entire lifetime and decommissioning of the installation) the direct responsibility for the management of the spent nuclear fuel and radioactive waste in view of their final disposal;
- to finance the own activities of collection, segregation, treatment, conditioning, intermediate storage and transport in view of final disposal of spent nuclear fuel and radioactive waste generated during operation, maintenance and repairing activities, including during decommissioning of the nuclear installation;
- to finance the own research and development activities regarding the management of the spent nuclear fuel and of the radioactive waste.

3. Short presentation of Cernavoda Spent Fuel Dry Storage

The CANDU-6 fuel bundle is composed by 37 elements of 495 mm length, containing natural uranium, in zircaloy cladding. The nominal weight of the bundle is 23.7 kg. The burnup of fuel is around 7 800 MW·d/tU. After 6 years of cooling, the mean residual decay heat is less than 6.1 W/bundle.

The design of the storage is the Canadian AECL “Monolithic Concrete Module” type MACSTOR. The spent fuel, after 6 years of cooling in the Spent Fuel Pond, is loaded, under water, in stainless steel storage basket, and lifted in the Spent Fuel Loading Station. Here the basket is dried and sealed by welding. The storage basket is then transferred in the transfer container that is placed above the Spent Fuel Loading Station, and sent to the storage module.

The transfer container is placed above the storage module with a gantry crane, the provisional protection plug of the storage cylinder is removed, and finally the storage basket is downloaded in the storage cylinder. The provisional protection plug is than reinstalled. After filling one cylinder, the provisional plug is replaced by the storage plug. The storage module is monolithic structure of normal density reinforced concrete. Each module contains 20 storage cylinders arranged in 2 lines, each cylinder containing 10 storage baskets with 60 fuel bundles. Thus the storage capacity is 12 000 bundles per module. In total, 27 modules have to be constructed. The storage cylinders are suspended from the superior plate of the module in a common cooling cavity. The passive cooling system of the cylinder is formed by this cavity, having several air inlets placed at the lower part of the module and several air outlets placed at the higher part of the module.

As CANDU reactors use natural uranium, there is no criticality issue in the dry storage of spent fuel. Also it has to be noted that due to low decay heat, the temperature of the fuel is much lower than in the case of PWR fuel, so the air is used as cooling agent inside the storage basket (as oxidation of uranium dioxide is not an issue).

4. Authorization process of the Cernavoda Spent Fuel Dry Storage

As already mentioned, USA and Canadian regulations were used for safety assessment and authorization of the facility. According to Romanian legislation, the authorization process includes the following phases: siting, construction, commissioning, test operation, operation.

The siting authorization is granted by CNCAN based on the Initial Nuclear Safety Analysis and supporting documents, the construction authorization is granted based on Preliminary Nuclear Safety Report, and supporting documents. Among the supporting documents are included reports on various assessments, and the authorizations and agreements that are requested by law (sanitary authorization, environment agreement that is issued based on environmental impact assessment, etc.)

The Initial Nuclear Safety Analysis and the Preliminary Nuclear Safety Report were both prepared taking into account the applicable requirements of U.S. Regulatory Guide 3.48 "Standard Format and Content for the Safety Analysis Report (Dry Storage)" , with some modifications related to the characteristics of the CANDU type spent fuel as well as of Romanian regulatory framework. These modifications were established or approved upon utility request by CNCAN and transmitted through regulatory dispositions and licensing conditions put into the authorizations.

It has to be mentioned that besides safety requirements, requirements related to physical protection and safeguards were formulated according to Romanian legislation for siting and construction authorization; and that in order to construct the storage the utility and contractors needed quality assurance authorizations, issued by CNCAN.

For the commissioning authorization, the Final Nuclear Safety Report is requested. This document is observing also the requirements of U.S. Regulatory Guide 3.48 "Standard Format and Content for the Safety Analysis Report (Dry Storage)". If for the previous stages part of the requirements of the guide were not dealt in detail (for example the chapters on conduct of operations and on operating controls and limits were not previously detailed) for the commissioning authorization they shall be extensively presented.

Regarding the work performed at the spent fuel bay at adjacent areas, the regulatory approach is to consider them as modifications under the operational authorization of Unit 1. CNCAN requested, as a main condition for starting the work at spent fuel bay area, the demonstration of assurance of the nuclear safety during the modification work as well as latter, during the operation of the storage. At commissioning stage, once the preparatory and transfer operation will start (loading of fuel in storage basket, drying, welding, and loading in transfer container), the Cernavoda Unit 1 document "Operational Policies and Principles" shall take into consideration the new operations. Also other documents of the station shall be revised or completed: the emergency plan, training programmes, operation manuals. The operations related to the transfer of fuel to the storage module will be covered at the beginning by the commissioning authorization of the Spent Fuel Dry Storage, and letter by the test operation and operation authorization of the Spent Fuel Dry Storage.

After a relatively complex process, the siting authorization was issued on 12.08.2001, followed by the construction authorization, issued on 05.06.2002. The authorization for commissioning of the storage is expected in the first part of 2003. Actually the Final Nuclear Safety Report was submitted to CNCAN and is under review.

It has to be mentioned that for the test operation authorization, CNCAN intends to require a revision of the Final Nuclear Safety Report that needs to be improved related to some aspects like the operating control and limits. With this occasion, it will be requested that the structure of the revised document observe closer the structure of U.S. Regulatory Guide 3.48.

5. Main requirements formulated by CNCAN in the authorization process

As utility requested the regulatory body to establish requirements related to nuclear safety for the design of the dry storage of spent fuel, CNCAN has formulated, by regulatory disposition no. 96597/22.01.1999 the following conditions:

(a) The structure of general requirements is according to the document 10 CFR 72 Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste, 1998 edition;
(b) The specific requirements and the acceptable values are the same as those for the siting of Cernavoda NPP, as they were described by the Initial Nuclear Safety Analysis, (Units 1-5), the Preliminary Nuclear Safety Report and the Final Nuclear Safety Report (Unit 1) and approved by CNCAN, with a supplementary condition related to the effluent releases that shall result in a dose for a member of critical group lower than 100 μSv/year pt (for Unit 1 and the dry storage together);
(c) The general design criteria and the design basis accidents shall be in accordance with the requirements of the Romanian regulations in force and of the authorizations for siting and test operation of Cernavoda Unit 1, with the following interpretation: the exclusion zone and low population zone calculated for the Spent Fuel Dry Storage shall be included in those established for Cernavoda site (Units 1-5). In the case that this cannot be achieved, any modification of such zones for Cenavoda site that includes areas of the town Cernavoda will not be allowed.

By the regulatory disposition no. 10291/10.04.1999 CNCAN has transmitted a list of US regulatory guides that are applicable for dry storage of spent fuel. It was stated that, depending of the solution chosen, the applicable guide should be observed (together with all the guides and standards referred in that document).

The Initial Nuclear Safety Analysis and the supporting documents submitted to CNCAN were analyzed, and a set of requirements for revising the analysis were formulated by CNCAN by letter 4941/LB/17.07.2001. The revised Initial Nuclear Safety Analysis and the supplementary supporting documents were submitted to CNCAN, and based of them the siting authorization no. CNE DICA-06/2001 was issued on 12.08.2001. The authorization had the following main conditions:

- The foundation of the storage modules shall be realized exclusively on limestone;
- The Preliminary Nuclear Safety Report of the Spent Fuel Dry Storage shall demonstrate the fulfilment by the storage module buildings of the condition related to maximum load;
- The seismic entrance data shall be confirmed by the study regarding the seismic data of the Cernavoda site that is undergoing (the data shall be presented in the Preliminary Nuclear Safety Report of the Dry Storage);
- The authorization is granted for siting a repository of not damaged fuel; storage of a limited number of damaged fuel bundles requires special conditions that could be established latter;

- The Preliminary Nuclear Safety Report shall demonstrate the completeness of the Design Basis accidents and of Beyond Design Basis Accidents;
- The Preliminary Nuclear Safety Report shall the observance of the dose constraint of 0.1 mSv/year for members of the public for normal evolution and of the requirements of Romanian nuclear safety regulations in force regarding the maximum doses in case of Design Basis Accidents; the support documentation for the Preliminary Nuclear Safety Report shall precise the radionuclides releases and the maximum population doses for all Beyond Design Basis Accidents;
- The Preliminary Nuclear Safety Report and the supporting documentation shall analyze the cross influences of the activities of the Cernavoda units (under operation or construction) and the Spent Fuel Dry Storage.

The Preliminary Nuclear Safety Report and the supporting documents were submitted to CNCAN, and based on them the construction authorization no. SNN DICA-02/2002 was issued on 05.06.2002. In the letter no. 3022/LB/06.06.2002, CNCAN states the following main conditions:

- The Preliminary Nuclear Safety Report shall be revised in order to demonstrate the assurance of nuclear safety in the area of the Spend Fuel Pond and in the Extension Building of Spent Fuel Pond;
- In the Final Nuclear Safety Report the severe accident scenario of airplane crash shall be revised, justifying the source term, the emission height, and presenting the calculations regarding activity concentrations, doses and dose rates for all meteorological conditions and for all distances and heights relevant for emergency planning;
- The members of the public working inside the exclusion zone shall be taken into consideration for dose assessments in the Final Nuclear Safety Report.

Based on the revision of the Preliminary Nuclear Safety Report and on supplementary support documents, the work inside the area of Spent Fuel Pond and in the Extension Building of Spent Fuel Pond could be started, as CNCAN gave the approval for this work to be performed under the operation authorization of Unit 1.

During the construction stage, extensive inspection programme was performed by CNCAN, both through site inspectors and through headquarters experts. The inspections covered aspects related to nuclear safety, radiation protection, physical protection, safeguards and quality assurance. For the main construction phases, hold points and witness points established. The inspection reports have mentioned the findings, and corrective actions were requested. The utility closely followed the requirements formulated by CNCAN inspectors.

At this moment, the construction is in final stage, and the Final Nuclear Safety Report was submitted to CNCAN together with the supporting documents. CNCAN has asked for clarifications, and the commissioning authorization will be issued after receiving them and after the last construction stage issues will be closed. In the commissioning stage a similar inspection programme, with hold points and witness points will be in place.

6. Geological disposal of spent fuel

6.1. Spent fuel management strategy

Romania is a relatively small country, and subject to earthquakes. Nevertheless the Government considers that siting a geological repository within the country is necessary for secure implementation of the nuclear programme.

As presented above, the Governmental Ordinance no. 11/2003 on the management of spent nuclear fuel and radioactive waste, including final disposal (to be endorsed by the Parliament) establishes that the national strategy for radioactive waste and spent fuel management has to be established by the new National Agency for Radioactive waste (ANDRAD). Till now, as no specialised body was in place, CNCAN established a virtual strategy that was considered valid in the process of authorization, for safety reasons. Main provisions of the spent fuel management strategy are:

- Transfer of TRIGA research reactor highly enriched fuel back to USA, and long term wet storage of the low enriched fuel of the same reactor;
- Transfer of the WWR-S research reactor spent fuel back to Russia and assure the conditions for safe storage till the transfer will take place (alternatively, if the transfer will not be possible, assurance of long term storage of WWR-S spent fuel);
- Putting into operation the NPP Intermediate Spent Fuel Dry Storage (in 2003), considering the possibility of extension of the designed storage period from 50 years towards 80-100 years, and starting activities related to the geological disposal of NPP spent fuel. The geological disposal will assure the solving of the situation of the spent fuel from research reactors, if they will not be transferred outside Romania, and of spent fuel fragments from post irradiation examination. The limited quantities of long lived radioactive waste (except the uranium mining and milling waste) could be also disposed together with the spent fuel.

6.2. Research activities related to geological disposal of spent fuel in Romania

Due to CNCAN requirements, the National Atomic Energy Agency (within the Ministry of Education and Research, responsible for the promotion of nuclear activities), and the NPP and the institutes that operate research reactors (SCN Pitesti and IFIN-HH Magurele), have started to consider the spent fuel long term management issue. The foreseen strategy for siting a deep geological repository for spent fuel considers the assessment of the possible host formations (salt, granite, volcanic tuff, schist, may be even clay), the use of international underground laboratories for developing the concept and finally, after the selection of the site, the construction of an underground confirmatory laboratory on the selected site.

Studies were carried out and a set of general site selection criteria was developed for the NPP spent fuel geological repository. The work was performed by the company GEOTEC with the help of experts from other organizations, including the University of Bucharest. The results of these studies show that for 4 widespread geological formations in Romania there are candidate sites suitable to the developed criteria [9].

A Romanian MENER research contract co-coordinated by the Polytechnic University of Bucharest, on researches for selection and preliminary characterization of the host geological formation in view of the final disposal of irradiated spent fuel started in 2001.

The first stage of the contract had the purpose: Determination of the technical requirements and conditions for the final disposal of irradiated spent fuel. In this stage were realized the following objectives:

- Performance of comparative study related to the worldwide situation regarding the disposal of spent fuel;
- Establishing the technical requirements and conditions for the spent fuel disposal.

The second stage had the purpose: Analysis of geological host formations for spent fuel disposal.

A long term safety assessment of a repository has been performed for spent CANDU and spent LWR fuel elements in a salt formation. The work was performed by co-operation of Nuclear Research Branch –SCN Pitesti and GRS- Germany, under the NATO contract LG-972750/1998 [10, 11]. A hypothetical repository site has been considered, using data from the EU project PAGIS. Three scenarios have been taken into account: subrosion as the normal evolution of the salt dome, human intrusion into a cavern (representing future human actions) and a combination of brine intrusion from the overburden and undetected brine pockets. Spent fuel elements are assumed disposed off within large storage casks inside drifts. For the sake of comparison, the same source-term model has been applied for both waste types, but with different inventories of radionuclides and different heat productions. The key parameter for assessing long term safety was the radiation effective dose to a member of the critical group. The results of the calculations demonstrate that both types of waste can be disposed off safely.

Impact analyses of radionuclide waste on a hypothetical repository in granite were performed by co-operation of Nuclear Research Branch –SCN Pitesti and DBE Institute-Germany, under the NATO contract NATO ENVIR. LG 974513 [12, 13]. Thermal analyses for the spent nuclear fuel underground hypothetical repository in granite, using TEMPROC code (developed by SCN Pitesti) were performed within the contract. Also thermomechanical analyses using ANSYS code were carried out to evaluate the stability of the rock mass in the near-field of a spent fuel hypothetical repository in granite.

At the Nuclear Research Branch –SCN Pitesti, experimental research connected with future possible disposal of spent fuel in salt rock were performed. Salt convergence experiments were performed on rock salt samples and on cylindrical samples simulating the filling material, resulted by crushed salt pressing. Radionuclide migration experiments for Co and U were performed in rock salt samples and in salt compacts having different densities.

6.3. Implementation schedule for geological disposal in Romania

CNCAN, taking into account the complexity of the safety issues, and the needs for ensuring flexibility in considering of other options feasible in the future, expects that the programme for spent fuel and long lived radioactive waste management will be carried out considering that spent nuclear fuel dry storage for 50 years is required to provide a necessary time for implementing the final disposal programme.

We expect the siting of spent fuel disposal by 2030, and commissioning of the repository by 2050. The actual data shall be established soon by ANDRAD, and the provisions for the collection and management of necessary funds shall be proposed to the Government.

However, considering the benefits that an international repository can present, Romania would like to keep such an option open, under the condition that international safety and security standards are met and regulatory control at the receiver country is established.

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Power reactors spent fuel storage in Slovakia

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Abstract. As the Slovak Republic has an open fuel cycle, it is necessary to store spent fuel. Legislative requirements are given in Act No. 130/1998 Coll. and respective regulations. As help for operators Guide on Construction and Operation of Spent Nuclear Fuel Storages was developed by NRA SR. There are two sites with nuclear installations in the Slovak Republic. Spent fuel unloaded from reactor core is stored in at-reactor pools. After three years storage is the spent fuel placed into interim spent fuel storage facility for approximately 50 years. During this period a deep geological repository should be build.

1. Role of the Nuclear Regulatory Authority of the Slovak Republic

Nuclear Regulatory Authority of the Slovak Republic, as a central body of the state administration, performs the state supervision over nuclear safety of nuclear installations in accordance with Act No. 130/1998 Coll. on peaceful use of nuclear energy. Based on its commission, NRA SR supervises over nuclear safety in facilities for storage and transportation of nuclear spent fuel. In accordance with above mentioned act and relevant regulations NRA SR performs the state supervision during designing, construction, commissioning, operation, and decommissioning of nuclear installations.

2. Legislation

The legislative framework in the Slovak Republic is based on acts and regulations. Acts are at the highest legislative level. Based on general requirements described in the acts, the regulations describe more detailed requirement. Several guides, who can be used as help for operators were issued by NRA SR. Unlike acts and regulations, are guides for operators not binding.

2.1. Acts

Act No. 130/1998 Coll. on Peaceful Use of Nuclear Energy regulates the conditions for use of nuclear energy for peaceful purposes [1]; the obligations and rights of legal persons and natural persons in the use of nuclear energy; the classification of nuclear materials, the conditions for their production, processing, procurement, storage, transportation, use, accounting and control; conditions for management of radioactive waste from nuclear installations and of spent nuclear fuel and conditions for disposal of institutional radioactive waste; nuclear safety conditions; compensation for nuclear damage; state supervision of nuclear safety at nuclear installations, procurement and use of nuclear materials, management of radioactive waste and management of spent nuclear fuel.

Act No. 254/1994 Coll**.** on National Fund of Nuclear Facility Decommissioning as amended by later regulations establishes the respective state fund [2]. Management of spent fuel and radioactive waste means their shipment, storage, treatment and disposal. The Fund being an independent legal entity is managed by the Ministry of Economy and is funded from several sources: contributions from nuclear power plant operators, banks, state and other entities.

2.2. Regulations

Regulation No. 190/2000 Coll. on Radwaste and Spent Nuclear Fuel Management by which details of radioactive waste management and spent fuel management are regulated [3]. This regulation describes general requirements placed upon radioactive waste management and spent fuel management. Radioactive waste and spent fuel shall be managed to as to:

a) Minimize the effect of ionizing radiation exerted upon operators, population and environment;
b) To maintain subcriticality;
c) Remove residual heat;
d) Minimize generation of radioactive waste.

Regulation No. 284/1999 Coll. on the Details of Transport of Radioactive Materials and Radioactive Waste [4]. By this decree the process and methods of road, rail, water and air transport of radioactive material, radioactive waste from nuclear facilities and burnt-up nuclear fuel and the scope and content of the documentation required for issuance of approval for transport of radioactive material are regulated.

Other relevant regulations are **Regulation No. 186/1999** Coll. on Physical protection of nuclear installations, nuclear materials and radioactive wastes, **Regulation No. 198/1999** Coll. on Accounting for and control of nuclear materials etc.

2.3. Guides

Guide of NRA SR on Construction and Operation of Spent Nuclear Fuel Storages. The guide describes requirements for design and operation of spent nuclear fuel storage, especially fulfilment of safety functions [5]. Guide provides detailed information on realization and control of these functions during whole operating life. Guide is developed following IAEA requirements for spent fuel handling and in accordance with Act No. 130/1998 and Regulation No. 190/2000.

3. Spent fuel storage facilities in the Slovak Republic

3.1. Spent fuel storage pools adjacent to reactors

The spent fuel storage pools adjacent to reactors are used for temporary storage of the spent fuel after its unloading from reactor core. V-1 and V-2 NPPs have the same spent fuel storage pools. Each unit has one pool. The base grid of all units at SE-EBO was constructed to store 319 spent fuel assemblies and 60 hermetic casings for defective fuel assemblies. In Mochovce nuclear power plant, the spent fuel storage pools adjacent to reactors were made more compact, and the lower grate capacity is almost double compared to the V-1 or V-2 NPP. The capacity of the pool is 603 fuel assemblies and 54 hermetic casings. Both Bohunice and Mochovce NPPs have standby grate to store spent fuel from reactor during planned overhauls or in emergency case.

3.2. Interim wet spent fuel storage facility Jaslovské Bohunice

The ISFSF Jaslovské Bohunice was commissioned in 1988. In this facility the spent fuel from V-1 and V-2 nuclear power plants is stored. During 1997-2000, the ISFSF was subject to a reconstruction and seismic upgrade. The original capacity was increased from 5 040 to

14 112 spent fuel assemblies by replacement of the T-12 containers with the capacity of 30 fuel elements by the compact ones KZ48 with the capacity of 48 fuel elements.

3.3. Interim dry spent fuel storage facility Mochovce (project preparation)

In 2001, Slovak Electric joint stock company decided on the dry interim spent fuel storage facility construction on Mochovce site. Spent fuel from 40 years of operation of two Mochovce nuclear power plant reactor units will be stored in it. It represents 6 552 fuel assemblies in total. The interim spent fuel storage facility shall be commissioned in 2009, when the spent fuel storage pools adjacent to reactors will be full.

4. Future development

4.1. Burnup credit application in the criticality calculation of VVER-440 fuel

NRA SR warrants various research tasks under the R&D program. The Nuclear Materials Division prepared a task of the burnup credit application in the criticality calculation of the WWER-440 fuel assemblies in cooperation with Nuclear Power Plants Research Institute. Nuclear Power Plants Research Institute will perform this task in 2003 through 2005.

4.2. Long term spent fuel storage facility development

Development of a deep geological repository (DGR) in Slovak Republic for permanent disposal of spent fuel and high-level radwaste started to be dealt with systematically step-by-step in 1996. There were 5 sites selected in the process of the step-by-step assessment. Results of work to be done shall demonstrate all necessary conditions of the DGR development and implementation. The most important aspect of the above mentioned policy is the site identification, including the public acceptance.

5. Conclusion

The existing legislation [6] is compliant with IAEA recommendations. Even when the Act No. 130/1998 Coll. had entered into force in 1998 and relevant regulations in 1999 and 2000, since the Slovak Republic became an accessing country to the European Union we had to start amendment of our legislation to reflect European legislation. The new Atomic act will be issued in 2004 followed by respective regulations. We have sufficient capacity to store all spent fuel produced in our NPPs for at least 15 years. During this time period a new dry interim spent fuel storage facility will be build at Mochovce site. At the end of this time period a deep geological repository should have be put into operation.

REFERENCES

[1] Act No. 130/1998 Coll. on Peaceful Use of Nuclear Energy.
[2] Act No. 254/1994 Coll. on National Fund of Nuclear Facility Decommissioning.
[3] Regulation No. 190/2000 Coll. on Radwaste and Spent Nuclear Fuel Management.
[4] Regulation No. 284/1999 Coll. on the Details of Transport of Radioactive Materials and Radioactive Waste.
[5] Guide of NRA SR on Construction and Operation of Spent Nuclear Fuel Storages.
[6] National report, Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (April 2003).

An update on spent fuel and HLW management in Spain

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Abstract. The Spanish nuclear waste policy is established in the General Radioactive Waste Management Plan (currently in its 5th revision) issued by the Ministry of Economy. An overall amount of 6 800 tU of spent fuel will have to be managed in the country along with other minor quantities of HLW. Pool reracking was carried out in the nineties and dry storage technologies have been selected for implementing the next intermediate steps, before taking the decision for a final repository.

1. The matter

The Spanish nuclear program consists of nine operating Light Water Reactors (LWR) and one Gas Cooled Reactor (GCR), which was shutdown in 1990 and is currently in the end of its Level 2 decommissioning phase.

ENRESA (Empresa Nacional de Residuos Radiactivos, S.A.) is a state owned company that is responsible for managing all the nuclear wastes generated in Spain and also for the decommissioning of the nuclear installations in the country. ENRESA's main policy is stated in the General Radioactive Waste Management Plan, currently its 5th revision is in force, which is issued by the Ministry of Economy and is approved by the Cabinet. The main lines of action indicated in this document are:

- Forecast of the overall waste generation;
- Establishment of strategies and technical activities;
- Definition of the economic and financial aspects;
- Deployment of R & D activities.

ENRESA's role related to the spent fuel (SF) and high level waste (HLW) management has three main tasks:

- Designing, building and operating the facilities needed for interim storage and final disposal of SF and HLW;
- Conditioning and transporting of the wastes;
- Managing the operations derived from the decommissioning of nuclear power plants (NPP).

Since 1982, the strategy followed in the country for spent fuel management is the "open cycle", except for the Vandellós one, which was reprocessed. An amount of 6 800 tU will have to be managed, assuming 40 years of operation of the NPPs. Limited quantities of MLW and HLW (to be returned to Spain before 2010), from former reprocessing contracts will also have to be managed.

Fuel elements come from different manufacturers. Their main characteristics are shown in Table I.

Table I. Spanish fuel assembly main characteristics

FUEL ASSEMBLY CHARACTERISTICS	14 × 14	17 × 17	16 × 16	BWR (8 × 8, 9 × 9, 10 × 10)
Cross section (mm × mm)	198 × 198	214 × 214	230 × 230	141 × 141
Number of rods	179	264	236	62
Maximum length (mm)	2 855	4 064	4 293	4 475
Active length (mm)	2 415	3 658	3 400	3 810
Rod pitch (mm)	14.1	12.6	14.3	16.25
Total weight (kg)	396	672	733	284
Uranium weight (kg)	266	465	480	190

2. The past

2.1. The reracking projects

The first step for increasing the original capacities of the spent fuel storage was to rerack the NPP spent fuel pools. All pools were reracked between 1989 and 1998 using borated stainless steel (BSS) high-density racks. Details of these projects were presented in the preceding IAEA conference celebrated in 1998.

2.2. The dry storage programme

Dry storage was the technology selected for the pool supplementary SF storage. The needed additional dry storage capacity is dictated by the Trillo NPP demand between 2002 and 2013, and from 2013 onwards by the needs of several other plants and others that will be decommissioned. The strategy adopted was directed towards At-Reactor (AR) storage in dual-purpose casks for the Trillo NPP spent fuel (using the DPT cask), and Away From Reactor (AFR) Centralised Storage Facility by 2010.

The DPT cask is a dual-purpose cask with an overall heat dissipation capacity of 27.3 kW, licensed in Spain for storage and transport (see Table II). It can allocate 21 SF assemblies of the following characteristics:

- Burnup: ....≤ 40 000 MW·d/tU;
- Initial enrichment:...............≤ 4% U-235;
- Minimum cooling time: .....5 years.

The cask has a multilayered body of SS-Lead-SS-NS4FR (a neutron shielding polymer) with a basket of SS and aluminium disks and fuel tubes made of SS and borated aluminium as a neutron absorber. It has 2 main lids and uses full sets of redundant metallic o-rings in all lids and ports. Its overall length is 5 024 mm (without impact limiters in its storage condition) and its external diameter is 2 368 mm.

Table II. DPT cask data

MATERIALS	• Multilayered body: SS - Lead - SS - NS4FR - SS • Basket: SS and Al disks. Fuel tubes made of SS y BAl • 2 SS main lids • Metalic closure o-rings • Impact limiters: Wood - SS
DIMENSIONS	• Outer: 5 024 mm in height and 2 368 mm diameter • Inner: 4 331 mm in height and 1 679 mm diameter
WEIGHTS	• Body: 81.8 t • Basket: 6.9 t • Impact limiters: 8.8 t • Fuel assemblies: 15.4 t • Loaded in storage configuration: 104.6 t • Loaded in transport configuration: 113.1 t
CAPACITY	• 21 fuel assemblies KWU 16 × 16 - 20 used in Trillo NPP
FUEL ASSEMBLY CHARACTERSITICS	• Burnup: ~ 40 000 MW·d/tU • U-235 initial enrichment: ~ 4% • Cooling time: ~ 5 years • Overall heat dissipation capacity: 27.3 kW
LICENSING	• Inital 1/4 scale model drop testing: March 1993 • Storage license: July 1995, June 2001, June 2002 • Transport license: December 1996, June 2001, June 2002

The DPT casks are stored in a dedicated Cask Store Building, built in 2000 at the NPP site, like a temporary installation that can receive up to 80 casks. The store is a rectangular plan (see Figs 1 and 2), all reinforced concrete building whose main outer dimensions are 80.8 m × 43.5 m × 21.7 m. The store is divided in two main areas: the cask Store Area, where the casks are being vertically stored, and the Access and Maintenance Area, where the casks are unloaded from the transfer vehicle, by means of a 135 t bridge crane that crosses over the entire building.

The building main design characteristics are:

- Structural seismic design;
- Passive heat removal system (natural convection);
- Very low (< 1 μSv/h) dose rate outside the facility;
- Any cask can be removed without reallocating the others;
- Easy and safe cask handling operations by single failure proof crane;
- Permanent cask leak control.

The cask store licensing process was initially seen as a design modification to the Trillo NPP. The start up authorisation was requested in February 1996. The Consejo de Seguridad Nuclear (CSN), which is the Spanish Regulatory Body, approved the design in 1997 based upon final resolution of the some issues like: structural design, external fire hydrant extinction, fire risk analysis, measures to avoid oil spills from the crane, etc. All the questions were satisfactorily answered in 1998. In June 2000, the Final Safety Analysis Report (FSAR) was finally issued.

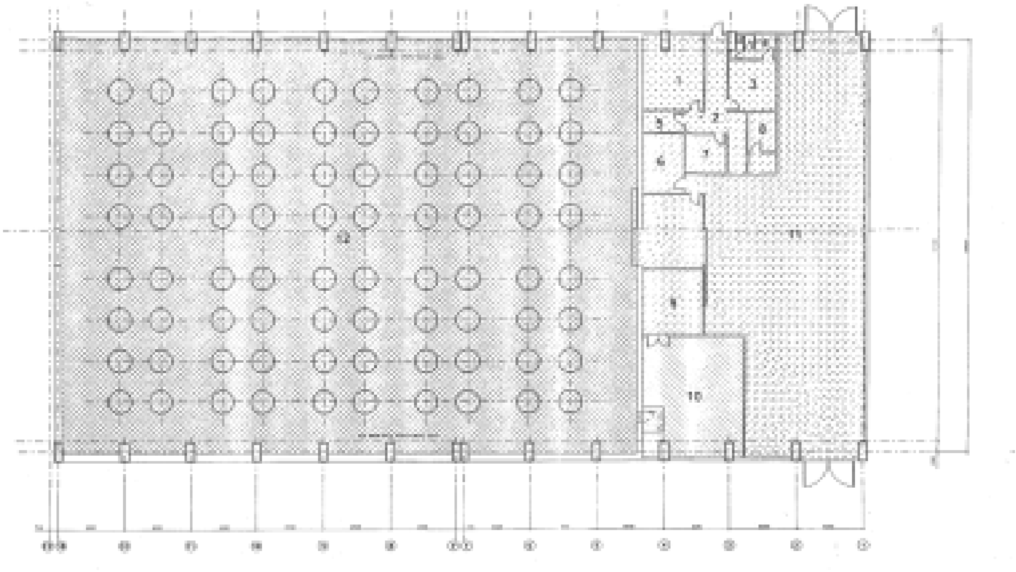

FIG. 1. The Trillo cask store plan.

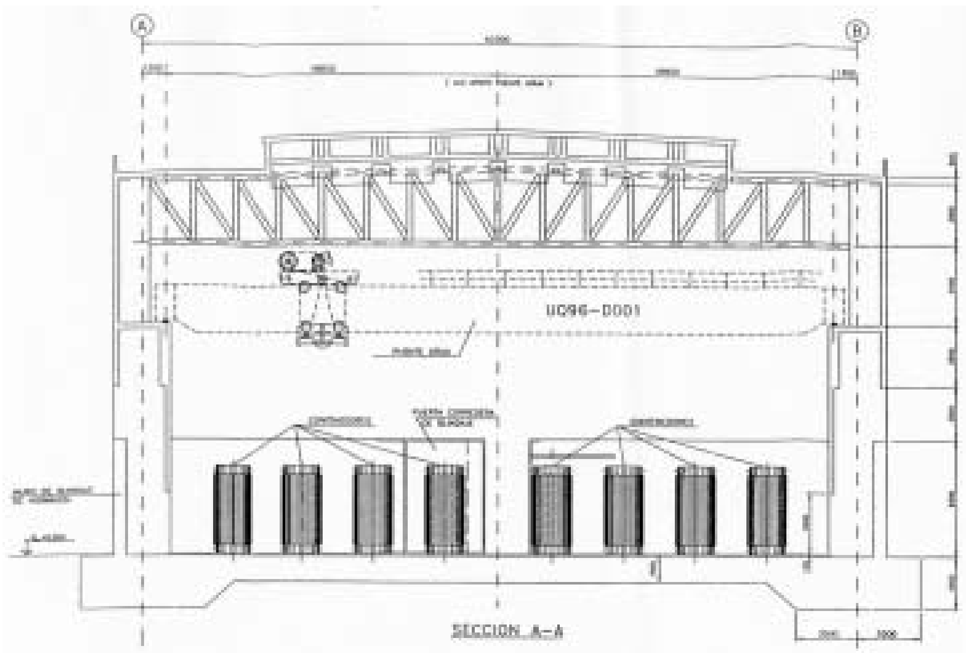

FIG. 2. The Trillo cask store cross section.

3. The present

The casks are being manufactured in Spain by the Spanish company ENSA and 2 units have already been loaded in July 2002 and four or six more will be loaded in 2003.

The Cask Store Building is very close to the exit of the Containment Building of Trillo NPP (around 200 m). After the casks are loaded in the spent fuel cask pit and prepared (drained, vacuumed and backfilled with helium) are taken outside trough the containment equipment airlock using a dedicated cart. Once outside the containment, the gantry crane lifts the cask and goes down it up to a special skid mounted over a lorry that transfers it to the Store Building. There the building bridge crane lifts the cask again and transfers it to the assigned storage position.

4. The near future

4.1. Trillo license for higher burnup

The current license conditions of the DPT cask allow to load as much as 320 SF assemblies of the existing ones generated by the Trillo NPP. An ongoing effort is dedicated to extend this license for higher burnup (up to 45 000 MW·d/tU without changing the design, just by extending the cooling time to 6 years). This approach would allow the storage of up to 462 spent fuel assemblies or otherwise to save the low burnup fuel for future mixed loadings with fuel with much higher burnup.

4.2. José Cabrera NPP dismantling

José Cabrera (Zorita) NPP will stop its operation by April 2006. The first activity that needs to be made before decommissioning will be to unload from the pool the 377 spent fuel assemblies that will remain in the plant by that time. The spent fuel from this Plant has relatively low burnup (< 45 000 MW·d/tU) and low initial enrichment (<3.6% in U-235). The main constraints for the design of the equipment to be used to support the storage system will be related to the plant features: the limited plant crane capacity (70 t) and its old design, the structural stresses of the floors and the small pool loading area.

The storage system to be used in the José Cabrera NPP is targeted to be ready by the time the plant will be stopped. A vendors pre-selection process has been performed and ENRESA is now requesting bids to the qualified vendors, for the design, licensing, manufacturing and delivery of the storage and transportable system for the spent fuel from this NPP.

4.3. The ATC project

The “Almacén Temporal Centralizado” (ATC) is the name for the Away from Reactor Spent Fuel Storage Centralised Installation, whose implementation is required by 2010, according the General Radioactive Waste Management Plan.

The ATC should be able to accommodate 20 233 fuel assemblies (6 875 tU), decommissioning wastes and should also receive minor quantities of reprocessing wastes (HLW glasses and MLW). The design life of the facility has been set at 100 years.

To comply with these requirements, former studies on the different technologies and alternatives (casks, pool, vaults, etc.) were performed in 1995 and 2002. As a result of these studies, the vault technology was selected and a preliminary conceptual project was performed in 1995. An ongoing effort consists in the development of the Generic design (site independent) which was started in December 2002 and is expected to be presented for Regulatory review by November 2003.

5. The far future

Deep geological disposal is the preferred option as final solution for the Spanish spent fuel and HLW. The following actions were undertaken up to now:

- Conceptual designs for granite, salt and clay formations;
- Performance Assessment (PA) exercises supporting these designs;
- Catalogue of favourable formations in the country.

The final decision for the disposal strategy is currently postponed until year 2010 In the meantime RTD studies on deep geological disposal and partitioning and transmutation will continue to provide the Administration with the information required for decision making.

National policy in the area of spent fuel management in Ukraine: *Current status and trends (prospective)*

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Abstract. The closed fuel cycle concept in relation to the WWER was adopted in the former USSR. The WWER-440 spent fuel assemblies (SFAs) were shipped to RT-1 plant ("Mayak" enterprise) for reprocessing. WWER-1000 SFAs were shipped to Krasnoyarsk-26 (Zheleznogorsk) for storage in the wet away-from-reactor (AFR) spent fuel storage facility (SFSF) of the prospective reprocessing plant. RBMK SFAs were transported to the wet AFR storage located on the NPP site. Reprocessing of RBMK spent fuel was considered inexpedient because of the low content of fissile nuclides. The closed fuel cycle for some countries at present and in the nearest decades is evaluated as economically unprofitable. Presently, Ukraine continuing to ship spent fuel to Russian reprocessing plants is developing the Intermediate Spent Fuel Dry Storage Programme (deferred decision). The dry SFSF on the basis of VSC-24 casks designed by the American company Sierra Nuclear was put into operation at Zaporizhzhya NPP in September 2001. The modular type SFSF is being constructed near the Chornobyl NPP site and will be completed in the year 2004. The Ministry of Fuel and Energy of Ukraine and the state nuclear utility "Energoatom" are performing preparation works for WWER SFAs storage during the period up to 50 -100 years in the centralized dry SFSF.

1. Introduction

On 20 April 2000, the Parliament of Ukraine (Verkhovna Rada) adopted the Law of Ukraine "About ratification of the joint convention about safety management of spent fuel and radioactive wastes " (Convention). Therefore, by joining the Convention Ukraine has undertaken the obligations to respect its clauses during implementation of the state policy in the field of nuclear energy. In order to implement the state policy in the field of spent nuclear fuel management (the main principles of which fully conform with the principles of the Convention) and to ensure the unified approach to its implementation the main activity directions in the area of spent nuclear fuel management are defined. They are as follows:

- Provision of safe long term storage of spent fuel (SF) in Ukraine. Advance accelerated investigation of spent fuel behaviour in hot cells are being performed (within 10 years) for forecasting spent fuel behaviour and safety justification under a long dry storage condition;
- Creation of the legislative base and of the financial mechanism for assurance of spent nuclear fuel reprocessing activity (in order to recycle valuable nuclear materials) and high-level radioactive wastes disposal or deep geological disposal of spent nuclear fuel after making an ultimate decision well-founded from economic and technical point of view safeguarding the interests of the country and the future generations;
- Allocations of duties, rights and responsibilities at all stages of the spent nuclear fuel management to the subjects of legal partnership in this sphere;
- Provision of scientific and technical support of the spent fuel management;
- Promotion of international cooperation and international experience to make spent fuel management practice in Ukraine conform to the world economic and technical accomplishments and meet the international safety standards.

2. General situation of nuclear energy in Ukraine

There are thirteen power water-cooled reactors in operation (six WWER-1000 at Zaporizhzhya NPP, three WWER-1000 at South Ukraine NPP, one WWER-1000 at Rivne NPP, one WWER-1000 at Khmelnytskyy NPP, and two WWER-440 at Rivne NPP) with a total installed capacity of 11 820 MW(e) in Ukraine. That is about 25% of the total installed capacity of the electric power plants. Unit # 3 with RBMK-1000 of Chernobyl NPP was shut down on 15 December 2000. One WWER-1000 unit at Rivne NPP and one WWER-1000 at Khmelnytskyy NPP are under construction. Construction of two WWER-1000 units at Khmelnytskyy NPP was suspended, but not cancelled. In 2002, the NPPs produced 78.0×10^9 kW·h of electricity (45.1% of electricity in the country). A forecast of the spent fuel generation from the Ukrainian NPPs, within the 2002 - 2015 year period, is shown in Fig. 1.

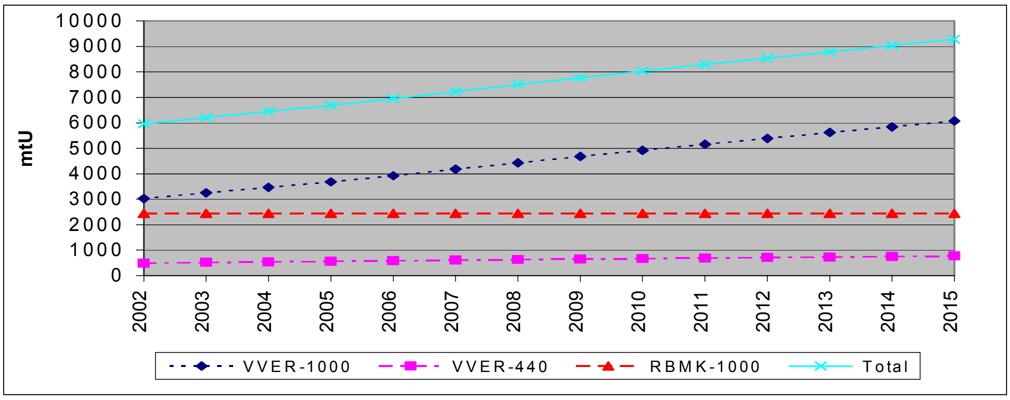

FIG. 1. The estimated spent fuel arising.

3. Spent fuel management

3.1. Current status

RBMK spent fuel assemblies (SFAs) are stored in at-reactor (AR) spent fuel pools and in the wet AFR storage located on the NPP site. WWER-440 SFAs are stored in AR spent fuel pools and than are shipped to RT-1 plant ("Mayak" enterprise) for reprocessing. WWER-1000 SFAs are stored in AR spent fuel pools and than are transported to RT-2 complex site for storage and reprocessing in future after completion of the fuel reprocessing plant construction. AR spent fuel pools were re-racked at all WWER-1000 power units to increase their storage capacity, except Zaporizhzhya -3, -4, -5. Since the year 2003, the WWER-1000 SFAs of Zaporizhzhya NPP are not shipped to RT-2 complex site but are stored in the dry SFSF on the design basis of VSC-24 casks of the American company Sierra Nuclear located at NPP site.

After the dissolution of the Soviet Union in 1991, spent fuel from Ukrainian WWERs was not transported to reprocessing plants until 1995, follow-up the ban that was in force in Russia. Since 1995, spent fuel from WWERs-1000 and WWERs-440 has been transferred to RT-1 and RT-2 plants, under the contracts. One of the conditions of the contracts is the return of vitrified wastes to Ukraine after spent fuel reprocessing.

The annual amount of shipped SF will be determined by the costs of SF reprocessing services and terms of dry storage commissioning. Spent fuel is shipped according to the Contracts concluded between the national nuclear utility "Energoatom" and reprocessing plants.General data on the spent fuel balance is shown in Table I.

Table I. General data on the balance of spent fuel of Ukrainian NPPs

(As of 01 January 2003)

Spent Fuel	WWER-1000, number FAs/tU	WWER-440, number FAs/tU	RBMK, number FAs/tU	Total, tU
Spent fuel generation:				
- during the whole period of NPP operation	7 858/3 250	4 260/490	21 284/2 447	6 187
Stored in spent fuel pools and in the wet AFR storage	3 551/1 468	877/101	21 284/2 447	4 016
Stored in the dry AFR SFSF	66/28			66/28
Shipped to reprocessing plant:				
- during the whole period of NPP operation	4 241/1754	3 383/389	-	2 143
- Including 1992-2001	3 408/1 400	1 767/204	-	1 604

3.2. Deferred decision rationale (justification)

The closed nuclear fuel cycle was evaluated as preferable for the option of large scale nuclear industry development involving fast breeder reactors into the nuclear fuel cycle. Nuclear energy has undergone essential changes during the last decades. The competitiveness of nuclear energy (NPPs) under the increased safety requirements, reliability of operation, was determined first of all by a nuclear fuel cycle cost reduction, NPPs construction terms reduction, and units' unification degree increase. Evolutionary water-cooled reactors (including WWER) have appeared most adaptable and competitive under natural uranium price reduction and enrichment capacities excess. The commercial use of fast reactors was deferred.

The cost of spent fuel removal services is increasing. The problem of spent fuel dispatch to Russian Federation needs consideration not only from the point of view of cost but also from the point of view of reliable operation and prospects for nuclear energy in Ukraine.

Ukraine's NPPs with WWER-1000 reliable operation assurance is a key component to ensuring the country's energy safety. The possibility of uninterrupted operation is mainly determined by the spent fuel storage system which must ensure that SFAs are unloaded from storage pools in time to provide for emergency FAs discharge conditions.

In accordance with the above it should be noted that the construction of RT-2 reprocessing plant is not planned until 2020. The capacities of the existing wet SFSF at the RT-2 complex are limited and will fill up by the year 2007 if the current rate of spent nuclear fuel intake from Ukraine, Russia and Bulgaria is kept. The construction of additional dry storage facility is planned but there is no guarantee that this work will be completed by the year 2007. Thus, spent fuel dispatch to Russia even in the nearest future could be limited due to circumstances beyond Ukraine's control. The national nuclear utility "Energoatom" advances payments for spent fuel reprocessing the results of which it would probably never get because the strategic decision on the structure of the national nuclear fuel cycle is not made. Several circumstances hinder the decision-making:

- the programme of nuclear energy development after the year 2010 and the programme of operating reactors life extension are not developed in Ukraine;
- the possibility of the safe operation of WWER-1000 using MOX fuel is not demonstrated and WWER-1000 necessary modernization programmes are not developed;
- WWER-1000 MOX fuel manufacturing is absent.

Since the nuclear fuel cycle strategy is not adopted in Ukraine a well-founded decision on long term spent nuclear fuel storage is of paramount importance. The decision should give the time for the country's long term nuclear energy strategy development, appropriate nuclear fuel cycle creation and adoption, and be competitive in comparison with the existing practice of spent fuel dispatch to Russia.

3.3. RBMK-1000 spent fuel long term dry storage implementation

It was admitted that the wet AFR SFSF at the Chornobyl NPP site didn't meet the safety requirements and has to be reconstructed or decommissioned. Further on, the existing storage facility design lifetime expires in 2016. It was decided to construct the new dry storage "SF-2" in the area of Chornobyl NPP site instead of the wet "SF-1". The project financial support was provided by the Grant Agreement signed by the Ukrainian Government, EBRD and Chornobyl NPP. The Grant would be available if Ukraine provides:

- financing related to licensing;
- construction site;
- construction of roads;
- power supply lines;
- administrative office and so on.

The choice of the interim dry SFSF project was put out to the international tender (bid) in 1999. The modular type (horizontal concrete modules "NUHOMS" designed by the Pacific Nuclear, USA and Framatome ATEA) SFSF is being constructed near the Chornobyl NPP site and will be completed in the year 2004. The NUHOMS modules are being built in two parallel lines of 116 modules. Each module contains 1 canister. Each canister contains 196 spent fuel bundle cartridges (98 RBMK FAs divided into halves).

The new storage capacity is planned for 21 356 RBMK SFAs and approximately for 2 000 discharged absorber rods for 100 years. According to the Programme, the transport of all spent fuel from the wet SFSF to the dry SFSF should be completed by 2012. The basic equipment for hot cells for dividing SFAs into two parts and their preparation for loading into canisters as well as the considerable part of construction materials and components of the reloading unit and modules are imported to Ukraine. Fig. 2 shows the Chornobyl NPP modular type SFSF.

3.4. Interim dry SFSF implementation on Zaporizhzhya NPP site

It should be noted that the construction of the dry SFSF at the Zaporizhzhya NPP site was foreseen by the design. It was understandable that, when the fifth and the sixth units of the Zaporizhzhya NPP would be put into operation, the difficulty with spent fuel shipment would arise. Primarily the spent fuel storage was planned at the NPP site in the TK-13 containers. Spent fuel was not shipped to Russia from 1993 until 1995. Due to that, the situation became complicated and the decision concerning the construction of dry SFSF, which is capable of long term storage of all spent fuel from Zaporizhzhya NPP, was made.

FIG. 2. Chernobyl spent fuel storage facility (general view).

After the proposals from different companies were addressed, the decision was made to build the SFSF on the design basis of VSC-24 casks of the American company Sierra Nuclear. Zaporizhzhya NPP signed an agreement with the engineering company Duke Engineering & Services to perform the design of the container to store spent fuel from WWER-1000 and escort the design of SFSF. The designing of the rest of the components was carried out by the Kharkiv Design Institute “Energoproject”. The full design capacity of the SFSF of Zaporizhzhya NPP is 380 containers. Each container can consist of 24 SFAs.

The VSC dry SFSF at Zaporizhzhya NPP was put into trial operation and the first three containers were loaded in September 2001. The trial operation was completed successfully. Three containers more (72 SFAs in total) were loaded in January 2003. Up to the end of 2003, other 14 containers (336 SFAs in total) are planned to be loaded. The first 14 baskets, equipment for container sealing-in and spent fuel transportation were manufactured and procured by the company Duke Engineering & Services. The rest of the containers will be manufactured in Ukraine.

The average capital component of the specific cost of spent fuel storage counting on the complete storage facility fill up (380 containers) is estimated to be not more than 40 $/kg U. The running costs of the container storage in the specific cost of spent fuel storage during 50 years will be insignificant because the NPP will be at the stage of operation and afterwards at the stage of decommissioning and practically there won’t be any need in additional personnel. Fig. 3 shows the Zaporizhzhya NPP dry SFSF.

3.5. Centralized dry WWER SFSF

Central dry SFSF is apparently the most economically viable decision for Ukraine [1]. The Ministry of Fuel and Energy of Ukraine and the state nuclear utility “Energoatom” are performing preparation works for WWER storage during the period up to 100 years. One of the most important problems is spent fuel storage technology selection on the basis of option assessment and based on the utilization experience.

FIG. 3. Zaporizhzhye NPP spent fuel storage facility (general view).

On the basis of the preliminary assessments performed by Kyiv “Energoproject” Institute, the following conclusions are suggested:

- the preferable spent fuel management strategy for South Ukraine NPP, Rivne NPP, Khmelnytskyy NPP is centralized storage;
- the preferable technology for centralized spent fuel storage is modular or container storage;
- the preferable centralized SFSF site location is the exclusion area of Chornobyl NPP adjacent to SF-2 of ChNPP.

The approximate capital component of the spent fuel storage specific cost in the centralized SFSF of modular type with two barrier concept in the exclusion area of Chornobyl NPP adjacent to SF-2 including transportation costs for shipment of SF from Rivne NPP, Khmelnytskyy NPP and South Ukraine NPP equals 40 – 45 $/kg U.

Advantages of the centralized SFSF are:

- centralization of nuclear materials monitoring and control;
- simplification of IAEA safeguards procedures assurance and physical protection on condition of one-site spent fuel storage;
- licensing procedures quantity decrease;
- relatively cheaper spent fuel storage in comparison with at-reactor spent fuel storage;
- nuclear facilities quantity decrease.

The main problems of individual SFSF strategy implementation are:

- For every individual (separated) SFSF individual licensing process, including site selection should be provided (according to the Law of Ukraine not less than three sites should be considered);
- For every site engineering and investigative works complex is necessary;
- For every NPP public hearings must be held, the results of which are hardly predictable today;

- According to preliminary individual SFSF site option studies it is impossible to locate such storage facility on any NPP industrial site, thus land allotment for SFSF would be necessary as well as the security perimeter creation;
- In connection with different NPP site geological conditions different engineering solutions for SFSF sites organization would be necessary, i.e. it is impossible to unify SFSF designs completely;
- Reconstruction works at operational units would be necessary which could be determined only after selection of definite storage technology;

Advantages of the site in Chornobyl exclusion area, in the SF-2 are:

- the site has been planned, engineering investigations have been performed;
- SF-2 infrastructure is created (technological environment provision, the viaduct and the roads have been built);
- there is a railroad from Yanov railroad station which is connected to state railroads;
- there is no population in the exclusion area;
- highly qualified personnel is available after Chornobyl NPP shut down (it will allow to create jobs for Slavutych residents);
- the cement plant is operating on the constructed SF-2 site;
- on the SF-2 site auxiliary buildings are foreseen which could be used for centralized SFSF.

Further centralized SFSF organization activities are:

- Definition of the phases of centralized SFSF project development;
- First stage performance of the technical and economic justification of centralized SFSF, including site selection;
- Solution of SFSF licensing organization problems taking into account normative provisions and ecological justification taking into consideration public participation;
- Solution of problems connected with the CFSF creation funding mechanisms;
- Preparation of the bid (tender) on the best offer for the CFSF construction;
- Putting SFSF construction out to the bid;
- Signing the contract with a winner of the tender;
- Technical and economic justification (rationale) of centralized SFSF second stage performance;
- Study of Ukrainian enterprises involvement in equipment manufacturing and supply;
- Decision-making on the state level about the location, designing and construction of the centralized SFSF for WWER NPPs in Chornobyl NPP exclusion area;
- More precise definition of CFSF design normative base provisions;
- designing and construction of the centralized SFSF.

In August of 2002, the state nuclear utility “Energoatom” sent letters to companies and organizations owning technologies both for container and for modular storage with the proposal of WWER spent fuel interim storage technologies implementation in Ukraine to store 11 000 WWER-1000 SFAs and 3 300 WWER-440 SFAs. The tender is planned to be conducted by October 2003. The second stage performance of the technical and economic justification (rationale) of centralized SFSF is planned to be completed by October 2004. According to the Plan the centralized SFSF construction may be realized during 2006- 2008.

4. Advance investigation of spent fuel behaviour under a long dry storage condition

4.1. Research basis

Advance accelerated investigation of spent fuel behaviour in hot cells is required for forecasting spent fuel behaviour under a long dry storage condition. In the State Scientific Centre of Russia "Research Institute of Atomic Reactors" (RIAR, Dimitrovgrad, Russia) the advance investigations of three fuel assemblies from Zaporizhzhya NPP are being performed (within 10 years) under the contract between "Energoatom" and RIAR.

4.2. Main objective

The main objectives of the research are [2]:

- To develop the safety criteria.
- To evaluate optimum and ultimate storage terms (for the specified storage conditions) by the calculating codes on the basis of research results.

4.3. Main specifications

Main specifications to the programme of the research are: The research of fuel rods and fuel assemblies skeleton (structure) fragments behaviour under long term dry storage conditions is performed with a special stand placed in the hot cell.

