IAEA-TECDOC-1286

Factors determining the long term back end nuclear fuel cycle strategy and future nuclear systems

Proceedings of a Technical Committee meeting held in Vienna, 8–10 November 1999

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

May 2002

The originating Section of this publication in the IAEA was:

Nuclear Fuel Cycle and Materials Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

FACTORS DETERMINING THE LONG TERM BACK END NUCLEAR FUEL CYCLE STRATEGY
AND FUTURE NUCLEAR SYSTEMS
IAEA, VIENNA, 2002
IAEA-TECDOC-1286
ISSN 1011–4289

Printed by the IAEA in Austria
May 2002

FOREWORD

The Technical Committee Meeting (TCM) on Factors Determining the Long Term Back End Nuclear Fuel Cycle Strategy and Future Nuclear Systems was held in Vienna on 8–10 November 1999. The TCM was organized by the International Atomic Energy Agency and attended by 26 participants from 16 Member States.

The purpose of the meeting was to exchange information among experts on the back end fuel cycle strategies adopted by many Member States; to identify key factors determining the longterm back end fuel cycle strategies; and to assess the applicability of these factors to future nuclear systems.

Issues associated with the back end fuel cycle supporting a country's nuclear power programme are technical, economic, institutional and political. To a great extent, these issues are also reflections of the public opinion of the country toward the acceptance of nuclear power. This TCM provided an opportunity to address these issues and their impacts to the back end fuel cycles, as well as to identify and assess factors affecting the back end fuel cycle strategies.

The IAEA wishes to express its gratitude to all the participants who contributed to the success of this meeting. The IAEA officers responsible for the organization of the TCM and for the completion of this publication were J.S. Choi and L. Angers of the Division of Nuclear Fuel Cycle and Waste Technology.

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SUMMARY

1. INTRODUCTION

Issues related to the back end of the nuclear fuel cycle which supports a country's nuclear power programme are of a technical, economic, institutional, and political nature. To a great extent, these issues are also reflections of the public opinion of the country toward the acceptance of nuclear power as a whole. The back end issues begin with the spent fuel discharged from the nuclear power reactors. Technically, 95% of all the radioactivity resulting from nuclear electricity generation ends up in the spent fuel itself. Whether reprocessed or not, the volumes are small by any modern industrial waste standards, and there are demonstrated methods (dry or wet) for interim storage of spent fuel. The challenge remains with the institutional and economic aspects of spent fuel management, i.e., interim storage for how long? and at what costs?

Regardless of back end fuel cycle options: once-through or reprocessing/recycling, there remains a challenge in the final disposal of spent fuel and/or radioactive wastes. Several countries have embarked on their respective national repository programmes to dispose of the spent fuel and radioactive wastes in geologic repositories. Here, the challenge is how to engage the public and stakeholders for their acceptance of the wastes management programme including a geologic repository, on a national basis, as well as on a co-operative regional or international arrangement.

Several countries reprocess their spent fuel. As a result, they produce and hold separated plutonium. Some separated civil plutonium are currently recycled as MOX fuel in light water reactors (LWRs), and some are simply stored pending a decision on disposition. The separated plutonium presents a challenge to the back end fuel cycle because of the proliferation concerns.

There is an increasing interest among several countries in pursuing research and development (R&D) programmes on advanced and innovative fuel cycles to deal with these fuel cycle challenges. The new concepts should propose unique features and special attributes to enhance proliferation-resistance, resource conservation, environmental preservation, safety, and wastes reduction. Nevertheless, the challenge is to ensure that the technological maturity and readiness of these concepts could help resolve the issues associated with the back end.

The back end fuel cycle issues are complex. Resolutions of these issues are of great interest to and will depend on the co-operation and participation from the Member States.

This Technical Committee Meeting (TCM) was convened to exchange information among experts on the back end nuclear fuel cycle strategies currently adopted by the Member States; to identify key factors determining the long term back end fuel cycle strategies; and to assess the applicability of these factors to future nuclear systems.

2. SUMMARY OF THE TECHNICAL COMMITTEE MEETING

The TCM on Factors Determining the Long Term Back End Nuclear Fuel Cycle Strategy and Future Nuclear Systems was held in Vienna, 8–10 November 1999. The discussion was organized in the following topical sessions:

(1) The nuclear fuel cycle

1. The front end nuclear fuel cycle: uranium resources and requirements

The uranium resources and requirements were summarized in a critical review of uranium resources and production capability to 2020. It concluded that uranium mine production will continue to be the primary supply meeting up to 78% of the cumulative requirements through 2020. As supply from excess inventory ends, uranium from other sources, such as uses of MOX fuel in LWRs, uses of low enriched uranium (LEU) blend-down from highly enriched uranium (HEU) would have to increase to meet the cumulative requirements. There may be significant market instability unless all of these supply sources are developed according to the projected schedule. Delay in the development of any of these sources would result in a market shortfall leading to price increases.