4.4. Research problems

The research problems/tasks are:

- To evaluate the corrosion processes and their influence on properties of the fuel cladding and FAs structure materials;
- To develop the methodology of forecasting of WWER -1000 SFAs condition after long term dry storage with the substantiation of safety criteria;
- To define allowable fuel cladding temperatures in the beginning of storage;
- To define allowable fuel cladding temperatures for emergency case and allowable time of overheating;
- To evaluate FA behavior as an integrity under long term dry storage with the calculated codes;
- To evaluate allowable term of spent fuel assemblies storage from the point of view of ensuring the possibility in the future to unload FAs from the storage facility and to load them in the transport container, then to transport, to reprocess or to dispose of;
- To develop the recommendations for the option of the safe mode of spent FAs.

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The German policy and strategy on the storage of spent fuel

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Abstract. The German Government has decided, in accordance with the utilities, a phase out of nuclear power. An "Agreement between the Federal Government and the Utility Companies" was initiated on 14 June 2000 and finally signed on 11 June 2001. On the basis of this agreement the phasing out was fixed 2002 in the "Act for the Regulated Termination of the Commercial Use of Nuclear Power" (Gesetz zur geordneten Beendigung der Kernenergienutzung zur gewerblichen Erzeugung von Elektrizität - Atomgesetz). In this law, a remaining operation time is fixed for each nuclear power plant. In context with the change in energy policy a new waste management concept was developed. Concerning the spent fuel management the separation of uranium and plutonium of the spent fuel was no longer desired politically with the consequence that only the direct disposal of the spent fuel will be allowed. A sudden stop of reprocessing, however, was not possible because there have been contracts between the operators of the nuclear power plants and the reprocessing plants in La Hague and Sellafield being granted by changing of notes between the government of the Federal Republic of Germany and the governments of the two reprocessing states. Taking this situation into consideration the abandoning of reprocessing is demanded in the middle of 2005 by law. After this date no transports to the reprocessing plants will be allowed with the consequence that beginning from this time only a direct disposal of the spent fuel remaining or further arising in Germany will be accepted. As from 1 July 2005, transports to the reprocessing plants at La Hague in France and Sellafield in the United Kingdom will be prohibited. The goal is to establish a deep geological repository for radioactive waste starting operation by 2030. Until the commissioning of the repository, the spent fuel has to be stored which has to be carried out at the site of its arising for reasons of avoiding transports and of burden sharing. The Federal Government and the Utility Companies agreed, that the utilities should set up interim storage facilities on the site or in the vicinity of the power plants as soon as possible. The storage takes place in casks in a dry way. Both parties assume that the onsite interim storage facilities will be ready for operation within a period of five years at the most. To avoid the transport of spent fuel to the two central storage facilities in Gorleben and Ahaus the licence for 12 new onsite storage facilities for dry storage has been granted or applied for. Until the commissioning of onsite storage facilities at five sites will be completed, it is necessary to install five temporary storage facilities, where up to 28 dry transport/storage-casks will be stored under prefabricated concrete elements. Apart from these five temporary storage facilities the spent fuel will be stored in 18 storage facilities (including two wet storage facilities) up to the commissioning of a repository.

1. Historical development of policy and strategy on the storage of spent fuel

Germany's policy and strategy on the management and storage of spent fuel has undergone a number of modifications and changes also with reaction upon the concepts of storage of spent fuel.

- From the beginning of the commercial use of nuclear power up to 1994, the relevant legislation, the Atomic Energy Act, included the requirement of reusing the fissile material contained in the spent fuel elements. Thus, consequently leads to the necessity of establishing a nuclear fuel cycle expressively with the installation of a reprocessing facility. In Karlsruhe, a pilot plant started operation. All efforts to install an industrial reprocessing facility integrated in a centre where all activities connected with the fuel cycle and waste management would be concentrated on one site failed. Alternately, the utilities turned instead to plans for abandoning this project and have concentrated their attention on reprocessing abroad. Until today, German spent fuel elements are reprocessed in France and the UK. Following the concept of reuse, the storage of spent fuel took only place in the pools of the nuclear power plants (NPPs) or in the reception pools of the reprocessing plants.

- At the end of the 70ties, there had been discussions between the government and the utilities to open the way of direct disposal of the spent fuel. This discussion lead to a decision of the heads of the governments of the federation and the states to examine the technical and economical aspects of this way. In this decision, the installation of so called central storage for spent fuel was agreed, which should be used as a buffer before moving the spent fuel abroad in case of bottlenecks in the storage capacity of the NPP pools or the reception pools of the reprocessing plants. This storage could be used as well for storing the spent fuel destined for direct disposal if this way would be opened.
- In 1994, the Atomic Energy Act was appropriate modified in a way that in this law both options, either the reuse of the fissile material in the spent fuel or its direct disposal, is regarded in the same ranking. The amendment of the law opened only the possibility of following these two options, the decision to use one of these options or both rested solely with the operators of nuclear power stations. Preparative to this situation, two central storage facilities (Gorleben and Ahaus) were commissioned for dry storage of casks for spent fuel elements and in addition of casks for vitrified HAWC (high active waste concentration) repatriated from the reprocessing abroad.
- After change of the federal government in 1998, the new German Government decided, in accordance with the power utilities, to phase out nuclear power by limiting the standard lifetime of the nuclear power plants to 32 years from the date of their commissioning. With an agreement negotiated between the Federal Government and the power utilities of 14 June 2000 [BUN 00] (ratified on 11 June 2001), in spite of the prevailing differences of opinion on the use of nuclear power, the German power industry has demonstrated that it respects the Federal Government's decision to phase out the production of electricity from nuclear energy in a carefully coordinated process and to work towards implementation of the new energy policy. The key points of this agreement relevant to the policy and strategy on the storage of spent fuel are as follows:
    - Reprocessing will be discontinued and replaced by the direct disposal of spent fuel elements;
    - The delivery of spent fuel elements to La Hague and Sellafield for reprocessing will be terminated by the middle of 2005. Thanks to this move, and by setting up local interim storage facilities at the sites of the German nuclear power plants for the remaining spent fuel elements generated until the time of closing operation, the number of nuclear transports will be considerably reduced. In future, this number will be reduced by up to two-thirds by storing the spent fuel elements at interim storage facilities on the NPP site and only transporting them to the repository;
    - The licensing procedure for a pilot facility for the disposal conditioning of spent fuel elements currently under construction will be completed, but the use of the facility will be restricted to the repair of defective casks;
    - The exploration of the Gorleben salt dome (started in 1979) as a repository for heat generating wastes including spent fuel will be interrupted to allow sufficient time in order to clarify conceptual and safety-related issues during a moratorium period of up to 10 years. Until that date, the site will be preserved and made safe in its current state.
- As a result of this agreement an amendment of the German Atomic Energy Act was decided by the German Parliament (Bundestag) on 14 December 2001 and entered into force on 22 April 2002. This amendment of the Nuclear Atomic Energy Act has also consequences on the German policy and strategy on the storage and disposal of spent fuel and its implementation. The actual situation will be described in the following.

2. Overview of the actual situation

As of 30 June 2005 transports to reprocessing will be prohibited in accordance with the mentioned amendment to the Atomic Energy Act (AtG) of 22 April 2002. Until that date, the quantities of spent fuel elements contractually agreed with the reprocessing facilities should be shipped to these facilities by the nuclear power plant operators. For the rest of the spent fuel then existing in Germany and that generated in the remaining lifetime of the NPPs, only direct disposal will be possible.

As there is as yet no repository available for spent fuel elements, they will be stored intermediately at the site where they were used until such time as the repository is commissioned, in order to avoid the transportation of spent fuel and help spread the burden. If, in exceptional cases, interim storage at the site of the nuclear power plants is not possible, there are two central storage facilities available at Ahaus and Gorleben which are operational and on stand-by.

Usually, the spent fuel elements from research reactors will be returned to their country of origin for disposal. If that is not possible, these too will be intermediately stored until their final transportation to the repository.

The Federal Government is aiming to establish a repository in deep geological formations for the disposal of all kinds of waste including spent fuel elements by the year 2030. The moratorium period for Gorleben of 3 to 10 years does not mean that this site will be abandoned as a possible repository for spent nuclear fuel.

3. Interim storage facilities for spent fuel

The set up of interim storage capacity at the site of production of spent fuel results in a minimization of the number of transports. When a repository will be available, the spent fuel elements can be transported directly from the site of their production to the site of the final repository. With central storage facilities, additional transports from the power plant to the central storage facility and from there to the future repository site would be necessary. An overview of storage facilities for spent fuel (interim storage, conditioning) can be found in Table I. The storage facilities can be classified as follows:

- the dry interim storage facilities at the reactor sites, including temporary storage facilities (so-called *Interimslager*);
- the central interim storage facilities at Gorleben (TBL-G) and Ahaus (BZA);
- the interim storage facilities at Greifswald (ZAB, ZLN) for spent fuel from the nuclear power plants at Rheinsberg and Greifswald, and the storage facility at Jülich for spent fuel from the high-temperature reactor AVR.

The spent fuel elements unloaded from the reactor core are first placed in cooling ponds within the reactor building. These pools allow the required decay in activity and heat generation until the fuel is shipped for reprocessing or placed in a storage cask for interim storage, and provides the operator with sufficient flexibility to operate the plant. The additional wet storage facility outside the reactor building at Obrigheim is an exceptional case. As this facility, like the cooling ponds inside the reactor buildings, is considered part of the power plant operation from a licensing point of view, it will not be considered in any further detail. It is, however, included in the tables for the sake of completeness.

Table I. Storage facilities for spent fuel elements (March 2003)

Site	Storage capacity	Storage capacity	Status	
	(number of storage positions)	(t HM)	Applied for	Licensed
Onsite storage facilities				
Biblis	135 cask positions	1400 t HM	X	
Brokdorf	100 cask positions	1000 t HM	X	
Brunsbüttel	80 cask positions	300 t HM	X	
Grafenrheinfeld	88 cask positions	800 t HM		X
Grohnde	100 cask positions	1000 t HM		X
Gundremmingen	192 cask positions	2250 t HM	X	
Isar	152 cask positions	1500 t HM	X	
Krümmel	80 cask positions	800 t HM	X	
Lingen/Emsland	125 cask positions	1250 t HM		X
Neckarwestheim	151 cask positions	1600 t HM	X	
Obrigheim[a]	980 fuel element positions	286 t HM		X
Philippsburg	152 cask positions	1600 t HM	X	
Unterweser	80 cask positions	800 t HM	X	
Temporary storage facilities				
Biblis	28 cask positions	300 t HM		X
Brunsbüttel	18 cask positions	140 t HM	X	
Krümmel	12 cask positions	120 t HM	X	
Neckarwestheim	24 cask positions	250 t HM		X
Philippsburg	24 cask positions	260 t HM		X
Central storage facilities				
Gorleben	420 cask positions[b]	3800 t HM		X
Ahaus	420 cask positions	3960 t HM		X
Local storage facilities outside the reactor sites				
ZAB Greifswald	4680 fuel element positions	560 t HM		X
ZLN Greifswald	80 cask positions	585 t HM		X
Jülich	158 casks	0.225 t nuclear fuel[c]		X

[a] The storage facility at Obrigheim is a wet storage facility outside of the reactor building that was commissioned in 1999. Consequently, an additional onsite temporary storage facility is not necessary.
[b] Including the positions for HAW canisters
[c] Excluding thorium

With regard to direct disposal, a remaining period of several decades still needs to be bridged, depending on the availability of a repository and the length of time required for heat generation to subside until disposal. The Federal Government's concept envisages that in future, spent fuel elements should be placed in interim storage at the reactor sites where they are generated, and should remain there until duly conditioned and disposed of in a repository. For casks, an application has been submitted for a maximum storage period of 40 years from the date of loading. Interim storage at the site means that the number of fuel element transportations will be reduced.

For twelve reactor sites, licence applications for the interim storage of spent fuel elements have been submitted to the authorities in accordance with Section 6 of the Atomic Energy Act (AtG). The storage facilities are designed as dry storage facilities where with spent fuel elements loaded casks are placed in new storage buildings to be constructed on site. The casks are cooled by passive air convection, which removes the heat from the casks without any active technical systems. The leak-proof and accident-resistant casks ensure safe enclosure as well as the necessary degree of radiation shielding and criticality safety during both normal operation and in the case of incidents. The heat is released from the outer surfaces of the casks into the environment by means of cooling fins. Protection against external impacts, such earthquakes, explosions and aircraft crashes, is ensured by the thick walls of the casks. Basically, there are three design options for interim storage facilities:

- Storage building;
- Storage tunnel;
- Temporary storage facilities.

In the first option with the WTI concept and the STEAG concept two different concepts are available. In the WTI concept, the storage building consists of two parts being separated by a wall in the middle of the building. The wall thickness is approximately 70 to 85 cm, and the roof thickness approximately 55 cm. In the STEAG concept, the storage buildings have no separating wall. The wall thickness is approximately 120 cm, and the roof thickness approximately 130 cm. The individual storage facilities each have a capacity of between 80 and 192 storage positions for suitable storage casks. The design option storage tunnel at Neckarwestheim, also a WTI concept, is a special case, where it is envisaged that the casks will be stored in two tunnels lined with gunite. This special underground solution was developed to accommodate the specific situation of the site. The Stade power plant has withdrawn its application for the storage of spent fuel elements under Section 6 of the Atomic Energy Act (AtG) because the reactor is due to be shut down in 2003 and the spent fuel elements will be transported to reprocessing. In the Obrigheim plant, an increase of the wet storage capacity was licensed in 1998, and this capacity be sufficient until the end of the reactor's operational life. Provided all the licences are granted, all nuclear power plants currently in operation, with the exception of Stade and Obrigheim, will have dry interim storage facilities at their sites in the future. If the licensing and subsequent construction proceeds on schedule, all onsite storage facilities are expected to be commissioned by 2006.

As a transitional solution until the onsite storage facilities are complete, and in order to avoid any disposal shortfalls, five nuclear power plant operators have applied for temporary storage facilities (so-called *Interimslager*) under Section 6 of the Atomic Energy Act (AtG). These installations have a capacity of up to 28 storage positions with a mobile concrete enclosure for each cask. The intention is that the casks will be transferred to the respective onsite storage facility within a limited period of time. The casks in the temporary storage facilities are likewise cooled by passive air convection. The casks, combined with the concrete enclosures, ensure compliance with the admissible dose limits stipulated by the Radiation Protection Ordinance (StrlSchV). For three sites (Biblis, Neckarwestheim, Philippsburg), the licences for the temporary storage facilities have already been granted, and a number of casks have already been emplaced.

The interim storage facilities at Greifswald/Rubenow and Jülich should be considered as special cases. Although constructed outside of the reactor sites, they are nevertheless closely linked to certain nuclear reactors. The dry "Interim Storage Facility North" (ZLN) only accepts fuel elements from the Soviet-type reactors at Rheinsberg and Greifswald, some of which are currently being stored in a nearby wet storage facility (ZAB). The storage facility at Jülich contains spent fuel elements (spheres) from the prototype high-temperature reactor AVR.

Central storage facilities containing fuel elements from various German nuclear power plants have been licensed at Gorleben and Ahaus. The facilities are designed as dry storage facilities. Here, too, the types of casks are in part identical with those already mentioned above in conjunction with onsite storage facilities. The Ahaus facility is licensed for HTR and LWR fuel elements, whilst the Gorleben facility is licensed for LWR fuel elements and HAW canisters. A certain number of storage positions has been allocated to the respective utility companies, but with the exception of the storage of HAW, only a small number of these are to be used so as to avoid the need for transportation.

4. Regulatory framework for interim storage facilities

The basis for the licensing of decentralized and central interim storage facilities is § 6 Atomic Energy Act. The license for the storage pursuant to § 6 Atomic Energy Act must be granted if the licensing requirements are met. Such a license is a pure storage license. For the construction of the storage facility a license pursuant to planning and building law according to the respective federal state building regulations is required additionally. Both temporary and onsite storage facilities undergo an environmental impact assessment (EIA) when applied for after 15 March 1999. The course of the licensing procedure pursuant to § 6 Atomic Energy Act is shown in Fig. 1. It mainly consists of:

- Procedure for the involvement of the public;
- Verification of the licensing requirements;
- Issuing of the notice of approval.

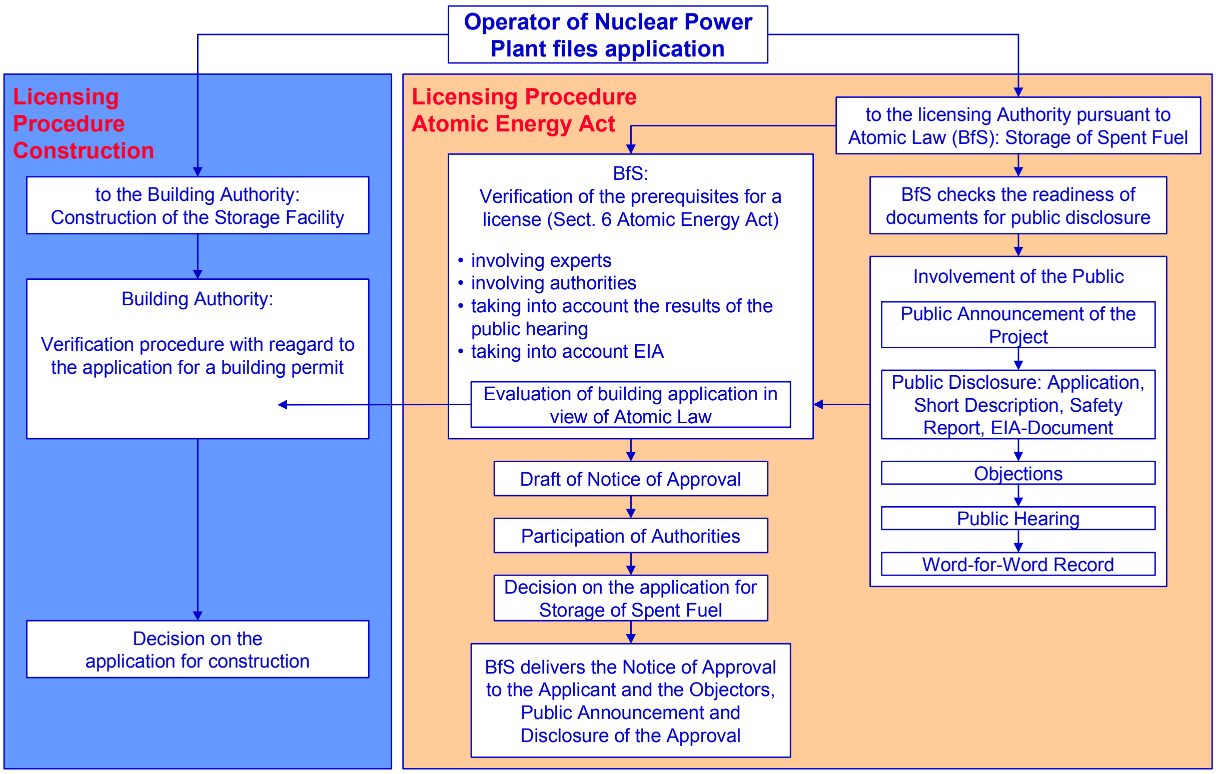

FIG. 1. Course of licensing procedure.

4.1. Responsibilities in interim storage

The utilities as producers of radioactive waste are responsible for the development, construction and operation of interim storage facilities. They decide and determine when, at which site and with which concept they file an application for the licensing of an interim storage facility to the competent authority.

The responsibility:
- for the licensing of nuclear fuel storage rests with the Federal Office for Radiation Protection (*BfS*);
- for the licensing of construction of a facility (storage building, storage tunnel, storage area and concrete encasements) rests with the respective federal state building authority;
- for the supervision during operation rests with the supervisory authority of the respective federal state.

4.2. Verification of the prerequisites for licensing

Pursuant to § 6 Atomic Energy Act a licence shall be granted if there is a need for such storage and if

1. there are no known facts giving rise to doubts as to the reliability of the applicant or of the persons responsible for the management and supervision of such storage, and the persons responsible for such management and supervision have the requisite qualification;
2. the necessary precautions have been taken in the light of the state of the art in science and technology to prevent damage resulting from the storage of nuclear fuel;
3. the necessary financial security has been provided for covering the legal liability to pay compensation for damage;
4. the necessary protection has been provided against disruptive action or other interference by third parties.

The "precaution against damages according to the state-of-the-art" is the most extensive complex to be verified. The following subjects have to be regarded, examined and verified:

- Safe enclosure of the radioactive inventory;
- Sufficient shielding;
- Subcriticality;
- Sufficient dissipation of decay heat.

Aircraft crashes and - following 11 September 2001 - attacks with a hijacked big passenger plane are examined as well. Before 11 September 2001, only the crash of smaller military jets with much less amounts of kerosene have been put into consideration.

4.3. Environmental impact assessment

The Act on the Assessment of Environmental Impacts (UVPG) and the EIA Amending Guideline 97/11/EC are the legal bases for the implementation of the environmental impact assessment. Possible effects of the project on man, animals, plants and their habitat as well as on soil, water, air and climate are assessed. Possible effects on the scenery and cultural assets as well as interactions are to be considered additionally.

As already mentioned an interim storage facility requires a building permit pursuant to the respective federal state planning and building regulations as well as a license to store nuclear fuel pursuant to § 6 Atomic Energy Act. According to the European guidelines a standardized environmental impact assessment is required in the licensing procedures. Since the duty to perform an environmental impact assessment results from the storage of nuclear fuel and not from the construction of a building, the Federal Office for Radiation Protection (BfS) acts in overall charge pursuant to § 14 of the Act on the Assessment of Environmental Impacts. This leads to a standardized consideration of environmental impacts in the various licensing procedures.

Since only one environmental impact assessment has to be performed for the project as a whole, and a unique decision (regarding the assessment of environmental impacts) has to be made, both licensing procedures pursuant to Atomic Law and to Planning and Building Law are linked and completed almost simultaneously.

4.4. Involvement of the public

The public is involved in the performance of licensing procedures for interim storage of spent fuel on the basis of the Ordinance on Procedures under Atomic Law (AtVfV). The following steps are required:

- Announcement of the project in the Federal Gazette (Bundesanzeiger) and in local daily papers;
- Disclosure for public inspection: Certain documents concerning the project (application, short description, safety report and the EIA-report) have to be disclosed for public inspection for a period of two months. During the period of disclosure for public inspection objections against the project can be raised;
- Public hearing: The licensing authority must discuss the objections with the applicant and the objectors. The objectors have the opportunity to extend their objections and to explain them in greater detail. The discussion is not public.

The objections expressed and reinforced in the public hearing are considered by the Federal Office for Radiation Protection (BfS) and taken into account for the examination of the prerequisites and the decision expressed in the notice of approval.

4.5. Involvement of the Republic of Austria

A public participation for citizens of the Republic of Austria is carried out for the six interim storage facilities located in the south of Germany with regard to transboundary environmental impact assessment.

5. Conditioning of spent fuel for its disposal in a repository

The German reference concept for direct disposal envisages that the spent fuel elements should be packaged in sealed thick-walled casks and emplaced in deep geological formations. In order to demonstrate the conditioning technique, a pilot conditioning plant (PKA) has been planned and constructed at Gorleben. Pursuant to the agreement between the Federal Government and the utilities the licensing procedure is complete, but use of the facility is restricted to the repair of defect casks. It is restricted to a maximum throughput of 35 tHM/a and the handling of other radioactive materials. At present, the granted licence cannot be utilised because a number of actions have been brought against it.

6. Disposal in a repository

In the Federal Republic of Germany, the intention is that all types of radioactive waste should be stored in deep geological formations. The intention to dispose of all types of radioactive waste in deep geological formations also makes it unnecessary to differentiate between waste containing radionuclides with comparatively short half-lives and waste containing radionuclides with comparatively long half-lives. In accordance with the German approach to disposal, the definition and categorisation of radioactive waste (i.e. its classification) must therefore comply with the requirements for safety assessment of an underground repository. In this respect, the effects of heat generation from radioactive waste on the design and evaluation of a repository system are particularly important, since the natural temperature conditions may be significantly altered by the deposited waste. In order to meet the requirements concerning the registration and categorisation of radioactive waste from the point of view of disposal, the authorities have chosen a basic subdivision into:

- Heat-generating radioactive waste; and
- Radioactive waste with negligible heat generation.

Examples of heat-generating radioactive wastes include the fission product concentrate, shells, structural components and feed sludge from the reprocessing of spent fuel elements, and the fuel elements themselves if there are no plans to reprocess them but instead to dispose of them directly as radioactive waste.

Since 1979, the Gorleben salt dome has been explored for the disposal of heat-generating radioactive waste. In the opinion of the Federal Government, there are doubts with regard to its suitability as a final repository. In February 1999, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) established the Committee on a Site Selection Procedure for Repository Sites (*Arbeitskreis Auswahlverfahren Endlagerstandorte AkEnd)* (hereinafter referred to as “the Committee”) in order to lay down criteria for the identification of sites that are both suitable for safe disposal and at the same time accepted by the general public. Until planning and safety-related issues have been clarified, the exploration of the Gorleben salt dome has been interrupted in October 2000 for at least 3 years, at most however 10 years.

The Committee’s recommendations serve to provide the Federal Government with a proposal on how the repository site selection procedure should be structured in accordance with the coalition agreement of 2002, to enable the Federal Government to meet its responsibility to establish facilities for the disposal of radioactive waste in accordance with Section 9a, para. 3 of the AtG.

7. Summary

In summary, it may be stated that the German policy and strategy on the storage and disposal of spent fuel has changed over the last years and now encompasses the following key elements:

- end of transports of German spent fuel elements to reprocessing facilities in 2005;
- licensing and construction of interim storage facilities for spent fuel elements, until a repository becomes available in order to minimise transports;
- disposal of fuel elements in a deep geological repository.

Spent fuel management strategy in Japan

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Abstract. Japan has consistently based its policy on its nuclear fuel cycle option with only peaceful purposes, in order to ensure a stable and reliable energy supply and to minimize environmental loads. Since the start of operation of the first commercial nuclear power plant in 1966, nuclear power generation in Japan has steadily increased. At present, 52 commercial nuclear power plants are in operation. Nearly 1 000 tU of spent fuel is discharged annually by these plants. JNFL is constructing the RRP, planned to be commissioned in July 2005. The reprocessing capacity is 800 tU/year. Therefore, more than 200 tU of surplus spent fuel will accumulate every year. Building up the interim storage capability outside of nuclear plant sites is now a focal point for the flexible, practical, and safe management of the surplus amount of spent fuel predicted to arise over the next several decades. In February 1997, a Cabinet decision on "Policy to Promote Nuclear Fuel Cycle" was made and the need to build more AFR interim storage capacity was acknowledged. A law concerning interim storage was enacted in 1999. Thus, the spent fuel storage business has legally become possible in Japan. Since then, the utilities have been endeavouring to site ARF storage facilities, with the goal of commencing operation around 2010. TEPCO is now preparing for the first AFR in Japan, Mutsu Recycle Fuel Storage Center, located at Sekinehama in Mutsu City. In April 2003, TEPCO released a report on the site feasibility investigation, which verified the suitability of the site for the AFR storage facility and also announced the framework of this project.

1. Fundamental policy for nuclear energy in Japan

Japan has consistently based its policy on its nuclear fuel cycle option with only peaceful purposes, from the beginning of nuclear development in the 1950s [1]. Japan has been pursuing the closed cycle to ensure a stable and reliable energy supply and to minimize environmental loads. Our policy has been in compliance with the IAEA full-scope safeguard.

Figure 1 shows the nuclear fuel cycle in Japan in the near future. The completion of the nuclear fuel cycle has been desired in Japan because the country is poor in energy resources, with a national energy self-sufficiency ratio of only 4%. Three commercial facilities - a reprocessing plant, a MOX fabrication plant, and an interim storage facility - are scheduled to operate by 2005, 2009 and 2010 respectively, in order to make full use of uranium resources. Interim storage of spent fuel provides an adjustable time period until the spent fuel is reprocessed and thus enables the nuclear fuel cycle to flow flexibly. A law concerning interim storage was enacted in 1999 and the private sector is now making preparations for commercial operation of the storage facility by 2010.

2. Current status of spent fuel management in Japan

Since the start of operation of the first commercial nuclear power plant in 1966, nuclear power generation in Japan has steadily increased. At present, 52 commercial nuclear power plants are in operation. Their total installed capacity is 46 GW, about one third of the total generated electric power. Nearly 1 000 tU of spent fuel is discharged annually by these plants.

The Tokai Reprocessing Plant (TRP) for research and development had reprocessed about 1 000 tU of spent fuel from nuclear power plants by the end of December 2002. A total of 7 100 tU of spent fuel has been shipped to reprocessing plants in France and the United Kingdom.

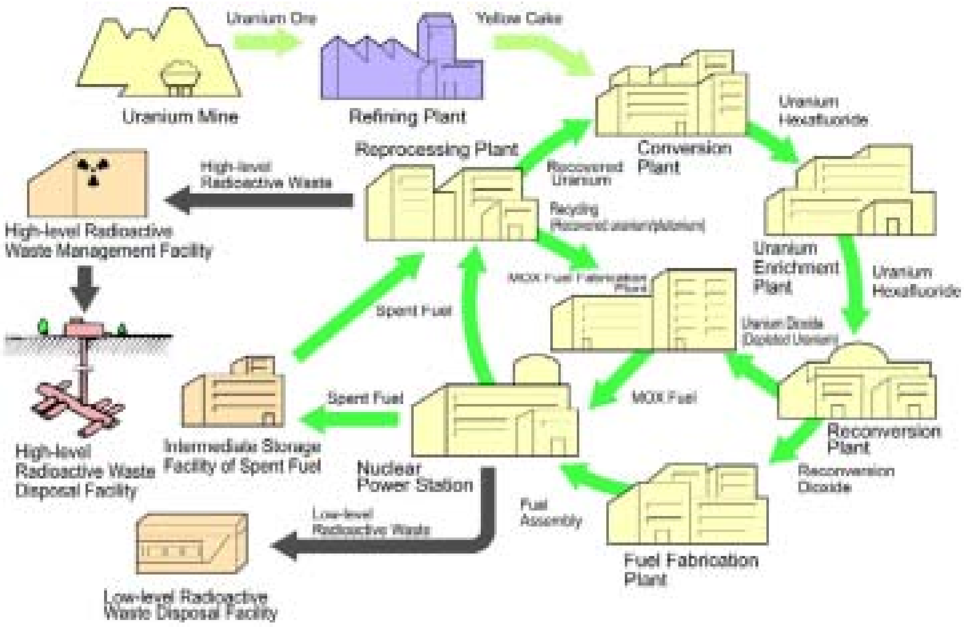

FIG. 1. Nuclear fuel cycle in Japan in the near future.

Japan Nuclear Fuel Limited (JNFL) is constructing the Rokkasho Reprocessing Plant (RRP), planned to be commissioned in July 2005. The reprocessing capacity is 800 tU/year. As of February 2003, 93%of the facility has been completed. The cold chemical check test is currently in progress.

3. Spent fuel balance and demand for interim storage

A total of 18 800 tU in spent fuel from nuclear power plants in Japan had been discharged by the end of September 2002. About 53% (10 000 tU) of the spent fuel is stored in nuclear power plants, while 38% (7 100 tU) has been shipped to European reprocessing plants. The rest (9%) has been shipped to TRP and RRP.

Now let's look at the spent fuel balance in Japan in the future. The total storage capacity in nuclear reactor sites is 16 000 tU. However, 10 000 tU of spent fuel has already been stored. Therefore, the remaining storage capacity is 6 000 tU. The total storage capacity in RRP is 3 000 tU. Of this, the storage capacity remaining is about 2 000 tU, because 780 tU of spent fuel has already been stored. About 1 000 tU of spent fuel is discharged annually at present. The discharge rate is scheduled to increase, as shown in Figure 2 [2]. Meanwhile, the reprocessing capacity of RRP is 800 tU/year, as described above. Therefore, more than 200 tU of surplus spent fuel will accumulate every year. So measures will be needed to cope with the increasing accumulation of surplus spent fuel. The following are measures that have been, or will be, introduced to cope with the increasing spent fuel accumulation:

(1) Burnup extension;
(2) Increase of reactor-site storage capacity by:
  - Re-racking;
  - Common use of pools (within the same site);
  - Expansion of pools;
  - Installation of on-site dry storage facilities;
(3) Construction of away-from-reactor (AFR) interim storage facilities.

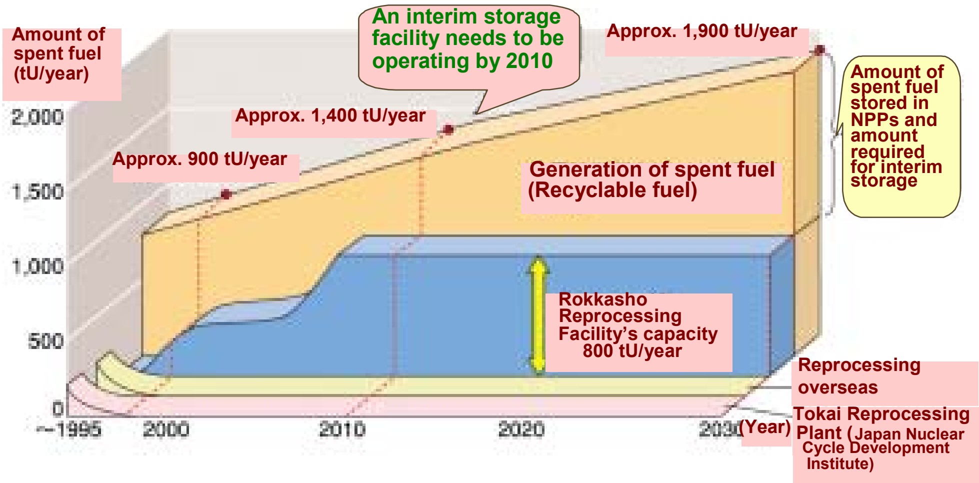

Fig. 2. Projected spent fuel balance [2].

Fuel burnup is now being extended to achieve a maximum burn-up of as high as 55 000 MW·d/t in most BWRs and PWRs, resulting in a degree of suppression of the spent fuel discharge rate. In many of the nuclear power stations, re-racking and common use of pools have already been implemented, and in some of them, the pools have been expanded. At present, dry cask storage facilities are in operation at only two sites: Fukushima Dai-ichi of Tokyo Electric Power Company (TEPCO) and Tokai Dai-ni of Japan Atomic Power Company (JAPCO). Moreover, the discharge rate is projected to increase as more nuclear power plants are expected to come into operation in the near future. Under such circumstances, thoroughly devised management of the back end of the fuel cycle constitutes an essential part of the nation's nuclear energy program, especially from the viewpoint of assuring long term stability in the electricity supply [3].

Building up the interim storage capability outside of nuclear plant sites is now a focal point for the flexible, practical, and safe management of the surplus amount of spent fuel predicted to arise over the next several decades.

4. Preparation of the fundamental regulatory environment for AFR storage

In February 1997, a Cabinet decision on "Policy to Promote Nuclear Fuel Cycle" was made and the need to build more away-from-reactor (AFR) interim storage capacity was acknowledged. In June 1998, an advisory subcommittee of MITI (now METI, or the Ministry of Economy, Trade and Industry) issued an interim report, "Towards Realization of Storage of Recycle Fuel Resources" [4]. The report states: "the spent fuel (under long term storage) should be regarded as a nuclear fuel reserve for future recycling." The report also emphasized the necessity for the establishment of AFR interim storage facilities by 2010 through the coordinated efforts of the utilities and the government. The Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors was partially amended to cope with the need for AFR facilities in June 1999. Thus, the spent fuel storage business has legally become possible in Japan. Since then, the utilities have been endeavouring to site ARF storage facilities, with the goal of commencing operation around 2010. The Nuclear Safety Commission (NSC) finalized the safety review guideline for interim storage facilities based on dry metal casks in October 2002.

5. Outline of the regulations for AFR storage

The regulations for AFR storage were enacted for the dry metal cask storage system as a priority matter. The regulation policy requires that the safety of spent fuel be maintained throughout the storing and that the regulations should be based on the safety inspections that have already been carried out in dry cask storage facilities. The regulations concerning storage operators require that storage operators intending to conduct spent fuel storage are obliged to obtain a license, as power plant operators are. The Atomic Energy Commission and the Nuclear Safety Commission will make a double-check on applicants for this license. The licensing standards regulate the following, among other matters:

- Not to be used for purposes other than peaceful ones;
- Technical capability and financial basis are required.

The safety review guideline for AFR using dry metal casks for both transportation and storage was established in October 2002 under the law enacted in 1999 [5]. The safety of the facility is secured by the following four basic safety functions according to the safety review guideline:

1. Confinement;
2. Radiation shielding;
3. Heat removal; and
4. Prevention of criticality.

The functions of confinement and prevention of criticality are assured by dry metal casks. The safety review guideline also requires that the metal casks must have sufficient structural strength to meet the regulation requirement for transportation outside nuclear facilities.

6. Mutsu Recycle Fuel Storage Center

TEPCO is now preparing for the first AFR in Japan, Mutsu Recycle Fuel Storage Center, located at Sekinehama in Mutsu City, Aomori Prefecture in the northern part of Honshu Island, Japan. Figure 3 shows an imaginary picture of the Center.

FIG. 3. An imaginary picture of Mutsu Recycle Fuel Storage Center.

Mutsu City expressed interest in hosting an AFR storage facility at Sekinehama area and requested TEPCO to initiate a feasibility investigation in November 2000. TEPCO conducted a site feasibility investigation for two years starting from 1 April 2002.

In April 2003, TEPCO released a report on the site feasibility investigation, which verified the suitability of the site for the AFR storage facility and also announced the framework of this project as shown in Table I.

Table I. Framework of Mutsu Recycle Fuel Storage Center

Implementing entity	New company established with TEPCO & other electric power companies
Beginning of storage	Around 2010
Storage system	Dry metal cask storage system
Storage capacity	Initially 3 000 tU Finally 5 000 to 6 000 tU
Storage period	50 years

7. Summary

Japan has consistently based its nuclear policy on the nuclear fuel cycle with only peaceful purposes. As of February 2003, 93% of the Rokkasho Reprocessing Plant has been completed. The fundamental regulatory environment for AFR storage has been prepared. TEPCO conducted a site feasibility investigation for Mutsu Recycle Fuel Storage Center and announced the framework of this project.

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Canada's national policy on the long term management of nuclear fuel waste

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Abstract. Nuclear energy is an important part of Canada's diversified energy mix. There are 22 CANDU reactors in Canada located in Ontario, New Brunswick, and Quebec. Like any other industry, nuclear fuel cycle operations produce some waste, and for this paper, we will focus on nuclear fuel waste, i.e., the irradiated fuel taken out of nuclear reactors at the end of their useful life. Canada has no plans to reprocess and recycle this fuel, so current plans are based on direct long term management of the waste fuel. Although nuclear fuel waste is currently in safe storage, steps are now underway to develop and proceed effectively with the implementation of more long term management solutions. A deep geological disposal concept was developed by the federal crown corporation Atomic Energy of Canada Limited (AECL) and Ontario Hydro, and, in October 1988, it was referred for review by a federal independent environmental assessment panel. AECL submitted the Environmental Impact Statement to the Panel in 1994. The Panel released its report with conclusions and recommendations on the acceptability of the concept in March 1998. It found that "from a technical perspective, safety of the AECL concept has been on balance adequately demonstrated for a conceptual stage of development, but from a social perspective, it is not. As it stands, the AECL concept for deep geological disposal has not been demonstrated to have broad public support. The concept in its current form does not have the required level of acceptability to be adopted as Canada's approach for managing nuclear fuel waste". Thus it was clear that Canada should increase public confidence before proceeding with any general approach on the long term management. With the Panel's recommendations in mind, and with further consultations with stakeholders, including the public, the Government of Canada developed the Nuclear Fuel Waste Act (NFW), which came into force on 15 November 2002. The NFW Act is a stand-alone piece of legislation with some 30 articles and without regulations. The Act deals essentially with social, financial and socio-economic considerations of the long term management of nuclear fuel waste. It complements the health, environment, safety and security requirements under the Nuclear Safety and Control Act. It provides for 1) the nuclear industry to set up a waste management organization to manage the full long term waste management activities and to establish trust funds to finance long term waste management responsibilities; and 2) the waste management organisation to submit for government decision long term waste management options within three years of coming into force of the NFW Act. On 24 October 2002, the nuclear industry formed the Nuclear Waste Management Organization (NWMO), which is now proceeding with preparing the options study. On 25 November 2002, the nuclear industry deposited initial amounts into the required trust funds. On 28 March 2003, the NWMO submitted to Government its first annual report.

1. Canadian context

Nuclear energy is an important part of Canada's diversified energy mix. The nuclear fuel cycle in Canada includes the mining and milling of uranium, the fabrication of fuel elements, the use of that fuel in nuclear power plants, and the safe management of the radioactive waste by-products. The development and control of the nuclear energy option falls within federal jurisdiction and the Government of Canada wants to ensure that the public has confidence that operations at each step of the nuclear fuel cycle are carried out in the best interest of Canadians.

Canada is rich in uranium deposits mainly in the northern part of the province of Saskatchewan. Uranium mined in Canada is used to fuel nuclear reactors around the world, including the CANDU (CANada Deuterium Uranium) reactor. Canada is the world's leading producer of uranium, accounting for roughly one-third of total global output. The mining and milling of uranium generated $500 million in revenues in 1998 and provided employment for over 1 000 Canadians.

There are 22 CANDU reactors in Canada located in Ontario (20), New Brunswick (1) and Quebec (1). Canada also has a successful CANDU export industry led by Atomic Energy of Canada Limited, a federal crown corporation. The nuclear industry contributes to the Canadian economy by generating thousands of jobs in the uranium industry, in the three provincial nuclear utilities and in approximately 150 Canadian manufacturing, supply and services companies.

Like any other industry, nuclear fuel cycle operations produce some waste, and more characteristically, radioactive waste. Concerns about radioactive waste increased sharply during the 1970s, as did awareness of the environment of nuclear safety issues, and of the hazards of radiation in particular. In the 1970s and 1980s, the Government recognized that the management of these wastes to currently acceptable standards would involve considerable scientific and engineering preparation, and extensive public review processes.

The Government also realized that past methods of dealing with radioactive wastes had not been adequate in some cases by current environmental standards. Work was needed to resolve historic waste situations and to ensure that future wastes would be adequately looked after during their hazardous lifetime. It was also necessary to assign responsibility for both past and future wastes, so that requirements could be established, preparations made and funding set aside.

In general, federal policy is now to manage these wastes so that the health of people and the environment is protected, as well as so that those who benefit from the wastes bear the costs of long term management. The owners are responsible for establishing the waste management organization, funding and carrying out acceptable waste management plans. However, where no owner can be identified, or held responsible, the federal government recognizes its residual responsibility.

Although Canada's radioactive wastes are currently in safe storage, steps are now underway to develop more permanent long term solutions for all types of radioactive waste, namely, uranium mine and mill tailings, low-level radioactive waste and nuclear fuel waste.

2. The government of Canada radioactive waste policy framework

By the mid-1990s, progress was underway on a range of initiatives leading towards more permanent solutions for the long term management of Canada's radioactive wastes. Most of those initiatives involved public or governmental processes, which tended to be somewhat protracted. While there was progress on cleaning up some sites and putting the wastes in storage, there was not much actual progress toward permanent solutions. In a number of cases, it was still not clear who would actually pay for and carry out long term management of radioactive waste and how the needed institutional structures would be organized.

Anticipating the conclusion of the various processes, and the need to move ahead with long term solutions, the Auditor General of Canada in May 1995 concluded that Canada should get on with the implementation of long term radioactive waste management, and ensure that appropriate funding arrangements be put into place. It noted that Natural Resources Canada (NRCan) should also reach an agreement with major stakeholders on their respective roles and responsibilities.

NRCan embarked on a series of discussions with the major stakeholders, notably the owners and producers of the wastes and the concerned federal and provincial departments and agencies. Some basic principles for radioactive waste management, and some proposed assignments of responsibility were part of the basis for discussion. There was general agreement on the principles and on the roles and responsibilities, and in 1996, the Government of Canada announced the Radioactive Waste Policy Framework clearly indicating that:

- The federal government will ensure that radioactive waste disposal is carried out in a safe, environmentally sound, comprehensive, cost-effective and integrated manner;
- The federal government has the responsibility to develop policy, to regulate, and to oversee producers and owners to ensure that they comply with legal requirements and meet their funding and operational responsibilities in accordance with approved waste disposal plans;
- The waste owners and producers are responsible, in accordance with the principle of "polluter pays", for the funding, organization, management and operation of disposal and other facilities required for their wastes. This recognises that arrangements may be different for nuclear fuel waste, low-level radioactive waste and uranium mine and mill tailings.

The Policy Framework is consistent with Canada's traditional nuclear regulatory practice in putting the onus of safe operations on the licensees or owners.

3. Nuclear fuel waste

In Canada, nuclear fuel waste is the irradiated fuel taken out of nuclear reactors at the end of their useful life. There are no plans to reprocess and recycle this fuel, so current plans are based on direct long term management of the used fuel.

The nuclear fuel waste refers to the irradiated fuel bundles that come from the twenty reactors owned by Ontario Power Generation Inc (OPG), and the other two owned by Hydro-Québec and New Brunswick Power. In addition, Atomic Energy of Canada Limited (AECL), a federal Crown corporation, produces a small amount of waste from its prototype and research reactors. OPG produces about 90 per cent of the total amount of waste, the other two nuclear utilities about 4 per cent each, and AECL 2 per cent. Other waste owners, e.g., universities, produce a much smaller quantity of nuclear fuel waste.

In total, about 1 million bundles of nuclear fuel waste are currently in safe storage at the reactor sites, where it can be kept for decades, in pools or in dry concrete canisters. Canada's entire nuclear power programme produces about 60 000 bundles i.e. less than 2 000 tonnes of nuclear fuel waste, annually.

As required by the federal government, a deep geological disposal concept for nuclear fuel waste was developed by AECL and Ontario Hydro, and, in October 1988, was referred for review by a federal independent Environmental Assessment Panel. Guidelines for the Environmental Impact Statement (EIS) were published in 1992, and the EIS was submitted by AECL in 1994. The Seaborn Panel, named after its Chairman, Blair Seaborn, released its report with conclusions and recommendations on the acceptability of the concept in March 1998. The concept was found to be on balance acceptable technically but not socially, and Panel proposed next steps to remedy the situation.

The 1998 Government of Canada response to the Seaborn Panel recommendations conformed with the 1996 Policy Framework for Radioactive Waste and set the stage for developing institutional and financial arrangements to implement the long term management waste management. The challenge was to ensure that the public would be confident that the next steps for the long term management of nuclear fuel waste would be carried out in the best interest of Canadians. An important part of the answer to this challenge was the development of the Nuclear Fuel Waste Act which received Royal Assent on 13 June 2002, and which entered into force on 15 November 2002.

4. The nuclear fuel waste act (NFW)

The *NFW Act* is a stand-alone piece of legislation with some 30 articles and without regulations. The Act deals essentially with financial and socio-economic considerations of the long term management of nuclear fuel waste and complements the health, environment, safety and security requirements under the *Nuclear Safety and Control Act.* It provides the legal framework to ensure that:

- nuclear energy corporations set up a waste management organization as a separate legal entity to manage the full range of long term waste management activities;
- nuclear energy corporations establish trust funds with an independent third party trust company to finance long term waste management responsibilities;
- the waste management organization submit long term waste management options to Government; and,
- the Governor in Council select a long term management option from those proposed by the waste management organization.

Under the *NFW Act,* waste owners will establish a waste management organization, incorporated as a separate legal entity, with a mandate to manage and coordinate the full range of activities relating to the long term management of nuclear fuel waste. The waste management organization will:

- within three years of coming into force of the Act (i.e. 15 November 2005), submit a Study to the Government which includes:
  - practicable long term management options for Canada, including the following: a modified concept for deep geological disposal; storage at reactor sites; and centralized storage, either above or below ground;
  - a comparison of risks, costs and benefits of the options; these options would need to be analysed within the context of proposed siting areas;
  - an analysis of ethical and social considerations;
  - an Aboriginal consultation plan;
  - a comprehensive public participation plan.
- have an Advisory Council reflecting a broad range of scientific, technical, social sciences disciplines as well as representatives from affected populations;
- implement the government-approved long term waste management approach using monies set aside in the trust funds.

On 24 October 2002, the nuclear industry announced that, as required under the *NFW Act*, it had formed the Nuclear Waste Management Organization (NWMO). The NWMO is a new organization created to recommend a long term approach for managing used nuclear fuel produced by Canada’s electricity generators. “It is engaging stakeholders, the best experts in the world and interested Canadians to develop a solution that safeguards the public in a way that is sustainable, ethically and socially acceptable, and respectful of the environment now and in the future”.

On 28 March 2003, the NWMO submitted its 2002 Annual Report to the Honourable Herb Dhaliwal, Minister of Natural Resources for Canada, *"From Dialogue to Decision: Managing Canada's Nuclear Fuel Waste,"* which is the first Annual Report prepared by NWMO, covering the organization's first three months of activity since its establishment in fall 2002.

One of the first accomplishments of the NWMO was the establishment, by the Board, of an Advisory Council responsible for reviewing and providing written comments on the NWMO study. Another early activity was the commissioning of public opinion research to benchmark Canadian perspectives about nuclear waste. NWMO also gave priority to the design of an interactive website to support the consultation process it will undertake over the next three years. The NWMO Annual Report, which includes the organization's financial statements for 2002, was made public simultaneous with its submission to the Minister of Natural Resources. This report together with other information on the NWMO can be viewed at www.NWMO.ca.

5. Federal oversight

The Government identified three key policy objectives for federal oversight:

- to have waste owners establish a segregated fund for fully financing the long term management of nuclear fuel waste;
- to establish a reporting relationship between the federal government and the Waste Management Organization for reviewing progress on a regular basis;
- to establish a federal review and approval process including access to the segregated fund.

The federal Minister of Natural Resources is designated under the *NFW Act* as the minister responsible for its administration. The nuclear fuel waste bureau, a group within the department of Natural Resources Canada, is charged with ensuring that ministerial responsibilities are carried out appropriately. The *NFW Act* and other information related to the long term management of nuclear fuel waste can be found on the Bureau's web site at www.nfwbureau.gc.ca.

One of the important aspects of this control and monitoring authority is the requirement that, pursuant to the *Act*, the NWMO must take into account the social concerns created by the project over its entire implementation period. Even before the Canadian government chooses among the proposed management solutions, the NWMO is compelled to take into consideration "the ethical, social and economic considerations associated with that approach" and "avoiding or minimizing significant socio-economic effects on a community's was of life or on its social, cultural or economic aspirations". In its triennial reports, the NWMO must then assess serious socio-economic effects. The wording of the *Act* states that the social effects of the project are a constant concern for the government - and, through the government, for Canadian society - and one of the key components of its proper implementation.

In this regard, the *Act* is innovative in that it compels, on the one hand, the NWMO to determine the social and ethical consequences of each management proposal even before the beginning of the environmental effects assessments resulting from the implementation of the *Canadian Environmental Assessment Act* and, on the other hand, the Minister of Natural Resources to take responsibility for the quality of the work done. Of note, the ethical dimension is an important one, and for the first time in Canadian legislation in the impacts field an act recognizes it officially.

The entrenchment in the *Act* of such a social provision meets the concerns expressed by the Nuclear Fuel Waste Management and Disposal Concept Environmental Assessment Panel in its report (Seaborn Report). The "ethical and social assessment framework" that the Panel had hoped for is in a way drafted into the obligations given to the Minister and to the NWMO in their controller-controllee relationship.

6. Social and ethical concerns

The *NFW Act* incorporates, at the legislative level, requirements which establish a process for due effort in addressing social impacts. What are the concerns and interests of an affected population? How to mitigate effectively these concerns? How to respond to public interests? These issues will be addressed on the same footing as technical matters both throughout the development and also the implementation of a solution for the long term management of nuclear fuel waste. Addressing these concerns is front and central in the *Act*.

In addition, the *NFW Act* specifically requires "taking into account ethical considerations" for each of the options proposed by the NWMO. Because of the long term nature of the management of nuclear fuel waste, one of the main "ethical" issues raised so far by the public during the Seaborn Panel public hearings and in international fora, such as the Nuclear Energy Agency (NEA), is that of generational fairness, both intragenerational (i.e., this current generation's interests versus those of the next) and intergenerational (i.e., interests among different groups of the same generation). The outcome of this debate will have an important impact on technical decisions, for example, whether to store or dispose of the waste, or whether to permanently isolate the waste or store it in a 'retrievable' form.

Other ethical aspects such as the protection of humans versus the protection of other species, the extent of protection required in the face of potential acts of terrorism will need to be addressed. In addition, equity aspects such as individual interests versus society's interests; burden to existing generation versus future ones; impact on the host communities versus impact on more distant ones; need to be examined, in the context of the societal principles of the day such as the precautionary principle. These considerations will influence the decision making process regarding storage or disposal.

7. Conclusion

The *Nuclear Fuel Waste Act* implements a key component of the Government of Canada's 1996 *Policy Framework for Radioactive Waste* - that the federal government, through effective oversight, would ensure that the long term management of radioactive waste is carried out in a comprehensive, integrated and economically sound manner. The key elements of the *Act* include:

a) requiring the major owners of nuclear fuel waste to establish a waste management organization to carry out the managerial, financial and operational activities to implement the long term management of nuclear fuel waste;

b) requiring the major owners of nuclear fuel waste to establish trust funds and to make annual payments into those trust funds to finance the long term management of nuclear fuel waste; and

c) authorizing the Governor in Council to make a decision on the choice of approach for long term management of nuclear fuel waste for Canada to be implemented by the WMO.

The *Act* also requires that the waste management organization carry out public consultations, that it's Study and reports (which are submitted to the Minister) be made public, that they establish an Advisory Council, whose comments on the Study and reports are to be made public, and that the Minister make public statements on all of the waste management organization's reports.

By 15 November 2005, the Nuclear Waste Management Organization (NWMO) will submit its proposed options for long term management of nuclear fuel waste to the federal government with social and ethical considerations as well as technical considerations. The federal government will then choose from the proposed options, including: deep geological disposal in the Canadian Shield; centralized storage, either above or below ground; and surface storage at nuclear reactor sites.

With that decision, 3 years from now, the long term strategy for nuclear fuel waste management in Canada will be made clear. What will also be made clear is that the need and requirement for nuclear fuel waste storage will be delineated within the context of that strategy.

Some aspects of the Russian nuclear fuel cycle development

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Abstract. The paper summarizes the status of spent fuel generated in the Russian Federation. It describes the spent fuel management infrastructure, current legislation and projects of the industry. The role of enterprises affiliated to Minatom's Department of Nuclear Fuel Cycle is illustrated.

1. Management of spent nuclear fuel in Russia

The existing scheme for handling of different spent nuclear fuels (SNFs) from power reactors, research and naval reactors in the Russian Federation, provide safe storage of irradiated fuels and radiochemical reprocessing of some part of them. The report illustrates the role of enterprises affiliated to Minatom's Department of Nuclear Fuel Cycle:

- The RT-1 complex of the "Mayak" Production Association carries out radiochemical reprocessing of SNF from VVER-440 reactors in Russia and Ukraine, BN-600 reactor, SNF from research reactors and nuclear power plants of sea vessels;
- The Mining and Chemical Complex carries out centralized intermediate storage of SNF from VVER-1000 reactors in Russia and Ukraine; some SNF from VVER-1000 and RMBK-1000 is currently stored in water medium in on-site pools.

Therefore, we could state that the Russian Federation has actually implemented two nuclear fuel cycles for different types of reactors:

- Closed nuclear fuel cycle for SNF from VVER-440, BN-600, research and naval reactors – the cycle includes radiochemical reprocessing of fuel and partial use of the recovered products (uranium, plutonium and other nuclides) and different technologies for waste treatment and storage;
- Deferred decision for SNF from VVER-1000, RBMK-1000 and some other activities.

The annual volume of SNF unloaded from a single NPP unit and cumulative volume of SNF generated by Russian NPPs are shown in Table I. Annual volume of SNF generated by Russian NPPs is relatively small (below 10% of the world SNF discharge). The existing capacity of AR and AFR storages for VVER-1000 spent fuel will be sufficient for up to 8 years, and the capacity of the storages for RBMK-1000 SNF will be exhausted in 3 to 5 years depending on site.

Most part of fuels from research reactors and critical test facilities is in store at Russian nuclear centres sites, e.g. Research Institute of Atomic Reactors (RIAR), Institute of Physics and Power Energy (IPPE) and Kurchatov Institute (KI).

Table I. SNF generated in the Russian Federation

Reactor type	Units in operation	Total generation of SNF t HM/year	Estimated SNF storage t HM/year[a]
BN-600	1	16	Under reprocessing
VVER-440	6	70	Under reprocessing
VVER-1000	8	170	~ 6 500
RBMK-1000	11	450	~ 10 000

[a] early 2003

It is worthy to note that Minatom has created a reliable industrial framework for cost-effective and environmentally safe implementation of the SNF management activities. All Minatom's enterprises involved in back-end nuclear cycle are licensed by the supervisory body – the State Committee on Surveillance of Nuclear Safety (Gosatomnadzor) in compliance with the Russian legislation.

"The Strategy of Nuclear Power Development in Russia in the first half of the 21st century" approved by the RF Government in May 2001 and the "Concept for the nuclear sector advancement up to the year 2010" stipulate development of enterprises for SNF management on site and at fuel cycle facilities, a gradual transition to completely closed cycle is envisaged in terms of recycled uranium and civil plutonium providing minor transmutation of minor actinides and long-lived fission products in fast neutron reactors. Minatom carries out preparatory work along various trends of promoting services and infrastructure related to nuclear fuel cycle activities.

2. Priority targets for improving the SNF management system at Minatom's enterprises

Since its creation in the 1970—1980s the Russian industrial infrastructure for SNF transportation, storage and reprocessing was oriented on fresh and spent fuel management for USSR (Russia) designed reactors (VVER-440, VVER-1000 and research reactors). Entering the world market of storage and reprocessing of SNF from foreign NPPs will require additional investments for development and improvement of industrial infrastructure of Russian nuclear cycle back-end.

The Minatom's concept of management of SNF in Russia stipulates:

- Modernization of the existing RT-1 complex at the "Mayak" Production Association and creation of a state-of-art scheme for management of all types of radioactive waste;
- Increasing the capacity of the existing wet storage facility for VVER-1000 type fuel at the Mining and Chemical Complex from 6 000 to 8 400 tHM;
- Construction at the Mining and Chemical Complex of a vault-type dry long term store for SNF from VVER-1000 and RBMK-1000 reactors with the design capacity of 33 000 t HM and manufacture of the necessary means for SNF transportation;
- In the longer-term perspective – construction of RT-2 reprocessing complex at the MCC for reprocessing of SNF from VVER-1000 with a set of installations for conditioning and disposal of radioactive waste.

The first stage of implementing the concept of a closed fuel cycle (up to 2010) is related to modernization of the existing RT-1 complex, expanding the capacity of "wet" repository of SNF at the RT-2 and dry storage of SNF from RBMK-1000 reactors. It is considered reasonable to postpone construction of the RT-2 complex till commissioning of the new generation fast breeders (estimated up to 2020). These trends for improving nuclear fuel cycle are described in the "Sectorial concept for SNF management" approved by the Minatom's Board in January 2001.

3. Modernization of the SNF reprocessing plant at the "Mayak" Production Association

The first Russian plant on radiochemical reprocessing of SNF from NPPs – the RT-1 complex at the "Mayak" Production Association - was based on the military facility with inadequate scheme of waste management. Today Minatom takes extensive efforts to rehabilitate water reservoirs in which large volumes of radioactive wastes were discharged (Karachay, Techa river cascade). Similar to other radiochemical plants the RT-1 complex uses a kind of Purex process that results in formation of large volumes of solid and liquid waste. Comparison with foreign plants shows that the RT-1 complex is unique in terms of SNF variety as well as the capacity for industrial-scale partitioning of nuclear fuel components for further use. The extracted radionuclides are used for manufacturing of different sources of ionizing radiation at the adjacent radioisotope plant.

Vitrification facility for treatment of liquid radioactive waste (EP-500/3 ceramic melter) allows solidification of 300 to 310 litres of liquid radioactive waste per hour, the vitrified waste are stored in a vault-type dry store. It should be noted that throughout the operation of the RT-1 complex (over 25 years) not a single serious accident was reported at the "Mayak" Production Association that was classified above one unit on the International Nuclear Emergency Scale (INES).

Today the amount of fuel sent for reprocessing at the RT-1 complex of the "Mayak" Production Association is considerably below the design output:

- Design output of the RT-I plant: 400 t/year;
- Limitation of SNF reprocessing volume by the local authorities: 250 t/year;
- Actual reprocessing volume: 120-150 t/year.

In order to improve technical and economic performance of SNF regeneration at the RT-1 complex an integrated investment project has been prepared with the aim of partial modernization of the production, particularly:

- To create capabilities for reprocessing of SNF from VVER-1000 (and PWR) aimed at increasing the load of the reprocessing complex;
- To improve technological flowchart for reducing specific volume of the liquid radioactive waste formation;
- To construct installations for waste reprocessing and conditioning.

The current technological capabilities of the RT-1 complex are shown in Fig. 1. Following modernization (by 2008) the plant will reprocess up to 300 t/year of SNF with environmentally acceptable parameters of emissions and discharges.

4. Enrichment of recovered uranium

The "Mayak" Production Association carries out re-enrichment of recovered uranium up to 2.6% content of U-235 for fabrication of fuel for RBMK-1000 through mixing of solutions of uranyl nitrate obtained through reprocessing of different types of irradiated fuel assemblies including those filled with medium enriched uranium (~20%). No concentration of even uranium isotopes is performed. As is known, a higher burnup of SNF in the reactor lowers the quality of unburnt uranium due to an increased content of U-232 and U-236 isotopes.

Given the planned increase of reprocessed SNF volume at the RT-1 complex and future incorporation of recycled uranium in the VVER fuel, the Minatom enterprises are faced with the target of elaborating a cost-effective technology for direct enrichment of recovered uranium. The most suitable for this purpose is the Siberian Chemical Plant (Seversk, Tomsk region) having the industrial potential for radiological purification, conversion of uranyl nitrate into hexafluoride, state-of-art installations for isotope separation, and also a unique geological repository for liquid waste disposal for hundreds of years.

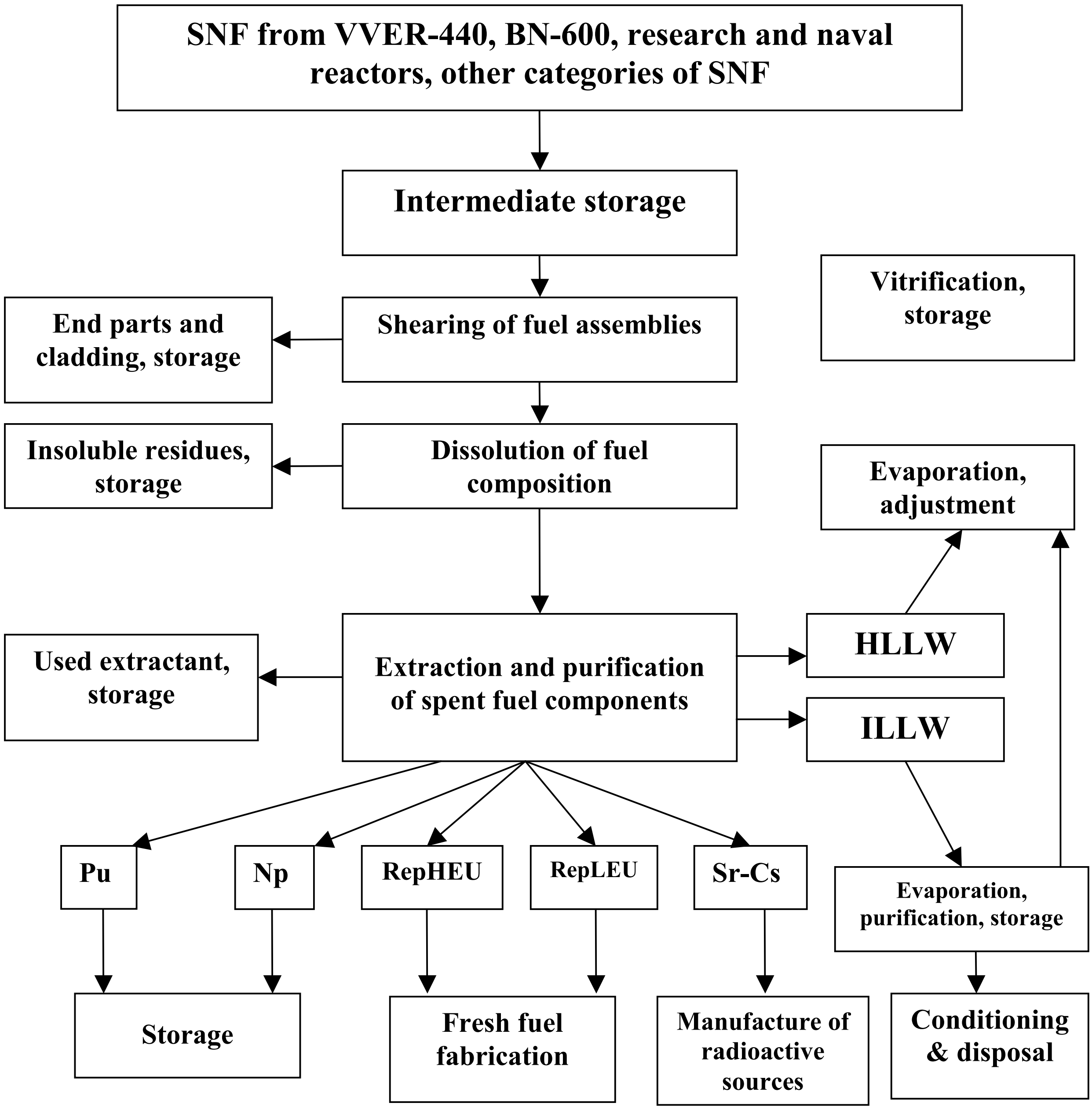

FIG. 1. Principle technological flowchart and potential capabilities of PT-1 complex.

On request by Cogema in the 1990-ties, the Siberian Chemical Plant successfully implemented a classical scheme for direct enrichment of recycled uranium obtaining enriched UF6 of the following composition: U-235 up to 4.95%; U-232 up to 10-6% and U-236 up to 1.4%. The plant has technological schemes that allow a substantial reduction of U-232 and U-236 isotopes in the low-enriched marketable product. In our opinion these options are of interest to foreign users having inventories of reprocessed uranium for use in nuclear power.

5. Construction of a dry storage facility at the Mining and Chemical Complex site

Russian planning and design organizations prepared a design of SNF repository in 2002. Today the project documentation is at the stage of approval and expertise. Construction of Stage 1 will start when the supervisory bodies finish the project expertise and issue a license for construction. Commissioning of Stage 1 is expected in 2005-2006 together with auxiliary structures and buildings, heat supply network and railroads.

During construction of Stage 1 it is planned to strengthen physical protection of the repository, modernize the infrastructure (railroads and motorways, power supply, etc.) and launch production of transportation means for delivery of SNF from RBMK-1000 at the NPP.

The investor in the repository construction is “Rosenergoatom” Concern responsible for operation of all NPPs in the Russian Federation. With the total cost of the repository of ~12.5 billion rubles, the Concern’s investment in construction will amount to about 950 million rubles (over US$ 30 million). The design capacity of the repository is slightly above the requirements of the Russian NPPs (up to 38 000 tons of SNF) that would allow in perspective to offer the interested foreign partners services related to SNF storage at the Mining and Chemical Complex.

The above-described activities are supported by the Government of the Russian Federation and included in the Federal Target Program “Energy Effective Economy” that envisages a framework for the Government support of investment projects, particularly R&D. As a rule, construction of new facilities is financed by profits of enterprises or target assignments incorporated in product price.

6. Conclusion

- Today the Russian nuclear industry is to a great extent adapted to the conditions of the national market economy;
- The Russian legislation allows offering foreign partners a broad spectrum of services in back-end nuclear cycle and SNF management;
- The Government of the Russian Federation supports the plan for development of the infrastructure and technological potential of Minatom’s enterprises with the aim of promoting to the world market of services related to the nuclear fuel cycle in compliance with safety and environmental regulations.

SESSION 2

TECHNOLOGIES

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Dry storage technologies: Keys to choosing among metal casks, concrete shielded steel canister modules and vaults

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Abstract. The current international trend towards expanding Spent Fuel Interim Dry Storage capabilities goes with an improvement of the performance of the proposed systems which have to accommodate Spent fuel Assemblies characterized by ever increasing burn-up, fissile isotopes contents, thermal releases, and total inventory. Due to heterogeneous worldwide reactor pools and specific local constraints the proposed solutions have also to cope with a wide fuel design variety. Moreover, the Spent fuel Assemblies stored temporarily for cooling may have to be transported either to reprocessing facilities or to interim storage facilities before direct disposal; it is the reason why the retrievability, including or not transportability of the proposed systems, is often specified by the Utilities for the design of their Storage systems and sometimes by law. This paper shows on examples developed within companies of the AREVA Group the key parameters and elements that can direct toward the selection of a technology in a user specific context. Some of the constraints are ability to dry store at once a large number of spent fuel assemblies, readily available, on a given site. No urgent need for further move of the fuel is foreseen.

1. Introduction

Defining a back end policy is a challenging question to which there can be a combined array of answers depending on such basic questions like: Who are we? An electricity producer ? An agency in charge of a country's long term back end management policy? Are my goals short or long term?

Clearly decision-makers in terms of back end policy for spent fuel, and more specifically in term of interim storage, can work on the subject from many perspectives that will in their turn evoke diverse choices or a combination of such choices. The Areva Group has the unique feature of having developed techniques and solutions that cover the full spectrum of state of the art interim storage technology:

- Pool storage followed by reprocessing and recycling and immobilization of the long lived actinides;
- For dry storage of spent fuel:
  - vault type systems of the Cascad type;
  - dual purpose (i.e. transport and storage) metal casks of the TN 24 ™ family;
  - concrete shielded welded canisters (NUHOMS).

The present paper will not dwell on the well-known comprehensive and integrated system that reprocessing offers, and rather concentrate on dry storage. After describing succinctly the features of each system, we shall discuss approaches and reasoning that orient chosen combination of storage systems. The main features of our dry storage systems:

- A multiple containment barrier;
- Passive cooling, while the Fuel Assemblies are stored in an inert atmosphere and under conditions of temperature preventing from the degradation of rod cladding;
- Sub-criticality meeting ICPR 60 requirements as well as all applicable regulations (including severe weather conditions and earthquake);
- Safe handling operations;
- Future decommissioning of the facility through design optimization;
- Construction and operating cost-effectiveness.

The Vault Type Storage system developed and implemented by SGN is an excellent solution: It combines passive safety with immediate large capacity, which allows quick amortization of fuel receiving equipment. In addition the versatile storage position can easily accept in the same facility different fuel types, and also intermediate and high level waste. This is the reason why a vault system is often a preferred solution for a long term dry interim centralized storage, for a multiplicity of spent fuel.

It can be also a choice solution when the ISFSI stands on a site that is dedicated permanently to many different nuclear activities. In most cases, the producers of spent fuel require a large capacity that is cumulated over many years, each reload at a time. Then the key criterion is maximum modularity.

Furthermore, the upfront capital costs requirement for this type of solution is minimal, so depending on the chosen discount rate of the investor, they have an additional attraction. Those smaller modules allow changing course in back end policy more easily.

Priority of modularity yields two other solutions, dual-purpose metal casks of the TN24™ family or dual purpose or single purpose concrete shielded welded canisters such as NUHOMS®. These solutions, implemented by COGEMA LOGISTICS, TRANSNUCLEAR Inc. and FRAMATOME-ANP, are very flexible and have been adapted also to quite different fuels.

What influences the choice, we can consider:

- In favour of metal casks:
  - Minimal ancillary equipment.
  - Ready to move to final or centralized repository or reprocessing or other ISFSI.
  - Compact systems.
  - Easy rearrangement.
  - Easy handling.
- In favour of concrete shielded canisters based systems:
  - Economics when initial quantity is sufficient to spread out up front equipment.
  - Significant cost – Shielding advantage.
  - Easy local production of the relatively light canisters.

Both approaches, when transportable, are also a factor for public acceptance because of the non-permanent characteristics and because transport licensing refers to internationally recognized rules, standards and methods.

2. The Cascad vault system

The design comprises 2 main facilities: The unloading unit and the interim storage modules, see Fig. 1. Containment is ensured by a double barrier:

- The first barrier is formed by the canister, in which the fuel elements are accommodated. The canister is inerted, tightly sealed and checked for integrity;
- The second barrier is made by the leaktight well into which the canisters are introduced.

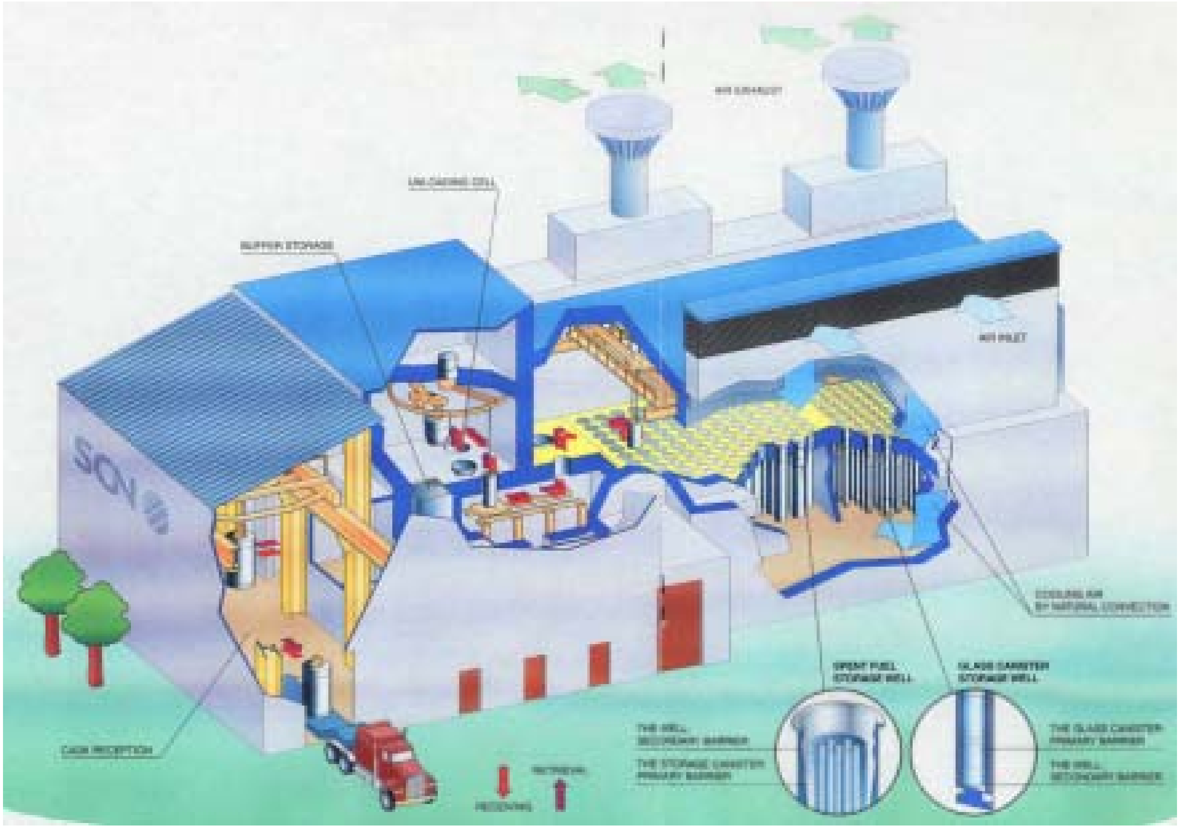

FIG. 1. Cascad schematic.

2.1. The unloading unit

This unit offers maximum flexibility, accommodating all types of casks and fuel element and HLW as well. The design of this unit is based on T0 spent fuel dry unloading facility of the COGEMA-La Hague Plant. Since 1986, T0 has unloaded more than 13 000 tU of PWR and BWR.

In the handling cell, spent fuel is inserted in canisters dimensioned and adapted to the fuel to be stored, irrespective of their dimensions or nuclear properties -residual- power, enrichment, etc.). After the interim storage period has elapsed, this unit also serves to remove the fuel canisters to their final destination. This operation requires no complementary installation.

2.2. The interim storage modules

These modules are built and added as the need arises. The fuel canisters are transferred from the unloading unit to the modules by means of a shielded transfer equipment or a crane. The fuel canisters are stored in a concrete structure, which protects both personnel and the public against radiation, but also the fuel against external phenomena, such as earthquake, aircraft crash explosion, etc.

3. Cascad vault technology worldwide

3.1. The reference facility: CASCAD, Cadarache, France

Cascad, located on the Cadarache site (France), has been operating since 1990, for a storage period of 50 years. Fuel stored in this facility originates from the CEA (French Atomic Energy Commission) research reactors and, in particular, from the Brennilis EL 4 Heavy Water Reactor as well as spent fuel from the French Navy. The cooling air, which enters the bottom of the wells is heated along the wells and discharged to the atmosphere through a stack.

3.2. Other facilities: EVSE, R7, T7, AVM, TOR

On the COGEMA-La Hague site, the vitrification of fission products generated by reprocessing and associated glass canister storage takes place in the **R7** facility for the UP2 plant and in the **T7** facility for the UP3 plant. These facilities were respectively put into operation in 1989 and 1992.

To increase the overall storage capacity, the **EVSE** facility (extension of glass storage for T7) was built and commissioned in 1995, see Fig. 2. In the EVSE facility, the released heat is removed by natural convection. A liner around each well forms a double jacket and the cooling air circulates in the annular space thus formed. The leaktight well in which canisters are inserted provides the 2nd containment barrier.

FIG. 2. EVSE facility.

Like R7 and T7, **AVM** is a vitrification facility with an interim storage for canisters, located at Marcoule (France). It has been commissioned in 1978. **TOR** Facility, also in Marcoule, was commissioned in 1986 for the reprocessing of FBR fuel from Phenix. The fuel elements is unloaded under dry conditions and transferred to storage pits, cooled by forced ventilation.

3.3. The Habog facility, the Netherlands

Final waste resulting from reprocessing of Dutch fuel from Borssele and Dodewaard as well as HEU reactor fuel (Petten and Delft) and research centre waste (Petten) will be stored for 100 years in Habog see Fig. 3, a facility built by SGN and based on the Cascad concept where commissioning is underway. The design complies with the American standard ANSI-ANS 57-9 rules and specific events like flooding, earthquake, aircraft crash (F16-A Falcon fighter), pressure wave resulting from external explosion and whirlwinds (velocity 125 m/s) have been considered.

FIG. 3. Habog, a new multi-purpose storage facility is ready to start-up – 2002.

4. Dual purpose casks TN™ 24 Family

The TN™ 24 concept is precisely adapted to the Transport and Storage of a wide range of fuel for both PWR and BWR types.

4.1. Design basis

The casks belonging to the TN™ 24 Family are basically constructed as follows, see Fig. 4:

- the basic structure is a thick steel cylindrical forging with a welded on forged bottom and one or two bolted forged steel lids equipped with metallic gaskets. These three main components make up the containment. The thick steel forging provide the main gamma shielding;
- surrounding the cylindrical cavity, a resin layer encased in a smooth steel outer shell acts as neutron shielding;
- surrounding the cylindrical cavity, a resin layer encased in a smooth steel outer shell acts as neutron shielding;

- inside the cylindrical cavity, a boron aluminum basket supports the Spent Fuel Assemblies and guaranties their subcriticality; it consists of mechanically assembled partitions in boronated aluminum defining an array of cells, one for each fuel assembly
- trunnions are attached to this structure for handling, tilting and tie down;
- a set of shock absorbing covers is fitted to the cask for transport operation, as well as lateral impact limiters;
- The possibility of easily adapting the boron content in the aluminum basket allows to consider increasing U-235 initial enrichments (see Table I);
- As shown in the Fig. 5, the TNTM 24 family covers a wide range of dimensional characteristics coping with heterogeneous fuel types and different nuclear facilities.

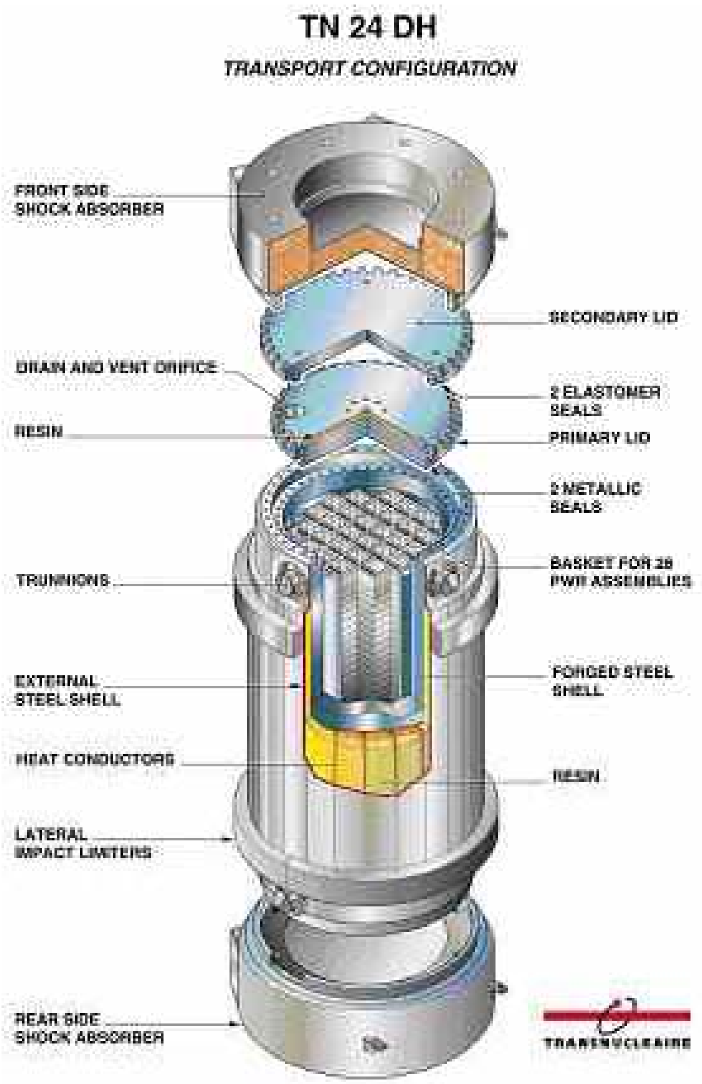

FIG. 4. Transport configuration of cask.

Table I. Cask characteristics of the TN™ 24 Family

Cask	N° of assemblies	Max Burnup (MW·d/tU)	Cooling time (years)	Max. Enr. (%)
TN™ 24 D	28 PWR	36 000	8	3.4
TN™ 24 DH	28 PWR	55 000	7	4.25
TN™ 24 XL	24 PWR	40 000	8	3.3
TN™ 24 XLH	24 PWR	55 000	7	4.25
TN™ 24 SH	37 PWR	55 000	5	4.25
TN™ 24 G	37 PWR	Average 42 000	10	3.8
TN 52 L	52 BWR	53 000	Min. 2.5	4.95
TN 68	68 BWR	40 000	10	3.3
TN 97 L	97 BWR	Average 26 000	10	3.95
TN™ 24 BH	69 BWR	50 000	6	5
TN 32	32 PWR	40 000	7	3.5
TN 40	40 PWR	40 000	10	3.5
TN 24 ™ P	24 PWR	33 000	5	3.5

FIG. 5. Various designs of the TN™ 24 Family.

4.2. The TN™ 24 casks worldwide

Delivered 15 years ago, the TN™ 24 P is the "eldest" cask of the TN™ 24 family cask. This cask has been used by USA Virginia Power and US DOE Idaho Falls National Laboratory. Since then, more than 100 TN™ 24 units have been licensed for transport and storage in Europe, the USA and Asia as shown in Table II.

Table II. Status of ordered and delivered casks

	ORDERED	DELIVERED
USA	91	33
BELGIUM	41	29
SWITZERLAND	16	9
JAPAN	9	9
TOTAL	**157**	**80**

5. NUHOMS®_canister based system

5.1. Design basis

The NUHOMS® system provides a comprehensive technology to store and transport spent nuclear fuel. It has been optimized by standardization of design, fabrication, and operation.

The NUHOMS® system is canister based, utilizing stainless steel canisters as spent fuel waste package and horizontal concrete storage modules as storage overpacks. The canister is loaded, transferred to the storage module, and ultimately transported to a repository or interim storage facility by means of a transfer or transport cask (see Fig. 6).

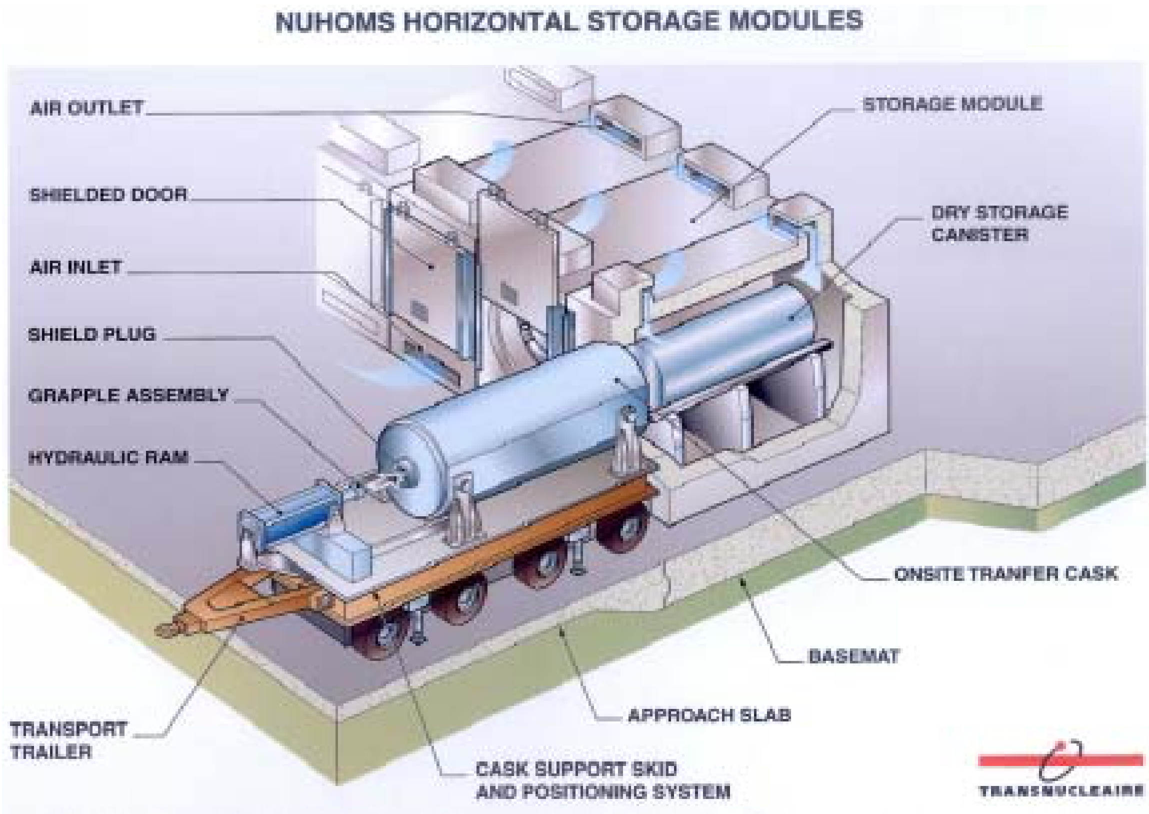

FIG. 6. NUHOMS® canister based system.

5.2. Canister

The NUHOMS® dry shielded canister (DSC) consists of a stainless steel shell and end covers with a basket assembly which provides structural support and criticality control of the spent nuclear fuel assemblies.

The basket assembly design can be adapted to handle different types of spent nuclear fuel as shown in Table III. Additionally, there are different basket designs for storage only and for the dual purpose systems. While the basket assembly can be varied, each separate basket is designed to fit into the standard canister configuration.

Table III. Characteristics of the NUHOMS® canister

Cask	N° of assemblies	Max. Burnup (MW·d/tU)	Cooling time (years)	Max. Enr. (%)
NUHOMS® 24 P	24 PWR	40 000	5	4
NUHOMS® 32 P	32 PWR	40 000	5	4
NUHOMS® 52 B	52 BWR	35 000	5	4
NUHOMS® 61 B	61 BWR	45 000	5	4
NUHOMS® 56 V	56 VVER	42 000	5	3.6 (U-235 equivalent)
NUHOMS® RBMK	95 RBMK	25 000	5	2,4

5.3. Transfer equipment

Once the fuel is loaded and all draining, drying and sealing operations are completed, the cask is placed on its skid in a horizontal orientation. When the cask and its canister waste package is moved outside the reactor building there are no operations, which require a heavy lift of the canister. The canister is transferred or retrieved from the cask to the storage module by means of a hydraulic ram, which pushes or pulls the canister out of or into the cask. The operations are simple and provide for an additional safety benefit by the exclusion of heavy lifts.

5.4. Storage module

The NUHOMS® storage overpacks are concrete horizontal storage modules (HSMs) which provide the overpack for the canister in its storage mode. The storage module is designed to handle the standardized diameter canister and has the flexibility to accommodate different lengths. The HSM is designed as a stand-alone unit consisting of two prefabricated pieces – a base unit (floor and walls) and a roof unit. These HSMs are transported to the site storage location and set in place. They can be arranged in various configurations to minimize the radiation dose associated with the site (Fig. 7).

5.5. Transport cask

The NUHOMS® dual-purpose storage and transportation cask utilizes the canister for its operations. This cask is also designed to be compatible with all NUHOMS® storage system components. This cask can be used for fuel loading and transfer operations, which places the canister into the storage module. The cask mates up with the storage module to allow for retrieval of the canister for shipment to another location – repository or interim storage facility.

5.6. Standardization and flexibility

The NUHOMS® system uses standard product dimensions so that various spent fuel canister waste packages can be designed using the same transfer and transport casks, auxiliary equipment, and storage modules.

5.7. The NUHOMS® technology worldwide

About 500 NUHOMS® systems have been ordered worldwide, see Table IV.

FIG. 7. Typical horizontal storage module configuration.

Table IV. Status of ordered NUHOMS® systems

Client	NPP	System	Loaded	Delivered	Ordered
CP&L	Robinson	NUHOMS®-7P	8	8	8
Duke	Oconee	NUHOMS®-24P	57	65	84
BG&E	Calverts Cliffs	NUHOMS®-24P	33	34	56
Toledo Ed.	Davis-Besse	NUHOMS®-24P	3	3	3
GPUN	Oyster Creek	NUHOMS®-52B	0	8	8
SMUD	Rancho Seco	NUHOMS®-24PT	0	1	21
SCE	SONGS1	NUHOMS®-24PT1	0	0	17
PP&L	Susquehanna	NUHOMS®-52B	8	14	26
DOE INEEL	TMI-2	NUHOMS®-12T	6	29	29
ARMATOM-ENERGO	Medzavior	NUHOMS®-56V	11	11	11
CH. NPP	CHERNOBYL	NUHOMS® RBMK	-	50	232
TOTAL			**126**	**173**	**495**

6. The parameters and factors of choice

We shall first display important parameters, then come back to address each of them. The first array of parameters is comprised of the legislative and regulatory context, its possible foreseen evolution, and the identity of the decision-maker.

6.1. Key questions

Legislative and regulatory context

1. What may we do (reprocess, store at reactor, send to a centralized interim storage facility)?
2. What can be licensed?
3. What is the timeframe involved from decision to commissioning?

Identity of the decision maker

4. What is my mission? What are my core competencies ?
5. What alternative do I possess?
6. Whence come my resources for that job ?
7. How does my choice influence my longer term costs?
8. Must my decision involve elements of local production?

The second array of parameters may be more site specific

9. Is changing course in the medium term valuable to me?
10. What quantities/qualities are involved?
11. How much space do I have?
12. What is my handling capability?
13. When is my need?
14. What is the quality of the relationship to the neighborhood?

6.2. Significance of the parameters for exercising a choice

Legislative and regulatory context

1. What may we do (reprocess, store at reactor, send to a centralized interim storage facility)?
   In some countries, like Germany, the recent evolution of the legislative context makes it a priority to avoid transport of spent fuel and facilitates at reactor interim storage. This means that the chosen system should be acceptable for storage on site, consider future unavailability of the NPP for casks reopening, and therefore leads to a dual purpose system such as the TN 24 ™ E chosen by E.ON Kernkraft and EnBW.
2. What can be licensed?
   Depending on regulations, impositions such as aircraft crash impact, acceptability of a mode of sub criticality justification or not can influence chosen technologies : for instance the boron credit used in the USA to justify subcriticality upon loading, plus moderator exclusion thereafter may create difficulties if one also wishes to have the system licensed for transport.
   Typically, some storage only systems that are presently implemented in the USA may require substantial additional effort if they have to be moved off site. This is why today, most procurement of NUHOMS® system chose to go for the transportable version.

3. What is the timeframe involved from decision to commissioning?
   Choosing a system that is already licensed (possibly with a different authorized content) by the authority in charge means additional confidence on ability to be licensed and timeliness of said license. This by no means precludes other choices, but precautions must be taken when the licensing route is being chosen, so that overall time schedules are compatible.

Identity of the decision maker

4. What is my mission? What are my core competencies?
   In the United States, responsibility for used fuel rests with the US DOE, who collects $ 0.001 per produced kWh to cater to this responsibility. In other words, US spent fuel producers are in the business of generating electricity or doing research, and wish to go for interim storage only to be able to hold on their operation until DOE does indeed take over the spent fuel.
   Conversely in Switzerland the four operators of NPP have regrouped their effort to create ZWILAG, a centralized interim storage facility, because it is their part to cater to the fuel as long as the geological disposal (or another solution) is not available. They combine this approach with reprocessing, for instance at the COGEMA LA Hague plant, and receive in return dual-purpose casks loaded with concentrated vitrified high level waste. So they chose to delegate the core competency of dealing with dry storage on the operator they created.
   In the HABOG facility, COVRA, based on its mission to cater to nuclear waste or spent fuel generated in the Netherlands, has to be an operator of that facility that can accept multiple shapes and forms of radioactive material in a very compact format. That brought them to choose the flexibility of a vault system, able to take in the same facility HLW, Spent Fuel from power and research reactor.
5. What alternative do I possess?
   A multi NPP utility may want to combine flexibility of optimizing the occupation ratio of their decay pool by performing transshipment between pools: in that case it may be an adequate choice to have a fleet of dual purpose casks first used for transport and then for interim storage.
   In other cases, the alternative may be to combine reprocessing and at-reactor interim storage.
   In other cases again, time can be a leading concern, and short lead time items such as NUHOMS® can be the solution.
6. Whence come my resources for that job?
   The US utilities, for instance, just like the Spanish ones, have no real concern on volume of spent fuel when considering back end solution, since a state entity is in charge of taking over their spent fuel in exchange for a fee based upon energy production. In that case, resources should be kept to the minimum necessary to keep the fuel pool with full core reserve since disposal of fuel is already taken core of . They generally chose the simplest equipment like NUHOMS®.
   If conversely the utility pays in proportion of volumes of SF, then it has an incentive of both raising burn-up and keeping a very dense interim storage solution, such as metal casks, that also cater well to issues linked to high burn-up.
   If large quantities of fuel are immediately available, choice of a vault system will be the prime choice for scale savings.
7. How does my choice influence my longer term costs?
   In line with the preceding point, the question of who is in charge and who is bearing costs for the longer term may influence choices:

As long as the NPP is running, the additional operational costs and concerns about operating an ISFSI are relatively small. Once no pool remains available, then transportability becomes a question of prime importance, since remedying a defect may involve shipping the system away. In addition, there may a significant premium in making sure that fuel is removed then stored outside the NPP soon after it reaches its final shutdown.

8. Must my decision involve elements of local production?
   There are many advantages of producing elements of a dry interim storage system locally.
   - Qualification of suppliers for the longer term;
   - Public acceptance aspect of having a benefit for local economy;
   - Easier follow up by local competent authorities.

   For that purpose, a large part of the Cascad Vault system and of the NUHOMS® system are easier to transfer to local industry than heavy casks.
   The example of Chernobyl, where all production is made locally, is a strong point of the world's largest dry ISFSI (Fig. 8).
   The longer term also involves the decommissioning of NPP for at reactor storage, and the question may be what of the costs of operation and back up solution once the NPP pool is not available any more.
   Vault systems and transportable systems are autonomous and provide solutions in such case. Storage only casks or canister systems are more problematic.

FIG. 8. The alley of FRAMATOME's Chernobyl NUHOMS®.

Site specific parameters:

9. Is ability to change course in the medium term, valuable to me?
   The modularity and up front investments associated with the system are a key parameter to help answering that question:
   With dual purpose metal casks, not only can the operator decide not to procure the next one, it could also use those delivered to ship the fuel elsewhere. The impact

of short term change is harder for canister systems, since up front equipment can represent a sizable amount, unless it can be sold to another user.

Vaults are of course committing the operator for a longer period before it becomes reasonable to change course.

One may also observe that this parameter may also point to the advantage of choosing from day one a strategy involving transportable systems. This keeps open at all times reprocessing options, or facilitate using them in parallel. The TN 52 L dual purpose cask was developed and used in order to perform routine transport of BWR fuel to reprocessing plants then to store spent fuel at the Zwilag facility.

10. What quantities /qualities are involved?

The inventory of spent fuel to be stored is of course another prime parameter in choosing a system.

Large inventories make the up front investments for a vault system worthwhile: the system initial units represent an relatively large percentage of the overall costs, while actual storage space, especially for a large batch readily available, does not represent such a high contribution, when expressed in kg of heavy metal stored.

Diversity of material and medium quantities can also justify the vault systems, that can offer different accommodation pits for different material under the same roof.

The combination of distance and inventory is also to be pondered: investing in a cask transport fleet in order to feed the vault if different sites are involved may also be costly and involve issues of public acceptance.

Small inventories, or progressively increasing inventory (i.e. a batch every year or so, corresponding to the reactor reload) may call for additional modularity such as that of dual-purpose casks or concrete shielded canisters like NUHOMS®. Thus it is possible to spread out the investment, and to profit from the financial discounting rate that favor differed investments.

11. How much space do I have?

Footprint, space available are also influencing the choice of system : maximum density can be achieved by metal casks, irregular shapes of sites can be best occupied by casks and canisters systems, vaults prefer rectangular sites. The South California Edison ISFSI choose NUHOMS® among other things because it required a canistered system with strong seismic resistance combined with a small footprint compatible with their available space.

12. What is my handling capability?

Metal casks are more competitive when they can maximize capacity and diameter, that is also when they can be quite heavy: the TN™ 24 G casks weighs 135 tons. The issue of handling and bearing capacity within the facilities where the system will be loaded has therefore a strong influence. In the case of the TN™ 24 G, it is the proper combination of size, mass and loading plans that authorized the loading of 37 PWR SFA per casks.

In canister-based systems, the relative standardization of peripheral loading equipment creates minimum and maximum performances for the systems in term of capacity.

Vault systems are relatively independent from these considerations, because they can always rely on a compatible transfer shuttle between NPP and storage.

13. When is my need?
    This parameter is connected to the question of ability to get a license on schedule, of choosing or not to qualify a local fabricator, of operational issues linked with the NPP such as core and decay pool management, burn-up, contingencies etc.
    There are two reference times:
    First is time to initial loading on a new ISFSI, and this is relatively long and second is time to procure additional units.
    These reference times are directly affected by necessity or not of new developments on the storage system itself, whether the license is generic or not, whether new investments for production are necessary or not.
    The shorter possibility for initial loading is the dual-purpose metal cask. The shorter renewal time is the canister system case.
14. What is the quality of the relationship to the neighborhood?
    As our Czech friends know well, neighborhood can be extended to neighbor countries, where any new development in the nuclear field is followed by intense anti activity from Austrians.
    Then choices pertaining to references that are internationally accepted such as the IAEA recommendations for transportation of radioactive materials, may become important:
    - they offer the fact that regulations are not a local choice, but the choice by experts coming from all horizons;
    - they also pass the message that if the system is transportable, then it will not stay there forever.

    The ability also to choose systems that can be seen elsewhere, discussed with other communities can be a parameter for choice.

6. Conclusion

Choosing a dry interim system technology is not an easy choice, it involves a combination of technological, political, licensing, policies parameter for which the answer has to be carefully built.

The ability to choose from organizations like AREVA that are able to display a complete range of solutions and services guarantees:

- that one or another important parameter is not discarded for wanting an adequate answer;
- that altering course may receive an adequate support for that;
- that the long term maintenance of the chosen solution(s) in the medium long term is provided for.

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Conceptual design for an intermediate dry storage facility for Atucha spent fuel in Argentina

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Abstract. Different types of fuel storage were evaluated to be used as interim storage of the spent fuel of Atucha I for a life period of 30 years. The horizontal concrete superficial silo, with air natural convection cooling system is the best cost effective solution for the requirements. A conceptual design of the Atucha I horizontal dry storage silo is presented. Among others, the main characteritics are its modular building, each spent fuel element is sealed inside a metallic canister, and it can be loaded into the silo individually enabling partial silo loading.

1. Introduction

Argentina National Atomic Energy Commission (CNEA) and Nucleoeléctirca Argentina S.A. (NASA) are planning a new facility for the spent fuel of Atucha I, according with the national policy to fulfill the requirements of the National Plan of Radioactive waste management at the lowest cost, having the flexibility to evaluate the fuel back end strategy [1].

Nuclear power plants are typically designed to provide spent fuel storage capacity for 10 years, waiting the back-end steps. But many power plants have initiated dry fuel storage to provide capacity during the remaining plant life [2]. Spent fuel elements can be stored economically and safely in concrete for many decades as an intermediate step. This provides time to develop an integrated fuel disposal system, and facilitates final disposal with less residual heat. Moreover it allows to manage the time for different life periods of nuclear power plants in order to have less final back-end processing batches.

Atucha I is a pressurized heavy water reactor (PHWR) having fuel elements with 37 rods in a circular cluster geometry with an active length of 5.3 meters. Atucha I started at 1974 with natural uranium fuel and it changed to slightly enriched uranium (SEU) fuel in 1998 and to full core load in 2001. Nowadays the total life Atucha I spent fuels are in two wet pools, with spent fuels of 28 years old. Atucha I is going to complete its cooling pools capacity by 2008, and CNEA and the utility must find a place to store the cooled spent fuel to freeing up space to the new ones up to 2014.

Embalse, the other Argentinean nuclear power plant (NPP) in operation (CANDU type), has dry storage vertical concrete silos. It was successfully implemented for intermediate strategy in 1993.

A centralized storage for NPP fuel elements (Embalse and Atucha I) with two very different fuel elements and different enrichment was not considered. This was done in order to minimize the spent fuel transportation. Having two NPPs in operation and another under construction, the NPPs have different dates to shutdown. This spread time can be managed with interim storage to have a whole back-end processing.

2. Atucha I interim storage general requirement

The Argentinean National Plan of Radioactive waste management law gives to the CNEA the responsibility of spent fuel elements (SFE) after NPP´s shut down. An interim spent fuel storage strategy was selected. It will be the only step before final disposal according with today expectations, and spent fuels must be retrieved from the storage to go to the final step. The CNEA sets general design requirements for the Atucha I interim storage. Theses are:

- Assurance fuel elements integrity for 30 year;
- Ten years for the decay time in wet pool;
- Allows NPP full decommissioning;
- Modular building to avoid over dimension systems;
- Additional isolation barriers;
- Low cost radiation shield (concrete);
- Leak Monitoring system for the fuel containment integrity;
- Retrieve the fuel when the containment fails or at the end of the storage life;
- Enable the re-encapsulation and the reentry for the fuel containment;
- Transfer systems and hot cell facility near the pool storage to use the existing water processing systems;
- Minimize the auxiliary systems with high maintenance cost, selecting passive systems;
- Compatible with the National Regulatory Authority (ARN) regulation with radiation monitoring systems, similar with the implemented in our dry silos at Embalse;
- Minimize secondary waste during wet pool transfer to the interim storage;
- Minimize the land recovery cost;
- Minimize spent fuel transportation.

An Independent Spent Fuel Storage Installation (ISFSI) for Atucha I has to be built in 2008, with a whole capacity for more than 10000 fuel elements. It also has to be compatible with the CANDU dry storage silos strategy. The ISFSI must store two different Atucha I fuel elements (FE) type, one FE with natural uranium and the other with SEU (slight enrichment uranium), having a fuel extraction burnup of 6 000 – 11 300 MW·d/tU respectively. It also must be flexible enough to store different fuel designs, Atucha and CARA fuel. The last one has been designed to reach a burnup of 14 000 MW·d/tU [3].

3. Storage type selection

Dry storage and wet storage of spent fuel depend on the capital and the operational costs associated with the ISFSI strategy. Modifying the current spent fuel pool or constructing a new pool is a major effort. This involves significant expenses including design, planning, and it has high operational cost. It was dismissed because its cost is prohibitive. Different dry storage systems for intermediate time were evaluated (cask, cavern and silos).

The cask is modular and doesn't need additional systems, but the current designs were approved for a period at least of 20 years, and the requirement (30 years) are beyond the design base [2]. Moreover, to attain the licensing requirement for transportation, the resulting device is very expensive [4], and it has over dimensioning to be used only as storage for long times.

The cavern involves high infrastructure cost (on-site works) and it needs active auxiliary systems for cooling and very complex fuel handling. These features do not fulfill the general requirements.

The silo is the best cost effective solution [4], compatible with the selected strategy for CANDU fuel type. Its great flexibility allows modular construction and passive cooling systems. A surface storage had been chosen to minimize on-site works. The fuel elements will be placed horizontally in the silo instead of vertically in CANDU silos, in order to simplify the silo loading system due to the Atucha I fuel length.

4. Dry storage design criteria

The design of the Atucha I spent fuel storage takes into account the following topics:

- Five years for the decay time in wet pool for fuel element with 14000 MW·d/tU;
- The fuel elements must to be moved individually to have great flexibility, enabling partial vault filling;
- The radiation shielding and the spent fuel isolation of contaminant are implemented by different systems with standard materials. External concrete vault with steel gates for shielding and a thin metallic walled canister for sealing the fuel element, are considered;
- The canister material must be corrosion resistant, and its thickness is designed to support mechanical transport requirement and to keep integrity for a 30 years period. The isolation of encapsulated fuel element can be tested by helium leak;
- Minimize internal silo volume using a square array of fuel canisters;
- The decay heat is removed by natural convection with the surroundings. It was designed to have the heat dissipation with a central pin sheath temperature up to 200 °C and atmospheric inlet air at 40 °C [5];
- Makes a common block of horizontal silos to minimize concrete cost and surface, by sharing lateral walls and using common bases for transportation. The number of joined silos in the block is set by economical optimization, taking into account the operational needs in the time investment schedule;
- The shielding width is designed with the ALARA (As Low As Reasonably Achievable) to ensure personnel and public radiation dose bellow allowable limits according to international standards;
- The installation is not a permanent storage facility, thereby a special handling machine is needed to insert and to remove each canister horizontally from the storage in a proper array location. This system is placed with the special transportation system that enables the load from conditioning hot cell of fuel elements. This systems also allows an easy re-encapsulation movement via sending back the canister to the hot cell if this isolation fails;
- Dry-storage siting needs to be near of NPP to minimize spent fuel transportation;
- Active systems only required for safeguard and for ensure leaking monitoring;
- The conditioning of fuel element inside the canister and the QA tests are done in small hot cell beside the wet pool;
- Transference gate design take into account labyrinthine shape coupling to avoid straight radiation beam to external operators.

5. Atucha I dry storage conceptual design

In accordance with the general requirements and the design criteria, a description of the main characteristics of the conceptual design of the Atucha I spent fuel dry storage is presented. The fuel canister has a special hook on one lid for the input-output of the silo. Elliptic rolling pin supports are used to place and guide each canister in an array position inside the silo and the cask. The decay heat is removed from the silo by a natural convection cooling system (passive). The air comes into the fuel room by one air inlet near the floor, and goes out through two air outlets in the upper part of the silo. Two internal perforated plates serve as flow distributors and increase the heat transfer area.

The silo allows regular inspection to monitor the integrity of the canister as well as inspections of safeguards. The silo inlet has a steel gate and external labyrinthine shape for coupling with cask for the canister transference. Making a common block of silos, the lateral and the back concrete walls in each unit, have less width by sharing with the neighboring silos the radiation shielding. A lateral cross section view of two horizontal natural convection silos of a common block is showed in Fig. 1, having for example a 5×5 squared array of canisters. Fig. 2 shows the frontal view of the block of silos.

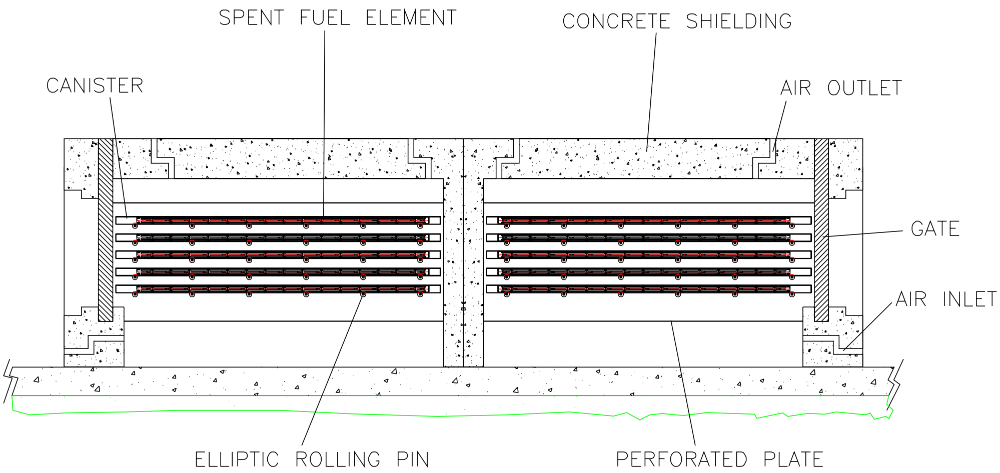

FIG. 1. Lateral cross section view of Atucha I common block silos.

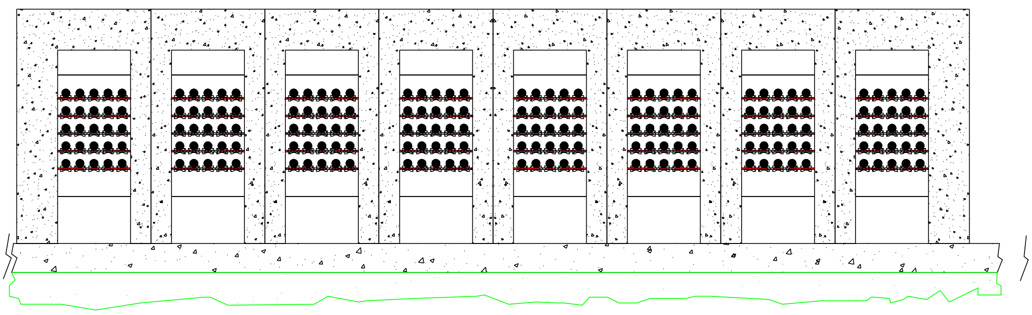

FIG. 2. Front cross section view of Atucha I common block.

Land surface and concrete road is minimized with the scheme showed in Fig. 3 using common blocks silos. The spent fuel transport system is a special cask with steel radiation shielding and it has a mechanical ram system to push and pull the fuel into the silo and hot cell. This system enables the spent fuel recovery from the concrete silo to return to the hot cell eventually for re-encapsulation, or for final spent-fuel treatment.

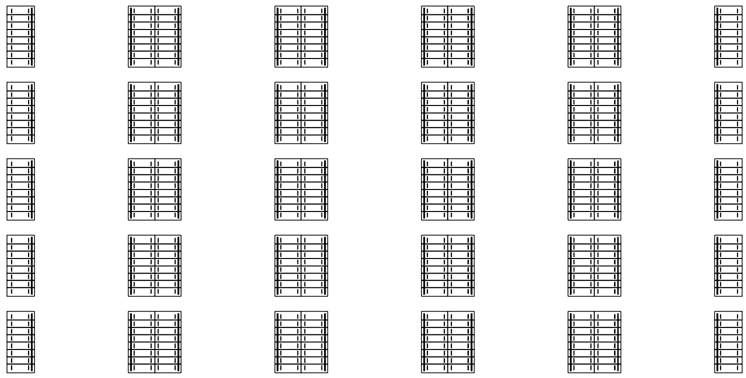

FIG. 3. General plan view for 10000 spent fuel element dry-storage installation.

The requirements of the mechanical ram system are to move each canister individually in both directions and the assuring of the ram position. The canister length and the array compactness are the major constrains. The mechanical ram system designed is composed by gears and cog rail with electrical drive for each canister position. Gears and cog-rails avoid direct outward radiation beam. This transport can be done with a truck and 14 axle trailer of 110 t. Fig. 4 shows details of the Atucha I spent fuel cask transport .

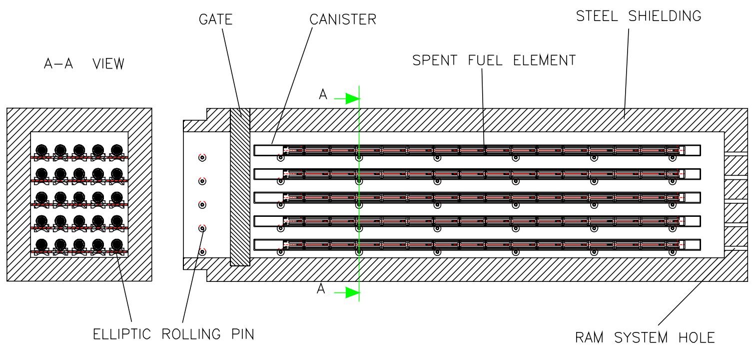

FIG. 4. Lateral cross section of cask system.

In a hot cell designed for individual spent fuel element handling, the fuel element is loaded inside a stainless steel canister, dried, backfilled with inert gas and hermetically sealed with a welded lid. To minimize fuel movement the hot cell is placed near the wet storage pool. A special cask is required for fuel element transportation from the wet pool to the hot cell. The short distance between the hot cell and the wet pool enable the use of auxiliary systems (air purification, water treatment). The cask can be drained before going to the hot cell.

The hot cell has a rolling cut machine (without small metal scarps) to remove Atucha I upper structural fuel assembly. The remaining water in the canister is evacuated with a vacuum dryer. The canister is pressurized with helium before welding the lid. This enables leakage test to be performed before sending it to the silo. The fuel element movement inside the hot cell is done by a remote handling system and a two dimension fuel position system, this one is used to transfer the encapsulated fuel element to the special transport cask. The hot cell and the transport cask allow the re-encapsulation of the canister failed. The final design will be economically optimized. Final thermohydraulics and shielding design is still on work. Preliminary evaluation gives a cost of 8 to 10 US$/kg HM.

6. Conclusions

The horizontal natural convection dry storage is the best cost effective solution to store spent fuel with prior decay in a wet pool for Atucha I spent fuels, compatible with the selected strategy for CANDU fuel type. On-site modular superficial concrete silos near the NPP allow great flexibility to build and fill them according with the demand.

The design can be implemented with standard materials and mechanical devices. Each spent fuel element is sealed inside a stainless steel canister with QA test. The cask transport with mechanical rams system can load and retrieve fuel element to/from the silo individually, given the posibility of partial loading of the silo or canister re-encapsulation.

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Advanced spent fuel storage pools

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Abstract: Spent fuel from power reactors is currently stored either in at-reactor spent fuel storage pools or in independent spent fuel storage installations using wet or dry storage technology. Most of the spent fuel generated up to now is stored in spent fuel pools. The assumption that wet storage is expensive due to the need of active cooling systems and generation of waste had caused vendors to look favorable at dry storage systems. One of the latest achievements in wet storage technology is used in FRAMATOME ANP's wet storage facility as currently designed for the new spent fuel storage building at Goesgen Nuclear Power Station Switzerland. It provides a passive cooling system which reliably removes the heat generated by the spent fuel by natural circulation through air cooled heat exchanger. Due to the passive nature of the operating system, the number of active components which require maintenance is substantially reduced. The frequency of maintenance activities can be determined under consideration of actual usage due to advanced acquisition methodology of operational data. This usually leads to a considerable reduction of human intervention and the time needed to act in radiation areas reducing considerably waste generation and dose burden to personnel. Due to the fact that maintenance and repair concepts are available, it can be predicted, if correctly applied, such pools to be operable for extended periods of time.

1. Introduction

Spent fuel from power reactors is currently stored either in at-reactor pools or in independent spent fuel storage installations (ISFSI) using wet or dry storage technology. During the past 15 years, storage capacity of at reactor pools was increased using high density spent fuel storage technology. To achieve maximum capacity, storage racks were replaced in many of the power reactors in operation at least once, some of them went through even various reracking cycles.

Independent spent fuel storage installations were established either at the site of power reactors of away from them. They use either wet or dry storage technology, the latter in form of metal casks and concrete silos or vaults.

Storage of spent fuel from power reactors must be safe for the public and respective facilities must be adequate to sufficiently protect the environment from its radioactive content. For this purpose, adequate regulations were developed and are available to be applied. However, advances in fuel and core design as well the need for extended storage periods require frequent re-assessment of the available spent fuel storage technology.

2. Requirements

The design of a spent fuel storage facilities must, as this is the case for any other nuclear facility, provide for sufficient means to maintain acceptable levels of safety under all feasible normal and abnormal operating conditions. The main safety goals to achieve this objective are:

- to remove of the decay heat generated by the spent fuel assemblies in storage safely;
- to maintain the spent fuel assemblies reliably in a subcritical configuration;
- to keep radioactive material sufficiently contained and separated from the biosphere.

To serve as a basis for such designs, there are three acceptance criteria which relay on the fuel to be stored. These are the heat generation, γ-dose rate and neutron dose rate. All three depend on the burn-up of the individual fuel assembly. Where the γ-dose rate increases moderate with increasing burnup, the temperature increases with higher gradient and the neutron dose rate increases sharply with burn-up. For dry storage casks for instance, this means the γ-dose rate is the governing factor to store fuel with less burn-up, while for fuel with higher burn-up the heat removal becomes the controlling design criterion (see Fig. 1). With further increase of burn-up the neutron dose rate is the factor which decides how long the fuel has to be decayed until it can be moved into dry storage. One can roughly say, each megawatt of additional burn-up per kilogram of U-235 requires one additional year decay time or additional shielding, exacerbating heat removal capability.

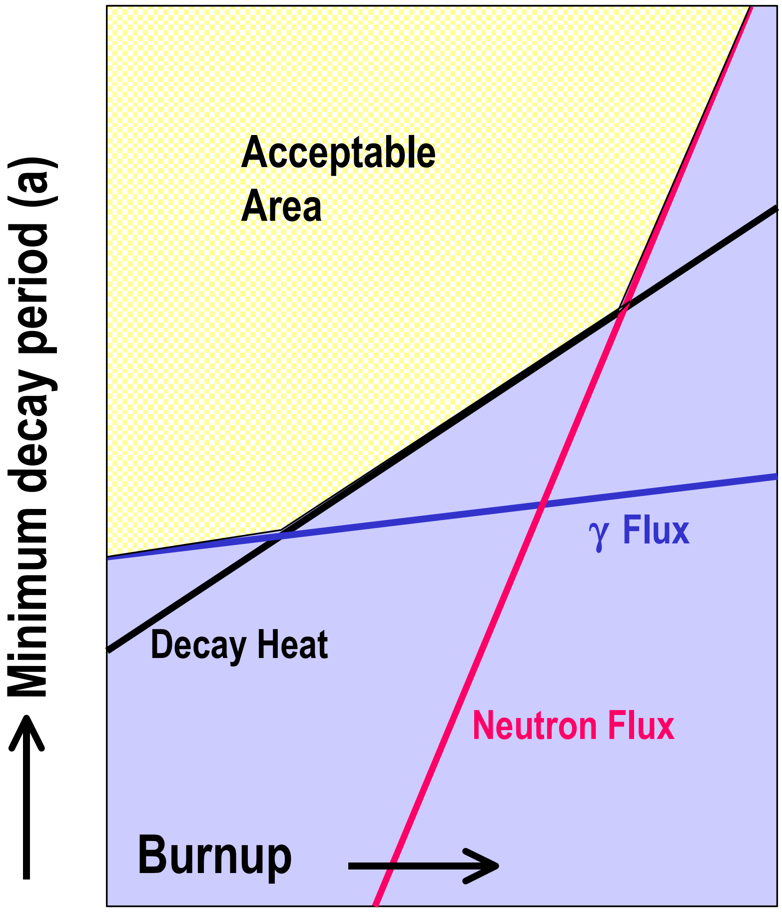

FIG. 1. Change of design parameter as a function of fuel burnup (schematic) [1].

Heat removal capability is of particular importance for the storage of spent MOX fuel as such fuel generates considerable more decay heat and neutron dose rate than spent uranium fuel. The difference in decay heat generation between uranium fuel and MOX fuel becomes more important, especially after longer decay periods (see Fig. 2). Usually a minimum of 10 years decay time is needed before spent MOX fuel can be loaded into dry storage.

As most of the mechanisms, which could endanger fuel integrity, are temperature dependent [2, 3], effective and reliable heat removal is one of the most important design criteria spent fuel storage systems have to comply with. Bearing these facts in mind, one has to consider that the last fuel assembly unloaded from the core of a permanently shut down power reactor may have to decay for another decade to be put into dry storage.

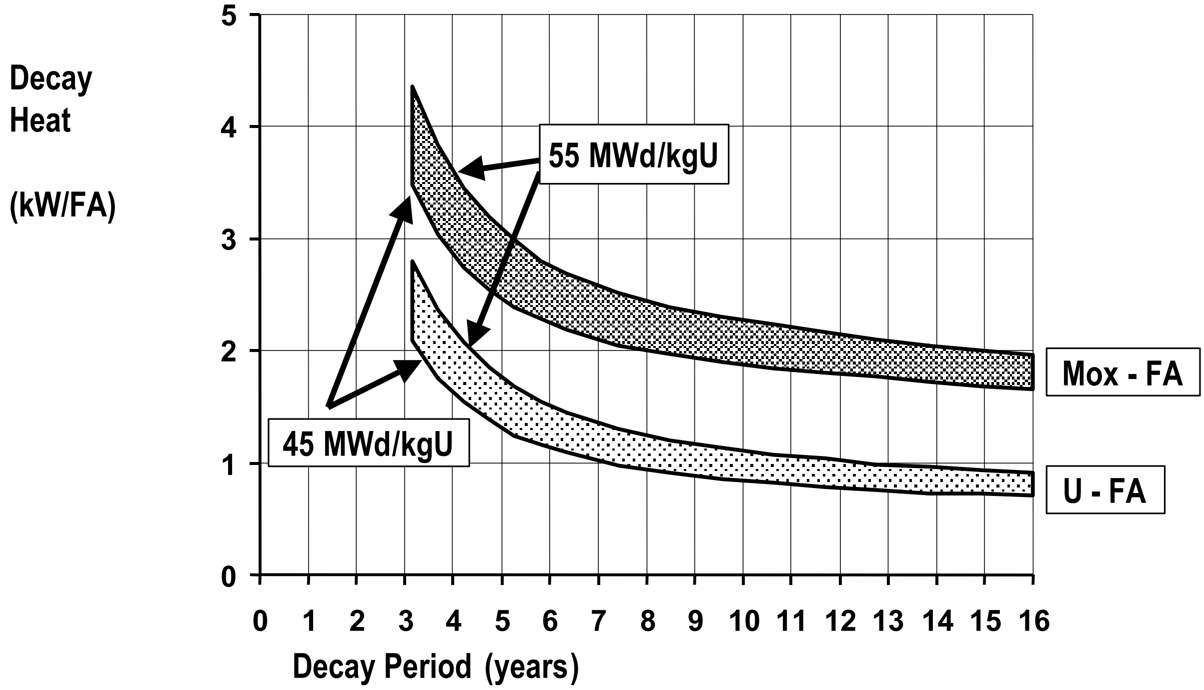

FIG. 2. Decay heat of uranium and MOX fuel as a function of the decay period [1].

Safety against criticality is the other important design requirement. Safe subcritical arrangement is either achieved by separating the fuel assemblies sufficiently from each other or, by using sufficient neutron absorbing material to be located between them as so called poisoned high-density storage facilities. In order to make most efficient use out of high density storage equipment, designer want to take credit from the actual burn-up of the spent fuel. Existing methodology to analyze burn-up credited spent fuel storage racks for criticality safety is being further developed to be applied in the design of dual-purpose casks and multi-purpose-canisters, respectively.

For poisoned high-density storage equipment, long term stability of the material as well as efficient neutron absorption is required. Degrading absorbers which contaminate the coolant of fuel pools and primary circuits and fuel assemblies getting stuck in swollen storage cells are well known problems which already caused considerable headaches to many operators and still continue to do so. Although limited in boron content, borated stainless steel has extensively proved as sufficiently effective and extremely stable neutron poison material.

Over and above the basic requirements for maximum safety for operators and the public, logistic in fuel reception is an important aspect to be addressed with the design of independent spent fuel storage facilities. The aspect to keep the operators' exposure to radiation as low as reasonably achievable requires expeditious reception of spent fuel and its transfer to the dedicated storage location.

3. Realization

Independent wet storage facilities are known for many years to comply best with most of the expectations as described above. Despite the remarkable development achieved in dry storage technology, most of the spent fuel generated up to now is stored in fuel pools, either at the reactors or in independent installations.

The advanced design of this independent spent fuel storage facility provides a passive cooling system, which reliably removes the heat generated by the spent fuel by natural circulation through air-cooled heat exchangers. This progressive design makes extensive use of well-balanced safety technology with largely passive safety features. Passive safety features are also used for the design of advanced nuclear power plants, e.g. as for Framatome ANP's Boiling Water Reactor SWR1000.

Due to the passive nature of the operating system, the number of active components, which require maintenance, is substantially reduced. In addition, due to advanced acquisition methodology of operational data, the frequency of maintenance activities can be determined under consideration of the actual usage. This usually leads to a considerable reduction of human intervention and the time needed to act in radiation areas reducing considerably waste generation and dose burden to personnel.

As a facility dedicated to receive and to store fissionable material, regulatory requirements on access controls as defined by the competent regulators are to be met. The advanced wet storage facility design of Framatome ANP meets safely with all the requirements for safeguards and physical protection, which may include protection against terrorist activities and actions of sabotage.

3.1. Design

3.1.1. Safety requirements

The requirements defined by national and international standards applicable for the choosen site have to be met. The three main targets are:

- Maintain subcriticality of fuel assemblies.
- Ensure sufficient cooling/water coverage of fuel assemblies.
- Ensure Activity confinement.

Following the different operating conditions the wet storage facility is designed for:

- Normal operation:
  - Normal operation
  - Inspection
  - Maintenance

- Category 1 events:
  - Loss of one cooling loop or cooling tower fan
  - Loss of power
  - Operating Basis Earthquake (OBE)

- Category 2 events:
  - Fuel handling accident

- Category 3 events:
  - Safe Shutdown Earthquake (SSE)
  - Airplane crash

- Spent fuel pool integrity:
  The conceptual principle of the fuel pool building allows for easy separation of the storage pool walls from the outer walls of the storage building. Thus, the risk of an direct impact to the pool structure caused by airplane crash is eliminated. As needed, either military aircraft or the impact of commercial aircraft, respectively can be considered with such design. Other credible scenarios for pool liner damage are excluded through appropriate selection of material, methods and arrangement of equipment and components. E.g. the fuel storage racks are arranged such that direct impact between racks and pool liner remains impossible.

- Accident management:
  Special provisions are foreseen to allow for water makeup and to assure sufficient heat removal after a reasonable grace period in case of accident conditions beyond design basis. In such unlikely event, mobile supply units can be connected to the system to resume removal of the decay heat from the fuel pool.

3.1.2. Fuel pool cooling system

The fuel pool cooling system is designed to remove all the heat generated by the stored fuel assemblies from the fuel pool water under all normal and credible abnormal operating conditions and to transfer it safely to the ambient air. It is based on the principle of natural circulation, hence providing for inherent operational reliability and safety (see Fig. 3) [4, 5].

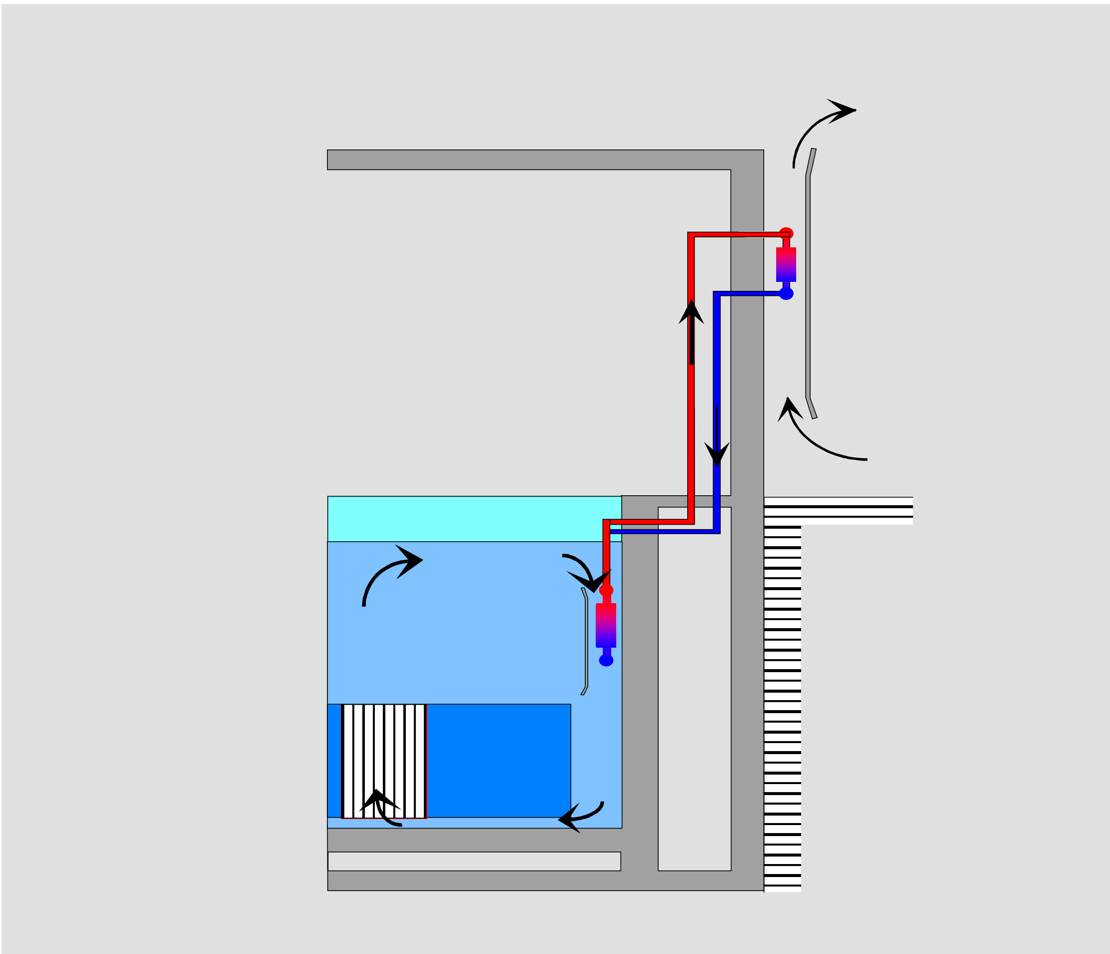

FIG. 3. Passive spent fuel pool cooling system (principle).

For this purpose, the fuel pool cooling system consists of four loops, two of which sufficient to provide 100% heat removal capacity in order to provide adequate redundancy. Each loop is equipped with plate heat exchanger(s) installed inside the fuel pool and water/air heat exchanger(s) installed in one of two cooling towers. No active components such as pumps are needed in this system. Natural circulation transfers the heat from the pool to the cooling towers (see Figs 4 and 5).

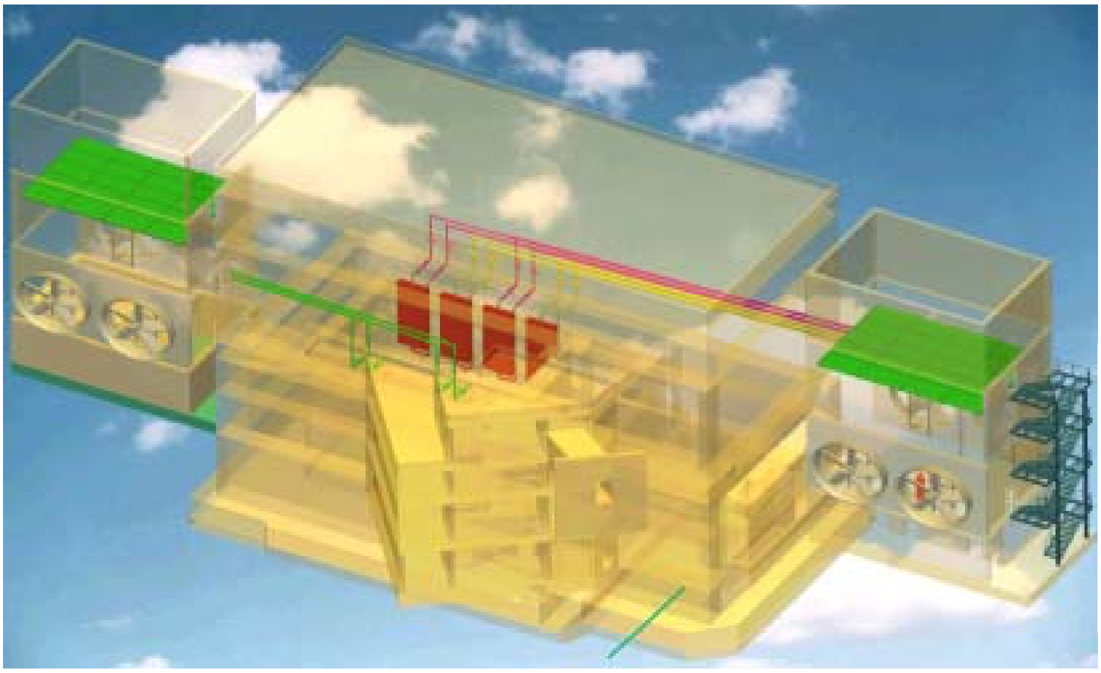

FIG. 4. Total view with highlighted cooling system.

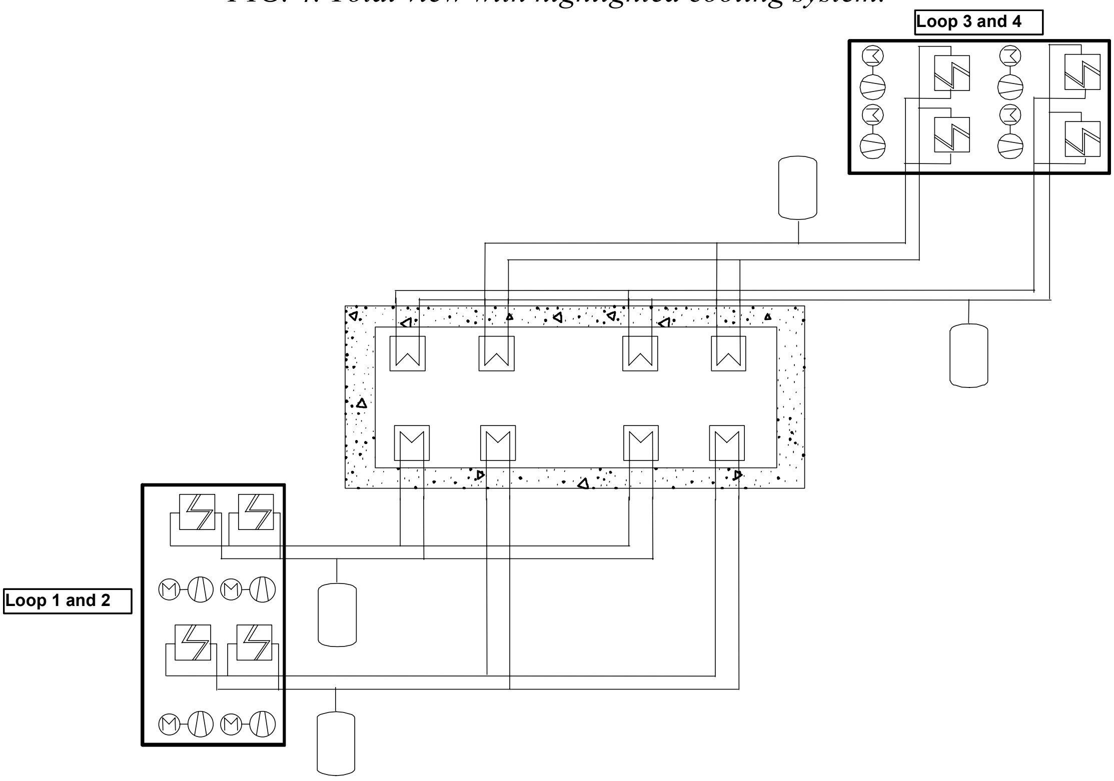

FIG. 5. Principle drawing of pool cooling system.

Natural air circulation ensures reliable heat transfer from the water/air heat exchanger to the ambient air. In order to provide comfortable conditions for the operators at the operating floor in case of high ambient temperatures during summer time simultaneous to maximum heat load in the pool, electrical fans can be used to limit the pool water temperature to the desired level. For normal operating conditions the pool water temperature is typically limited to 45°C. In case of abnormal operating condition the electrical fans are not needed. The system is designed for absolute passive operation. In case of category 1 events the maximum pool water temperature is typically allowed to rise to a maximum of 60°C. In case of category 3 events, the maximum coolant temperature is typically limited to 80°C.

In the unlikely event of a complete loss of cooling there is a grace period of 2 to 4 days to initiate the accident management system for cooling and/or water make-up. For this purpose, suitable connections are in place.

3.1.3. Pool water purification system

Pool water purification is needed firstly to achieve sufficient visibility and secondly, to reduce as reasonably as achievable, the radiation level at the operating floor of the fuel pool. The use of advanced equipment, recently developed from the submerged vacuum cleaner principle but allowing the use of ion exchange filter separate from mechanical filters, enables the designer to locate the fuel pool purification equipment inside the boundaries of the fuel pool, avoiding such the need for pipe penetrations through the pool walls typically vulnerable to damage and subsequent leakage. Loading of the mechanical filter module is monitored by separate differential pressure measurement. If differential pressure indicates that the filter is exhausted, it can be replaced underwater.

Consequently, the ion exchange unit is less affected by mechanical impurities and can be used for considerably longer periods of time. Hence, the total amount of spent ion exchange resin to be dealt with during the life of the fuel pool is considerably reduced.

3.1.4. Pool liner

Although different liner material such as epoxi coating or ceramic tiles were used as pool liner material, stainless steel liner systems are well established world wide and have proven best suitability for wet spent fuel storage pools. The stainless steel pool liner design allows ease installation of leak detection systems, which enable to identify affected areas in case of leakage.

Due to the selected arrangement of the components of the pool cooling and purification system inside the boundaries of the fuel pool, no pipe penetrations below water level are needed.

3.1.5. Fuel storage

A fuel pool storage facility can be designed to use any suitable type of fuel storage equipment. Typically, poisoned high-density fuel storage racks are used identical with such actually used in at-reactor storage pools. The use of neutron poison material allows optimizing the building's size versus storage capacity to achieve best economical benefit. Racks are typically designed to be free standing and free sliding in order to reduce the impact of the dynamic behavior of the racks when interacting with the fuel pool structure in case of seismic or other external events.

Depending on the result of respective economical studies, fuel racks can be designed either taking credit from the actual burnup and hence, consider the residual fissionable amount and certain fission products in the criticality analysis, or such to maintain the required subcriticality criterion for fuel at the point of its maximum reactivity. For burnup credited racks, the double contingency principle has to be maintained in order to safely comply with the subcriticality requirement even in case of erroneous misplacement of fuel not complying with the minimum burnup required. Soluble boron is typically used to provide the necessary safety margin in such hypothetical incident. Opposite to existing pressurized water reactor fuel pools, independent facilities would usually not use soluble boron in the coolant. Hence, fuel racks in independent wet storage facilities will typically be designed to receive maximum reactive fuel.

Neutron poison material used in many of the high-density fuel storage racks has caused numerous headaches to operators up to now. We all are well familiar with shrinking or swelling effects different materials used have experienced up to now. Fuel assemblies got stuck in storage cells deformed from swollen poison panels and pool water gets contaminated with silica from deteriorating Boraflex strips. As borated stainless steel has shown to be most durable under all relevant operating conditions of fuel pools, this neutron absorbing material is recommended to be used for poisoned high-density fuel storage racks.

3.1.6. Fuel reception and shipment

Operational efficiency of fuel reception is certainly important to a Spent Fuel Storage Facility for a number of reasons. One of the most important is to keep the operators exposure to radiation as low as reasonably achievable. It requires expeditious reception and transfer to the dedicated storage location. For this purpose, different options are available (see Fig. 6).

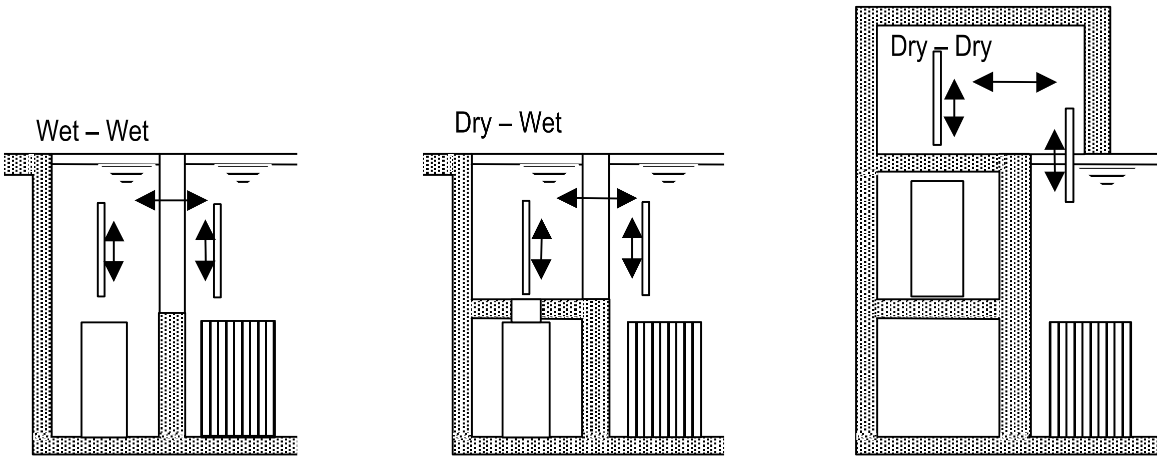

FIG. 6. Fuel reception methods.

Cask reception at pool storage facilities usually follows the logic applied when loaded at the Nuclear Power Station. Using the wet/wet loading process where casks are submerged into the fuel pool or separate cask handling pools, the arriving cask is cooled down to be refilled with water before opened. Then, unloading is performed by means of a similar fuel handling crane than it was used in the nuclear power plant where this cask was loaded. Special attention has to be paid to prevent the outer surface from becoming contaminated with radioactive substances contained in the pool water since contamination in excess of limit values found at some of the transport equipment demonstrated the need for improvement of cleaning processes or better, improvement of respective protective measures.

Prevention of surface contamination is certainly improved using the dry/wet method for cask loading/unloading. In this case, the fuel shipping cask is placed underneath a usually separate cask loading pool and sealed against the loading opening located at the floor of this pool. Suitable sealing and respective flow condition of the ambient air during placement of the secondary lid may prevent casks from becoming contaminated when carefully handled. A third option is available with dry/dry cask loading systems where only airborne contamination needs to be prevented from being transferred to the fuel cask [6].

3.1.7. Decommissioning

As no contaminated pool water is circulated through cooling or purification systems, radioactive contamination remains limited to the fuel pool area itself and the fuel transfer equipment in the fuel reception bay. Hence, the amount of radioactive material to be disposed off remains extremely low. After shipment of the fuel to its final destination, the spent fuel storage building can be decommissioned using available decommissioning techniques.

3.2. Economics

As a matter of fact, initial investment is somewhat higher for the construction of the fuel pool storage facility. However, expenditure for storage racks and increasing cooling capacity can be distributed over a longer period as the need for storage and consequently cooling capacity arises.

The initial cost for a storage facility per fuel assembly decreases rapidly as the number of fuel assemblies to be stored increases, where the costs per fuel assembly in dry storage remains nearly constant with increasing number of fuel assemblies. Optimizing investment programs, pool storage facilities may brake even with dual purpose cask storage facilities at total storage capacity of less than 1 000 tU.

Wet storage facilities provide very high storage densities. Hence, land consumption is considerable low, which makes it very attractive for on site storage where space might be limited. Wet storage facilities require considerable less space then dry storage facility using casks. This may be of particular importance in cases were the storage facility has to be built on the site of a power reactor which might be desirable in order to avoid the licensing of a new site away from the reactor.

As discussed in section 0 the need to store spent fuel assemblies in storage pools for longer periods rises because of increased heat and neutron dose rate generation due to higher burn-up. At the time the reactor is shut down at end of life, all systems like cooling, air conditioning, water purification, demineralized water supply, radiation control, power supply and power supply backup have to stay in operation for 5 to 10 years until the fuel assemblies can be removed from the pool for dry storage. The costs for keeping the systems in operation are substantial. Additionally the nuclear license has to be maintained valid for the whole plant.

The operating costs for a passively cooled fuel pool facility system is considerably lower then for the oversized systems of the complete power station. The major cost impact results from security, an effort that does in fact not differ from dry storage facilities.

3.3. Goesgen NPP

The latest achievements in wet storage technology as described above are used in Framatome ANP's wet storage facility design as currently being performed for the new spent fuel pool building to be constructed at the Goesgen Nuclear Power Station in Switzerland [7]. In 2002 the Goesgen Nuclear Power Plant and Framatome ANP signed a contract to design and construct an inherent safe, passive cooled spent fuel storage building (see Figs 7 and 8). The storage capacity is defined as 1000 fuel assemblies. In the first step a storage capacity of 500 fuel assemblies will be installed. The main data are listed in Table I.

4. Conclusion

The advanced version of a pool storage facility provides best for safe and reliable storage of spent uranium and MOX fuel. Maintenance and repair concepts are available which allow predicting, if correctly applied, such fuel pools to be operable for extended periods of time. A storage facility using the concept described above is an attractive alternative from technical and financial point of view compared with dry storage facilities.

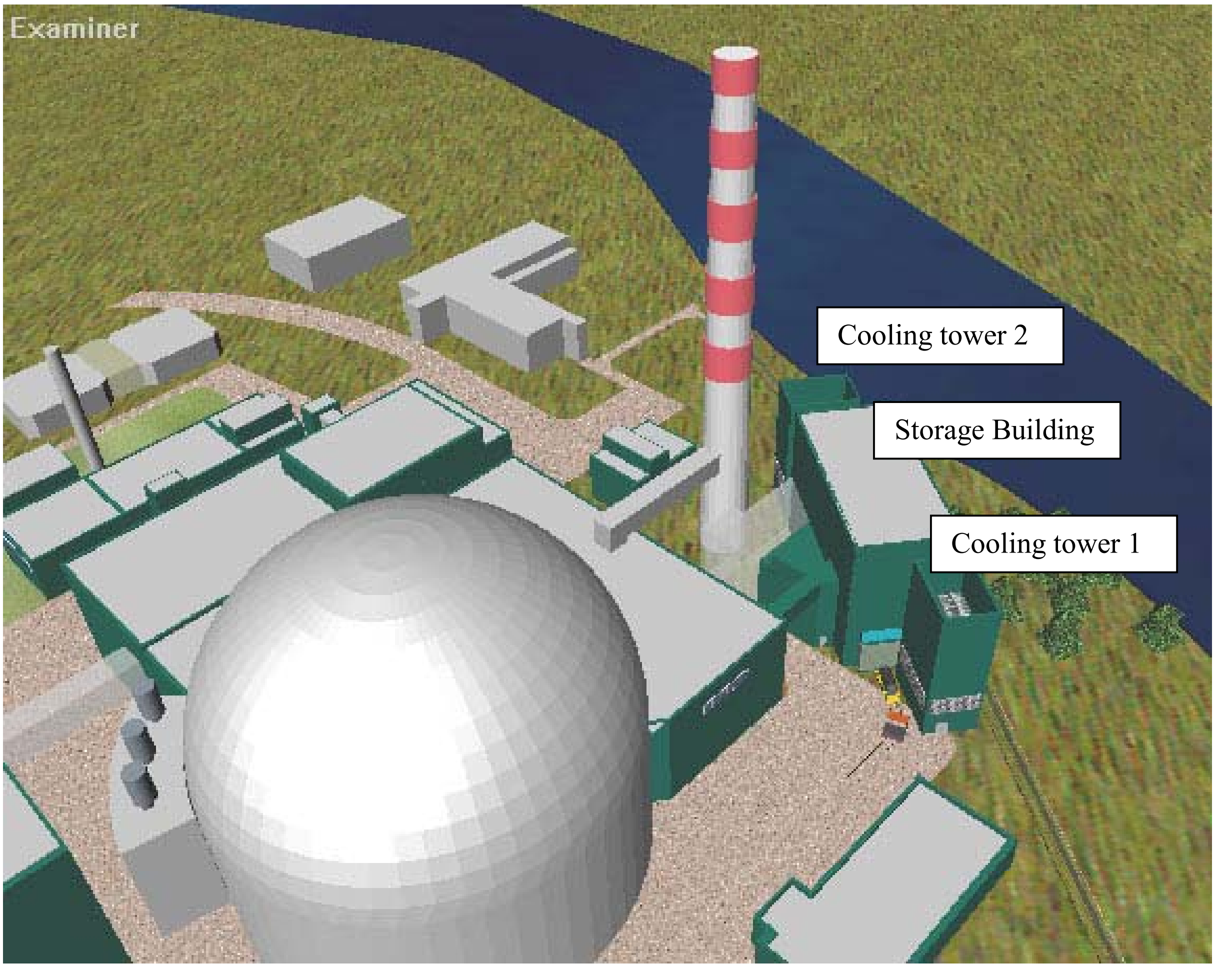

FIG. 7. Overview of the Goesgen NPP with the planned spent fuel storage building.

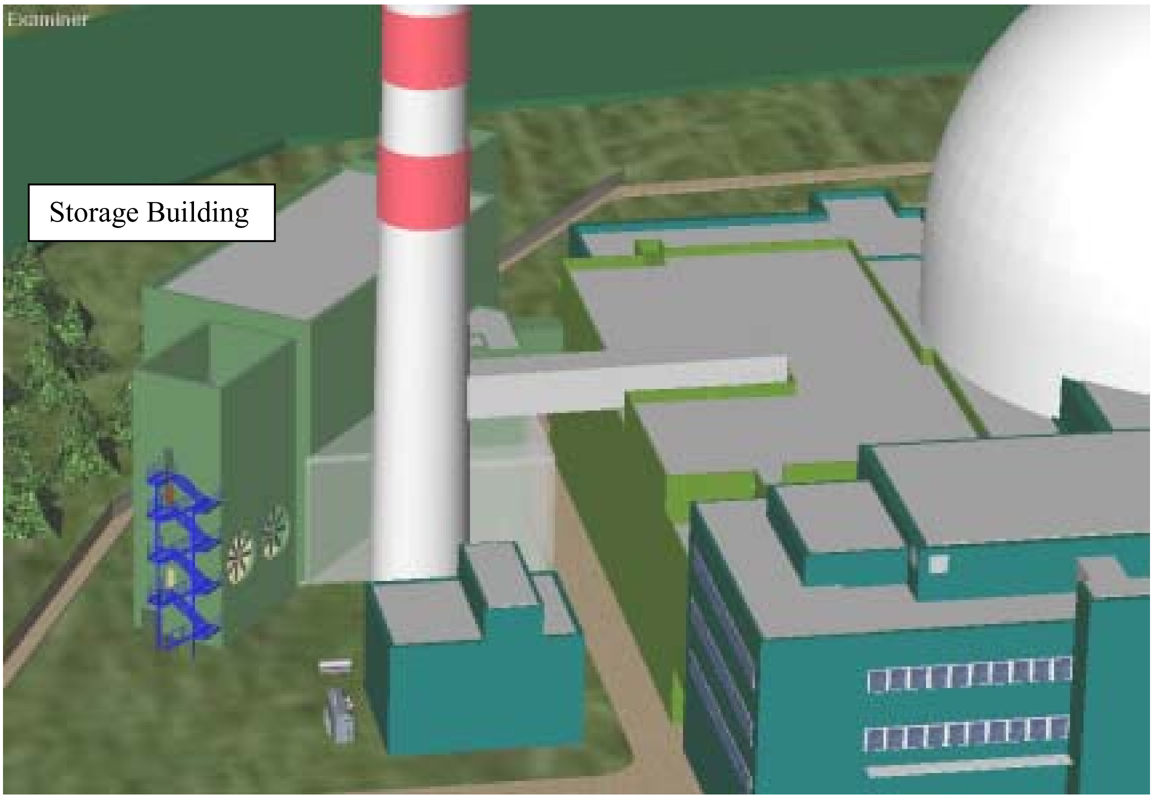

FIG. 8. Detailed view of the Goesgen NPP with the planned spent fuel storage building.

Table I. Technical data

Building size	35.5 m × 17 m
Pool size	13.2 m × 8,2 m
Storage capacity	Step 1: 504 fuel assemblies Step 2: 1 008 fuel assemblies
Enrichment	U-Fuel: max. 5% U-235 MOX-Fuel: max 4.8% Pu_{fiss}
Poison Material	Borated Stainless Steel (1.75 wt-% B)
Minimum decay time	U-fuel: 3 years MOX-fuel: 5 years
Cooling capacity	Step 1: 0.5 MW Step 2: 1.0 MW
Coolant	Demineralized water without soluble boron

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Management of spent fuel from power and research reactors using CASTOR® and CONSTOR® casks and licensing experience in Germany

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Abstract. The German company GNB has developed, tested, licensed, fabricated, loaded and transported a large number of casks for spent fuel and high-level waste. Meanwhile CASTOR casks are used at 19 sites on four continents. Up to now, more than 680 CASTOR casks have been loaded for long term storage. The two decades of storage have shown that the basic requirements, which are safe confinement, criticality safety, sufficient shielding and appropriate heat transfer have been fulfilled in each case and, of course, the experience of 20 years has resulted in improvements of the CASTOR cask design. Starting in the middle of the nineties, the new GNB cask line CONSTOR was developed with special consideration to an economical and effective way of manufacturing by using conventional technologies and common materials. The cask concept also fulfills all design criteria for transport and storage given by the IAEA recommendations and national authorities. By the end of 2002 forty CONSTOR casks have been delivered and 30 of them have successfully been loaded and stored. In the past the German disposal concept was mainly related to reprocessing of the spent fuel in France and UK and further storage of the high active waste (HAW) in the existing central intermediate storage facilities. However, in the consensus with the German Government the German utilities declared a change in their spent fuel policy from reprocessing to dry storage and direct disposal. For the intermediate on-site dry storage different storage facility solutions are planed, mainly storage halls, but in one case a subsurface storage tunnel. To overcome the time up to the erection of these long term storage facilities some utilities applied for and got short-term storage licences. For this, horizontally stored casks are shielded by mobile concrete covers.

1. Introduction

During the seventies a first idea of dry storage of spent fuel in casks arose at the GNB mother company GNS. The well-known CASTOR® cask design with ductile cast iron (DCI) as cask body base material was developed for the dual purposes of storage and transport. After only five years of developing and testing, the first storage license was granted for four types of CASTOR® casks at the centralized storage facility Gorleben.

Meanwhile, spent fuel assemblies of the types PWR, BWR, VVER, RBMK, MTR and THTR as well as vitrified high active waste containers (HAW) are stored in these kinds of casks. By now more than 680 CASTOR® casks have been loaded and stored, more than 900 have been delivered and more than 1 000 have been delivered and ordered.

2. GNB´s cask designs

2.1. General

In Europe, CASTOR® casks are used for transport and storage of spent fuel, as well as for the return of residues from reprocessing. The contents include Western and Eastern European fuel types. The largest number of one single cask type (459 CASTOR® THTR/AVR) is used for storage of the whole fuel of two Thorium high temperature reactors.

Vitrified residues from reprocessing have to be returned from France to Germany corresponding to about 2 500 high-level waste canisters or about 90 casks. At present, they are transported with the CASTOR® HAW 20/28 CG. An additional 700 high-level waste canisters will be returned from Great Britain.

To reduce costs for the customer, GNB started the development of a CONSTOR® cask, using steel and heavy concrete as basic material in the first half of the nineties.

2.2. Design characteristics of the CASTOR® casks

CASTOR® casks fulfill both the requirements for type B packages according to IAEA regulations and the requirements covering different accident situations to be assumed at the storage sites. The cask body is made of ductile cast iron. In order to improve the neutron moderation axial boreholes are drilled into the cask wall which contain moderator rods made of polyethylene (see Figure 1). As protection against corrosion, the inside surface of the cask and the sealing surfaces are provided with a nickel coating. The outside surface is protected by a coat of paint on an epoxy-resin basis. The basket for spent fuel assemblies basically consists of tubes made partly of borated stainless steel or combinations of steel and aluminium. On the outside wall of the cask, radial cooling fins can be machined to improve the heat transfer from the cask to the environment. The inner cavity of the cask is closed by a primary lid and by a secondary lid. The space between the lids will be filled by helium under overpressure; the control of this pressure delivers the tightness information during storage. Four trunnions are used for cask handling, fixing the cask in a transport frame. For transport purposes shock absorbers are used.

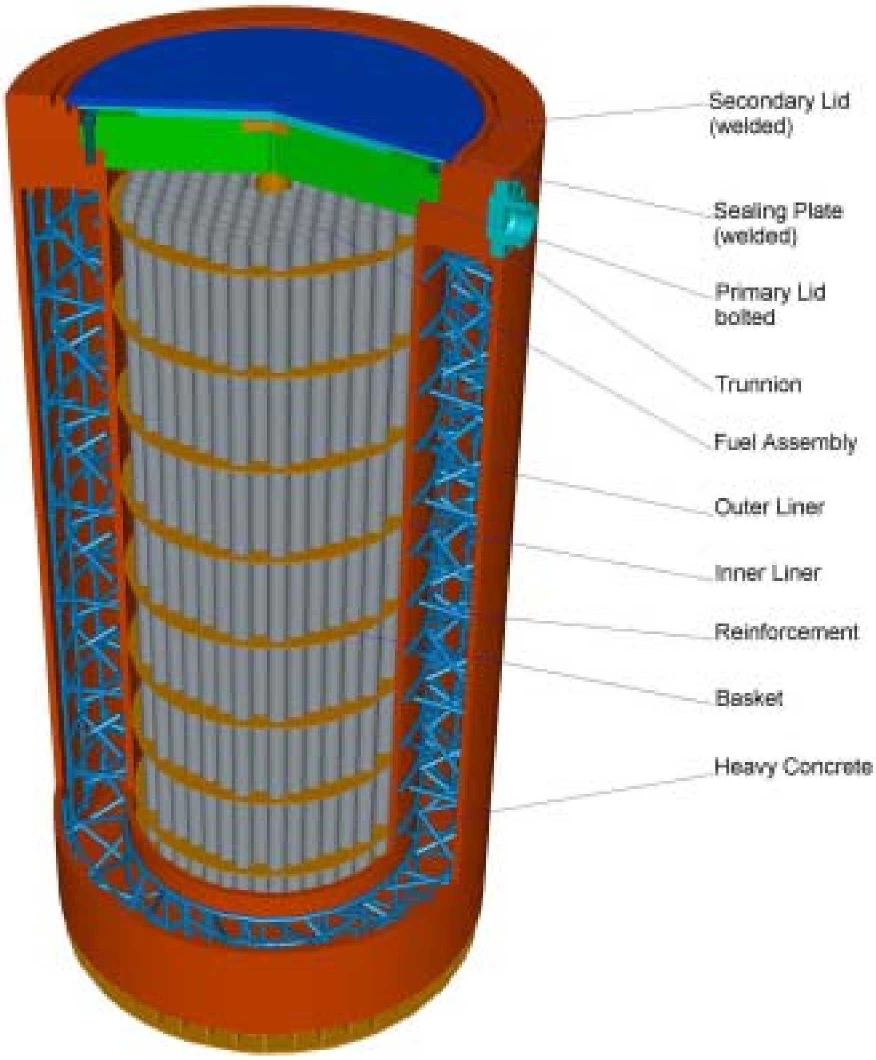

FIG. 1. Design characteristics of the CASTOR® casks.

2.3. The CONSTOR® concept

The CONSTOR® concept was developed especially for an economical and effective way of manufacturing by using conventional technologies and common materials. Nevertheless, the CONSTOR® sandwich cask concept fulfills both the internationally valid IAEA criteria for transportation and the criteria for long term intermediate storage.

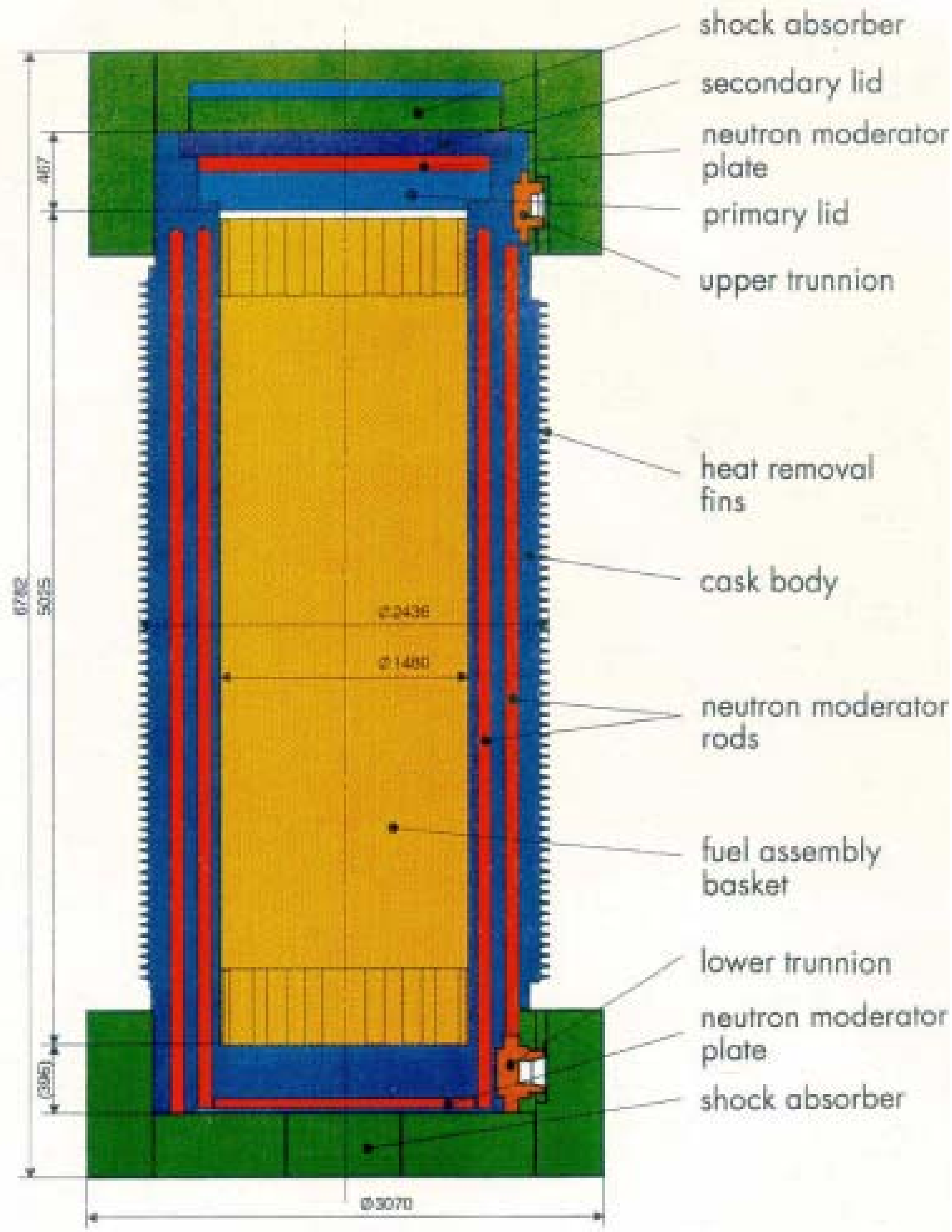

FIG. 2. Design characteristics of the CONSTOR® casks.

The CONSTOR® concept consists of a sandwich design with an outer and inner shell made of steel (see Figure 2). The space between the two shells is filled with heavy concrete for gamma and neutron shielding. The design does not rely on the concrete for structural integrity. Inside the concrete, steel reinforcement is arranged to improve strength and heat removal properties. The cask bottom has the same sandwich design as the wall. At the upper end, the shells are welded to a ring made of forged steel. The lid system is designed as a multibarrier system. The bolted and sealed primary lid fulfills strength, shielding and temporary sealing functions. The sealing plate and the secondary lid are welded to the forged steel ring after loading of the cask. These two welded lids together with the inner and outer shell constitute the double barrier system. Alternatively it is possible to bolt the primary and secondary lid.

The analyses of nuclear and thermal behaviour as well as of strength according to IAEA examination requirements (9 m drop, 1 m pin-drop, 800°C-fire test) and of the behaviour during accident scenarios at the storage site (drop, fire, gas cloud explosion, side impact) were carried out by means of validated calculational methods and programmes. In a special experimental programme, the mechanical and thermo mechanical properties of heavy concrete were verified and the reference values required for safety analyses were determined.

The results of the safety analysis after drop tests and the fire test according to IAEA-regulations as well as after 1m-drops at the storage site were confirmed by means of an extensive test programme using a 1:2 scale model. The post-test inspection programme of the model cask has shown that the cask integrity and leak tightness were maintained after the series of 6 drop tests.

2.4. Outlook on further development

The challenge for further development results from:

- higher technical specification, particularly related to fuel (enrichment, burnup);
- cost reduction;
- increase of licensing requirements.

The first two aspects are a clear consequence of the market condition the utilities are faced with worldwide. The latter aspect serves the need for keeping design and proof of the design state-of-the-art.

Concerning technical specification increase in enrichment and higher burn up are the most challenging issues along with disposal management of spent MOX assemblies in some countries. As a consequence higher heat capacity and sophisticated shielding measures have to be considered. Besides design and new materials even new methodologies for the proof of the design have to be developed and applied.

In addition, defect fuel disposal is increasingly requested on the market, which has a clear impact on design feature, not only of the cask internals, but also of handling equipment.
Related to cost an increase of the number of fuel assemblies per cask as well as a cost optimized cask design are the most important approaches of the cask vendors. Validated methodologies for the proof of the design during the licensing process is important even in terms of cost.

3. Cask loading and storage experience

Loading performance is a key issue to establish Spent Nuclear Fuel (SNF) management. For this, development of each GNB-cask includes the participation of plant operators and experienced loading staff starting with the first conceptual design phase, e. g. by elaboration of handling studies. Further verification is given by cold trials of both, casks and cask related equipment as well as design reviews and/or prototype loadings with extensive measurement programmes. The cold trials, as an example, can be performed first outside the reactor, e.g. at special test facilities available at the cask manufacturer's shop under participation of the relevant parties of the NPP operators and/or experts of the competent authority. This can be followed by performing of the complete handling sequence of the cask and the correlating equipment in the NPP for verification of proper function including fit of all plant specific interfaces. Furthermore, review of cold trials and prototype loadings open potential for further optimization of procedures e.g. with respect to minimization of collective dose per loading (ALARA).

This way for establishing standard loading procedures has been already implemented for the complete range of cask loading conditions, as there are:

- Dry remote loading;
- Wet loading / dry transport and storage with bolted and welded lid systems;
- Fuel loading via locking gates;
- Wet loading / wet transport.

In two decades CASTOR® and CONSTOR® casks have been successfully introduced for spent fuel management at many NPP sites. In Figure 3 storage of CASTOR® casks in the Gorleben storage facility is shown.

FIG. 3. CASTOR® casks in Intermediate Storage Gorleben/Germany.

Independent of the kind of storage there are general acceptance criteria, which have to be fulfilled after loading:

- Accordance of the loaded fuel with the licensed specifications;
- Drying of the cask cavity to the specified moisture limits;
- Testing of the primary and secondary lid with respect to specified tightness;
- Determination of the outer radiation dose rates;
- Determination of the contamination of the cask surface;
- Determination of the surface temperature.

Presently, altogether more than 700 casks will be stored at 19 sites. This results in storage experience of around 4 800 cask·years. Because there are two or three seals per lid (in addition to the main seal there are seals for closure lids for dewatering, drying and gas filling bore holes) GNB has acquired a seal experience of more than 24 000 seal·years.

All this experience has shown that the safety requirements for the casks will be fulfilled without any problems. Over the last two decades only one metallic seal lost the required tightness. For such a case maintenance actions are prepared.

4. Licensing experience of CASTOR® casks in Germany

Since the federal election in 1998 the German energy policy has changed. For instance the government and the operators of nuclear power plants reached a consensus regarding the disposal of spent fuel elements of the NPP in Germany.

In the past the disposal concept was characterized by the reprocessing of the spent fuel in France and UK and alternatively the storage of it in the existing central intermediate storage facilities Ahaus and Gorleben. The storage in these central storage facilities always required transports within Germany. Considering the fact that the transports were often a target of violent resistance it became a policy of the government to avoid such transports. To minimize the number of transports the operators of NPP declared in the consensus that they will build decentralized, on-site intermediate dry storage facilities for spent fuel elements. To overcome the time up to the erection of these long term (intermediate) storage facilities some utilities applied for and got short-term (interim) storage licences. For this, horizontally stored casks are shielded by mobile concrete covers. Seventeen storage facilities have been applied for.

The licensing procedure of a storage facility consists of 2 parts:

- Licence (Operating permission) according to the atomic law (§ 6 Atomic Energy Act) granted by the Federal Office of Radiation Protection (BfS);
- Licence (Building permission) according to the respective federal state building regulations granted by the local authorities.

The course for the licensing procedure pursuant to § 6 Atomic Energy Act includes the following steps:

(1) Application and submission of the required documents:
  - Safety analysis report;
  - Short description of the facility;
  - Environmental impact assessment documents;
  - Documents proving the fulfilment of the licence requirements including supporting documents, plans and drawings of the cask and the building;

(2) Public announcement and public hearing;
(3) Check of all licence requirements by the competent authority and external experts;
(4) Draft of licence;
(5) Coordination with the supervision authority and final coordination with the Federal Ministry of Environment, Nature Protection and Reactor Safety;
(6) Granting of the licence.

Fig. 4 shows two types of the storage buildings, which have been applied for. In addition to that one subsurface facility will be erected in a tunnel. The documents proving the fulfilment of the licence requirements for the cask are grouped into three parts (Table I):

- Cask description and safety analyses;
- Construction;
- Handling documents.

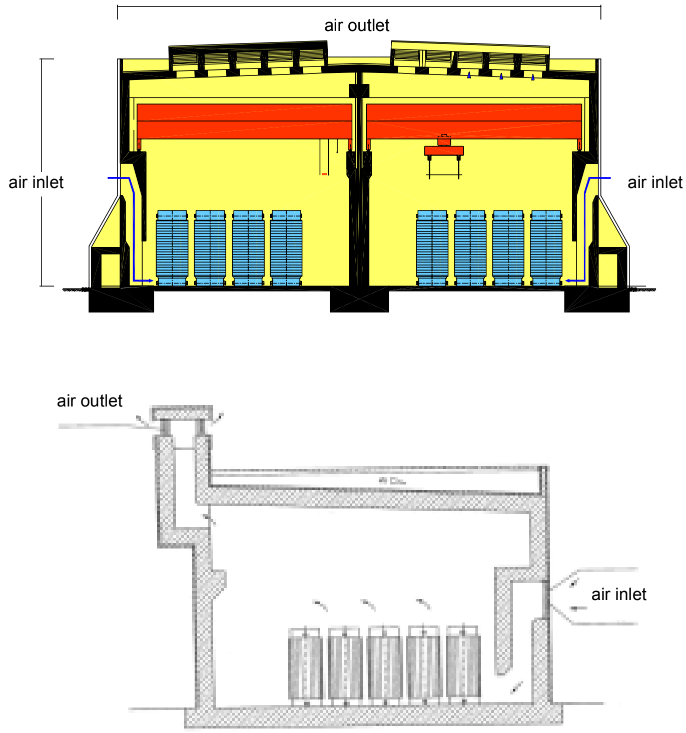

FIG. 4. Concept of the storage building.

Table 1. Structure of application documents

Cask description and safety analyses	**Construction**	**Handling documents**
- Description of the cask, the inventory and the handling - Structural analyses - Thermal analyses - Description of containment and release considerations - Cask shielding calculation - Subcriticality analyses - Long term behaviour of components - Safety analyses for normal operation - Safety analyses for accident conditions - Safety consideration for events in the residual risk range	- Part list - Drawings - Material specifications	- Work instructions · Cask drying · Protective coating · Contamination protection - Test specifications · Dose rate measurement · Contamination measurement · Temperature measurement · Tightness measurement - Assembly specifications - Operating plans

With additional documents, the differences between the types of storages, e. g. with respect to heat removal, will be taken into account. Since the consensus in 2000, three On-site Interim Storage Facilities and three Intermediate Storage Facilities have been licensed. One Interim Storage Facility got the 2nd partial licence. The licenses for the other facilities are expected in 2003, see Table II. Up to now three interim and one intermediate storage facilities have started operation. In these 37 casks are stored now.

Table II. Application for on-site storage facilities

NPP	Date of Application	Type of Storage Facility	Number of Storage Positions	Licence Date of Issue
Biblis	23.12.1999	Intermediate	135	
	30.11.2000	Interim	28	20.12.2001
Brokdorf	20.12.1999	Intermediate	100	
Brunsbüttel	30.11.1999	Intermediate	80	
	15.08.2000	Interim	18	
Grafenrheinfeld	23.02.2000	Intermediate	88	12.02.2003
Grohnde	20.12.1999	Intermediate	100	20.12.2002
Gundremminge	25.02.2000	Intermediate	192	
Isar	23.02.2000	Intermediate	152	
Krümmel	30.11.1999	Intermediate	80	
	15.08.200	Interim	12	
Lingen	22.12.1998	Intermediate	130	06.11.2002
Neckarwestheim	20.12.1999	Intermediate	151	
	20.12.1999	Interim	24	10.04.2001/21.12.2002
Philippsburg	20.12.1999	Intermediate	152	
	20.12.1999	Interim	24	31.07.2001/17.02.2003
Unterweser	20.12.1999	Intermediate	80	

5. Conclusion

GNB is one of the global leaders with a very large amount of worldwide experience with casks for transport and storage of spent fuel and high level waste. Transport and dry storage of spent fuel and high active waste in CASTOR and CONSTOR casks are a proven and extensively applied technology. Experience over two decades with almost all types of fuel and high active waste has been gained. GNB´s casks of CASTOR and CONSTOR type fulfil highest safety requirements in terms of both transport and storage. The future challenges can clearly be identified and will be coped by new designs of next generation of GNB´s casks.

Verification of dual-purpose metal cask integrity: *Verification tests*

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Abstract. The interim storage facilities planning in Japan will be off site of the reactors without hot-cell. Therefore, it should be important to confirm integrity of cask during storage and transport after storage. NUPEC has been conducting the dual-purpose metal cask verification test since 1999 under the sponsership of the Ministry of Economic, Trade and Industry (METI). The purpose of this verification test is to verify and establish the evaluation methods of structural integrity and safety functions of a dual-purpose cask during interim storage and under transport conditions after the storage. This paper presents "tentative requirements to be satisfied under the transport conditions after long term storage derived from the latest test data and the test contents to verify this requirements, in addition to general plan and current status of the test.

1. Introduction

Spent fuel which is generated by nuclear power plants (NPP) is designated as useful recycled resources and shall be properly stored until reprocessing according to the policy of Japan. Recently, the quantity of spent fuel stored at each NPP site is going to be increased due to finishing oversea reprocessing contract and delay of domestic reprocessing project. Therefore, the long term interim dry storage facilities of spent fuels using dual-purpose metal casks are expected to be early realized [1].

The interim storage facilities will be off site of NPP without hot-cell. After long term strage, dual-purpose metal casks will be transported to the reprocessing facility directly. Therefore, it is important to confirm integrity of casks during storage and transport.

NUPEC has been conducting the dual-purpose metal cask verification test project since 1999 Japanese fiscal year under the sponsorship of the Ministry of Economic, Trade and Industry (METI). The purpose of this project is to verify and establish the evaluation methods of structural integrity and safety functions of a dual-purpose cask during interim storage and under transport conditions after the storage.

In order to achieve the above purposes, this project is composed of material property tests and system verification tests. In this paper, verification test results and their evuluation methods on seal integrity under the drop test conditions after the long term storage are mainly focussed.

2. Material property tests

2.1. Outline

The purpose of these material property tests is to verify and establish the evaluation methods about the degradation of materials of which cask consists. In these tests, the composition materials of the cask were classified into body materials, basket materials, neutron shielding materials, and seal boundary (a metallic gasket). The factors of aged degradation were assumed to be three factors (decay heat, radiation, and ambiance).

The factors and materials that it was possible to evaluate by published data were excluded from material property tests by NUPEC. Test matrix is shown in Table I.

Table I. Composition materials - deterioration factors matrix

	Decay Heat	Radiation	Ambiance
Body Metals	-	-	Corrosion & SCC
Basket Metals	Over Aging & Creep	-	Corrosion & SCC
Neutron Shieldings	Change of Composition	Change of Composition	[a]
Seal Boundary	Relaxation	-	Corrosion & SCC

[a] Not public information.

Each test with its objectives and current status is summarized in Table II. More detail information was presented in [2]. Following disscussions are mainly focuused on tests of seal boundary.

Table II. Objectives and current status of material property tests

TEST	OBJECTIVES	CURRENT STATUS
Body & Basket metals	To confirm the effects on the strength due to corrosion and SCC	No significant corrosion and SCC by iodine compound in 60 year-storage.
	To confirm the effects on the strength due to over ageing and creep	No significant difference between aluminum and borated aluminum alloy for basket materials by over ageing treatment
Neutron Shieldings	To establish the evaluation method for the degradation of shielding	Regression equation for weight loss of epoxy resin predicted by LMP has reasonable precision during long term storage.
	To confirm chemical changes due to heating and irradiation	No significant weight loss for shielding due to heating and irradiation under closed circumstances for epoxy resin
Seal Boundary	To establish the evaluation method for the degradation of sealing	Refer to clause 2.2.
	To propose the requirements	

2.2. Seal boundary (metallic gaskets)

2.2.1. Tentative requirements

The tests concerning the stress relaxation property of metallic gaskets and the followability to the displacements of seal boundary assumed to be occurred under the drop and/or fire conditions were performed. The tests concerning the followability were performed by both dynamic and semi-static conditions. Detail description about these tests were presented in [3]. As a result, the tentative requirements under the drop and fire conditions were proposed in Table III.

Table III. The Tentative Requirements under the drop and fire conditions

Items	Requirements
Spring Back Force	1kN/cm (after storage)
Opening	(No requirement)
Sliding	Design target of movable range for semi-static displacement: 0.5mm(max. displacement:about 1mm)

Note: Surface roughness on sealing surface : Ra 0.8 or less

2.2.2. Relaxation tests

The study on aged gasket property was conducted focussing on leak tight performance under transport condition. From this study, the followings were obtained.

The relation between spring back force of aged metal gasket and Larson-Miluar Parameter (LMP) was obtained. Here, LMP is defined as follws:

$$\mathbf{LMP = T \times (C + \log t)} \quad \textbf{(1)}$$

where

T : Absolute temperature (K)
C: = 14 (Const.)
t: Time (hr)

From this test, the spring back force of age metal gasket was able to be estimated and it would be estimated to be more than 1 kN/cm, that is mentioned below, after actual 60 years storage.

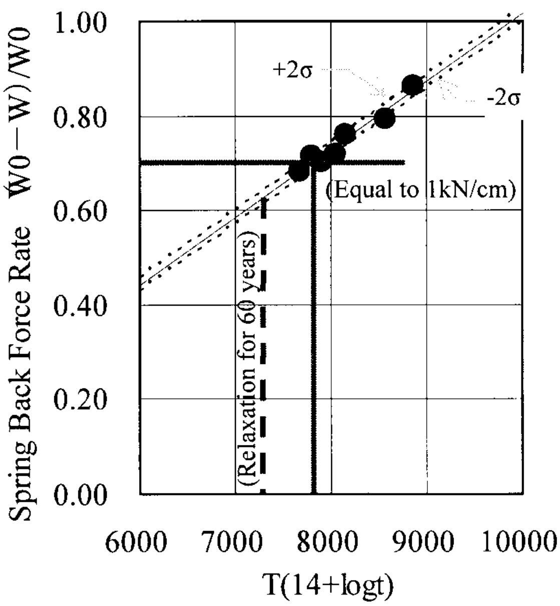


FIG. 1. Relaxation tests results.

* Results of Single Type Gasket
* Strength of Outer Shell Material (Al Alloy) 65 < Su < 95 MPa

Here, actual 60 years storage is equivalent to LMP = 7 380 that is derived considering the temperature decrease during storage. As shown in Fig. 1, the LMP value to keep spring back force over 1 kN/cm is about 7 800 taking range of 2 sigma into account.

2.2.3. Semi-static displacement tests (vertical direction /parallel direction)

Concerning vertical displacement, lid opening tests were conducted for various aged gaskets. Fig. 2 shows the relationship between the leak rate and spring back force after once opened flange. The leak rate was less than 10^{-5} Pam^3/sec against LMP equal to actual 60 years storage and over even though lid and gasket made a clear gap temporarily, if spring back force was more than 1kN/cm. It is well understood that the minimun spring back force is very important in order to keep the leak rate requirement.

Concerning parallel displacement, tests were performed changing the relative displacement between lid and flange as a parameter. Leak rate was steady for the displacements from 0 to 0.5 mm and it's rising gradually over about 0.5 mm of displacement against any LMP (see Fig. 3). Therefore, the range from 0 to 0.5 mm was set as the design target of movable range of lid for full scale cask model of drop test.

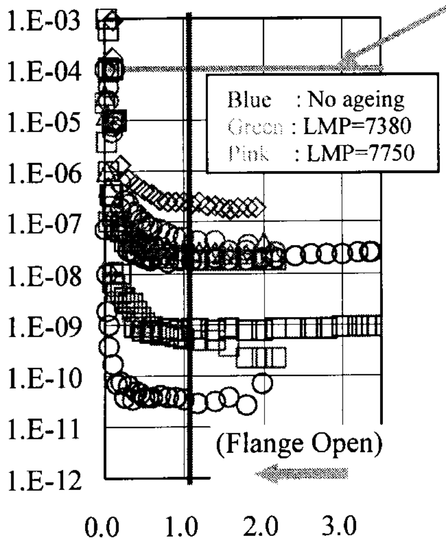


FIG. 2. Semi-static vertical displacement tests (after temporarily open).

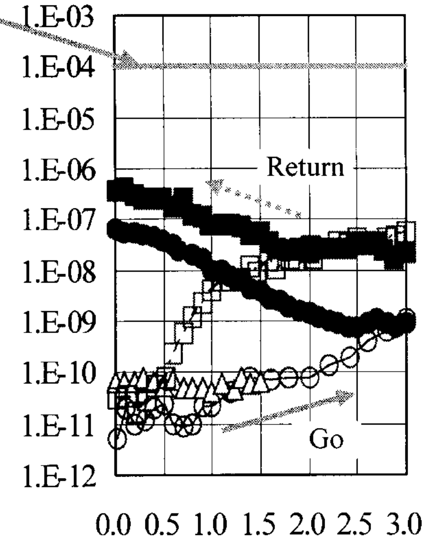


FIG. 3. Semi-static parallel displacement tests (LMP=7380).

[NOTE] * Results of Double Type Gasket

** This is 1/10 of an example of design leak rate of typical domestic transport dry cask because of the difference in mean dia. of gasket. Leakage of radioactive nuclides would be estimated less than 1/100 of A2 value even if the leak occurred the same rate as above under the condition II of that cask.

2.2.4. Dynamic displacement tests (vertical direction /parallel direction)

Dynamic displacement tests were conducted using a pendulum-type impactor and 1/10 scale model seal boundary. As shown in Fig. 4, concerning vertical displacement, leak rates were less than 10^{-5} Pam^3/sec against LMP equal to actual 60 years storage and over hardly depending on the displacement, of cause even though lid and gasket made a clear gap momentarily. Results about parallel displacement, leak rates were less than 10^{-5} Pam^3/sec until about 3 mm and were hardly depending on LMP values in stable. Besides, all flanges used the above tests were made the roughness (Ra) of sealing surface less than 0.8.

3. System verification tests

3.1. Outline

The purpose of system verification tests is to confirm the integrity of safety functions during storage and transport in comparison with regulation criteria and the evaluation methods. The applicability of the tentative requirement in clause 2.2 is scheduled to be confirmed under the special test conditions in these tests. For this purpose, system verification tests shall be performed with the full-scaled model under the special test conditions (9m drop test & fire test). Each test with its objectives and current status of system verification tests is summarized on Table IV.

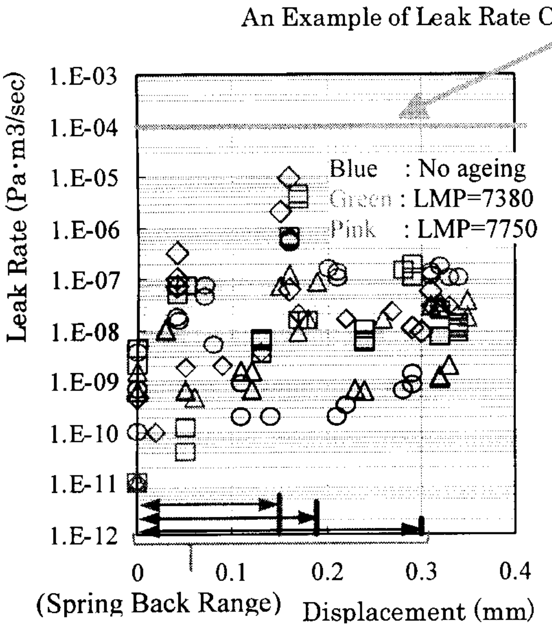


Figure 4 Dynamic vertical displacement

[NOTE] See NOTE in Figure 3 & 4

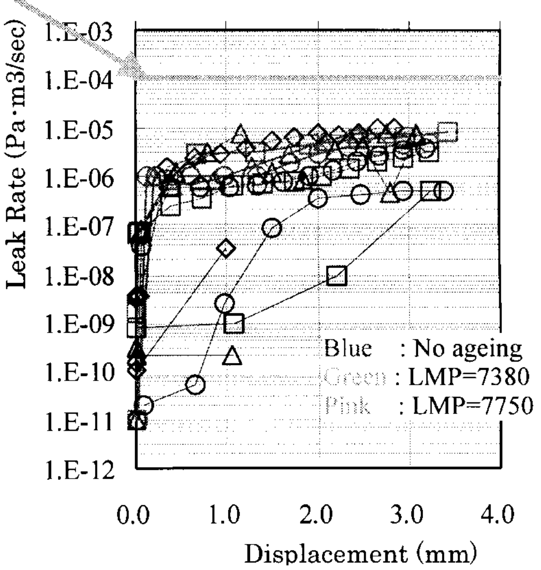


Figure 5 Dynamic parallel displacement

Table IV. Objectives and current status of system verification tests

TEST	OBJECTIVES	CURRENT STATUS
Drop test	Confirmation of the sealing ability with aged metal seals under the drop test conditions	1/3-scaled model drop tests Designing and manufacturing a full-scaled model
	Verification of assumptions, modeling and methods, which used in the dynamic impact analysis	3D dynamic impact analysis by LS-DYNA
Thermal test	Confirmation of the sealing ability with aged metal seals under the simulated fire test conditions	Design and manufacturing a full-scale lids section model Confirmation of the aging methods for metallic gaskets
	Verification of assumptions, modeling and methods, which used in the thermal and structural analysis	3D thermal and structural analysis by ABAQUS

In this paper, confirmation and verification of integrity of seal boundary under the drop test conditions are mainly disccussed in following section. For these purposes, it is necessary to confirm the accuracy of the analysis needed in evaluating safety. Especially confirmation of the accuracy of analysis for sealing behavior is interested and required.

3.2. Drop tests

3.2.1. Purposes and test contents

The purposes of the drop tests are the following two points for which the verification is necessary for the safety evaluation of the dual-purpose metal casks:

1. To comfirm the applicablity of the tentative requirements o actual casks;
2. To comfirm the accuracy of analysis.

The tests are executed in the following steps, in order to comfirm the aboves:

1. 1/3-scaled model drop tests;
2. Design and fabrication of a full-scaled model;
3. Full-scaled drop tests;
4. Analyses and evaluation.

Here, it is described 1/3-scale model drop tests executed in 2002 Japanese fiscal year.

3.2.2. 1/3-Scaled model drop tests

3.2.2.1. Purposes

1/3-scale model drop tests had been executed for the undermentioned purposes before the full-scaled model drop tests:

1. Accuracy of measurements: To confirm the accuracy of measurements about displacements;
2. Accuracy of analysis: To confirm the accuracy of impact analysis for main parts;
3. Control of sliding: To verify the controllability of lid against sliding tightening by bolts.

3.2.2.2. Outline of 1/3-scale model drop tests

Using 1/3-scale model of typical dry cask, 9m drop tests were conducted. A top side vertical drop test and two times horizontal drop tests were executed. Horizontal drop tests were demonstrated under 2 different bolt tightening force conditions to verify the controllability by bolt tightening. Sliding displacement (parallel direction) of the secondary lid and openings (vertical direction) of the primary and the secondary lids, in addition to the accelerations and the strains were measured. Items and positions of measurements are shown in Fig. 6.

3.2.2.3. Verification analyses

Verification analyses were performed by using analysis code LS-DYNA that was general purpose nonlinear finite element program. Key characteristics of modeling are as follows. For modeling of bolts, solids element is used. As lid and body contact conditions, “Slide & Void (considering Friction Effect)” is selected. And also damping is considered in vibration of lids, bolts and flange. Analysis model is shown in Fig. 7.

3.2.2.4. Results and evaluations of 1/3-scale model drop tests

Typical examples of verification results are shown in Figs 8 to 10. It was confirmed that accuracy of measurements and analyses had enough accuracy about the parts measured by 1/3-scale model drop tests in order to evaluate the applicability of the tentative requirements.

A main factor of sliding is oval shaped deformation of body as shown in Fig. 11. And also it is clarified that it is difficult to prevent the oval shaped deformation by tightening force of bolts. Therefore, the full-scaled model was designed to control sliding by limiting the gap between the lid and the flange in order to satisfy the requirements in Table III.

3.2.3. Schedule for the future

At present, the full-scaled drop test model which will satisfy the requirements is being manufactured. The top side vertical drop test, the horizontal drop tests and the top side corner drop test will be executed. It is included to execute a horizontal drop test with the simulated degradation metallic gasket. Degradation will be executed by the aging method confirmed in 2002 fisical year.

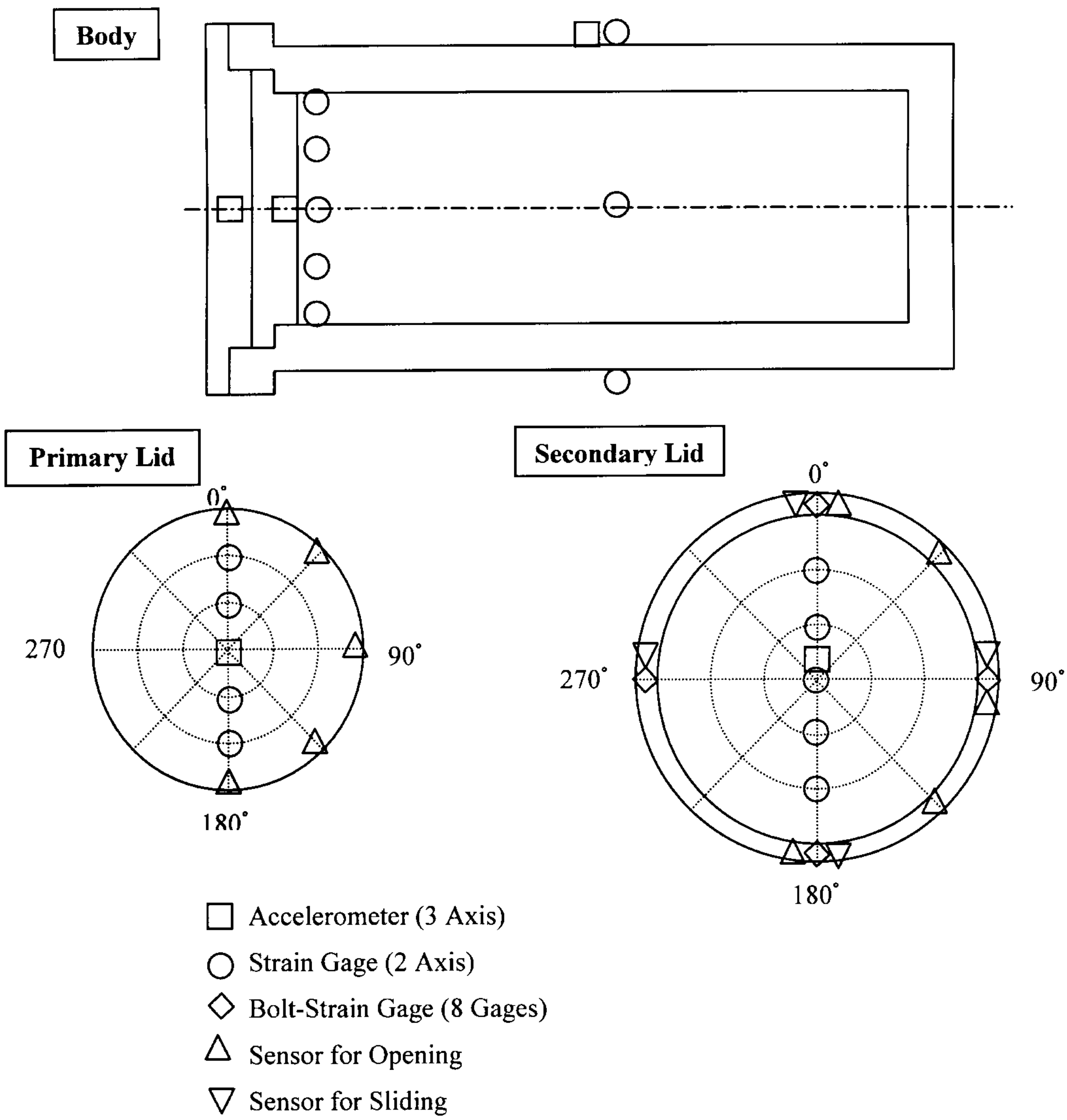
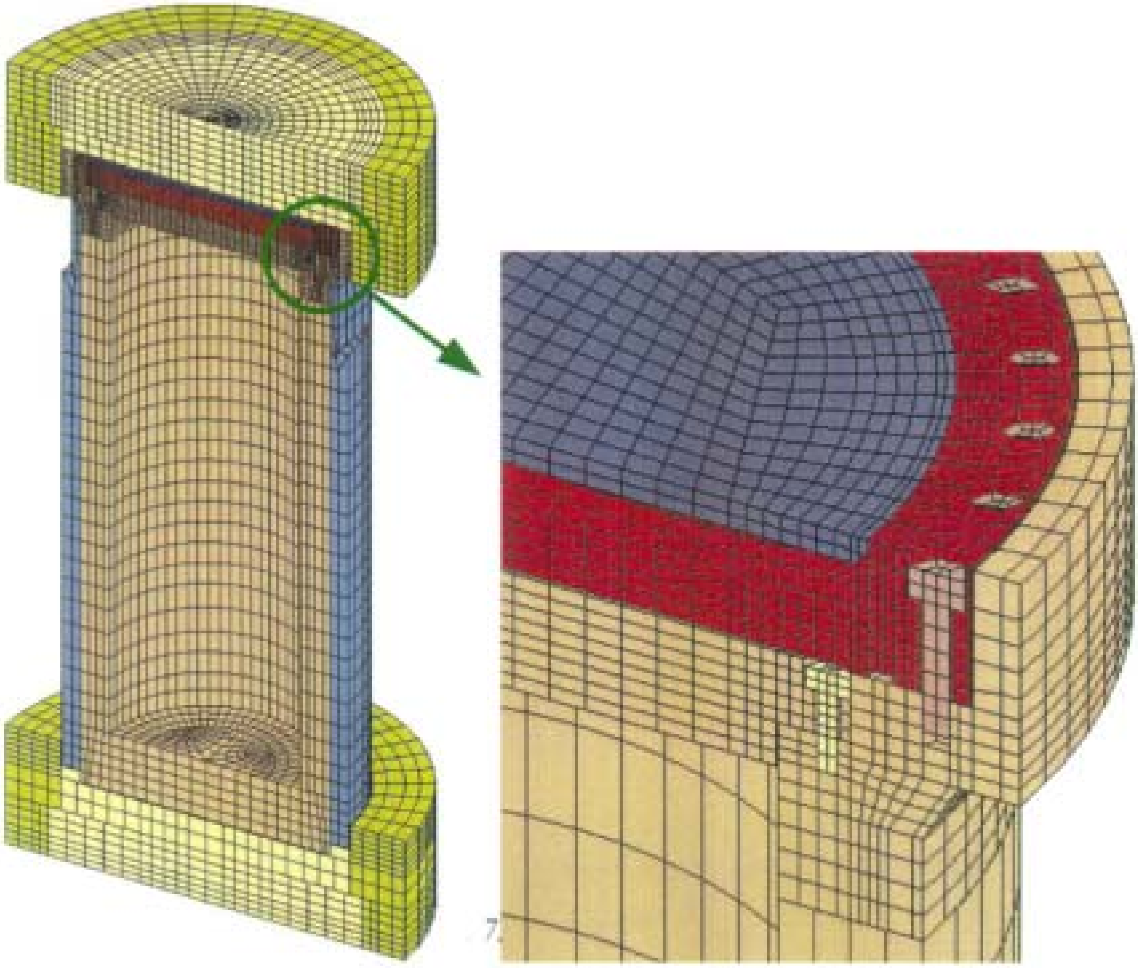


FIG. 6. Items and positions of measurements.



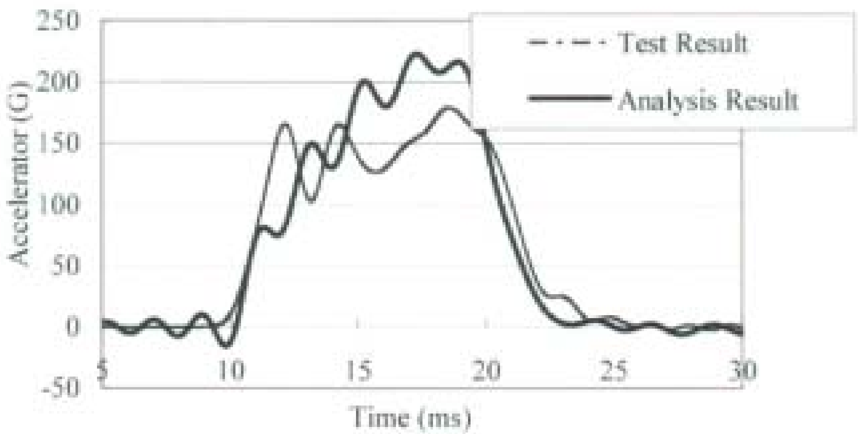


FIG. 8. Acceleration.

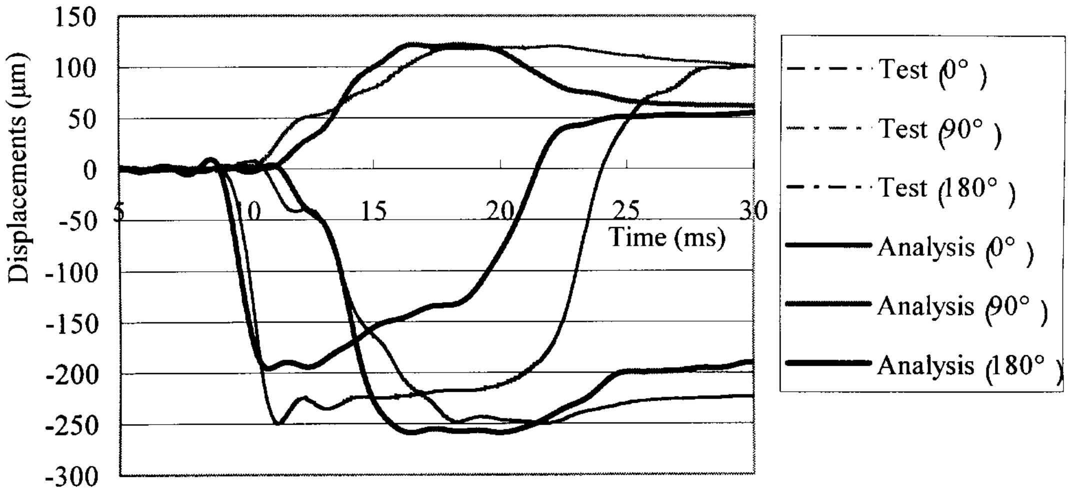


FIG. 9. Sliding.

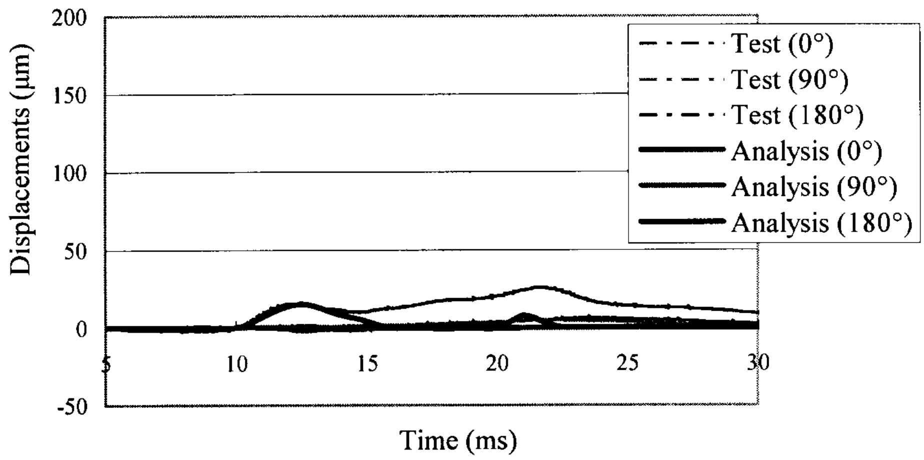


FIG. 10. Opening.

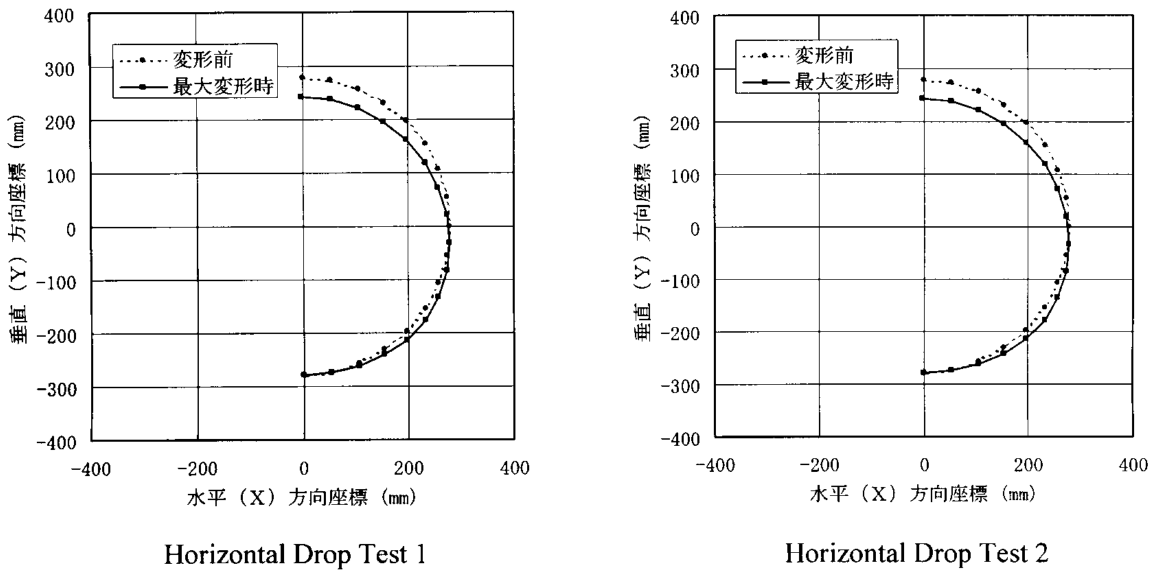


FIG. 11. Oval shaped deformation.

4. Conclusions

The following conclusions were obtained from the above-mentioned verification test results:

1. The performance of metallic seals is degradated because of the stress relaxation by a long term storage. However, if the one of the line diameter of φ10 is used, it is confirmed that a necessary performance will be maintained after a long term storage;
2. For transportation after a long term storage, it is necessary to suppress opening and sliding of the second lids taking into account of the performance degradation of the metallic gasket seal. By 1/3-scaled model drop tests, it is confirmed that it will be possible to satisfy the tentative requirements for opening and sliding behavior of lids. And also it is confirmed that the analyses have a capability to estimate these various key features;
3. The above-mentioned conclusions shall be finally confirmed by the full-scaled model drop tests in the future.

ACKNOWLEDGEMENTS

The authors are grateful to the members of “Spent Fuel Storage Technology Verification Test (Dual-purpose Cask Technology Verification Test)”, who gave us useful advice and comments about the plans, implementation and result evaluations of this testing.

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Multi-purpose canister storage of spent nuclear fuel in modular vault system
Technology development

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Abstract. The original Modular Vault Dry Storage (MVDS) technology was developed in the early 1980s leading on from the experience gained with the magnox fuel dry storage facilities at the Wylfa power station in Wales (UK). The Wylfa dry fuel stores were commissioned in 1969 and the MVDS can, therefore, rightly claim to be the only dry fuel storage technology that has an operational and technological background of over thirty years. The MVDS system was originally designed to store individual fuel assemblies within a Storage Canister. This system ensures minimum fuel storage temperatures and provides maximum flexibility for future off-site transportation. Individual fuel assemblies can be removed from their storage locations and placed into a transportation cask for either road or rail off-site shipment. However, this requires each fuel assembly to be re-handled and transferred into the transportation cask. A vault storage system based on the proven MVDS technology using a large Multi-Purpose Canister (MPC), is now being developed to provide a cost-effective interim spent fuel storage system. Integrating the MVDS technology with a MPC and adapting the cooling, shielding and handling systems, allows the new vault storage system to provide high storage efficiency in compact storage buildings suitable for a large spent fuel interim storage facility. It has been possible to re-configure the vault storage array from individual fuel assembly storage canisters to large diameter canisters, and to re-configure the handling equipment to transfer large canisters. By modifying the design of the MVDS to accept large, multiple fuel assembly, multi-purpose canisters, it has been possible to maintain the technical and operational benefits of the original MVDS design, with the additional benefits of multi-purpose canisters.

1. Introduction

Dry storage of spent nuclear fuel utilizes two main types of technology: casks or vaults. Both technologies are safe, proven and in commercial operation around the world. Vault storage systems are suitable for at reactor storage or for away from reactor storage. Typically vaults are able to store large amounts of spent nuclear fuel in a compact footprint and can be operated all year round independent of outside weather conditions. Spent fuel is stored securely within a vault, as access is security controlled into the facility, and specialized equipment has to be used to move loaded fuel canisters. Fuel can be delivered to a vault by use of an on-site transfer cask or by road/rail transportation cask, either as bare fuel assemblies or canistered fuel. Similarly, fuel can be transported away from a vault by loading the fuel from the vault storage locations into a road/rail transportation cask.

Examples of the Modular Vault Dry Store (MVDS) system are in commercial operation in U.S.A., U.K., and Hungary. Spent nuclear fuels in these facilities are stored within small diameter containers that are up to 0.61 metre diameter. The MVDS at Fort St Vrain in the USA utilizes a transportable canister that can be placed directly into a transport cask when the fuel is removed from the storage facility; while the MVDS at Paks, Hungary stores individual fuel assemblies in storage canisters that are not transportable.

The MVDS technology is now being developed to enable high storage efficiency and compact storage buildings with low building height, by using large Multi-Purpose Canister (MPC) and improvement of handling system. The vault structure can be located above ground, or below ground. The use of a subterranean vault can also be used to provide additional shielding benefit, as well as enhanced physical protection. The concept of a modular vault storage system with large MPC has been studied, and it is concluded that the design ensures that safety considerations for cooling, shielding, confinement, criticality and seismic are maintained. The main characteristics of this system are as follows:

- Cooling: Based on proven horizontal passive cooling system, the height of storage area is lowered by improvement of the cooling system suitable for large MPC;
- Shielding: Underground storage and radiation streaming reduction structures can reduce the radiation dose at site boundary. Also, providing thick concrete shielding of the vault roof above canisters minimizes expected dose of workers;
- Confinement: A fully welded MPC is used to ensure confinement;
- Canister Handling: Transfer operations use a seismically qualified Canister Handling system. The facility building above ground is minimized by compact canister handling system;
- Seismic qualification: The storage facility building and canister handling equipment are fully seismically qualified. For high seismic zones, the use of the subterranean vault minimizes seismic accelerations that have to be accommodated by the building and equipment design.

In recognition of the size of the canister that is stored within this style of Modular Vault Dry Store, and its ability to potentially store very large amounts of spent nuclear fuel, the generic name given to this style of vault is 'Mega-Vault Dry Store' (MVDS).

2. Design background to the Mega-Vault dry storage system

The original MVDS system as depicted in Figure 1 was designed to store individual fuel assemblies within an array of canisters in a vault. This system provides maximum flexibility for future off-site transportation as the individual fuel assemblies can be removed from their storage locations and placed into a transportation cask for either road or rail off-site shipment. Therefore, future changes or uncertainties in transportation requirements can be accommodated at the time that fuel is removed from storage. However, the development of multiple assembly, multi-purpose canisters, which can be designed for both storage and transportation, has now reached the stage where there are additional benefits of integrating the MVDS design with a MPC.

Figure 1 shows a cross section view through the MVDS vault and illustrates how the cooling system works. Fuel decay heat removal is via a once through airflow system through the vault structure as it passes around the canister arrays. The vault airflow results from a buoyancy induced thermosyphon. The buoyancy head is created by the difference between the inlet and outlet air densities together with the differential height between the inlet and outlet ducts. The pressure drops that result from the rate of airflow and the flow resistances created by the vault inlet duct arrangement, the canister bank array and the outlet duct arrangement balance this buoyancy head. This passive system is capable of rejecting approximately 450kW of heat from each vault module, before either the bulk air temperature causes the vault concrete temperature or the fuel temperature to exceed an acceptable temperature limit.

It has been possible to re-configure the vault storage array from individual fuel assembly storage canisters to accommodate multi-assembly large diameter canisters, and to re-configure the handling equipment to transfer larger canisters. By modifying the design of the MVDS to accept multiple assembly, multi-purpose canisters, it has been possible to maintain the technical and operational benefits of the original MVDS, with the additional benefits of multi-purpose canisters.

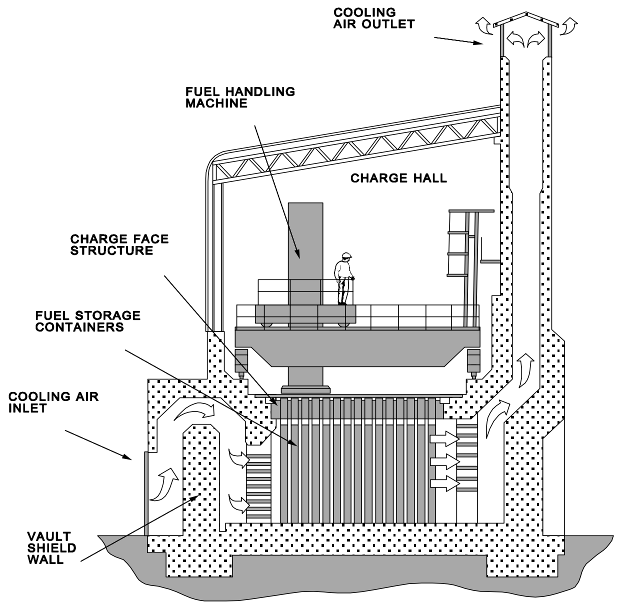

FIG. 1. Cross section view through Modular Vault Dry Store.

The benefits of using a multi-purpose (storage and transport) canister in a vault storage system are:

- Reduced number of fuel assembly transfer and handling operations (compared to individual fuel assembly handling) = lower dose uptake to operators, and lower operating costs;
- Confined transfer of fuel into and out of the facility = no bare fuel handling operations;
- Fuel is packed ready for transportation direct out of the facility = no re-packing required for transportation.

A storage canister for storing individual fuel assemblies is typically 0.2 to 0.4 metre outside diameter, whereas a multiple assembly canister is typically 1.6 to 1.8 metres outside diameter. The Mega-Vault Dry Storage system can be used to store a variety of MPC types that meet the design basis requirements.

3. Mega-Vault dry storage system design parameters

The primary design parameters for the Mega-vault dry storage system are selected to be suitable for a centralized storage facility that can be located in a high seismic area (Table I).

Table I. Primary design parameters for the Mega-vault dry storage system

MPC heat load:	22kW (assumes that the canister has to be transportable to bring it to the store)
Mean Fuel Irradiation:	45 GW·d/tU
Number of MPCs stored in each vault:	15
Design dose rate for operators:	< 5 μSv/h at a height of 1m above the charge face
Dose rate at the site boundary:	< 100 μSv/year at 100 metres from facility
Seismic, Design Basis Earthquake:	0.6g horizontal ground acceleration

4. Description of the Mega-Vault dry storage system

The Mega-Vault dry storage facility consists of three main systems that are shown in Figure 2:

- The Cask Receipt Bay and Transfer Tunnel, where canistered spent fuel is received and transferred to a port under the vault where it can be collected by the Canister Handling Machine. The Transfer Tunnel and Cask Receipt Bay are also used to despatch fuel at the end of storage life;
- The Storage Vault Modules, where canistered spent fuel is stored. The cooling air inlet duct and outlet duct are connected to the vault modules;
- The Canister Handling Machine, which raises and transfers canisters from the cask in the Transfer Tunnel to the storage position in the Storage Vault.

FIG. 2. Mega-Vault dry storage facility.

The Cask Receipt Bay is positioned at the inlet end of the Transfer Tunnel. The Cask Receipt Bay is an enclosed structure where the incoming casks are removed from their transport trailers and placed into the transfer trolley. A 150 tonne overhead crane fitted with a lifting frame lifts the incoming cask into a vertical orientation from the back of the transfer vehicle and lowers it into the transfer trolley that is positioned below grade. The construction of the Transfer Tunnel ensures the canistered spent fuel is protected during transfer operations.

The transfer trolley runs inside the Transfer Tunnel between the Cask Receipt Bay and the Storage Vault modules, as shown in Figure 3. The trolley is mounted on fixed floor rails and moves the loaded Canisters within the cask from the receipt area to the load / unload ports that are inside the vault buildings for access by the Canister Handling Machine. The transfer trolley contains a jacking system that raises the transfer cask into a recess under the load / unload port.

A vault module consists of inlet and outlet ducts, vault sidewalls, supporting foundation structure and the charge face structure. Each row of vault modules is covered by a continuous roof structure which provides a weather tight and illuminated enclosure for year round fuel loading, unloading and maintenance operations. The roof structure provides protection for the Canister Handling Machine and charge face structure and gives operating flexibility during adverse weather conditions and hours of darkness.

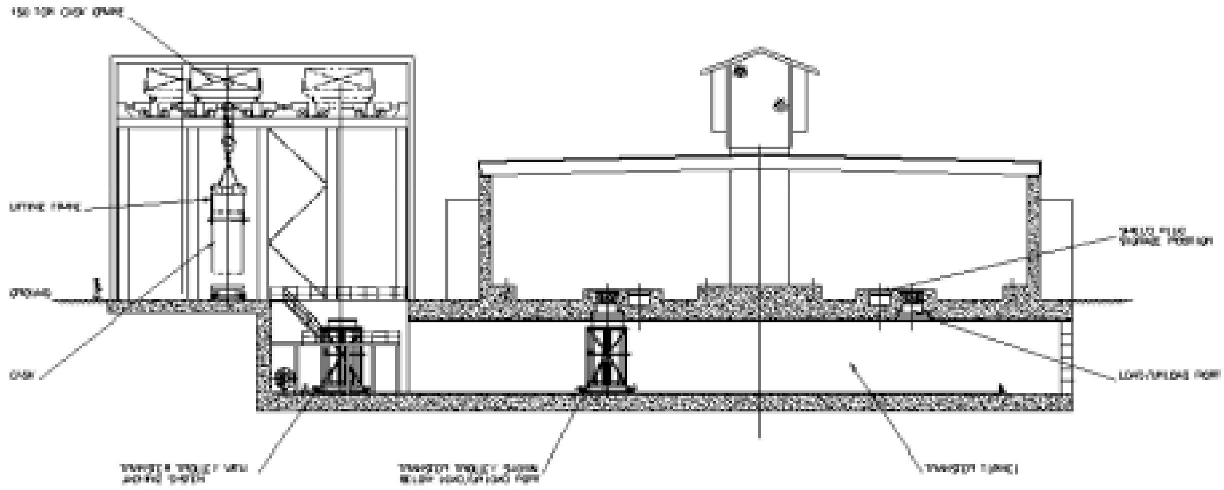

FIG. 3. Cask receipt bay and transfer tunnel.

Cooling air enters each vault through a louvered opening above ground level that is provided with a mesh to prevent the ingress of birds, debris, vermin, etc. The concrete and labyrinth arrangement of the inlet structure provides radiological shielding for the stored fuel. The cooling air leaves the vault and is exhausted to atmosphere through a concrete outlet duct. A steel canopy and mesh provided on the top of the outlet duct, prevents the ingress of rain, snow, birds, etc. The ambient cooling air does not come into contact with the fuel assemblies, which are sealed and confined inside the MPCs, ensuring the internal walls of the vault remain radiologically clean. The MVDS cooling system is thermally very efficient and designed to give acceptable concrete and fuel storage temperatures. The principle of operation of the Mega-Vault cooling system is illustrated on Figure 4. This figure shows the passive cooling air flow system, the vertical position of the canisters within the vault and the concrete shield walls that surround the canisters to form the vault structure.

The base of the canister locates into a support plate that is fastened to the floor of the vault. This plate is designed to withstand vertical and horizontal loads imposed by the Canister and its contents. Dropping a canister within the vault is not considered a design basis accident as single failure proof canister handling devices are used. The top of the Canister is located in the charge face structure. A concrete filled shield plug sits above each Canister in the charge face to complete the shielding of the charge face structure. The canister grapple and hoist of the Canister Handling Machine handle the shield plug by a lifting ring that is bolted to the top of the shield plug and which replicates the lifting feature on the Canisters.

The Canister Handling Machine (CHM) is a shielded cask assembly mounted on a bridge and trolley that runs on rails above the storage vaults, as shown on Figure 4. The CHM lifts loaded canisters from the cask and places them into the storage vaults. The CHM is designed as a single failure proof crane with a dual load path, this configuration ensures that dropping a canister is not a credible event.

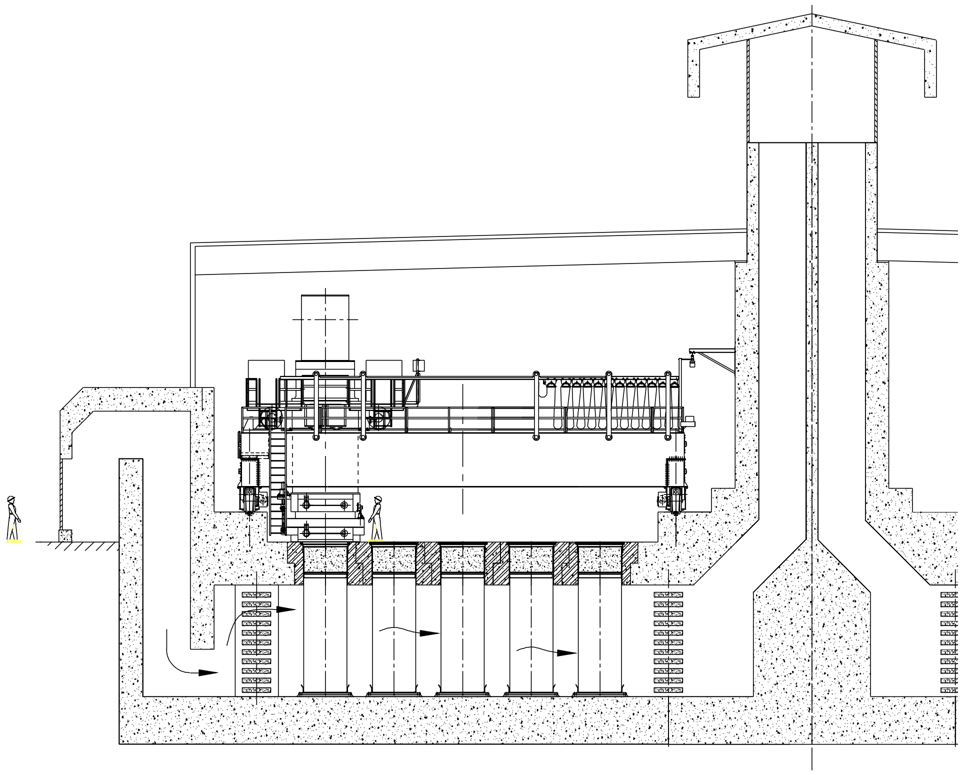

FIG. 4. Cross section view through Mega-Vault.

A wire rope hoist is used to raise and lower a power operated canister grapple. The canister grapple jaws engage with the lifting ring that is bolted to the lid of the canister. The design of the canister grapple ensures the grapple jaws cannot disengage from the canister lifting ring until the weight of the canister is fully supported, i.e. sitting down. The jaws pivot outwards to engage under the lifting ring, and as the canister is raised a mechanical lock prevents further jaw movement, the lock can only be released by supporting the weight of the canister and canister grapple.

The CHM is fitted with double shield gates, a retractable shield skirt and floating shielding blocks to its base. The shield gate doors are electrically operated and retract via twin screw drives. The lower shield gate is the charge face shield gate and is detachable from the CHM. It is normally positioned above the chosen storage position for loading / unloading operations. The charge face shield gate allows removal of the shield plug to prepare the storage position for canister loading or unloading without compromising the overall shielding provided by the charge face structure. The charge face shield gate is moved to the appropriate storage position by the CHM.

The upper shield gate is the CHM shield gate and is permanently attached to the base of the shielded body of the CHM. The CHM shield gate is used to close off the base of the CHM when carrying canisters or shield plugs. The retractable shield skirt is lowered during fuel and shield plug transfer operations to close the gap between the bottom of the CHM and the top of the charge face shield gate. The skirt is raised during CHM travel movements to provide a running clearance.

5. Loading a canister into a Mega-Vault storage position

Following the cross-site, or cross-country, transfer of the canistered fuel to the MVDS, the cask is removed from its transport vehicle and placed into position on a trolley in the Receipt Bay. Refer to Figure 3. After preparing the cask, the lid bolts are released to allow the removal of the lid. The lid is removed and the cask is then moved along the Transfer Tunnel to a position under the vault to allow access by the Canister Handling Machine.

In order to place a canister into the vault the charge face shield plug has to first be removed to create a path for inserting the canister, see Figure 5. The charge face shield gate is lowered onto the charge face using the CHM hoisting system. The CHM then lowers its shield skirt onto the shield gate to complete the shielding path. The doors of the shield gate are opened and the CHM lifts the shield plug from the charge face using its main hoist. The shield gate doors are closed and the CHM shield skirt raised to allow the CHM to move clear of the charge face shield gate. The shield plug that was removed from the charge face is transferred to a stowage position over the transfer tunnel, so that the CHM can accept the canister from the transfer tunnel.

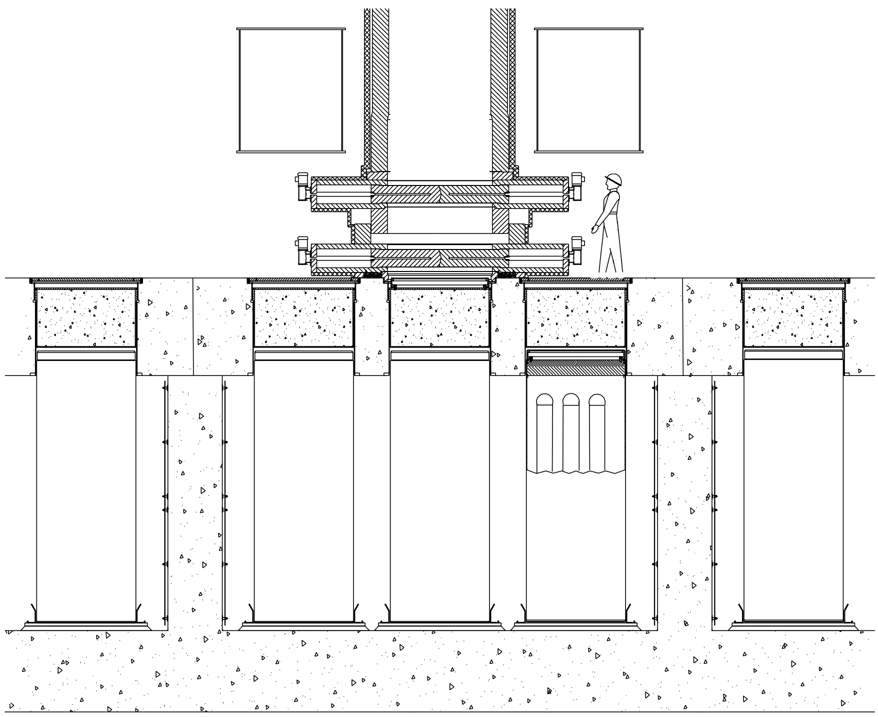

FIG. 5. CHM preparing to remove a shield plug from the charge face.

With the cask positioned under the load/unload port, and the CHM docked over the port, the canister is lifted by the CHM hoist system. The CHM is moved from the load/unload port location and re-docks over the charge face shield gate. The shield gate doors are opened to permit the CHM to lower the canister into its vault storage location. After the CHM hoist has been fully retracted, then the charge face shield gate doors are closed, and the CHM shield skirt is retracted. The CHM picks up the charge face shield gate from the charge face so that it can be moved to the next storage location.

6. Conclusion

The Mega-Vault design is a natural development of the Modular Vault Dry Storage system design. An international patent has been applied for. The system is primarily being developed for away-from-reactor sites for centralized storage of MPCs that may originate from several different source sites. The security and safeguards features inherent within the Mega-Vault system are an important aspect required by current spent fuel management systems.

The Mega-Vault system offers safe, secure and simple storage of spent nuclear fuel, in a seismicaly qualified facility that can be operated all year round. Doses to the operators and public are designed to be very low. Future transportation of the fuel away from the storage facility is assured by the use of transportable canisters.

International experience of storing spent fuel in NUHOMS® systems

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Abstract. International experience in the design, licensing and operation of the NUHOMS® system shows it to be a safe and reliable method for both the intermediate storage and off-site transportation of spent fuel. Its flexibility enables a wide range of fuel types to be stored and transported. It is also readily adaptable to local constraints such as handling weight limits, restricted access to fuel pool buildings and a wide range of environmental conditions.

1. Introduction

The NUHOMS® system for spent fuel intermediate storage has two main components. The spent fuel is contained in a stainless steel canister that is sealed by welding two stainless steel lids. The canister is stored with its main axis horizontal in a concrete module (HSM). The canister provides a containment boundary and includes an internal basket structure to ensure criticality safety and good heat transfer. The HSM has an access door, an internal support structure for the canister and air vents for the rejection of heat by natural convection. The thick reinforced concrete walls offer excellent radiation shielding properties. After spent fuel loading, canisters are transferred from the spent fuel pool to the storage site using a shielded transfer cask with an integral hydraulic ram to facilitate canister loading into the concrete module.

The first designs were conceived for US commercial light water reactor fuel, and the first storage licenses were obtained from the US Nuclear Regulatory Commission. The initial system for PWR fuel has a capacity of 7 spent fuel assemblies, but this was increased to 24 in the next design. A canister for storing 52 BWR fuel assemblies was soon added to the fleet. As the demand for dry storage systems in the US increased, Transnuclear, Inc identified a market need for higher capacity systems and two further designs were added with capacities of 32 PWR and 61 BWR assemblies. These designs are further evolving to match the trends in increased fuel initial enrichment and burnup.

In addition to meeting the US regulatory requirements for storage, the latest systems are designed to meet US NRC requirements for transportation. This latest development gives NUHOMS® users the additional flexibility of a dual-purpose system. Another NUHOMS® system was successfully developed to store fuel debris from the Three Mile Island reactor. Solutions have also been developed for safely storing and transporting damaged fuel.

Outside the US, the NUHOMS® system has attracted considerable interest. As a licensee for the NUHOMS® technology, Framatome has supplied a NUHOMS® system for storing VVER fuel assemblies at Metzamor in Armenia. Framatome is also supplying a NUHOMS® system for storing RBMK fuel at Chernobyl in the Ukraine.

The worldwide popularity of the NUHOMS® system is due to a combination of economic and technical factors. As a system for intermediate dry storage, the modular design allows owners and operators to defer investment by increasing the installed capacity incrementally. Relatively short fabrication times allow the installation to be timed to meet the operational needs for spent fuel loadings. Local fabrication of the concrete module is another factor that can influence system selection when the use of local labor is desirable. However, the overriding technical advantage of the NUHOMS® system is the inherent flexibility of having its two main components designed to fulfill specific technical functions. These components can be individually adapted to suit a specific need without altering the overall system concept. For example, reducing the external radiation dose rates at the storage site is a simple matter of increasing the thickness of the concrete walls of the storage module and this has no direct influence on the canister operations.

2. Nuhoms® system description and operational advantages

The canister consists of a stainless steel cylindrical shell, top and bottom shield plugs, inner and outer bottom closure plates, inner and outer top cover plates, and the internal basket. A typical DSC for PWR fuel assemblies is shown in Figure 1.

The fuel assemblies are supported by the stainless steel fuel compartments that extend through the entire canister cavity. Criticality control is achieved by a combination of geometric spacing of the fuel assemblies and a selective use of neutron poison material in the basket.

A typical NUHOMS canister (basket and shell) is fabricated primarily from high quality stainless steel. All of the canister shell materials are ASME code materials and are used consistent with code approved applications. The shell materials are resistant to corrosion and are not susceptible to other galvanic reactions. The DSC internals are enveloped in a dry helium inerted environment and are designed for all postulated environmental conditions.

The design of the NUHOMS® Horizontal Storage Module (HSM) allows the canister to be transferred and stored without performing a single critical lift at the Independent Spent Fuel Storage Installation (ISFSI) or anywhere outside the protected area. This design eliminates entirely the risk associated with such critical lifts of canisters loaded with spent fuel or the need for a heavy single failure proof crane at the ISFSI.

The NUHOMS® HSM design allows the highest shielding performance of any other system offered in the industry. It accomplishes such performance by surrounding the canister with massive concrete walls and by close packing the modules at the ISFSI, allowing adjacent units to shield one another. Figure 2 shows how the NUHOMS footprint can save up to 33% of the space needed for vertical systems.

The HSM is a reinforced concrete structure designed to shield and support the DSC while providing passive heat removal. Ambient air enters the module through the bottom inlet vents, circulates around the DSC and exits through the outlet vents at the top. The HSM is designed to protect the DSC from extreme environmental and geological conditions including tornadoes, earthquakes, and floods.

The HSM design uses passive ventilation for the removal of spent fuel decay heat from the canister. The currently licensed storage module ventilation system has a heat removal capacity of 24 kilowatts but new designs are under development to increase this capacity to around 34 kilowatts for extreme ambient temperatures ranging from –40°F to 117°F.

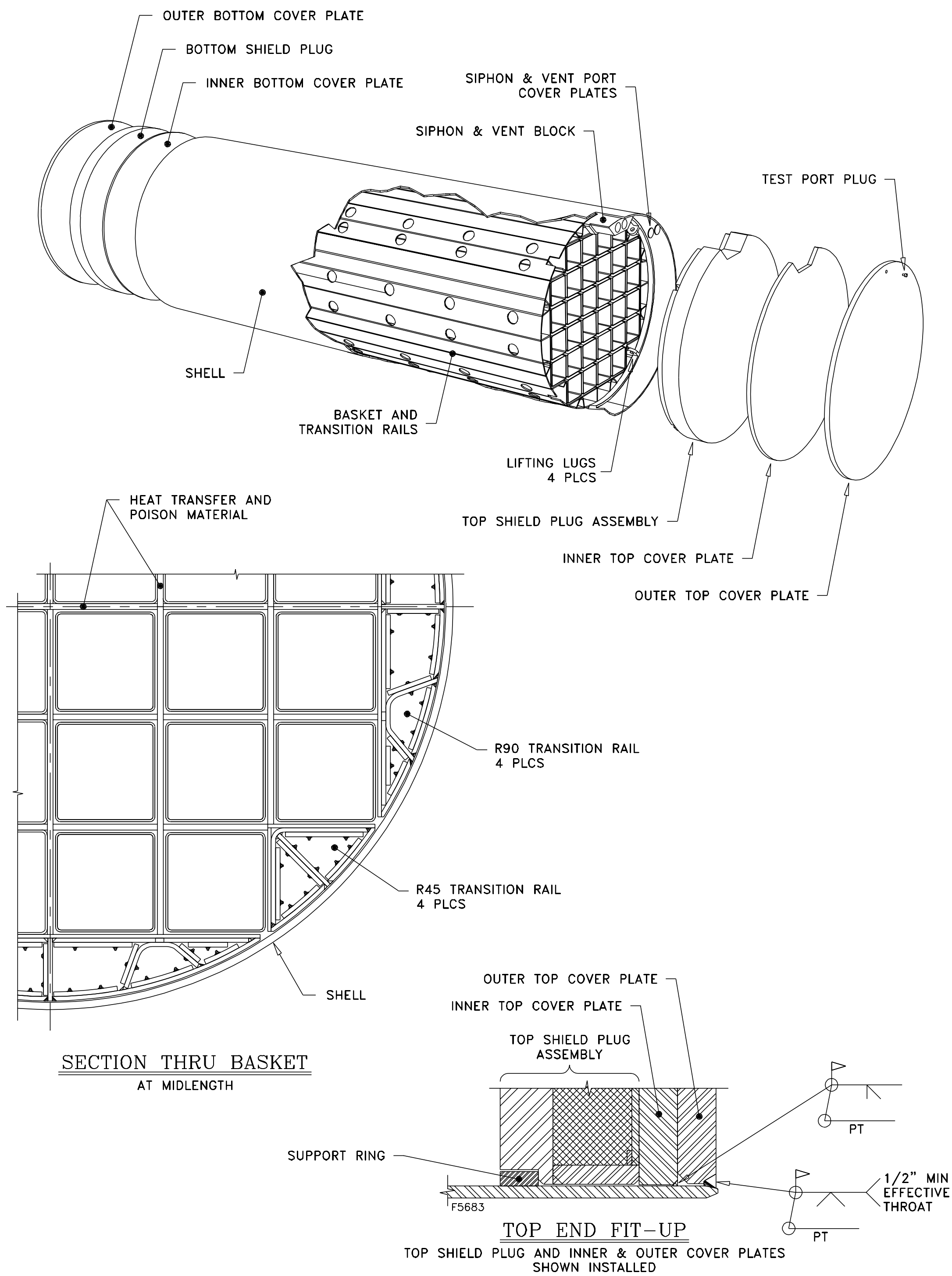

FIG. 1. A typical DSC for storing and transporting PWR fuel.

Each module includes hard-faced steel support rails that allow the horizontal sliding insertion and retrieval of the canister. The canister transfer operations at existing NUHOMS® ISFSIs have been highly successful in demonstrating the safety and simplicity of the horizontal sliding transfer technology.

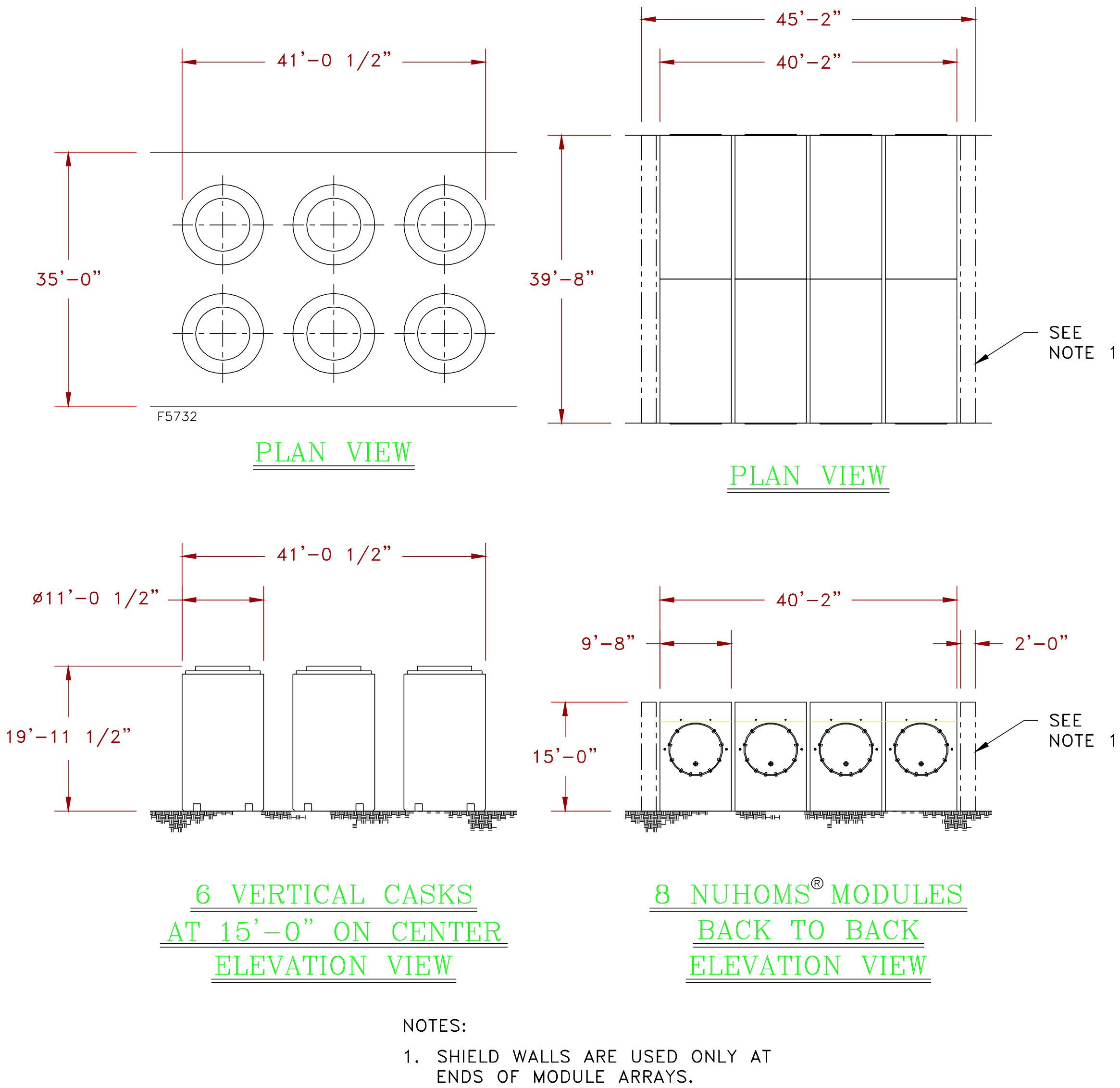

FIG. 2. NUHOMS® layout compared to vertical casks.

The NUHOMS® HSMs can either be cast in situ at the storage site or prefabricated off-site at a qualified concrete fabrication facility. Prefabricated modules are constructed out of two segments, a base unit and a roof slab, which are delivered separately and installed at the ISFSI. By fabricating these components off-site, the NUHOMS® System minimizes the impact on an operating facility. By delivering finished segments, and not requiring any major construction or concrete placements at the ISFSI, each NUHOMS® HSM can be fully erected and completed in approximately one day, using a crane and a 5-man crew (See Figs 3 and 4).

The NUHOMS® transfer cask incorporates gamma and neutron shielding materials. The exterior shell has a surface finish to facilitate decontamination.

As shown in Figure 5, the transfer cask is constructed from two concentric cylindrical stainless steel shells with a bolted top cover plate and a welded bottom end assembly. The annulus formed by these two shells is filled with cast lead to provide gamma shielding. The transfer cask also includes an outer steel jacket, which is filled with water for neutron shielding. The top and bottom end assemblies incorporate a solid neutron shield material. The transfer cask is designed to provide sufficient shielding to ensure that dose rates are ALARA.

FIG. 3. HSM fabrication.

FIG. 4. HSM Installation at an ISFSI site.

NUHOMS transfer casks have been used successfully in PWR and BWR spent fuel pools and have never exhibited an adverse interaction with the spent fuel pool water.

Canister loading includes physically placing the fuel assemblies into the canister, decontamination, draining, drying, and seal welding. An automated welding system (AWS) welds the DSC top closure plates to the DSC shell after fuel loading. The AWS is a fully-integrated, remotely-operated automated welding system, including remote viewing and motion control. Non-destructive examination of the closure welds is performed by Dye Penetrant (PT) examination. The AWS is show in Figure 7.

Canister transfer operations include transferring the loaded cask to the on-site transporter, transporting the cask/canister to the ISFSI, and inserting the canister into the storage module. These operations are illustrated in Figure 8.

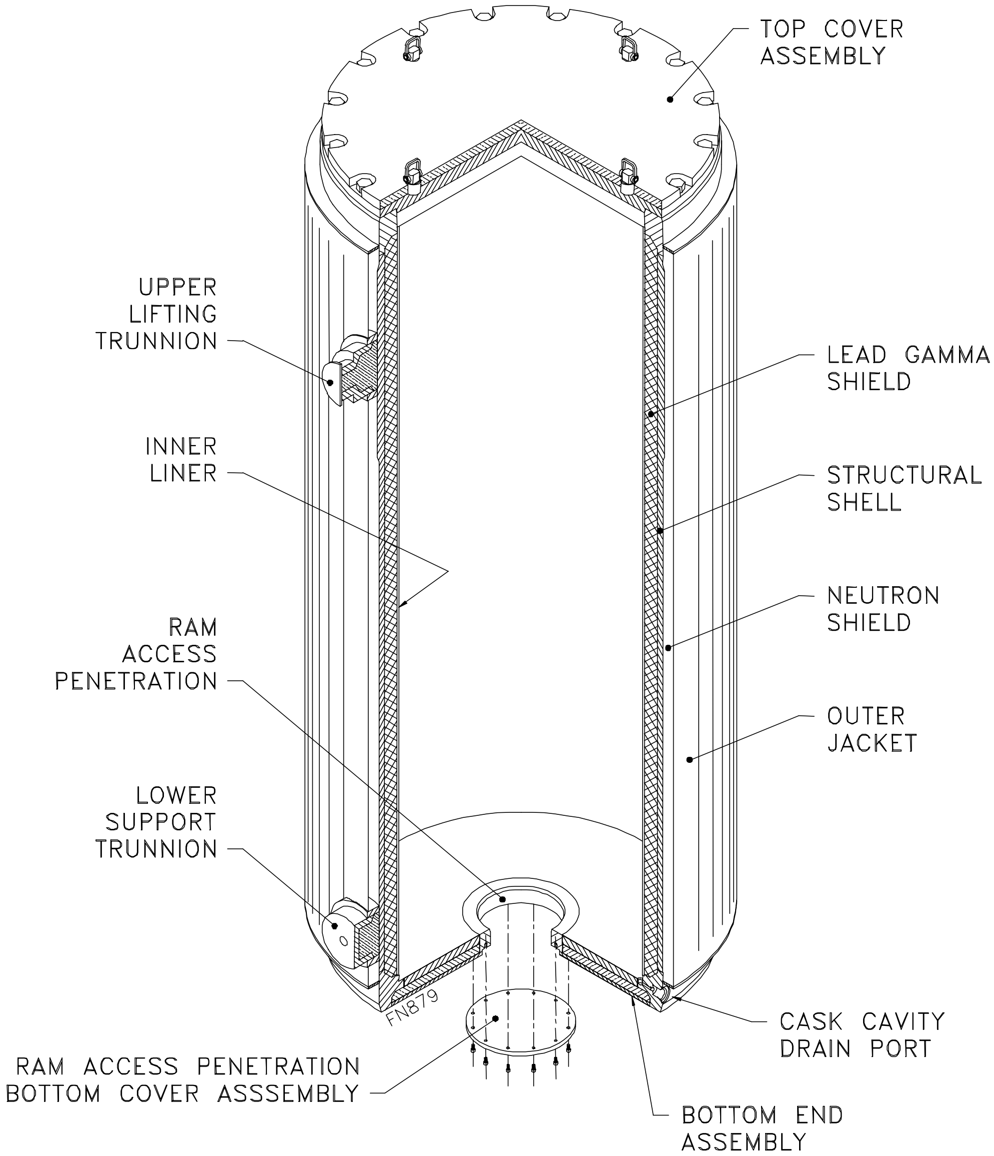

FIG. 5. The NUHOMS® transfer cask.

The welded canister provides a leaktight containment during the storage period and there is no need to perform any special monitoring or periodic leaktightness checks. Recovery of the fuel from the HSM for transfer to another site is performed using the same system of horizontal transfer but with a cask licensed for off site transport. Such a cask is shown in Figure 9. After horizontal transfer, the cask lid is secured and the impact limiters are fitted so put the package in its transport configuration for shipment either by rail, truck or ship.

3. Current development status

3.1. Nuhoms® in the USA

Since the USA adopted a ‘once through’ policy for spent fuel management, intermediate storage has become a necessity for many of the US commercial nuclear power stations as they reach the limit of their in poll storage capacity. 70% of fuel currently in intermediate dry storage in the USA is stored in Transnuclear systems, either TN24 type metal casks or

NUHOMS® systems. Up to March 2003, 258 NUHOMS® systems had been delivered to customers and the successful operational experience has enabled Transnuclear to reduce the lead time for new NUHOMS® systems to less than 24 months.

Licensing of NUHOMS® systems is under NRC certification, either on a site specific basis or under a generic license. The licensing status in March 2003 is shown in Table I.

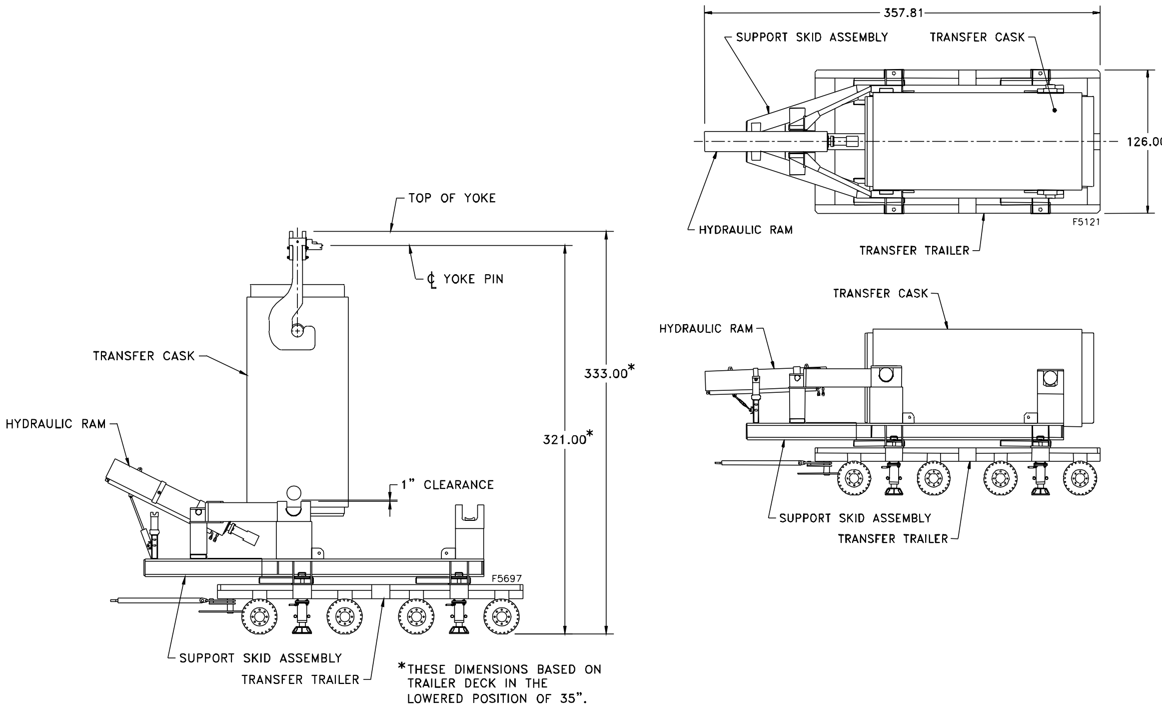

FIG. 6. The trailer and skid assembly.

FIG. 7. Automated welding system.

3.2. Nuhoms® in Armenia

The NUHOMS® system has been successfully implemented in Armenia by Framatome-ANP under license from Transnuclear and is operational since 2000. 11 systems have been delivered and loaded with VVER 440 spent fuel at the Medzamor site, each having a capacity for 56 spent fuel assemblies.

FIG. 8. Placing the canister in the HSM.

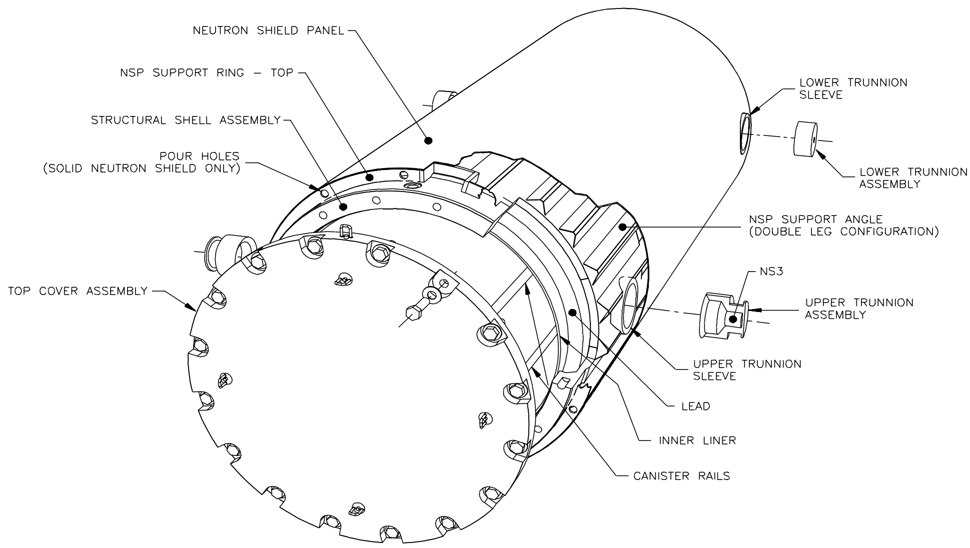

FIG. 9. MP 1197 transportation cask (shown without impact limiters).

3.3. Nuhoms® in Ukraine

The NUHOMS® system was chosen by Framatome-ANP to store 21355 RMBK fuel at Chernobyl. A canister for 95 assemblies has been developed together with a fuel conditioning system to prepare the fuel for loading. The 232 HSM fabrication has been fully completed on site as well as the conditioning facility.

The conditioning facility role is to provide two technological barriers. The first one is a stainless steel cartridge to receive each single fuel assembly with an inert gas. The second barrier is the canister itself where cartridges are inserted (both are manufactured in Ukraine). The particularity of this installation is due to the design of RBMK fuel assembly (length, two fuel bundles). Year 2003 will be dedicated to erection tasks of equipment manufacture in France and Ukraine. The full commissioning is scheduled end of 2004.

Table I. NUHOMS® licensing status in the US

System	Fuel type	Transport license	Storage license
NUHOMS® 7 P	PWR	No	Yes
NUHOMS® 24	PWR	No	Yes
NUHOMS® 24 PHB	PWR	No	Under review
NUHOMS® 24 PT 1	PWR	Yes	Yes
NUHOMS® MP 187	PWR	Yes	Yes
NUHOMS® 24 PT 2	PWR	Yes	Yes
NUHOMS® 24 PT 4	PWR	Under review	Under review
NUHOMS® 32 PT	PWR	In preparation	Under review
NUHOMS® 32PTH	PWR	In preparation	In preparation
NUHOMS® 24PTH	PWR	In preparation	In preparation
NUHOMS® 52 B	BWR	No	Yes
NUHOMS® 61 BT	BWR	Yes	Yes
NUHOMS® 12 T	TMI debris	No	Yes

4. Adapting Nuhoms® to specific customer needs

The NUHOMS® system is under continuous development to meet specific customer needs. Specific components can be adjusted to increase performance levels without changing the basic design concept. Examples of such developments are as follows:

4.1. Storage only and dual purpose systems

Early systems for customers in the US were conceived for on site storage because transportation needs were not defined at that time. Today, most clients are opting for a storage solution with transport capabilities and this can be easily achieved for the NUHOMS® system by placing the canister in a transportation cask. This uses similar technology as employed in on-site transfer casks to allow easy transfer of the canister from the HSM to the transportation cask.

4.2. Increased fuel enrichments

Higher fuel enrichments can be safely added to the canister contents by adding neutron poison material in the basket. Transnuclear uses a variety of poison materials and is able to optimize the amount of poison to the particular fuel characteristics for specific clients. Even if a range of enrichments exists for candidate fuel, the canister can be offered with a corresponding range of poison loadings. This approach ensures that criticality safety is assured at all times whilst minimizing the cost of specific canisters.

4.3. Minimizing dose rates

As burnups increase, the need for greater shielding capacity becomes more important to keep radiation doses as low as possible during interim storage. The NUHOMS® system has an inherent high shielding capacity by using closely packed modules in storage. The concrete walls of the HSM provide excellent shielding and thicknesses can readily be adapted to meet any required doses rate in storage. This feature has already been implemented for the advanced NUHOMS® system for San Onofre operated by Southern California Edison where the site geometry required very challenging dose rate targets.

4.4. Short cooling times

Early dry storage needs in the US were typically for low burn up, long cooled fuel with correspondingly low heat loads. At some dry storage sites, the ‘stock’ of such relatively cool fuel is being used up and future needs will require a capability to safely store higher heat load fuels. Current NUHOMS® systems are licensed for total heat loads of 24 kW but new systems are under development to increase the heat load capacity to around 34 kW. This is achieved by careful design of the basket to improve heat transfer within the canister and improved air circulation systems within the HSM’s.

4.5. Maximizing payloads

The number of assemblies stored in a particular dry storage system plays a major role in the overall economics. Higher capacity canisters can reduce the storage cost per assembly by minimizing the number of storage units and reducing the total size of the interim storage facility. Transnuclear has increased NUHOMS® systems capacities from 24 to 32 PWR assemblies and from 52 to 61 BWR assemblies. This was achieved by making full use of the high performance basket design technology available within the COGEMA LOGISTICS group of companies.

4.6. International fabrication

HSM fabrication uses industry standard concrete technology which can easily be applied close to the storage site. This allows HSM fabrication to be performed either directly on the site using cast in place methods or at a local concrete fabricator using performs. NUHOMS® system components have successfully been fabricated in the USA, Japan and Europe. This proven reliable suite of high quality international fabricators allows COGEMA LOGISTICS and Transnuclear, Inc. to fully optimize customer needs in terms of localization, scheduling and optimizing costs.

5. Nuhoms® international statistics

The NUHOMS® system is well established in the US and Europe and is currently under consideration by other countries who are approaching the time to introduce dry storage capacities. Table II shows current worldwide order and delivery status for NUHOMS® systems.

Table II. NUHOMS® Worldwide Distribution

COUNTRY	SYSTEM	ORDERED	DELIVERED
USA	BWR	67	36
USA	PWR	270	193
USA	Others	29	29
ARMENIA	VVER	11	11
UKRAINE	RBMK	232	-
TOTAL		609	269

6. Conclusions and future development plans

NUHOMS® systems are well established in the USA and Europe as a reliable, safe and well proven option for dry storage of spent fuel.

NUHOMS® systems can be used for either on-site storage or away from reactor storage because the canisters can be transported in a B(U) packaging.

The proven NUHOMS® flexibility can readily be adapted to meet new customer needs in terms of handling limitations (size and weight), fuel characteristics, local regulations etc. Transnuclear, Inc and Cogema Logistics are continually looking for worldwide opportunities for the NUHOMS® system.

Discussions with potential clients are in progress and new design concepts are in preparation, including a vertical version of the NUHOMS® system.

Current status of R&D programme of spent fuel storage technology in CRIEPI

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Abstract. In 1997, a new research programme of demonstrative tests for interim storage of spent fuel had been started, which is mainly related to concrete cask storage technology, particularly aiming at the realization of dry storage away-from-reactor in 2010. Concrete cask storage system is considered to essentially have an economical advantage. To propose safety standards for concrete cask structures, systems, components, the demonstration programme for qualification of concrete cask performance, such as heat removal tests under the normal, abnormal and accidental events with the full-scale casks, impact tests with the full-scale canisters and seismic tests with scale-model cask are in progress. This paper introduces the current status of CRIEPI's R&D programme on concrete cask technology for spent fuel storage.

1. Introduction

Spent fuel generated by nuclear power plants (NPP) is designated as a useful resource and shall be properly stored, according to the policy of Japan, until the time of reprocessing and recycling. Recently, the quantity of spent fuel stored at each NPP site will increase, due to concluding overseas reprocessing contracts and delay in the domestic reprocessing project. Therefore, the construction of spent fuel interim storage facilities on- or off-site of the NPP is envisaged. The dual-purpose metal cask that can be used for storage and transportation has been receiving highest priority in implementing storage facilities for the short and medium term, with its superb economics compared to water pool facilities. In Japan, two dry storage facilities using metal casks are being operated at the Fukushima-Dai-Ichi site of the Tokyo Electric Power Company and the Tokai-Dai-Ni site of the Japan Atomic Power Company.

With a longer term perspective, research on the concrete modular dry storage technology is continuing, aiming at better economic performance. Key issues of this research include safety standards in operation and maintenance during storage and loading/unloading for transportation, long term integrity of metal canister and concrete materials, and so on. In 1997, a new research programme for demonstration tests of interim storage of spent fuel commenced, mainly involving concrete cask storage technologies, with the aim of obtaining basic data for regulating safety[1] [1, 2].

2. Demonstration programme for qualification of concrete cask performance

In the demonstration programme, the following studies (see Fig. 1) are currently in progress:

a. For concrete material and structures:
   i. Long term durability of concrete material (carbonation and salt damage);
   ii. Dynamic strength of concrete materials under high temperature and in the event of an accident;

[1] This work has been being executed under contract with Ministry of Economy and Trade Industry.

   iii. Characteristics of heat transfer and cracking due to thermal stress;
   iv. Shielding performance of concrete structures;

b. For metal canisters:
   i. Impact and corrosion resistance of multipurpose canister with welded components;

c. For spent fuel:
   i. Development of non-destructive monitoring method;
   ii. Characteristics and long term performance of high burnup and MOX spent fuel;

d. Programme of demonstrations for determining concrete cask performance (a schedule of this demonstration programme is shown in Fig. 2):
   i. Basic design of Japanese concrete cask:
      Two types of concrete cask, a reinforced-concrete cask (RC cask) and concrete-filled-steel cask (CFS cask) to store the high burn-up spent fuel, were designed.
   ii. Manufacture:
      Two types of full-size concrete cask and multi-purpose canister were manufactured.
   iii. Demonstration tests:
      Heat removal tests of the concrete cask are in execution taking into consideration normal, off-normal and accidental events, and well as impact tests on the metal canister. Seismic tests using a scale-model cask and streaming tests with the air inlet components were carried out;
   iv. Safety analysis:
      Safety analysis will be performed using the information obtained in the demonstration tests, to contribute to safety standards for concrete modular structures, systems, components.

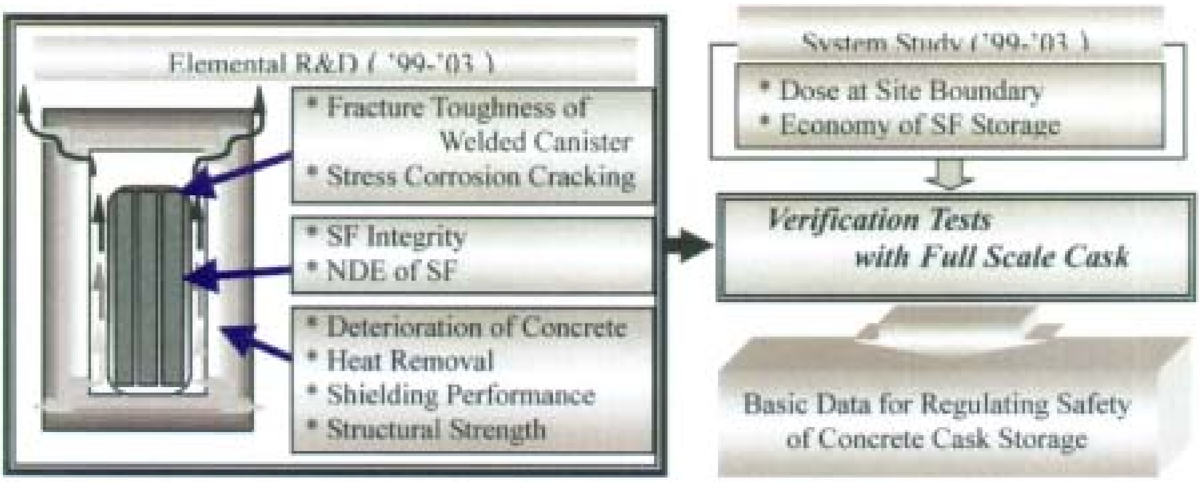


FIG. 1. Schematic showing performance of reinforced concrete components in dry storage.

3. Programme of demonstration tests for determining concrete cask performance

3.1. Demonstration test facility in Akagi Test Center

The demonstration test facility as shown in Fig. 3 was constructed in the Akagi Test Center of CRIEPI, located in the north of about 130 km from the centre of Tokyo. In this facility, there are heat removal test area and drop test area.

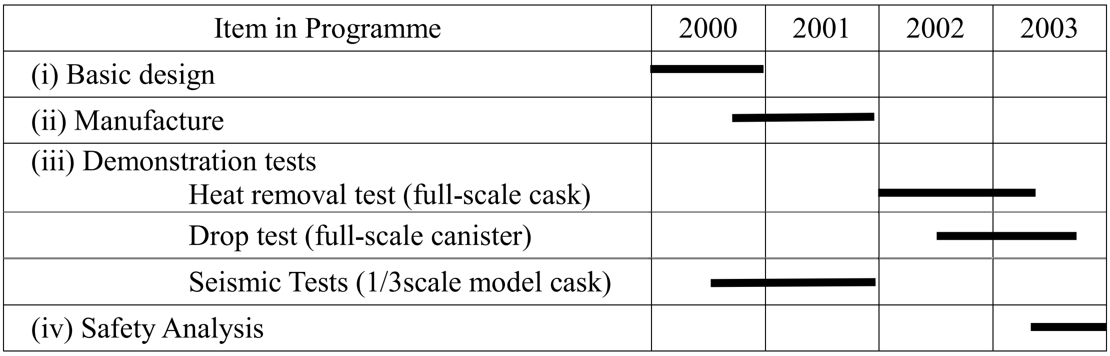


FIG. 2. Schedule of demonstration programme for concrete cask.

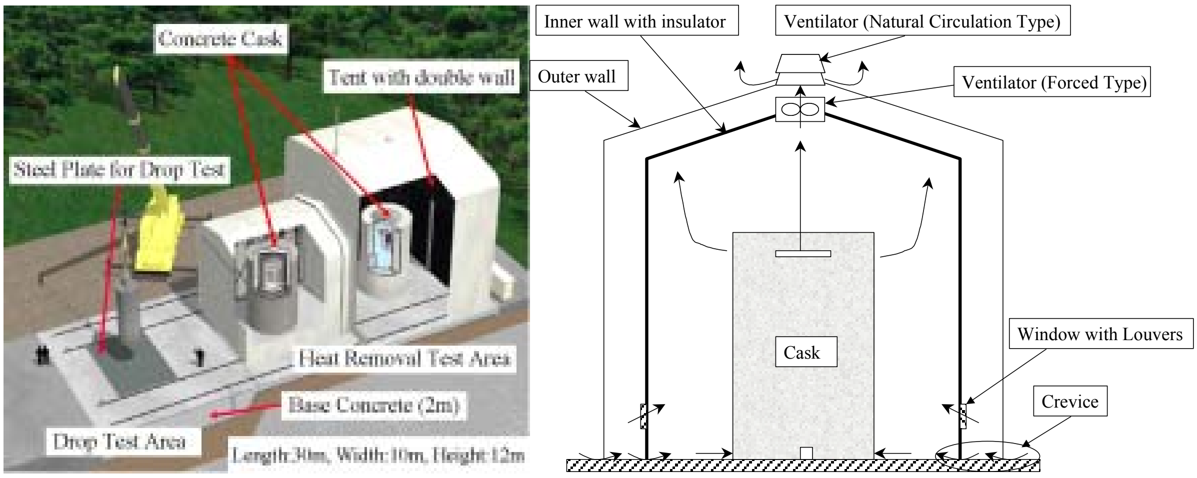


FIG. 3. Overview of the demonstration test facility in Akagi Test Center of CRIEPI.

In the heat removal test area, there are two movable tents on the rail. One tent is used for avoiding wind and rain for the preparation of the test and the other is used for the heat removal test. The heat removal test tent has the outer wall and insulated inner wall to decrease the influence of fluctuation of ambient temperature. During the heat removal test, the concrete cask is located in the middle, and the cooling air goes inside through the four windows with louvers and the crevice between the base concrete and the wall and goes outside through the ventilators attached on the roof as shown in Fig. 3. The horizontal and vertical distances between the cask surface and the inner wall are about 2 m and 4 m, respectively. In the drop test area, a steel plate is fixed on the base concrete. Size of the steel plate is 7.5 m length, 4.5 m width and 50 mm thickness. Thickness of the base concrete is 2 m and its weight 400 t.

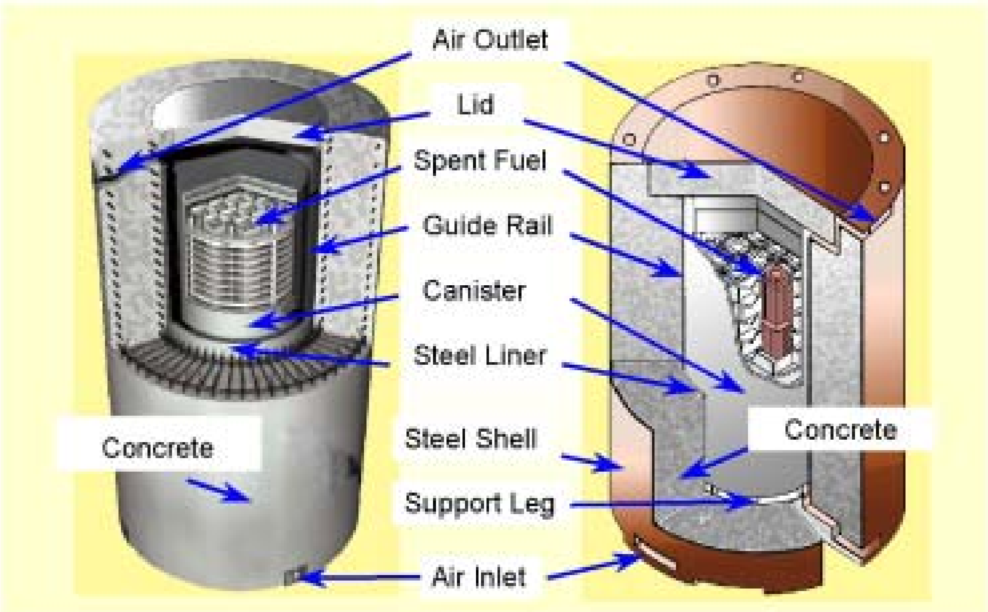
3.2. Basic design of Japanese type concrete cask

Strength and safety must be maintained to the load when considering the conditions under which casks are used (size of the site, installation on the shoreline, seismic factors) which is peculiar to our country about the structure and the use material of the cask and to be assumed during the design storage period. Preliminary design items and parameters are shown in Table I. The concrete cask was assumed to be for indoor use.

Preliminary designs for two types of cask, an RC cask and CFS cask were employed as the basic structure as shown in Fig. 4. The RC cask is made from reinforced concrete storage container and the reinforced concrete becomes a structure strength part to the assumed load. On the other hand, at the CFS cask, concrete storage container consists of concrete covered with a steel sheet, creating a steel structure; concrete is not a structure strength part and is a radiation shielding material.

Table I. Preliminary design item and parameter

Design Item	Condition	Evaluation Item	
Thermal	Normal	Heat generation rate, Air flow rate	
	Off-Normal	Integrity of the fuel cladding, Temperature	
Containment	Normal	Quality assurance of the welded structure of the canister	
Shielding	Normal	Dose rates, Cooling air activation	
Criticality	Normal	Wet condition, dry condition	
Structural strength	Normal	Durability, Thermal stress, Internal pressure, Seismic ability	
	Off-Normal	Drop of canister, Tumble of cask, Blockage of air inlet	
Design storage period		40~60 years	
Design parameter		Fuel type	17 × 17 array for PWR
		Enrichment (wt % U^{235})	4.9
		Burnup (MW·d/kgHM) (Max)	55
		Cooling time (year)	10
		Environmental temperature	33



[RC (Reinforced Concrete) cask] *[CFS (Concrete filled steel) cask]*

FIG. 4. Outline of the concrete cask.

Two types of canister were designed as shown in Fig. 5. Each canister can store 21 PWR spent fuels, and for each canister body, high corrosion-resistant material is used. The basket of type I consists of guide tubes and stainless steel plates. The stainless steel plate fixed at constant intervals of distance by steel rod has 21 square holes for the guide tube. The guide tubes are placed in the hole and fixed to the plate. To increase thermal conduction, aluminium plate is fixed to the stainless steel plate. The basket of type II is the assembly of rectangular hollow block made of aluminium alloy.

For these casks, preliminary safety evaluation of thermal, confinement, shielding, non-criticality and structural strength properties was conducted. Tables II and III shows the summaries of the thermal and shielding evaluation. A thermal analysis was conducted under normal and off-normal (half air inlets blocked) conditions and the temperature of the concrete cask and air flow rate calculated. The temperature of each section fell below the allowable value confirming thermal safety. Moreover, the dose rate of each section also fell below the allowable value confirming occupational exposure safety.

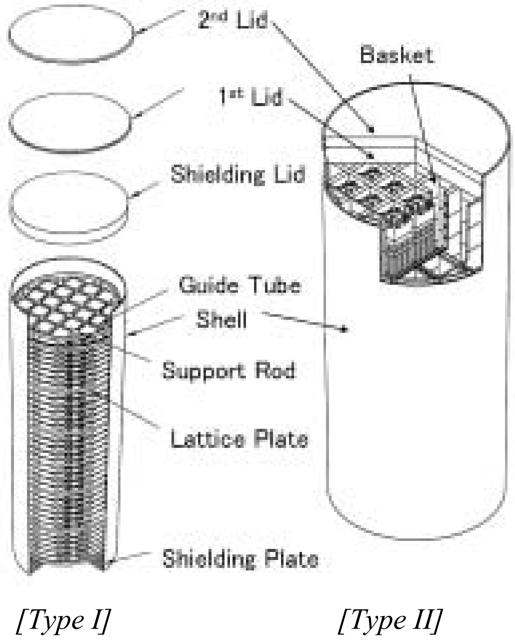


FIG. 5. Outline of the canister.

Table II. Summery of thermal evaluation

Item / Cask type	RC cask		CFS cask	
	Normal	Off-normal	Normal	Off-normal
Air flow rate	0.29 m^3/s	0.25 m^3/s	0.289 m^3/s	0.208 m^3/s
Outlet air temperature	92.5 ^oC	100 ^oC	81 ^oC	100 ^oC
Concrete temperature (Max)	74 ^oC	79 ^oC	72 ^oC	85 ^oC
Canister surface temperature (Max)	190 ^oC	199 ^oC	216 ^oC	233 ^oC
Fuel clad temperature (Max)	318 ^oC	326 ^oC	315 ^oC	322 ^oC

Table III. Summery of shielding analysis (Max. dose rates at 1 m, μSv/h)

	RC cask	CFS cask	Occupational Exposure
Body Center	71.3	28.4	
Lid Center	80.9	79.5	200
Air Inlet	189	28.3	(provisional limit value)
Air Outlet	336.0	1.9	

3.3. Fabrication of full-scale concrete cask

Based on the design, two types of full-scale concrete container and canister were fabricated for the demonstration tests. Main specifications of these casks are shown in Table IV.

The ratio of reinforcement for the RC cask was 1.7% from the point of view of ensuring good durability for the long term storage. We used high quality concrete (water cement ratio is smaller than 50 %) including a highly efficient AE water reducing agent for the casks. Concrete container was fabricated without the placing joint. For the CFS storage container, the studs are welded on the inner surface of the outer shell of the cask and we used the same high quality concrete as described above.

Fig. 6 shows the arrangement of reinforcing bar of RC storage container, the RC storage container and the SC storage container. Fig. 7 shows the basket and canister body for each cask.

Table IV. Specifications of the concrete cask

Storage Container	Type of storage Container	RC	CFS
	Height	5 787 mm	6 120 mm
	Outside diameter	3 940 mm	3 800 mm
	Inside diameter	1 850 mm	1 838 mm
	Weight (without canister)	150 t	154 t
Canister	Type of canister	Type I	Type II
	Height	4 630 mm	4 470 mm
	Outside diameter	1 676 mm	1 640 mm
	Weight (with spent fuels)	35 t	30 t
	Body	Super stainless steel	Austenitic-ferritic stainless steel
	Basket	Stainless steel	Aluminium alloy

[Arrangement of reinforced bar] *[RC cask]* *[CFS cask (without Lid)]*

FIG. 6. RC and CFS cask.

[Basket for Type I] [Canister body for Type I] [Basket for Type II]

FIG. 7. Baskets and canister body.

3.4. Heat removal test using the full-scale RC cask

Two types of concrete cask have been fabricated and the heat removal tests using the RC cask have been finished. The contents obtained in this test are summarized as follows.

3.4.1. Test cask

During actual storage, two lids are welded to the canister body to maintain the confinement. However, during the thermal tests, only one lid is welded to the body taking account of the opening of the lid after the test.

In the canister lids, there are 21 holes for heaters and 3 holes for measurements as shown in Fig. 8. The heater was inserted to dummy weight steel structure and fixed on the top of the secondary lid by the flange consists of a sheath heater, and generates heat in the same length as the spent fuel.

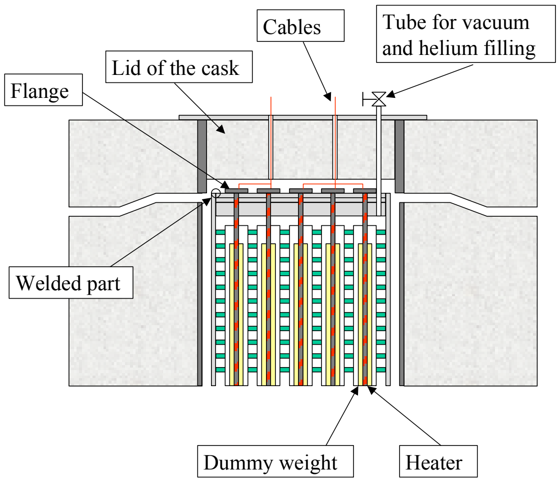


FIG. 8. Schematic of the RC test cask.

3.4.2. Test condition

Table V shows the test condition. Test parameters are heater power, closure rate of air inlet and cask position. The canister is sealed and filled with helium gas at 0.1MPa approximately in the ambient temperature. In the beginning, the tests were performed in the vertical position and then the cask position was changed to the horizontal position. During the tests, the ventilator of the tent was in operation so that the stratification boundary does not go down to the level at the air outlet.

3.4.3. Measurement

Table VI shows the items of measurement and Fig. 9 shows the measuring points of temperature and strain in the representative cross sections. The cooling air removes the most part of the heat energy discharged from the spent fuel in the concrete. As it is important to evaluate the heat balance, the air velocity and the flow rate at the air inlet were measured precisely. During measuring the air inlet flow with the anemometer, a rectangular pipe with bell-mouth structure was used to regulate three-dimensional air inlet flow to one-dimensional flow. As the temperature exhausting from the outlet duct is very high, the air velocity at the outlet duct is measured with the propeller flow sensor.

Table V. Test condition

No.	Cask position	Cavity gas	Total heat Power (kW)	Closure rate of the air inlet (%)	Situation
1	Vertical	He	22.6	0	Steady state
8	Vertical	He	16.0	0	Steady state
2	Vertical	He	10.0	0	Steady state
3	Vertical	He	22.6	50	Steady state
4	Vertical	He	22.6	100	Transient
5	Vertical	Leak condition	22.6	0	Transient
6	Horizontal	He	22.6	0	Steady state
7	Horizontal	He	22.6	100[a]	Transient

[a] In this case, all the inlet and outlet are closed.

Table VI. Items of measurement

Item	Sensor	Subject and number
Temperature	Thermocouple (Sheath type)	Inside of the cask: 134 Surface of the cask: 158 Inside of the canister: 118 Surface of the canister: 54 Heater: 25 Cooling air: 67 Tent and ambient t: 23 Inside of the base concrete: 9
Air velocity	Anemometer flow sensor	Inlet of the cask: 4
	Propeller flow sensor	Outlet of the cask: 16
Flow pattern	Smoke and Laser sheet	Outlet of the cask
Strain	Strain gauge	Inside of the cask: 74×2 (direction) Surface of the cask: 55×2 (direction) Reinforced bar: 121×2(direction)
Crack Width	Contact gauge Microscope	Surface of the cask
Acoustic Emission	AE sensor	Inside of the cask: 10
Pressure	Pressure gauge	Inner pressure of the canister: 1
Heater power	Wattmeter	Heater: 2[a]

[a] 21 heaters are divided in two regions for the electrical control.

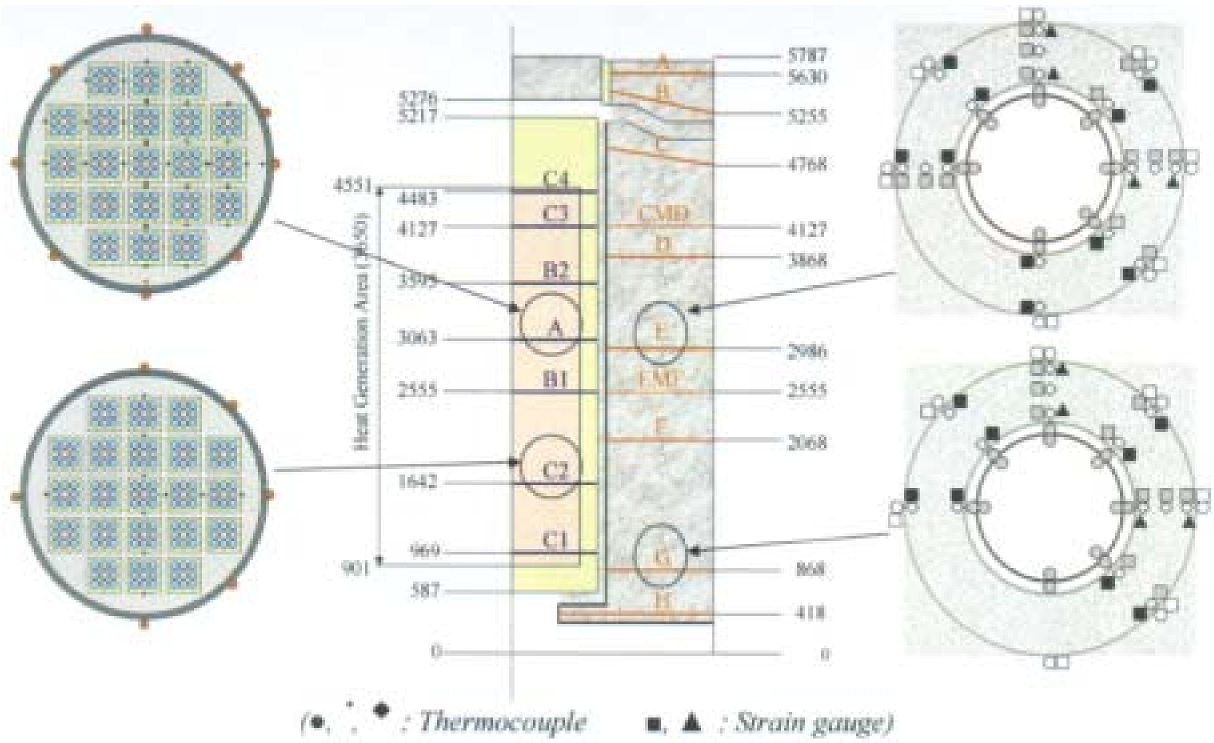


FIG. 9. Measuring points (temperature and strain).

3.4.4. Test results

3.4.4.1. Case 1

(a) Flow velocity and temperature

In case 1, the normal storage condition corresponding to the initial state during the storage is considered. The inlet air temperature was 23°C in the steady state. Fig. 10 shows the velocity distribution in the cross section of the 180°-inlet duct. Average velocity is 0.84 m/s, and there is not so much difference in the velocity distribution. The total flow rate of the cooling air is 0.28 m^3/sec.

Fig. 11 shows the velocity distribution in the cross section of the 90°-outlet duct. Steel plates for radiation shielding are welded near the exit of the outlet duct and divided into four areas. The velocity is measured in the centre of each area.

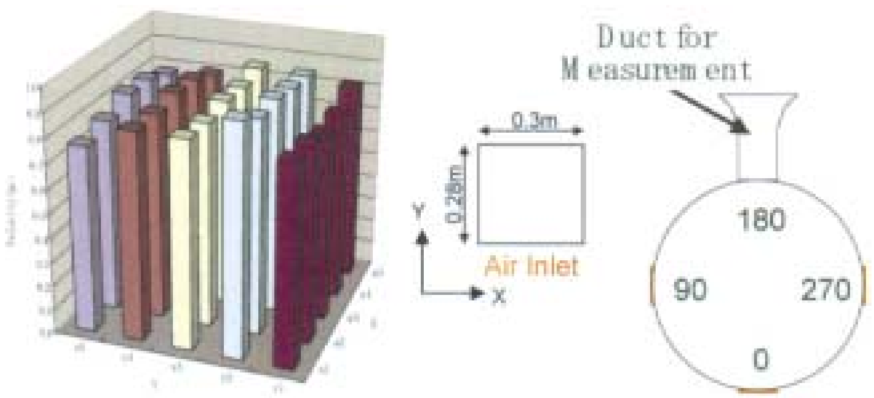


FIG. 10. Velocity distribution in the cross section of the 180° inlet duct.

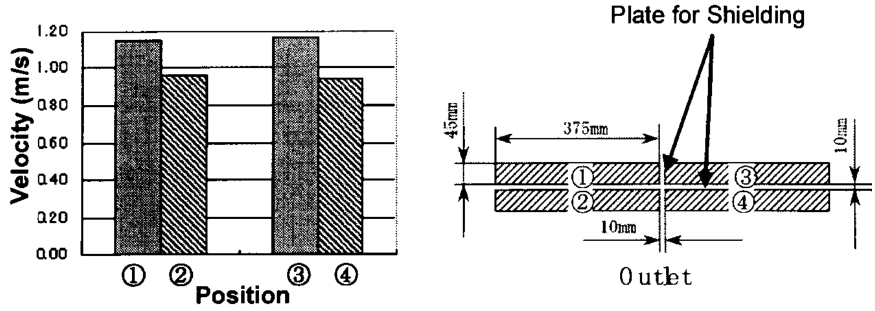


FIG. 11. Velocity distribution in the cross section of the 90° outlet duct.

Fig. 12 shows the air temperature measured at the each outlet duct. There is a large temperature difference between the upper and lower parts at the inside of the outlet duct. The air temperature of the upper part is considerably affected by the hot air going along the canister surface by buoyancy force. The air temperature of the lower part is only affected by the air going up through the flow area between the thermal shielding plate and the inner liner. On the other hand, at the outside of the outlet duct, the air temperature is almost the same because the exhausted air is highly mixed throughout the outlet duct. Temperature increase of the bulk air is about 65°C.

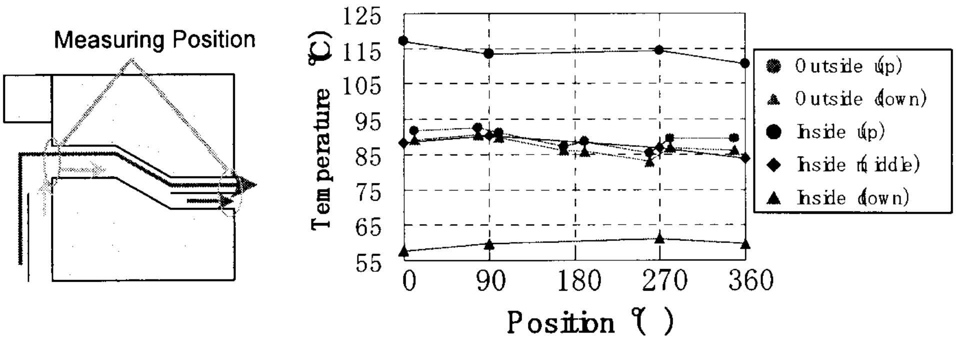


FIG. 12. Air temperature measured at the each outlet duct.

(b) Concrete temperature

Fig. 13 and Fig. 14 shows the temperature distribution of the concrete container and the picture measured by the thermo-viewer. The temperature distribution along the radial direction is almost linear and the maximum concrete temperature around the outlet duct is about 81°C. As this value seems to exceed the estimated value obtained in the pre-thermal evaluation, it is necessary to modify the evaluation method and preliminary design.

(c) Canister temperature

Fig. 15 shows the circumferential surface temperature of the canister comparing with the pre-calculation value. The circumferential surface temperature in the 45° direction is lower than that in the other direction because of the contact between the canister and the guide rail, and furthermore, the basket may also contact with the canister body in the 180° direction. Concerning to the longitudinal distribution of the surface temperature, there is not so good agreement between the experimental value and the pre-calculation results, especially in the upper part of the canister. Because of this temperature difference, the temperature of the concrete container (around the outlet duct and the bottom part of the lid) obtained in the pre-calculation value is considerably smaller than the test data. Therefore, it is very important to take account of the contact condition and the longitudinal heat conduction model in the preliminary design and evaluation.

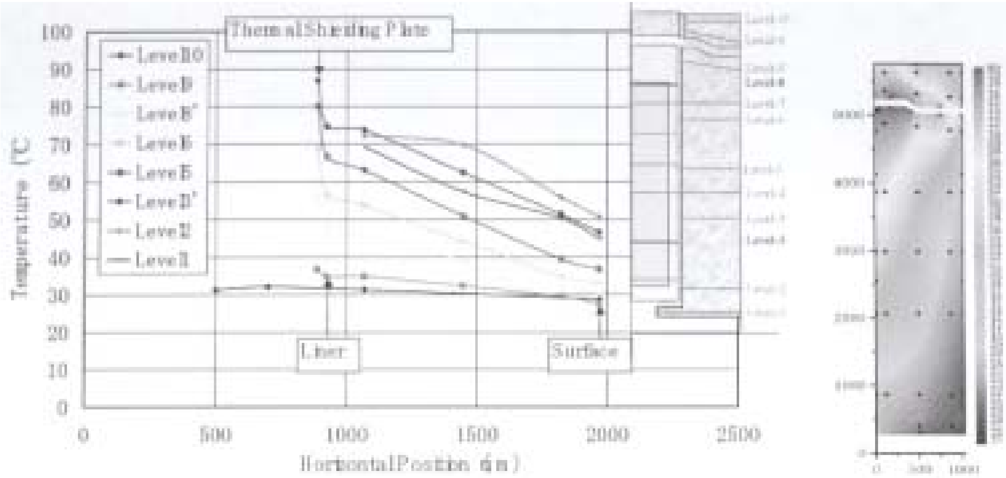


FIG. 13. Temperature distribution of the concrete container.

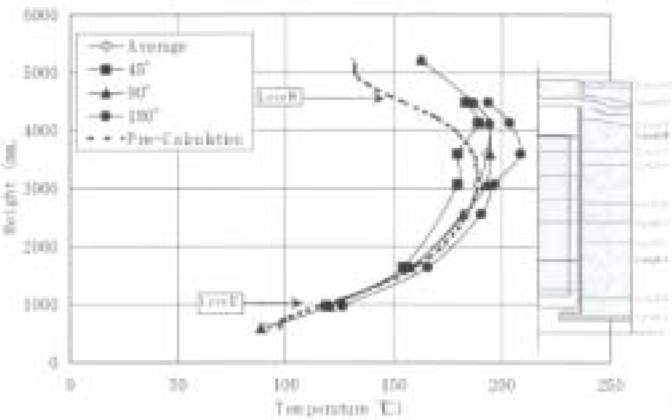


FIG. 14. Picture measured by the thermo-viewer.

FIG. 15. Temperature distribution of the canister surface.

Fig. 16 shows the temperature distribution inside of the canister in the radial direction. As the shape of distribution is almost symmetry, the temperature of distribution in the canister is not affected to the contact condition as mentioned above. Moreover, maximum temperature of the canister surface, basket and so on is lower than the allowable value.

(d) Heat balance

The heat discharged from the concrete cask to the environment is attained by the cooling air and heat transfer on the cask surface. In order to obtain the heat balance, the amount of heat removed by the air and heat transfer on the cask surface is calculated using air and temperature distribution data in the inlet and outlet ducts and temperature gradient in the concrete container. Fig. 17 shows the ratio of the heat balance, 80 % of the heat is removed by the cooling air.

(e) Strain and crack

During the test, the crack of the concrete surface is occurred. In the upper part of the cask, number of crack and its width are larger than that in the lower part. Fig. 18 and Fig. 19 shows the crack on the top surface of the cask and the relationship between the temperature difference and the crack width. The crack occurs and the crack width increases as the temperature difference between inside and outside of the concrete container increases, and moreover tension stress is appeared on the outside region, and compression stress on the inside region.

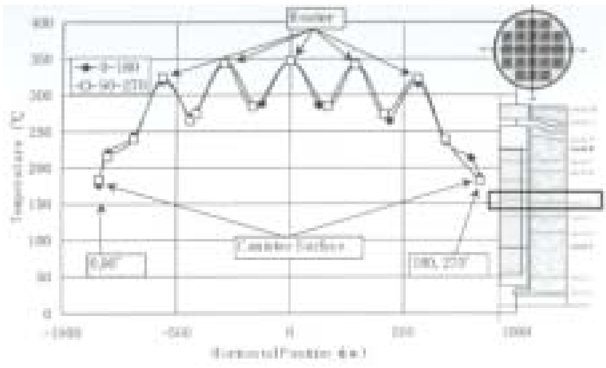


FIG. 16. Temperature distribution inside of the canister.

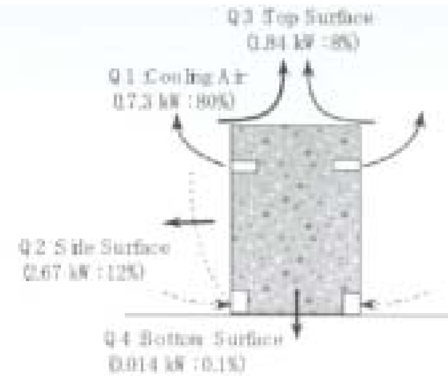


Fig. 17. Schematic diagram of the heat balance.

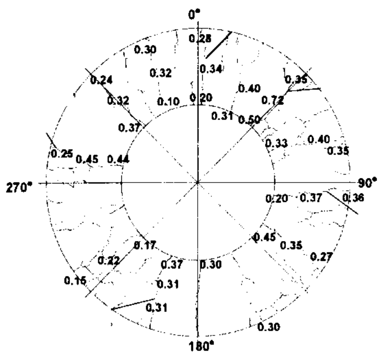


FIG. 18. Crack on the top of the cask.

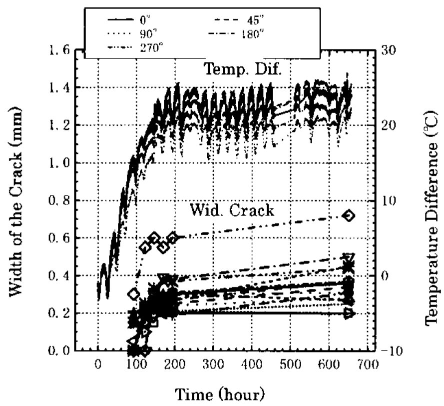


FIG. 19. Relation between the temperature difference and the width of the crack.

3.4.4.2. Case 3

This case is the condition of 50 % blockage of the inlet. After closing the inlet, the condition reaches the steady state as shown in Fig. 20. In this case, air flow rate decreases and air temperature of the outlet increases compared with the case 1. Judging from the temperature distribution, drift flow in the flow area which effects upon the temperature of the cask has not been observed. As the temperature increase is only 5°C, the influence of the 50 % blockage on the temperature seems to be small. Fig. 21 shows the temperature distribution of the canister and the cask body in the axial direction by comparing the test results between case 1 and 3.

3.4.4.3. Case 4

In this case, after the steady state of normal condition (case 1), all inlets were closed. The test was continued for 48 hours, but the condition does not reach the steady state. For the test period, cooling air does not go out from the outlet. The temperature of the canister and the cask continues to increase. Fig. 22 shows the temperature distribution of the center of the guide tube.

3.4.4.4. Cases 2 and 8

For test cases 1, 2 and 8, the heat power is considered as a test parameter. Data with small heat power is necessary to evaluate the condition in the middle and final state of the storage. Especially, as the temperature of the canister surface goes down by the heat power decrease, it is important to evaluate the cold part of the canister surface from the point of view of stress corrosion cracking.

Fig. 23 shows the temperature distribution of the canister surface among test cases 1, 2, and 8. According to these test results, it is found that temperature of lower and upper part of the canister is relatively low.

3.5. Seismic test

The concrete cask will be preferable to be oriented vertically in the freestanding condition [3]. In order to evaluate the tipping-over phenomena under strong earthquake motion, the excitation tests were performed with a scale model concrete cask using two-dimensional shaking table test, and the applicability of the energy spectrum approach was discussed.

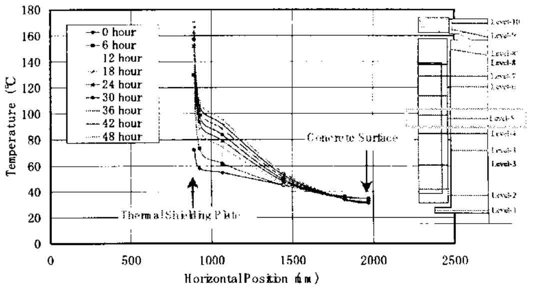


FIG. 20. Temperature distribution of the cask body.

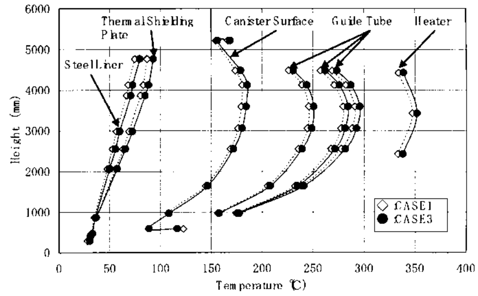


FIG. 21. Comparison of temperature distribution in the axial direction between Case 1 and 3.

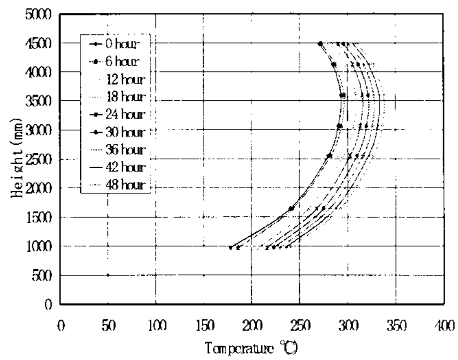


FIG. 22. Temperature distribution of the guide tube (Center of the guide tube).

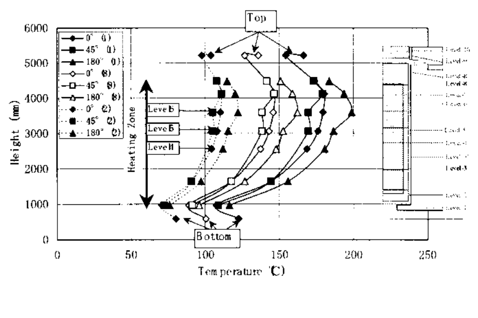


FIG. 23. Temperature distribution of the canister surface (Comparison among Cases 1, 2 and 8).

3.5.1. Scale model cask

The scale model cask and model floor were set on a two-dimensional (horizontal and vertical) shaking table as shown in Fig. 24. A scale model cask including the canister model was fabricated based on the similarity law referring the configuration of the RC type to simulate the effect of the gravitational acceleration on the tipping-over condition of the cask. The scaling ratios for acceleration, geometry and bottom stress were set to 1, 1/3, 0.95, respectively. A 30 cm thick reinforced concrete slab was used as the floor model. During the seismic excitation test, the angle, angular velocity, acceleration and displacement of the cask body and the canister were measured.

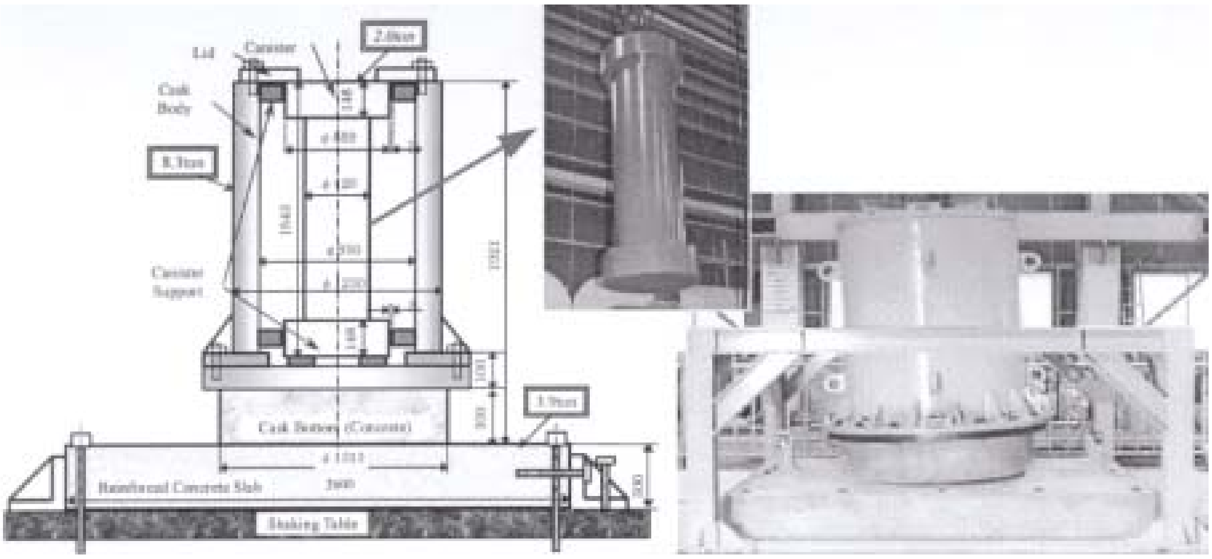


FIG. 24. Scale model cask.

3.5.2. Test condition

For input of the seismic excitation test, recorded waves during typical natural earthquake waves and artificial seismic waves were employed. Time duration of the input wave was scaled (1/1.73) according to a similarity law and the acceleration levels were varied according to the test conditions. Test condition includes the cases considering horizontal and vertical motions simultaneously. Moreover, the effect of the gap distance between the canister model and the cask body on the overall response of the scale model cask was also investigated.

3.5.3. Test results

3.5.3.1. Rocking response

Before the test, the damping ratio for the rocking vibration and kinetic coefficient friction between the scale model cask and the model floor were measured and set to 0.066 and 0.7, respectively.

During seismic response of the scale model, three-dimensional behavior like top-spinning was observed. However, the residual sliding displacements were very small. Fig. 25 shows the example of the test results using the wave recorded during Hyogo-ken Nanbu earthquake occurred in 1995.

The increase of maximum response angle by the effect of the vertical motion was up to 20%. It is also found that the existence of the gap between cask body and canister decreases the rotational angle response of the model cask.

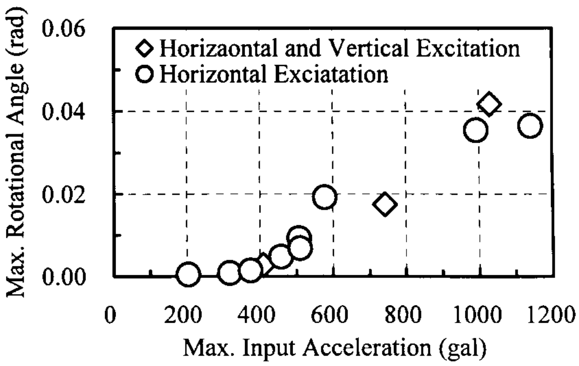


FIG. 25. Maximum rotational angle response of a scale model cask for excitation tests.

3.5.3.2. Tipping-over criteria by energy spectrum

Akiyama et al. [4] proposed the estimation method for tipping-over of the two-dimensional rigid rectangular body based on the energy spectrum approach. If V_{Ereq} and ${}_{ou}V_E$ are defined as the equivalent velocity calculated from the critical potential energy of the rigid body and the equivalent velocity calculated from input energy to rigid body, respectively, the criteria for the tipping-over of the rigid body with energy spectrum is defined by equation (1).

$$ {}_{ou}V_E(a) < V_{Ereq} \tag{1} $$

Fig. 26 shows the relationship between the equivalent velocities for input energy and for potential energy necessary for uplifting the scale model cask during the excitation tests with the Hyogo-ken Nanbu recorded wave. As the input acceleration level increases, the experimental value is approaching to "${}_{ou}V_E$=V_{Eresp}-line", and all of the energy accumulated to the model cask by the earthquake wave are consumed for uplifting the model cask. Therefore, it seems that the energy spectrum criteria is very useful and practical to estimate the possibility of the tipping-over of the real cask during the storage subjected to the strong earthquake motions.

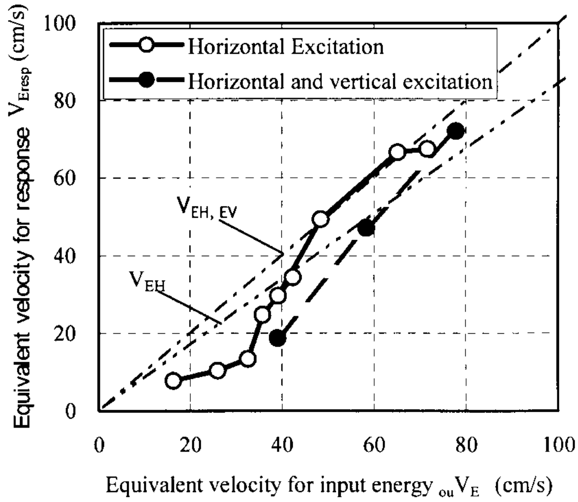


FIG. 26. Relationship between the equivalent velocities for input energy and potential energy.

4. Conclusion

The demonstration tests with the full-scale concrete cask are successfully in progress. The heat removal test using the full-scale RC type cask and the excitation tests using the 1/3 model have been carried out and evaluated.

Until the end of 2003, also the heat removal test using the full-scale CFS type cask and two drop tests with full-scale Type I and Type II canisters in horizontal and vertical orientations will be implemented.

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Optimization of cask capacity for long term spent fuel storage

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Abstract. Within the framework of the IAEA Subprogramme of Spent Fuel Management, a new project was conceived, focusing on issues associated with the optimization of cask/container loading (capacity) with respect to long term storage and the related integrity of fuel. An initial Consultants Meeting held in November 2002 identified and discussed principal issues regarding the optimization of cask/container assembly capacity and burnup/age capability in the design of systems for long term spent fuel storage and the related integrity of fuel. Based on resulting working materials, a Technical Meeting was held in March 2003 to obtain country-specific views from both regulators and implementers on this topic. Discussions focused on the following issues relevant to cask loading optimization: fuel integrity, retrievability, zoning, burnup credit, damaged fuel, computer code verification, life of cask components, cask maintenance, performance confirmation, and records management. Follow-on actions and meetings will be pursued to develop a TECDOC on this subject.

1. Introduction

Long term storage of spent fuel is a priority topic within the Member States of the IAEA. Technical meetings held by the Agency in 1999 and 2000 resulted in TECDOC-1293 [1], which focused on the challenges to extending the life of existing and new storage facilities for much longer periods of time. In addition, TECDOC-1343 published earlier this year [2] reported on the Co-ordinated Research Project on Spent Fuel Performance Assessment and Research conducted from 1997 to 2001. That report identified areas of technical interest as storage durations extend, while noting that dry cask storage of spent fuel is playing a steadily increasing role. In this context and within the framework of the IAEA subprogramme on spent fuel management, a new project was conceived, focusing on issues associated with the optimization of cask/container loading (capacity) with respect to long term storage and the related integrity of fuel [3].

An initial Consultants Meeting held in November 2002 identified and discussed principal issues regarding the optimization of cask/container assembly capacity and burnup/age capability in the design of systems for long term spent fuel storage and the related integrity of fuel. Working materials developed during that meeting noted that cask designers currently face a number of new challenges including storage of high burnup fuel with correspondingly higher enrichments, the use of mixed oxide (MOX) fuel, and obtaining regulatory approval for the use of burnup credit. Optimization might have different meanings for the cask vendor, the cask owner, the cask operators, and the institution having the ultimate responsibility for the storage, the Licensing and Supervisory Authority.

The working materials resulting from that consultancy were then provided to participants in a Technical Meeting held in March 2003 to obtain country-specific views from both regulators and implementers on this topic. Participants in the technical meeting reviewed the results of the consultancy and then provided country-specific perspectives on the topic. Thereafter, participants formed two working groups to focus on implementer/regulator views of the role of optimization in cask design and open issues identified both during the consultancy and during the technical meeting.

2. The role of optimization in cask design

The purpose of this section is to describe the optimization process within the context of cask design, in general, and storage cask design, in particular. The general objective of cask design is to provide a cask with the largest possible capacity of spent nuclear fuel (SNF) assemblies with the largest burnups and shortest cooling times that are practicable. The cask must also meet the weight and dimensional limitations defined by the application, and must meet regulatory requirements with appropriate design margins. Optimization occurs because many of the cask safety limits such as dose and fuel temperature limits can be met only by imposing limits on the various cask performance measures such as assembly capacity or burnup/age capability. Optimization is thus the part of the design process in which the combination of application objectives, regulatory limits and design margins are innovatively addressed and judiciously balanced in the final design. A primary result of a successful design optimization is a cask of superior assembly and burnup/age capacity that minimizes the total number of required cask loadings. An equally important and parallel benefit is that this process also results in reduced overall radiation exposure, thereby contributing significantly to ALARA objectives [4]. In this sense, both cask designers and regulators have the common ultimate goal of improving cask performance within regulatory limits, and thus of facilitating the optimization process.

There is an additional optimization consideration that is specific to storage casks: assurance of the integrity of the fuel cladding and assembly structural components. This is required both to validate intact configurations used in criticality analyses and to assure subsequent retrievability. In many storage applications, this issue is complicated by uncertainty as to the duration of the storage period that will be required prior to the ultimate disposition of the stored SNF. The expected useful life of the fabricated materials that make up the physical cask components is generally expected to be more than 100 years, assuming reasonable monitoring, care, and maintenance. However, the general requirement for storage is that the SNF contents of the cask remain isolated from the environment and maintain their cladding integrity, and as a minimum be mechanically removable from storage without significant complication. In the latter regard, the experience with dry storage is not long but gives encouragement that extended periods of dry cask storage are a realistic expectation. However, extrapolation of current data on storage performance to longer storage periods and higher burnups must include an uncertainty that increases with the degree of extrapolation. In that regard, the concept of “long term storage” can be defined by the following time periods:

- Initial period with little uncertainty as to storage safety. Design basis conditions would fall within this period;
- An intermediate period during which the predictability of performance has larger, but reasonable uncertainty;
- The subsequent period during which the predictability of performance is much less certain, requiring greater analysis and more data.

It is noted that storage monitoring programmes can be used in conjunction with “long term storage” to provide the basis for continued storage, or remediation, if necessary.

Optimization has always been a part of the design process for storage and transport casks. However, prior designs were frequently for the storage of long-cooled, lower-burnup fuels.

Test casks were also loaded with these fuels and periodically checked as a part of government-funded national programmes, resulting in much of the data currently available on fuel performance. The situation is now progressively changing:

- fuel burnups are steadily increasing and will continue to do so;
- storage requirements for spent MOX fuels are developing;
- the available storage performance data applies principally to the lower-burnup fuels;
- as more cool fuel is moved into dry storage, shorter-cooled fuel may be transported first since the fuel in dry storage tends to be less resource-intensive.

These factors increase the challenges of cask design, demanding innovation in cask technology and requiring better analytic methods and benchmarking data to reduce uncertainty in design margins as regulatory limits are approached.

3. Methods for achieving optimization

Taking into account the storage cask design issues and the sometimes-conflicting requirements raised by the above analysis of the cask design requirements, it becomes evident that there is both a strong need and a significant benefit to be realized by designing to achieve improvements in cask performance.

These improvement could be realised in different ways:

- **improvement in terms of assembly capacity** of the cask and meeting all weight and interface requirements for loading at the nuclear power plant (NPP). This could result from reduced assembly spacing within the cask basket, and/or improved burnup/age capability;
- **increased material performance** of the cask or use of new package materials (e.g. shielding, structural, thermal), including the qualification of these materials at high ranges of temperature and radiation.

An approach to reach these goals may consist of reducing uncertainty in design margins using more calculations of greater precision. This would take advantage of additional benchmarking performed so as to improve the qualification of codes and calculation methodology.

Once identified, the reduced design margin uncertainty can be achieved via increased sophistication of both cask design and content definition (inventory list) in the areas of shielding, structural, thermal and criticality design. This increased sophistication can occur in both the software and hardware areas, with software referring to the methodology of analysis (e.g., assumptions, definition of the content) and hardware referring to the tangible design itself (e.g. physical properties of the design).

Several of the design methods that illustrate the increased level of design sophistication are discussed in the following paragraphs.

The first optimization method involves a zoning approach in which defined zones of the cask radial cross section are loaded with spent fuel of different characteristics, with the objective of increasing the burnup/age capability of the cask. It could be implemented by the definition of loading patterns in the design license that contain fuel of defined characteristics, achieved and verified during the loading of the fuel elements into the cask. This approach could be simultaneously used for:

- increasing shielding effectiveness, with fuel elements with higher source term in the central area of the cavity so as to use the self-shielding properties of fuel elements;
- criticality analysis;
- thermal analysis.

In implementing the zoning approach, a balance should be made between the advantages (in term of capacity) given by the zoned loading pattern and the flexibility needed by utilities for loading the cask with a large variety of fuel elements. The difficulties raised by a potential lack of flexibility are avoided in the case of a fuel inventory that is known and fixed prior to cask design. Cask optimization in this particular case could be very efficient in term of an increase of cask capacity.

With regard to thermal optimization of the cask, a primary goal is to improve heat transfer so as to decrease the temperature of the cladding as much as possible, thereby avoiding or reducing the risk of cladding degradation due to high temperature. Particular attention should be paid to:

- the reduction of gaps and interfaces in heat conduction;
- the choice of adequate material;
- the adaptation of the external shape of the cask for thermal control.

For thermal optimization of the cask, full advantage should be taken of all cask material properties including materials designed for other primary functions. For example, take advantage of thermal and mechanical benefits by considering material designed for criticality control.

The concept of loading patterns could be extended to storage site management. A dynamic site emplacement pattern for cask storage could be developed to maximize heat removal from recently-loaded casks, with subsequent repositioning of casks so as to maximize site cask storage capability within thermal and radiation dose limits.

A second general optimization method focuses on refinement of regulatory practices. The impacts of increases in the fuel initial enrichment could be counteracted by:

- reducing the stack-up of conservative assumptions in the analysis;
- validating and implementing burnup credit in a qualitative way;
- validating and implementing burnup credit in a quantitative way (such as actinide and fission product approachs). The individual fuel assembly characteristics must be confirmed within the established quality assurance system, with the possibility of additional controls at the time of loading (i.e. confirmatory burnup measurement).

Current regulatory practice for cask storage does not generally account for transport occuring many years into the future, and that heat and source term decay could be significant. There are potential benefits in cask performance if regulatory review accommodates such decay in design concepts.

Refinement of regulatory practice can also result in realistic long term dose rate management on the site. If there is a deliberate bias in favour of neutron shielding relative to gamma shielding in the original design, the subsequent slower decrease of neutron sources relative to gamma sources will result in lower total long term doses than if the normal neutron/gamma shielding tradeoff had been made in the original design. Ultimately, the foregoing approaches open the possibility to admit more casks over the storage period without increasing dose rates or total thermal power allowed in the facility.

Additional possible solutions in particular circumstances are:

- Any physical limits on the cask (overall dimensions and weight) that are imposed by the required compatibility with nuclear power plant (NPP) interfaces could be bypassed by not loading the cask in the NPP. A dedicated smaller cask and additional cask handling operations could be used to transfer the fuel from the NPP pool to the storage cask in a controlled loading area;
- The use of qualified internal criticality control material in pressurized water reactor fuel elements might be a cost-effective way to accommodate outlier assemblies with insufficient burnup relative to initial enrichment;
- Typical approaches for dealing with damaged fuel are the use of a sealed bottle, or the use of a vented canister. These choices may affect the cask design with respect to heat transfer and mechanical design, as well as the handling and drying procedures. Since experience with storage of damaged fuel is limited, the need to store damaged fuel elements could have an impact on cask and basket design and on loading patterns;
- There are tradeoffs to be made in selecting between welded and bolted lid systems for long term containment. In the case of welded lid system, more initial effort has to be spent on quality control during welding and periodic inspection of weld integrity is needed. In the case of bolted lid systems, specific monitoring systems and operational monitoring programmes are needed to ensure uninterrupted control of the tightness of containment barrier.

It is clear from the foregoing that the cask design optimization process requires increased sophistication in both the software and hardware aspects of design. Because the potential benefits from reductions in design margin uncertainty on material performance are not unlimited, a current focus of optimization is on software developments. The ultimate limit on design optimization is likely to have been achieved by reducing design margin uncertainty to the point beyond which there are increased operational controls at loading, and increased restrictions on utility flexibility with respect to the selection of assemblies for cask loading.

4. Open issues

1) Fuel Integrity: As the design/manufacturing of fuel assemblies (and in-reactor conditions they experience) evolve, further research and development is needed to assure fuel integrity during storage, specifically dry storage. Since the requirements for cladding features for long term dry storage are evolving, cladding features unique to long term storage are not specified in fuel designs, and the predictions of the long term behaviour of cladding may have significant uncertainties. For example, work related to the following parameters is required: creep, cladding absorption of hydrogen, stress corrosion cracking, oxidation, internal gas pressure (helium build up).

2) Retrievability: Retrievability requirements must be defined in national contexts with specific requirements defined as early as possible in any project. Depending on national policy, retrievability requirements could vary significantly. As storage durations lengthen and fuel designs/conditions evolve as noted above, subsequent retrievability involves more uncertainty. Accordingly, appropriately targeted monitoring programmes may help address related concerns.

3) Zoning: While the potential benefits of zoned cask loading should be investigated with respect to criticality, shielding and heat removal, meeting representatives noted that there needs to be a balance between the potential advantages of zoning and operator flexibility in dealing with a variety of fuel assembly characteristics. They further noted that zoning must consider the total cask system and related requirements (e.g. both storage and transportation if dual-purpose).

4) Burnup Credit: Burnup credit is an important consideration in this optimization process. In order to pursue the storage-related advantages of burnup credit, it is necessary to have good knowledge of spent nuclear fuel characteristics, from both measurement and calculations. Quality assurance associated with supporting data remains a key prerequisite to burnup credit implementation. In parallel, the methodology for the measurement and confirmation of assembly burnup will require development/refinement. Both meetings endorsed the importance of considering burnup credit, with national representatives noting that related costs and benefits must be evaluated in specific national contexts.

5) Damaged Fuel: Regulatory conditions for storage of damaged fuel, including containment, should be established in a clear, consistent context (i.e., no “universal” definition exists). Regulatory representatives noted “tightness” of potentially damaged (mechanically) fuel rods required further definition

6) Internal Moderator: The use of internal criticality control was identified in November as an optimization aid. Technical meeting representatives concluded that further development of this topic should be in the context of specific concepts.

7) Computer Code Verification: Additional qualification of codes may be needed for specific designs, with increased detail and precision based on appropriate additional benchmarks. As an example, improved data for source term definition is needed for use at higher burnups and enrichments.

8) Life of cask components: Specific cask components critical to extended storage life should be identified, with a view to reducing needed maintenance. Components of interest include e.g., lifting trunnions, seals, neutron moderator, monitoring equipment.

9) Long term cask maintenance: It is necessary to evaluate cask design implications to ensure appropriate maintenance during the extended storage period.

10) Long term performance confirmation: To assure that facilities and components operate as expected, monitoring programmes for radiation, temperature, etc. may be carried out.

11) Long term records management: Effective management and protection of storage-related data is a key condition for long term spent fuel management in general and for optimization efforts in particular. As data storage technologies evolve and as personnel rotate, continuity of knowledge will require continuing attention.

5. Conclusions

The above meetings served as key steps toward developing a technical resource available to IAEA Member States on this subject. Follow-on actions and meetings will be pursued to develop a TECDOC on this subject.

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Selection of AFR facilities for spent fuel storage

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Abstract. With the current trends toward extended storage of spent fuel in most countries, the demands for AFR facilities for interim storage of spent fuel is to continue to increase in the future. In addition to the classical method of storing spent fuel in water pool which has been reliably used for many decades, the successful development and commercialisation of dry storage methods by the industry in the past couple of decades has brought some powerful options for AFR storage, with particular advantage for long term storage, among others. The dry storage technology has begun to dominate recent market for new builds of spent fuel storage systems. Although it can be said with confidence that competitive services and products are currently available for a variety of technical options from the globalized market, it is often not evident how to choose the best option because of the complex factors to be considered in the decision. Whereas there are some given requirements imposed to all options, some other criteria are open to the choice of customers. It should be noted that the surrounding issues at local, national or international dimensions are dynamically changing in the present world of globalization, which might emanate some overriding impacts to the choice of option and implementation strategy. All those important factors will have to be systematically assessed and taken into account in the selection of a best option for AFR facilities for spent fuel storage.

1. Introduction

In the past, the shortfall in temporary pool storage capacity at a number of nuclear power plant (NPP) sites have been mitigated by increasing storage densities with such methods as re-racking of fuel in the pools or trans-shipment of fuel among pools. These temporary measures taken at reactor (AR) pools have substantially contributed to the efficiency improvement of storage and prolongation of the need to build additional facilities. Since a couple of decades ago, however, those easier methods have begun to be used up requiring additional storage by AFR type of facilities, often in response to capacity building measures for lifetime operation of the reactors. The successes in increasing the nuclear capacity factor and reactor life extension initiatives in several countries have further increased the need for storage. Furthermore, as more and more reactors get decommissioned in the future, the spent fuel currently stored at these reactors would also require to be removed from the NPP sites to AFR storage. In countries that do not intend to remove the spent fuel from storage until definitive plans are firmed up, an adequate AFR storage facility would be required to provide for interim storage until the future destination of the spent fuel in the fuel cycle is known [1].

In the light of these global factors, it can be predicted that the extent and duration of spent fuel storage will likely increase in this century, increasing in turn the need for AFR storage. It is also recognized that current trends to privatisation and globalisation of the nuclear power sector are likely to further spread with dynamic impacts on spent fuel management. Political, socio-economic and public acceptance factors are known to play a significant role. Competitiveness and economic factors would influence AFR storage to be brought into service where needed safely, economically and in a timely manner. As a result of these factors, there have been significant developments recently in the spent fuel storage business. Of importance among these developments are:

- Dry storage technologies have emerged to become a mature international industry offering a wide range of options, including leasing of equipment and services, with an increasing degree of innovation for the growing need of AFR storage in many countries.

Maturing of technology and competitive markets have positioned dry storage technologies for accelerated growth in the coming years;

- Public involvement as a criterion in the selection process has been given higher priorities in many Member States. There has been increasing public input in many countries into decision-making, and particularly in areas such as legislative bases and policies, site selection, assessment of impacts, and their relation to decision-making;
- Trend towards bid competition due to globalization of the market economy now requires greater effort towards evaluation of alternatives and preparation and evaluation of bids in line with modern contract management methods. Project and contract management have therefore become a major task in the efforts for selection and implementation of cost-effective AFR storage for majority of projects today.

The selection of AFR storage facilities is in fact a very critical step for successful implementation of a spent fuel storage project, due to the subsequent penalties involved in changing the option especially after the construction of facility. It should be noted that the focal issues in selecting an AFR storage facility can be shifted from time to time as spent fuel management strategies and technologies advance and can differ from one country to another due to considerations particular to those countries. Many of the issues that may arise in the process of acquiring AFR spent fuel storage relate to a variety of areas, such as need assessment, feasibility studies, design, licensing, environmental assessment, public consultation and contract management [2].

When considering the AFR storage as a solution to manage spent fuel, it must be recognised that this is not a final solution. Spent fuel would have to be eventually retrieved from AFR storage. The ultimate solution for the spent fuel would either be direct disposal or reprocessing (or perhaps other emerging options like transmutation). In instances where these solutions have not yet been put in place, there will likely be requirements to foresee these needs at the time the AFR facility is being designed or licensed and have necessary arrangements in place in order to cope with this future situation. Plans may also be required for extended management of the spent fuel when a decision is made to retire the AFR storage facility in circumstances where ultimate solution (disposal) may entail a very long wait period (a number of decades). An interesting question in this regard is the technical option for developing dual or multiple purpose cask or container. In spite of the significant benefits anticipated by standarization of container design for dual multiple purposes, the absence of disposal package design and its compatibility with existing systems is a pending issue to be resolved in the future.

2. Selection process

The procurement cycle for the AFR storage is initiated generally by the nuclear power plant (NPP) owner (or operator), who recognizes the existence of the need and has the authority and resources to fulfil such a need. Because of the long-range system planning and strategic studies normally carried out by most NPPs on a continuing basis, the need for AFR storage is well foreseen in most of these organizations, and assessment studies for AFR storage are often initiated well in advance of the project. The nuclear power plants take the necessary steps to meet the need and generally become the project sponsors or proponents for AFR storage as well as its ultimate customers and operators. Key resources required to put in place an AFR storage system relate to various project activities such as technical studies, licensing, design, construction and commissioning. In some other countries, some public or private service organizations are designated for the management of spent fuel and often together with other types of radioactive waste.

A project management organization could be hired to carry out the initial planning process. Alternately, key individuals could be brought together to form a team and a project manager appointed. The project manager in turn will act on behalf of the NPP or the waste management organization and establish a project delivery strategy identifying the key components such as a project plan, technology selection, regulatory approvals, bids invitation and evaluation, awarding the contract, detailed design, procurement, construction, commissioning including training of staff for operating the AFR storage system, and turning over the system once completed to the organization charged with the task of operating the AFR storage system. The project manager could also establish expert advisory groups to advise the project organization in various specialized areas.

The steps involved in AFR storage implementation can be broadly divided in two phases: one, the technology selection phase which focuses on the selection of the appropriate conceptual alternative and its use in an AFR storage facility leading to a contract award, and the other, engineering and construction phase which focuses on the implementation of the physical systems by the contractor. The general process of selection includes the following steps:

1) Defining the need and scope;
2) Selecting the technology:
    a. Clarifying what is needed;
    b. Identifying technology options;
    c. Identifying criteria and methodology for the evaluation of options;
    d. Preparing a feasibility assessment and identifying feasible options;
    e. Selecting site;
    f. Selecting transportation system;
    g. Obtaining regulatory approvals;
    h. Making the business decision and assessing risks;
    i. Preparing Functional Specification;
3) Inviting bids;
4) Evaluation of bids;
5) Awarding the contract.

3. Technical options

The technologies currently available for spent fuel storage fall into two categories so distinguished according to cooling medium used. The technologies are distinguishable by their major characteristics, namely, the predominant heat transfer methods, type of shielding, transportability, location with regards to the geological surface, degree of independence of the individual storage units, and the storage structure [3].

3.1. Wet storage

Water pools are the most common option for storage of spent fuel immediately upon discharge from reactors, since they provide excellent heat transfer essential in the early phase of cooling. At the nuclear plants, these pools are generally integrated with the plant design and spent fuel management in these pools is part of the plant operation. For a long period, the wet storage of spent fuel using water pools was the predominant storage method. As an established practice since the early days of nuclear power, water filled pools have been used almost exclusively for initial shielding and cooling of spent fuel discharged from reactors for some technical and economical features. Water pool storage is however also being considered for AFR storage facilities by virtue of the large amount of experience available with this technology in addition to some inherent merits of water as a medium for spent fuel storage. Water pool storage, however, requires active process systems to ensure satisfactory performance and continuous attention to preserve water purity. Protection against the possibility of air crash from the sky is a newly raised issue, to which some studies on storage systems have been conducted. A new design of water pool with such advanced features as passive cooling and protective roof against airplane crash, with a view to amend the drawbacks of water pools has recently appeared [4].

3.2. Dry storage

The spent fuel assembles become suitable for naturally cooled dry storage after a few years of initial cooling in the water pool. The minimum required time of initial cooling in pools is mainly related to the burn-up and the irradiation history. Taking into consideration the 20-50 years or even longer period required of storage, it is obvious that the naturally cooled dry storage facilities could be an attractive alternative to water pools.

A review of spent fuel storage facilities implemented during the last 10 years show that the storage in a dry environment is becoming more common. There are several generic types of these technologies available from vendors on the international market. There are also a large number of facility designs based on these generic technologies that are now available. These technologies differ largely in terms of materials of construction, size, modularity, spent fuel configuration, layout of the storage containers (horizontal, vertical etc) and methods for fuel handling. Multi-purpose technologies (i.e. a single technology for storage, transportation and disposal) have also been studied in some countries [5]. Further differences could be in terms of their placement above or under the earth's surface. An increasing number of storage facilities are coming into operation in each of these types. Although there is no clear favourite technology world-wide, dry storage of spent fuel in casks is being particularly recognized as a flexible option with the advantages of transportability in case of future need, and the option of leasing of casks from vendors.

4. Requirements

Any plan to acquire AFR facilities for spent fuel storage should be based on feasibility study by looking at the associated constraints including the given requirements to which the facility is subject. The key among these requirements are the functional ones from which a variety of associated requirements are derived.

It is important in the feasibility study to develop the practice of documenting all the assumptions made in the formulation of requirements such that there is a clear understanding of how the requirements were arrived at during the early stages of the project. The listing of assumptions allows the project staff to review the requirements during the project if necessary and also helps the bidders at a later stage to understand the requirements and propose changes if they find themselves unable to meet them.

The functional requirements would have to be based on the information on the spent fuel including its characteristics, amount to be stored, with which the facility in need can be designed in consideration of other associated requirements such as site and infrastructure, transportation, resources, project management. These are conditions requiring considerations in the development of the project. There are another categories of requirements set up by institutional decision: safety and licensing, environmental impacts, public involvement.

4.1. Spent fuel information

As a first step in the above process, it is necessary to identify general requirements determined by the quantity and the characteristics of the spent fuel. It may not be possible or necessary to foresee fully the envisaged AFR spent fuel storage capacity since nuclear programs in most countries are continuously changing. Decision will have to be made according to available projections of spent fuel arising and remaining pool capacity at the nuclear plants. Allowance would be required at the at-reactor pools for contingencies such as removal of reactor fuel load in emergencies (referred to as core discharge) and pool operational contingencies. Allowances would also be required to deal with potential project delays in planning AFR storage. A staged, modular approach may well be more appropriate to satisfy immediate needs (i.e. several years of storage) of capacity building and for planning provisions for future extensions. Future requirements could be included at the initial design stage at a preliminary conceptual level and refined at the time of modular expansion of the AFR storage systems.

It is important to recognise that spent fuel is made of reactive materials and will be subject to physical and chemical changes over time. These changes may affect the integrity of the spent fuel in storage and there from the overall safety of system. Therefore, adequate provisions must be made to take account of these changes that may arise both during irradiation and following discharge from a reactor. Defective fuel may require special attention in terms of canning them prior to storage in an AFR (if it is not done already at the reactor pools). This may require special size containers for storage as well as transportation if they do not fit into standard containers. An agreed upon determinant or criterion (usually based on sipping procedures) would be required to control the identification and canning of defective fuel. There may also be plans to consolidate the fuel (i.e. remove non-fuel components and increase storage density) for which canning is also required at the time of AFR storage or after for disposal purpose (encapsulation).

Nuclear fuel designs have been changing over time in attempts to improve their characteristics. Utilities have been increasing their use of higher burnup fuels, a trend that is likely to increase and envelope different types of fuel such as MOX fuel. Based on the characteristics of the spent fuel existing at the time of the selection of the AFR storage facility, some allowance may have to be made to make room for the future development of the fuel used in the reactors. The modular approach for AFR storage will allow the required flexibility to take into consideration any unforeseen changes that could take place in the future including changes to fuel characteristics, containers, regulatory requirements, and the knowledge base of storage systems. A modular approach will also allow future improvements in storage systems themselves to be accommodated based on lessons learned in the initial stages and from feedback from storage operations.

Overall, some effort may be required to define acceptance conditions for spent fuel in the AFR storage such that AFR storage design requirements can be developed such that they are compatible with the received spent fuel. This will require cooperation between the NPPs and the project staff such that any extraordinary technical difficulties can be identified in advance and resolved in the best possible manner.

4.2. Siting and transport conditions

Any design and construction of AFR storage is closely tied to the site where the facility is to be located. Consideration to siting options is therefore an important part of any AFR storage selection. Site conditions must fit the initial intent for the AFR storage facility that may consist of alternatives such as a single national facility, several facilities at various local sites or even regional locations shared between two or more countries. Preference may be given to on-site storage at sites already involved in nuclear activities (such as NPPs) for the reasons of sharing existing infrastructure. Moreover, local communities at such sites may already be familiar with nuclear undertakings and may be more favourable to hosting an AFR storage facility than communities at non-nuclear sites. Some countries may have other preferences, such as collocation with an eventual disposal site or reprocessing sites. In the case of regional locations, it is important to recognise and give proper attention to the international obligations that may apply to such locations.

Site characteristics are essential features that may take considerable attention in making a proper decision on an AFR storage selection especially in case of a greenfield facility. These are not only important for engineering design of an AFR storage facility, but also for safety assessments and environmental impact assessments. Of importance are site data that are required for constructing a facility, site-related natural phenomena pertaining to storage safety (such as earthquakes, floodplains etc), and environmental and social factors. Site selection and decisions could involve in most cases a range of stakeholders, particularly local governments (municipalities) and affected communities.

Transportation of spent nuclear fuel may require considerable attention in any AFR storage project since the storage facility may be hundreds of kilometres away from the nuclear plant site. Discussion may be required with the shipping agents to ensure that transportation plans are practical and logistics are implementable. On-site operations for preparation of transportation containers, loading and unloading, contamination control and inspection would require significant effort both at the AFR storage site and the NPPs. Degree of attention required may differ from country to country depending on its familiarity with this technology and public acceptance factors. Transportation regulations may be different from one country to another, although with respect to transportation containers, Member States have generally adopted IAEA regulations. Details of transportation regulations are beyond the scope of this document. The reader may take advantage of existing IAEA literature in this regard [6.7].

The general requirements of transportation and handling should be identified at the beginning of the selection process including the accessibility for rail/road/water transport from the NPP to the AFR storage site. If there is any preference related to the fuel handling and preparation before storing the fuel in the AFR facility (such as spent fuel drying, inert gas filling and sealing of containers before placing in dry storage), it should be defined. Such preferences may also relate to the location where such activities are carried out, i.e., NPPs versus AFR storage site. Licensed transportation containers are usually readily available from a variety of suppliers or can be readily developed to meet specific requirements if necessary. However, one will have to pay attention to docking arrangements and systems for handling of fuel from transportation containers. Depending on the type of fuel involved, containers may have to be customized in some cases. Leasing of these containers and subcontracting of transportation services are also available options for consideration. It appears that transportation requirements can be met after either 20 or 100 years as long as galvanic action, which would take place only in the case of an incomplete cask drying, does not significantly deteriorate the condition of the basket.

With regard to transportation, considering public protection and safety, readily available measures must be in place to take into account any off-site transportation emergencies that may arise. These could include off-site emergency organizations and staff, such as police, fire, environment and public health emergency personnel [8].

In either case, it would be necessary to identify the type of emergencies that can occur, methods to identify and mitigate their consequences, and appropriately trained personnel and organizational systems to deal with such emergencies.

4.3. National policy and future strategy

The lifetime of the AFR storage facility should be determined based on the necessary interim storage period prior to any future treatment, be it reprocessing or direct disposal. In cases where such period is indefinite or very long, one may be constrained by the achievable design life of the facility, in which case the spent fuel may have to be transferred from one facility to another during the storage period. Transferring of stored spent fuel from one facility to another may take several years, even decades, depending on the amount of fuel and loading and handling constraints at the facility. Such limitations would have to be given consideration in developing AFR storage, particularly in terms of facility durability, licensing conditions with regard to facility design life, and any licensing agreements with respect to extended use of the storage facility beyond the licensed period.

One must take into account the ageing mechanisms of the facility and its equipment. In cases where it is planned to retrieve the fuel from storage containers, the integrity of the fuel during storage conditions will also be an important consideration. It may be necessary to implement storage and spent fuel monitoring plans to provide ongoing information on the structures and the fuel. Although it is usual practice to consider lifetime of a few decades for storage (and perhaps 50 to 100 years), longer periods might require caution because of uncertainties involved. Extended storage periods may also augment the need for a carefully designed monitoring plan and proper provisions to handle possible contingencies. Although behaviour of spent fuel during storage has been studied to some extent, experience with long term integrity of storage structures is generally not available in the nuclear industry. For instance, consideration should be given to having readily usable extra capacity for spent fuel should an emergency occur within the facility that may require the removal of some spent fuel.

4.4. Regulatory and licensing

Some countries, having been engaged in spent fuel management for many years, have set up comprehensive national standards, safety regulations, emergency response and licensing procedures, etc. for activities involving spent fuel. These systems can serve as a model in countries where the relevant national regulations are not yet fully developed. However, a careful analysis has to be made in order to identify the limitations of such practice in meeting particular national expediency for putting regulations in place.

Licensing requirements should be identified in the beginning of any project to ensure timely compliance and to take this factor into account in selecting technologies. Licensing could involve several regulatory authorities, and the extent of licensing, agencies involved and the coordination effort required will have to be clearly known. Licensing of an AFR spent fuel storage facility encompasses many aspects involving:

- The nuclear power plant site where necessary modifications may be required to support the AFR storage (such as changes to reactor pools and transportation access). Changes required are often complex due to the reason that these involve an operating facility and require additional licensing effort that involves the operating nuclear power plant.

- The spent fuel transport system, including interfacing systems at the nuclear plant and the AFR storage site, and along the transportation route, which may need to take into account appropriate risk assessments and the involvement of all affected stakeholders (transportation workers, communities along the transportation route, etc).

- The AFR storage facility itself, including siting, design, construction, commissioning, and operation.

Each of the licensing stages requires preparation of an appropriate Safety Analysis Report to support the application for the relevant stage. The content will reflect the particular stage of licensing, gradually increasing in scope to support an application to operate a constructed facility. Licensing is often a time-consuming activity due to the extensive analysis required for supporting safety design of the facility. Some planning may be required to assess the timelines and ensure that licensing activities are taken up sufficiently in advance and in parallel with other project activities, where feasible, such that any negative impact on project schedule is minimized.

4.5. Other considerations

4.5.1. Safeguards

The objective of the safeguards is the timely detection of diversion of nuclear material for non-declared purposes and deterrence of such diversion by early detection. The IAEA safeguards system is based primarily on the use of materials accountancy as a safeguards measure, with containment and surveillance as major complementary measures. In the context of AFR storage operations, arrangements shall be made to ensure that the facility operator is aware at all times of the location and quantities of nuclear materials in storage and to provide the necessary reports defined within the particular Safeguards Agreement between the Member State and the IAEA.

4.5.2. Physical Protection

Physical protection of AFR storage must comply with the needs of safeguards provisions and provisions for physical security of the storage system with associated spent fuel. Physical protection measures not only include designed features but also various administrative controls in the facility such as on-site security staff and procedures. Physical protection of spent fuel storage facilities has recently become an issue of mounting concern due to the possibility of becoming a target of terrorism.

4.5.3. Environmental assessment

In many Member States, a decision to build AFR storage would trigger an environmental assessment process legislated by the government. Environmental assessment is a focused response to the protection of the human and natural environment. It is a process that may differ among countries in its details if not in intent. Due to the long term nature of the AFR storage system, environmental stewardship requirements over the storage period would be a key consideration. An environmental assessment process would generally include assessments of environmental impacts of the facility over its life cycle from the range of activities involved, primarily construction, operation and decommissioning. The process would be designed to provide opportunities for the public and affected communities to participate in the decision-making processes through consultation, which may include public hearing. It could include elements of other assessments such as feasibility and licensing assessments, but would be in response to different legislative requirement stipulated by the countries. The environmental decisions to proceed with the project are given by the regulatory bodies in charge of the environmental assessment process.

4.5.4. Stakeholder involvement

Decision to construct an AFR spent fuel storage facility cannot be made without the full participation of all relevant stakeholders. Depending on the intent, this could include the need to meet local community concerns, concerns of the general public, or concerns expressed at national or even regional levels (if bi-national or international facilities are considered). Implication of stakeholder involvement must be envisaged at a very early stage as it could deeply influence storage plans, degree of regulatory and political support to storage plans, public and community support, etc. Therefore it is important to identify early who might be the stakeholders involved, and design a process to involve all stakeholders in order to reduce risks related to stakeholder acceptance in various stages of the project life cycle.

It might be a daunting task to obtain public participation where needed if proper attention is not given to public involvement. Lack of public support could delay or even prevent the implementation of any AFR storage solution. There may be specific requirements to involving the public in necessary consultation activities and decision-making. This area is currently subject to many discussions at various local, national and international levels that could result in evolving future requirements

4.5.5. Quality assurance

All activities related to an AFR storage facility shall be subject to a quality assurance programme encompassing the entire procurement cycle including the selection process and the various stages such as the detailed design, construction and the operation. The objective of the quality assurance is to ensure with confidence that the storage system will perform satisfactorily during service. To that end, quality assurance will include all planned and systematic actions necessary to assure that all aspects of the project, covering activities, systems, components and materials meet the quality requirement. The quality assurance requirement shall always be commensurate with the safety and licensing requirements. There are relevant IAEA publications elaborating quality assurance needs in nuclear projects that may be taken into account in developing a quality assurance program [9].

4.5.6. Project management

An AFR spent fuel storage project requires proper management. Since the main responsibilities of the utilities are the operation and the maintenance of the nuclear power plants to generate electricity, project management capability may not already exist within such organisations to handle such large projects as the AFR storage. This situation may also exist in the case of waste management organizations (WMOs) charged with the task of providing AFR storage. These organizations may acquire such expertise through the hiring of project management resources. The project management organization, so appointed, will be generally responsible to carry out the initial feasibility studies, technology selection, and selection of suppliers/contractors to design, procure, construct, commission, and train staff for the AFR storage. This organization may include on its staff experienced consultants and architect/engineers (A/Es) integrated in its structure so as to provide appropriate support to the overall project contract strategy [10].

Successful AFR storage projects are also successful partnerships between the project and its stakeholders. The stakeholders include project staff, NPP management and plant engineers, WMO senior management, regulatory authorities for licensing and environmental assessment, various government agencies that may have a stake in the project, site communities and the general public, vendors and suppliers and their subcontractors. Nurturing these partnerships would be an important consideration for a project and a criterion for success.

4.5.7. Risk management

Exposure to risk is a natural consequence in AFR storage projects as in the case of any other industrial projects. The project sponsor and project management organizations are accountable to their stakeholders, and have a range of obligations to be met, such as in terms of cost, quality, legislative compliance, safety and environmental protection, and political support. A mature organization would have a risk management strategy fully integrated into its procurement cycle.

Risks could arise both due to external factors (legal challenges, environmental causes) and internal factors (cost, schedule, safety, quality). A good strategy will have a continuous process to identify, assess and respond to risks during the entire project. Risks are generally the greatest in the early stages of the project and should diminish as the project evolves towards completion. Early risk assessments provide opportunities to take mid-course corrections and allow the project staff to change risk-causing uncertainties into opportunities. Risk assessment carried out prior to award of contract permits the project manager to recognize business and financial risks, and put in place measures to avoid, reduce or absorb such risks.

In an AFR storage project, unmitigated risks could lead, in the extreme, to unacceptable situations such as: inability to store the fuel making continued operation of the NPP difficult; serious challenges from the public or other stakeholder groups; inability to finance the project due to cost over-runs; unmanageable safety and environmental issues, or similar contingencies. Without consideration to avoidance strategies, alternatives and fallback positions, risks could have disastrous consequences on the project as well as the NPP. At the simplest level, risk assessment is carried out by the project staff through structured discussions about potential pitfalls and unusual occurrences that can be expected in the course of the project. Steps are taken to screen risk situations, quantify identified risks, mitigate, and apply the lessons learned to future risk situations. At a detailed level, a variety of comprehensive risk assessment and management methods including statistical and computerized techniques can be used. These techniques help the project team to reduce the drudgery and time required in handling large amount of risk scenarios and data, and to carry out “what if” risk modelling assessments.

4.5.8. International obligations

There are a number of existing international obligations that need to be considered which may vary from one Member State to another. These become particularly important in the case of bi-lateral or multi-lateral arrangements for AFR storage. Examples of various obligations are:

- IAEA Safeguards,
- EU requirements (in Europe),
- Bilateral agreements,
- Joint conventions
- Trans-boundary issues
- International treaties, etc.

International agreements are legally binding and can directly influence the project direction and technological choices to be made. With increasing private sector involvement and with globalisation of the nuclear industry, AFR storage industry is showing signs of increased international activity and may require greater attention in the years to come.

5. Selection criteria

From the various general requirements that need to be identified in the beginning of a project, the project organization can then identify the technology options that are available and generally develop some preliminary ideas as to how they can be incorporated into an AFR storage facility design or adapted to conform to the requirements. However, to make a selection, it becomes essential to carry out a feasibility assessment the aim of which is to narrow down the choices available to a few most suited choices (about 1-3 options). Final selection of the technology option is often left to a later stage, i.e. to the time of selecting a supplier. Taking more than one technology to the bidding stage facilitates larger participation from the bidding community, often beneficial to the project in terms of supplier selection.

The feasibility assessment usually tends to become a comprehensive study of the options available and an evaluation and screening of these options using a consistent set of criteria. In cases where the initial choices are many and somewhat poorly defined to start with, the feasibility assessment is carried out in stages such that the large number of choices is initially screened with a preliminary set of selection criteria at a conceptual level to yield a smaller set of alternatives. These are then further narrowed down with criteria specifically fine-tuned towards the final selection. To acquire an AFR storage system that is most appropriate for a given situation and that will interface best with the customer needs and requirement, the selection criteria for the AFR storage must be carefully established. The range of criteria must be broad enough to be of use not only in the comparison of technologies, but also in the selection of the AFR storage facility which includes besides technology, site, transportation and various infrastructure which are sometimes critical to successful implementation of the spent fuel storage project in question.

The selection criteria need to be effective and pertinent to the selection process. These indeed are detailed requirements with which one should be able to discriminate various options and be able to rank them after evaluating their merits and demerits. The key selection criteria relate to the acquisition of the site and transportation routes, overall system performance, licensing, facility construction, operation and maintenance, environmental impacts, decommissioning and of course associated cost which is determinant criterion in nornmal cases. All factors affecting these items must be identified and described to such a level where the different storage system alternatives could be compared on a common reference.

5.1. Site

Irrespective of the fact whether the site for the AFR storage has been selected or not, sufficient site investigation and assessment work requires to be carried out to quantify the characteristics of the site. The site information can then be used to compare the technology options available in terms of their siting advantages. If the site has already been selected then the site-related factors affecting the storage technology selection must be identified. If the site has not been selected, a selection process could be developed incorporating potential technology constraints and the site selection carried out with such a 'technology-based' selection process.

Traditional approaches of siting which rely heavily on scientific and technical criteria are now yielding to co-operative approaches where sites are sought in cooperation with volunteering host communities. In cooperative approaches, regional information meetings are first held to inform communities about the proposed facility. Those communities that show an interest in hosting the facility enter into consultation with the project organization. Screening of potential sites is then carried out and a site selected with community involvement throughout the screening process. Traditional approaches may suffice in instances where suitable sites already exist and community involvement is generally not required. In most instances where new sites have to be acquired, however, the cooperative approaches are preferred or mandatory. These approaches are suitable not only for the AFR storage site but also for making decisions on acceptable transportation routes for the spent fuel. Transportation is often a formidable task in terms of public acceptance, and requires considerable public and community interaction.

The accessibility of the site, availability of routes and selection of modes of transportation, availability of the infrastructure, and various other site characteristics of importance to AFR storage design shall be considered in the screening process. In the case of an existing site (such as a nuclear power plant site), locating a facility shall take into account existing site layout and any interference with the existing facilities and ongoing operation. In the case of a new site, additional factors such as community and public preferences have to be taken into account. Suitable modes of transportation (e.g. by road, rail or water) would have to chosen. Appropriate transportation corridor may have to be developed in either case if there is none available.

The area selected for the storage facility shall be sufficiently large to accommodate the anticipated amount of fuel storage and all ancillary equipment and facilities. Provision shall be made for any planned expansion. Some exclusion area, as determined by nuclear regulations in the Member States, may be required to meet safety requirements. For the study of public safety and environmental impact of the facility, knowledge of the basic site design factors is required. It should include site characteristics related to the geology, soil bearing capability, topography, hydrology, hydrogeology, meteorology, demography and civil design, including potential external hazards particular to the site. External hazards to be considered should include both natural phenomena (e.g. earthquake, floods, winds, snow, ice and lightning) and man-made hazards (e.g. aircraft crash and explosions).

Site conditions, processes and events described above will impose certain constraints and design requirements on the AFR storage system. The objective is to establish the normal or average situation and to identify the credible extreme events to be considered.

5.2. Safety/Licensing

The main nuclear safety issues of an AFR storage facility are: protection of fuel integrity; heat removal; radiological shielding; containment; environmental protection; prevention of criticality; and safe management of radioactive waste. The underlying policy is to reduce radiation exposures as low as reasonably achievable (ALARA) and take measures to avoid, reduce or eliminate any adverse effects on the environment and the public as well as to the workers of the facility from the storage activities during the storage timeframe. The layout and arrangement of the storage facility shall be such that inadvertent criticality is prevented by the use of geometrically safe configuration, sustained even under accident conditions.

Requirement for licensing differ from country to country and provide specific criteria for the selection of the storage system. Key regulations in most countries deal with allowable doses to the public and workers generally based on internationally accepted information of radiation effects such as the ICRP recommendations. A storage concept that has already been licensed, for example in the country of origin, would make the licensing procedure easier, since compliance against regulatory criteria has already been tested in the original country. Existence of an operating prototype or demonstration facility or facilities could also be desirable, since prototypes provide an opportunity to observe actual effectiveness of the design in meeting safety objectives.

5.3. Monitoring and inspection

The design of storage and handling facilities and equipment shall provide for adequate access in order to facilitate inspection, testing and maintenance of the equipment and to facilitate radiation monitoring and contamination control. Integrity of the storage and handling systems should be monitored and inspected. Such monitoring should include testing of seals, monitoring of dose rates, temperatures and contamination in the facility. Storage system should be designed such that the fuel integrity is not compromised during handling and storage.

Where the fuel integrity is not monitored within the separate storage units during operation, the system should include appropriate features for safe unloading of the fuel. Ageing of fuel cladding and structural material should be considered for monitoring over the design life of the facility.

5.4. Environmental impacts

Environmental impacts could influence the selection process. Effects during the construction, operation and decommissioning phases should be evaluated and checked against the established criteria such as radioactive releases, dose rates, heat emission etc. and various environmental regulations. Environmental regulations generally deal with a range of issues, such as biophysical impacts on biota, socio-economic impacts, and environmental quality (such as of air and water). Environmental assessment requirements differ from country to country and it would be necessary to review these requirements such that necessary environmental criteria for the selection process can be developed.

5.5. Criteria associated with facility

- Design, engineering, construction / fabrication;
- Operation & Maintenance [11];
- Decommissioning [12].

5.6. Cost and financing considerations

A proper cost analysis of the various options requires identification of the detailed costs. These costs may be grouped into different categories and described in such a way that they can be applied to all the options. The main cost categories are: capital investment cost, operation and maintenance (O&M) cost and decommissioning cost:

- Development Costs (not required in case of purchase);
- Capital Cost (purchase of vendor supplied storage systems, customer supplied premise and support systems, infrastructural construction, feasibility and engineering studies, licensing fees, etc.);
- O&M Cost (labour costs, material supplies, tax/insurances, administrative services, overhead, etc.);
- Decommissioning.

The assessment of costs required for an option and comparison of costs between different options are usually done by analysis based on life cycle costs (LCC) taking time value of monet into account by discounted cash flows, rather than by overnight costs. A convenient metric method for cost comparison is net present value (NPV), defined as the sum of a time series of discounted costs each of which occurs at the time of actual spending to the expenditure profile of each option. For the comparative purpose, it is essential to calculate the NPV on a basis of consistent time frame for the options to be compared. The LCC based on NPV is particularly important in cases of long time span involved in the project and at high discount rate [13]. The inflation rate can also be accounted for in the LCC based on discounted cash flow.

Another convenient method of cost comparison is to use unit costs, rather than total costs, as a measure of characteristic parameter. The levelized unit cost (LUC) is defined as the total cost divided by the total quantity (of spent fuel stored), both side of which are discounted over the same period of time.

From the NPV or LUC calculation, it is possible to analyze the sensitivity and uncertainties with various assumptions on the parameters including discount rate which is variable depending on the funding source.

6. Methodologies for selection

While the purpose of the selection method is to find the most suitable solution, there is no single rationale to select the best option for a storage concept. There are many factors and issues that significantly influence selection of a storage concept that are not directly related to its technical merit or its cost, and often, stakeholder groups could unduly sway the decision based on considerations other than these factors.

During the project-planning phase, the project organization will identify general technical requirements, technological options available, scope and schedules, and various management processes (such as stakeholder involvement, public consultation, regulatory processes, quality and risk management) required for the successful completion of the project. The role of the project management organization is to apply the appropriate knowledge, skills, tools and techniques to the AFR storage project in order to meet or exceed customer and stakeholder needs and expectations. By the end of the project planning phase, the project organization would have integrated schedule, budget and a comprehensive work description for the proposed AFR storage, all of which would continue to be updated and elaborated as the project moves forward, increasing the overall level of understanding of the project.

Following the completion of the technology selection phase, the project is taken to the tendering phase. A Functional Specification document, based on the outcome of the technological selection process, provides a contractual basis for the potential bidders.

The last stage in the AFR storage selection is a process that involves pre-qualification of suppliers, bid invitation and receipt, and evaluation of bids leading to the selection of a supplier and a technology. The selection concludes with the award of a contract.

7. Summary

The measures taken by each countries for interim storage of spent fuel (and for spent fuel management options in that context) are driven by a variety of complex factors, both technical and non-technical ones. While the decisions made are specific to national policies and regulatory constraints, some tendencies are notable in the approaches to implementations due to some common factors and issues visible. On the one hand, most countries with nuclear power production are involved in, or preparing for, interim storage of spent fuel of AFR type which can be stretched in some cases beyond a century, as a fallback for uncertainties in the future of fuel cycle backend. It is therefore predictable that a large inventory of spent fuel is likely to continue to accumulate calling for new builds of AFR facilities around the world in the foreseeable future. On the other hand, the technology for spent fuel storage has well matured in general and in particular the dry storage options developed in the past couple of decades are now available to offer reliable and competitive products and services required for long term storage.

In view of the current market circumstance of competitive supply and offers for more advanced products, it is customary for customers to select the attractive system market that would best fit in the AFR storage requirements based on the criteria of preference. Whereas it is obvious that the selection of the best system is critical for implementation of AFR storage of spent fuel, it is often not evident which option to choose due to the complex factors to be considered in the decision. The selection process involves a series of steps toward contractual award, taking into account those requirements and criteria applicable to the AFR storage, with a rational approach to the project goal.

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The Idaho spent fuel project
An update — January 2003

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Abstract. The Department of Energy awarded a privatized contract to Foster Wheeler Environmental Corporation in May 2000 for the design, licensing, construction and operation of a spent nuclear fuel repackaging and storage facility. The Foster Wheeler Environmental Team consists of Foster Wheeler Environmental Corp. (the primary contractor), Alstec, RWE-Nukem, RIO Technical Services, Winston and Strawn, and Utility Engineering. The Idaho Spent Fuel (ISF) facility is an integral part of the DOE-EM approach to accelerating SNF disposition at the Idaho National Engineering and Environmental Laboratory (INEEL). Construction of this facility is also important in helping DOE to meet the provisions of the Idaho Settlement Agreement. The ISF Facility is a substantial facility with heavy shielding walls in the repackaging and storage bays and state-of-the-art features required to meet the provisions of 10 CFR 72 requirements. The facility is designed for a 40-year life.

1. Introduction

During the last 40 years, the United States Department of Energy (DOE) and its predecessor agencies have generated, transported, received, stored, and reprocessed spent nuclear fuel at several facilities in the DOE's nationwide complex. This spent fuel was generated from various sources, including production reactors; research and test reactors; special-case commercial power reactors; and foreign research reactors. Some of the DOE's spent fuel is in storage at the Idaho National Engineering and Environmental Laboratory (INEEL).

The DOE ended reprocessing of spent nuclear fuel in the USA in 1992. Partly due to this decision, 235 metric tons of heavy metal (tHM) spent fuel is still stored at the INEEL in pools, dry wells and above ground storage pending disposal in a geologic repository. The current storage facilities are located over the Snake River aquifer, a major water source for the region. In addition, the INEEL is currently planning to receive an additional 70 tHM of spent fuel from sources including foreign and domestic research reactor programmes.

A Settlement Agreement signed on 17 October 1995 by the DOE, the U.S. Navy and the State of Idaho requires that all INEEL spent fuels be transferred to dry storage by 31 December 2023; and removed from Idaho by 1 January 2035. The agreement includes fuel from Peach Bottom and Shippingport reactors, and TRIGA fuel from various sources. Current spent fuel storage and handling facility capability is not considered adequate to meet this mission need for the next twenty to thirty years. The current contract scope includes the repackaging and storage of 20 tHM of spent fuel. With future facility modifications, enhancements, and appropriate license amendments, the DOE is planning on handling the majority of the ultimate INEEL spent fuel inventory through the core capability provided by the ISF Project including load-out for shipment to the geologic repository.

2. The Idaho spent fuel project

The contract for an additional interim handling and dry storage facility at the INEEL was awarded to TTFWI (Tetra Tech Foster Wheeler Inc.) on 19 May 2000. The contract is for the design, licensing, construction and operation of an Independent Spent Fuel Storage Installation (ISFSI) that will repackage and store Peach Bottom, TRIGA and Shippingport fuels. The project is known as the Idaho Spent Fuel (ISF) Project and the interior of the planned storage facility is shown in Figure 1.

The ISF facility is being licensed for interim storage by Nuclear Regulatory Commission (NRC) to 10CFR72 requirements. In addition the fuel storage canisters and baskets are being designed for transportation to 10CFR71 requirements and also to meet repository requirements. The facility design is based around the TTFWI/ALSTEC Modular Vault Dry Storage technology.

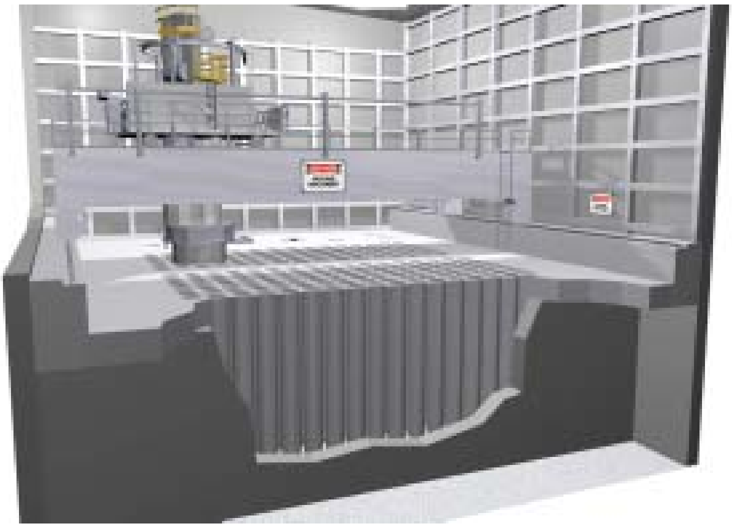

FIG. 1. The Idaho spent fuel project storage vault.

The ISF facility consists of three main functional areas: the Cask Receipt Area, the Transfer Area and the Fuel Storage Area. The Transfer Area consists of two main sub-areas: the Fuel Packaging Area and the Canister Closure Area. Fuel is delivered to the facility at the Cask Receipt Area and is repackaged into canisters within the Fuel Packaging Area. The canisters are then welded closed and inerted within the Canister Closure Area, and then placed into storage in the Storage Area.

The ISF facility uses the DOE standardized spent fuel storage canisters that are compatible with the requirements of the proposed national high level waste repository as currently defined in repository waste acceptance criteria. The preliminary canister specifications were designed by DOE and its contractors to accommodate a wide range of fuels currently being stored by DOE. The ISF will use two different diameter storage canisters, 18 inch and 24 inch. The 24 inch canisters are required to accommodate the Shippingport fuel assemblies, while the 18 inch canisters will be used for the remaining fuel types. The facility is designed to easily retrieve standard canisters from the storage area and deliver them to the load-out point for shipment to the geologic repository.

As part of the contract, TTFWI also provided a conceptual transportation system design compatible with the ISF facility and standard canisters. This provides a conceptual design of an NRC licensable shipping package for eventual offsite rail or road shipment to the geologic repository. Neither the detailed design nor manufacture of this shipping package nor retrieval of canisters and/or shipment to the geologic repository is in the scope of this project.

3. Current status of the ISF project

The ISF project contract has now completed the licensing design phase. The license application was submitted to the NRC on 19 November 2001, and after a pre-acceptance review of the license application the NRC formally accepted the documentation for licensing review in March 2002. The NRC issued their first Request for Additional Information (RAI) to TTFWI on 25 October 2002 and these were returned to NRC for continuing review on 24 January 2003. A second round of RAIs will be issued in May 2003 if required. Detail design and preparation of fabrication information is currently underway in parallel with the NRC review of the license application.

Key dates for the project are:

• Contract Award:	19 May 2000
• License Application submitted to NRC:	19 November 2001
• Part 72 license approval:	Nov. 2001 through June 2004
• NRC issue first round technical RAIs:	25 October 2002
• First round technical RAI responses to NRC:	24 January 2003
• NRC issue first round of Environmental RAIs	29 January 2003
• First round of Environmental RAI responses to NRC	14 March 2003
• NRC issue second round of technical RAIs (If required):	30 May 2003
• Second round of technical RAI responses to NRC:	29 August 2003
• NRC issues SER and Part 72 License	31 March 2004
• Start operations:	June 2005

4. The Modular Vault Dry Store System

The ISF facility utilizes the TTFWI/ALSTEC Modular Vault Dry Storage (MVDS) system to provide interim storage for approximately 220 canisters. The current status of TTFWI/ALSTEC's MVDS facilities are shown in the Table I. The MVDS has had an NRC Topical Report approved status since 1988. The Topical Report covers the interim storage of Light Water Reactor fuels, including both PWR and BWR, at any reactor sites in the USA.

The MVDS provides a simple passive design for dry fuel storage. The fuel assemblies are cooled by natural circulation, a self-regulating system in which higher spent fuel temperatures prompt increased airflow and thus heat removal. Criticality control is provided by the geometrical array of storage tubes within the vault array.

The MVDS was employed for the first time in the U.S. at the Fort St. Vrain plant for the storage of High Temperature Gas Reactor fuel elements (Fig. 2). The Fort St Vrain ISFSI was licensed under 10CFR72 and the unit went into operation in 1991.

In 1998, ownership of the Fort St. Vrain MVDS was transferred from Public Service Company of Colorado to DOE. DOE manages Fort St. Vrain ISFSI under NRC regulatory authority.

The Paks MVDS went into operation in December 1997 (Fig. 3). This facility is in Hungary and is designed for the storage of VVER 440 fuel elements. The Paks MVDS was built with an initial storage capacity for 1 350 fuel assemblies. In 1999 and 2002, two additional construction phases increased the Paks MVDS capacity by a further 3 600 assemblies. This brings the existing capacity of the Paks MVDS up to 4 950 assemblies and there are future plans to increase the storage capacity up to 15 000 assemblies.

FIG. 2. Fort St. Vrain MVDS.

FIG. 3. Paks MVDS – shown in 2000 after first extension vaults were added.

5. Description of the ISF facility

The storage design of the ISF facility is based on a vault storage system. The vault design provides radiation shielding as well as a passive cooling system where spent nuclear fuel assemblies are cooled with natural circulation. This is a self-regulating system in which higher spent fuel temperatures prompt faster air flow and thus heat is removed within established design parameters. Criticality control is provided by the geometry of the storage canisters and vault array. The ISF facility provides for year round operations. Due to the weather extremes in Idaho, all operations occur inside the facility. The ISF facility design provides for specific operations to occur within discrete areas of the facility. These areas include the Receipt Area, Fuel Packaging Area, and Storage Area.

Table I. Current status of TTFWI/ALSTEC MVDS facility

Facility	Type of Reactor/Fuel	Dry Storage Method	Licensing Authority and Date of License Approval	Date of Operation
MVDS Topical SAR	PWR and BWR, anywhere in USA	Concrete vault – MVDS	USA NRC 1988	n/a
Wylfa dry fuel cells 1 to 3 Anglesey, UK	Gas Cooled Reactor Magnox fuel	Concrete vault, tube storage	UK NII 1969	1969
Wylfa dry fuel cells 4 to 5 Anglesey, UK	Gas Cooled Reactor Magnox fuel	Concrete vault, tube storage	UK NII 1979 and 1980	Cell 4: 1979 Cell 5: 1980
Fort St Vrain MVDS Colorado, USA	High temperature gas reactor HTGR fuel blocks	Concrete vault – MVDS	USA NRC 1991	1991
Paks MVDS Paks, Hungary	VVER 440 VVER 440 fuel	Concrete vault – MVDS	Hungary OAH Feb 1997	December 1997
Idaho Spent Fuel Facility Idaho, USA	DOE owned fuels: Peach Bottom Core 1 Peach Bottom Core 2 TRIGA aluminum clad TRIGA stainless clad Shippingport modules	Concrete vault – MVDS	USA NRC Planned 2003	Planned 2005

The spent fuel storage canisters are shown in Fig. 4. The design is compatible with the requirements of the proposed national high level waste repository. The ISF will use two different diameter storage canisters, 18 inch and 24 inch. The 24 inch canisters are required to accommodate the Shippingport modules, while the 18 inch canisters will be used for the remaining fuel types. After loading with spent nuclear fuel, the ISF Canisters are placed into sealed storage tubes within the vault. The vault storage tubes provide a secondary confinement boundary around the stored fuel, and also ensure future recoverability of the canisters for off-site transportation. The canisters and storage tubes will be designed and fabricated to ASME Section III, Division 1; and N-stamped accordingly.

6. Fuel type summary

The fuel to be stored in the ISF facility consist of the following types:

- Peach Bottom Core 1 and Core 2 Fuel Assemblies;
- TRIGA Fuel elements;
- Shippingport Modules.

Each of the three fuel types has different physical, chemical and radiological characteristics that have been addressed in the facility design and license application (see Table II).

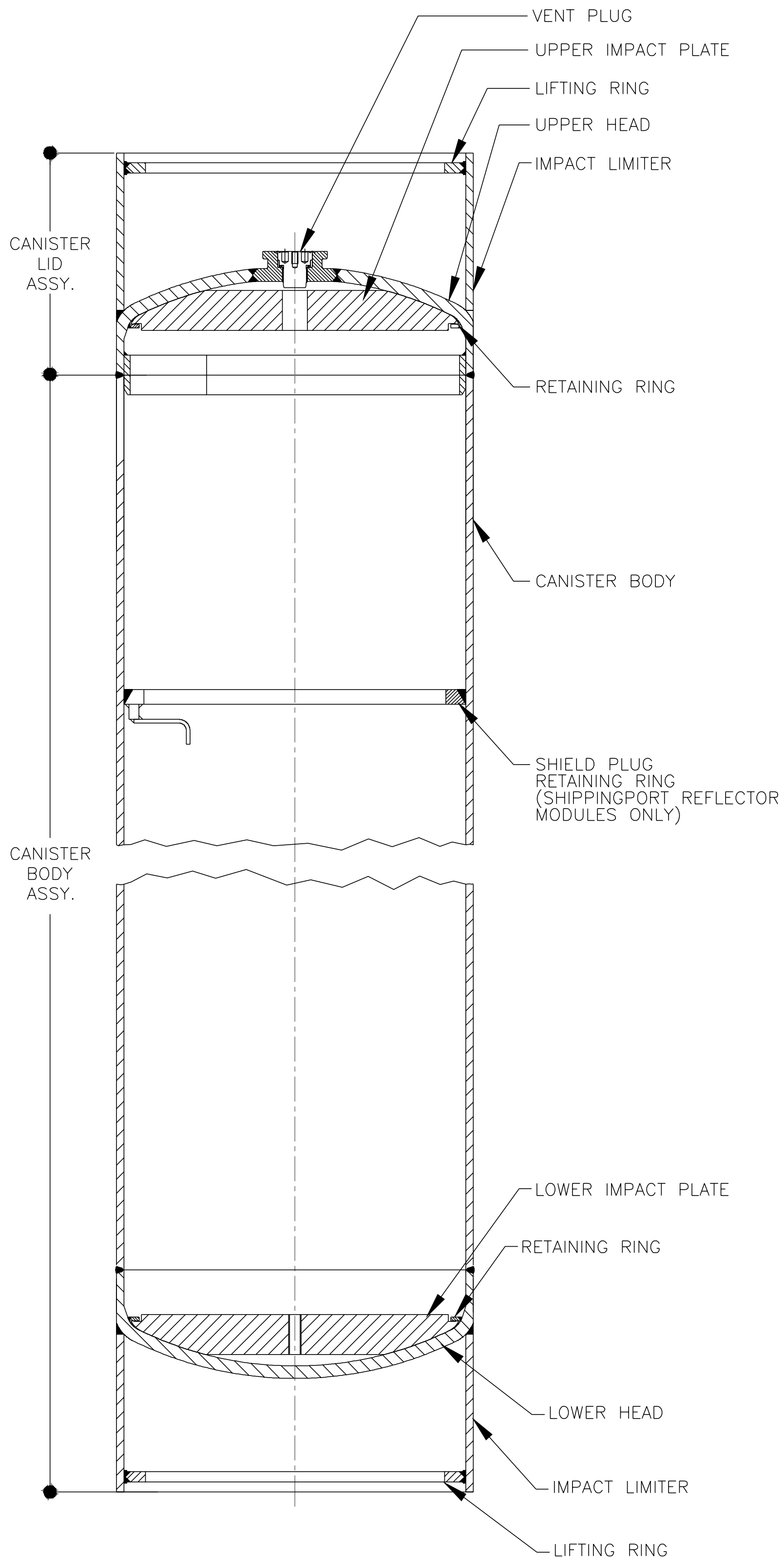

FIG. 4. ISF canister.

7. Fuel handling process flow

The ISF Facility is laid out to efficiently transfer incoming fuel to the Fuel Packaging Area, Canister Closure Area and into the Storage Area. This is achieved by a transfer tunnel that interconnects these areas. In the future the ISF canisters will be transported offsite using a 10CFR71 licensed transport handled within the existing structure. The process flow in Fig. 5 depicts the typical evolution through the facility.

Table II. Fuel Cladding Material

Fuel Type	Clad Material
Peach Bottom Core 1 and 2	Graphite
TRIGA	Aluminum or Stainless Steel
Shippingport Module	Zircalloy

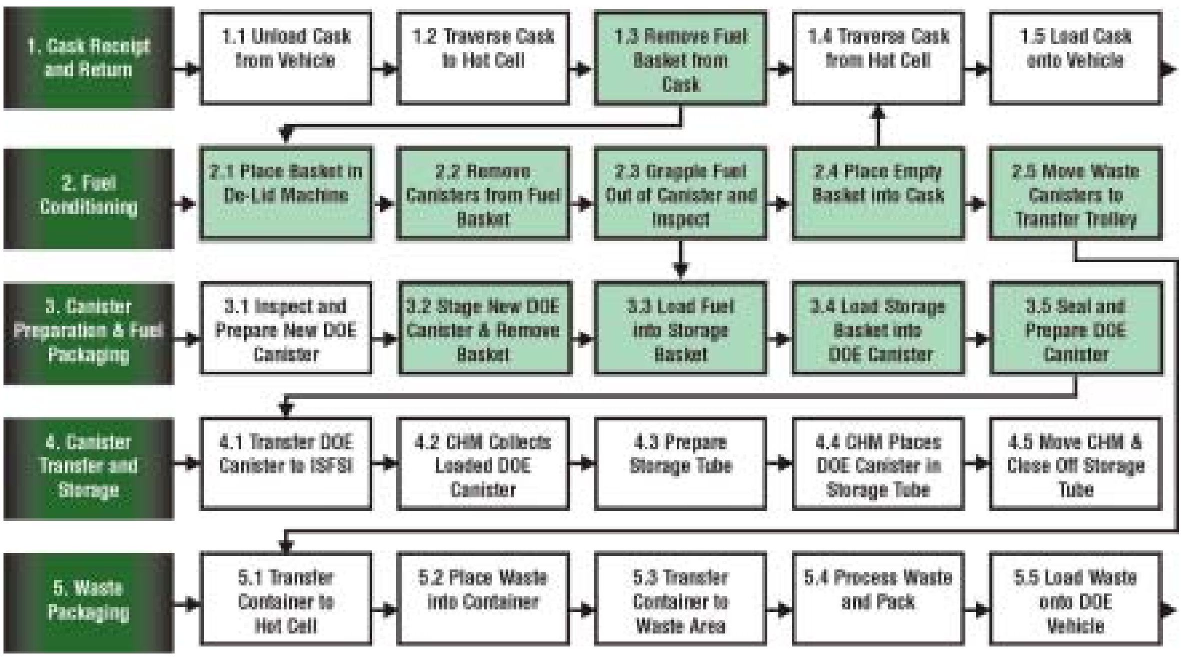

FIG. 5. Fuel handling process flow.

8. Cask receipt area

The Cask Receipt Area houses the equipment necessary to receive shipments of spent fuel from DOE. The major equipment in this facility consists of an overhead hoist and a cask transfer trolley. The fuel arrives at the ISF facility in a transfer cask, which was formerly designed for transportation of Peach Bottom fuel. The transfer of the fuel to the ISF facility does not traverse public roadways since the transfer is from an adjacent DOE facility.

The Cask Receipt Area hoist is a single failure proof hoist in order to minimize the probability of drop accidents associated with cask handling. The transfer cask is lifted from the transport vehicle and placed in a rail-mounted trolley, which will restrain the cask from tipping even in the unlikely event of an earthquake. The transfer cask provides radiation shielding for the fuel to reduce personnel exposure. The cask trolley moves the cask from the Receipt Area down a tunnel to the Fuel Packaging Area.

9. Transfer area

The Fuel Packaging Area within the Transfer Area is designed to allow remote handling and unloading of the transfer casks containing the spent fuel. The fuel will be remotely handled during inspection and repackaging into the storage canisters. The process involves removing the cask lid and removing the inner canister containing the spent fuel. This canister is placed in a port located in the floor of the fuel packaging area. The canister lid is removed to allow access to the spent fuel. The spent fuel assemblies are removed, inspected, and placed in a basket, which is designed to hold the fuel within a fixed configuration when placed inside an ISF canister.

The loaded storage canister is then transferred to the Canister Closure Area adjacent to the Fuel Packaging Area where a lid is welded to the canister. Once the canister lid is welded, an access port allows the interior volume of the canister to be evacuated. This vacuum drying process removes any residual moisture and air from the canister interior. The canister is then back-filled with helium to provide an inert atmosphere, which minimizes corrosion and improves heat transfer. Once the canister closure and inerting operations are complete, the canister is ready for storage.

10. Storage area

The canister is transferred to the Storage Area in the canister trolley. This trolley is moved to the Storage Area and located underneath the Canister Handling Machine (CHM). The CHM provides shielding and remote handling of the canister to minimize personnel radiation exposure. The CHM is also designed to preclude credible drop accidents, as it is designed as a single failure proof crane in accordance with NRC guidance.

The ISF facility CHM is a close copy of the MCO Handling Machine (MHM) that was supplied by Foster Wheeler/ALSTEC for the Hanford Canister Storage Building project. The Hanford MHM has a bridge span of 126 feet 6 inches, while the ISF Project CHM has a bridge span of 73 feet (Fig. 6).

The storage canister is placed within a mechanically sealed storage tube whose primary purpose is to provide a redundant confinement barrier for the stored fuel. The vault system provides for a self-regulating passive cooling of the fuel canisters as well as shielding to limit personnel exposure, and protection of the fuel canister confinement boundaries from all credible accident scenarios.

11. Off-site transport of casks

To prepare for the eventual transport of the fuel canisters to a national repository or other storage location outside the state of Idaho, TTFWI has prepared a conceptual design for a transport system. This conceptual design is compatible with the storage canisters and integrated with the ISF facility operations and equipment. The conceptual transport cask design will provide for off-site shipment of the packaged spent fuel in accordance with the requirements of 10CFR71. To accommodate the transport system, the ISF facility has a staging area for loading off-site transportation casks onto either a truck or rail car. It is contractually designed to handle a cask envelope 128 inches in diameter, 308 inches long, and weighing up to 300 000 pounds. These parameters will accommodate all known shipping packages that are anticipated to be used to ship INEEL spent fuel to the geologic repository.

FIG. 6. Hanford MHM

12. Summary

TTFWI is contracted to design, license, construct and operate an Independent Spent Fuel Storage Installation adjacent to INTEC within the INEEL, approximately 50 miles west of Idaho Falls, Idaho. The ISF facility will receive, repackage, and store spent nuclear fuel provided by DOE. The ISF facility will be licensed and operated under NRC jurisdiction in accordance with 10 CFR 72.

TTFWI will own and operate the ISF until the end of the contract when, in the absence of a contract extension, the facility and NRC license would be transferred to the DOE or its designated successor contractor. The ISF facility is being designed to provide for the interim storage of the spent fuel for a minimum of 40 years and accordingly will be NRC licensed for 20 years (with a 20 year license extension option).