2. The back end nuclear fuel cycle

Current options of the back end nuclear fuel cycle are: (1) direct disposal of spent fuel (the so-called 'once-through open cycle'), and (2) reprocessing of spent fuel and recycling of the recovered plutonium (the so-called 'closed cycle'). In practices, the open cycle operates with the strategy that spent fuel will not be reprocessed but stored, first on an interim basis, and later, on a long term basis pending on the availability of a geologic repository. The interim storage of spent fuel is required also by the so-called 'wait-and-see' strategy where final decision on spent fuel management has not been made.

Countries operating with a closed cycle will have separated plutonium. Due to the delay in the commercialization of fast reactors, large scale use of the separated plutonium is not foreseen in the near future. The separated plutonium can be fabricated into mixed oxide (MOX) fuel and recycled into the LWRs. Some countries have adopted such strategy to reduce the inventory of separated plutonium. The spent MOX fuel is currently not reprocessed. Instead, it is definitely stored, similar to those spent UO_2 fuel in the open cycle.

The fact that neither the open nor the closed cycle has demonstrated the completion of its intended purposes has resulted in many emerging fuel cycle issues, especially at the back end. "What are the most important factors determining the back end nuclear fuel cycle strategies?" "How to implement these factors, once identified to future nuclear systems?" and "Can advanced and innovative fuel cycles be developed to help resolve these issues?" These are relevant questions which set the stage for open dialogue and discussion in the following sessions of the TCM.

(2) Spent fuel management

The management of spent nuclear fuel is among the most pressing issues to be addressed in the back end of the nuclear fuel cycle. As of 1998, the total amount of spent fuel accumulated worldwide is about 220 000 t HM. Of this, about 75 000 t HM were reprocessed, and the remaining 145 000 t HM is presently being stored in at-reactor (AR) and away-from-reactor (AFR) storage. Over 70% of this amount is stored in at reactor pools, the remainder in away from reactor wet and dry storage facilities. All countries that generate electricity from nuclear power have to manage their spent nuclear fuel. There is a need to provide adequate storage capacity to ensure that (1) an operating reactor will not lose its full-core-reserve requirement, (2) a reactor at the end of its operating life can be decommissioned and the plant site can be returned to green-field usage, and (3) the spent fuel could be safely stored until a geologic repository is available.

Nearly all countries operating nuclear power plants have increased their original AR storage capacity by re-racking the spent fuel pools with high density racks and to some extent, by implementing burnup credit. Several countries commissioned additional AFR storage facilities, in wet or dry configurations. Regardless of storage technology, it is mature and commercially available. The challenge for spent fuel management is economical and institutional, e.g., spent fuel should remain in proper custody even after the nuclear power plant is shut down and revenue is no longer generated to defray the cost of prolonged storage.

(3) Waste management and repository

Ultimately, spent fuel and high level waste should be disposed of in a geologic repository. Several countries have embarked on their respective national repository programmes, at specific or demonstrating sites, e.g., the USA, Sweden and Germany, etc. The developed repository technology is site specific, and the timing for a repository is country dependent. The challenge for repository development is institutional and political, e.g., how to obtain public and stakeholders' acceptance and support for a repository site, especially the local public and governments.

For countries with small nuclear power programmes and therefore relatively small amounts of spent fuel and radioactive wastes, and for countries with dense population and small geographic areas, consideration of regional and multilateral co-operative arrangements for repository development may be attractive. These countries may have limited potentials to develop their own systems for the back end fuel cycles. Furthermore, it may not be in the interest of the international community that repositories are spread out all over the world which may constitute a proliferation risk. However, the challenge is again institutional and political, e.g., how to ensure that an attempt for a regional co-operative framework would not jeopardize individual country's national repository programme.

(4) Plutonium management

Countries reprocessed spent fuel produce separated plutonium. The separated plutonium was originally intended to be recycled back to the nuclear reactor, preferably the fast reactors. Due to the delay in commercialization of fast reactors, several countries recycle the separated plutonium as MOX fuel in light water reactors (LWRs), while a few with no plutonium utilization programmes simply store their stocks definitely.

The global separated plutonium inventory will continue to grow, due to an imbalance between its production and utilization. On separated civil plutonium alone, the total at the end of 1998 was about 200 tonne. It will be in excess of 250 tonne by the end of the decade. There is also a large inventory of separated weapons-grade (WG) plutonium in nuclear weapons countries. Dismantling of nuclear weapons and the subsequent introduction of the excess WG plutonium into the civilian nuclear fuel cycle for disposition could increase the burden for safe and secure management of separated plutonium.

TABLE 1. FACTORS DETERMINING THE LONG TERM BACK END STRATEGIES

	Technological	**Economic**	**Institutional**
Spent fuel management National	availability of surface storage conditioning extended cycle waste acceptance	storage costs life cycle costs	overall back end strategy ownership transportation
Regional	same as national	contractual agreements	custodian-ship transportation
Repository issues	engineering features geologic medium acceptance criteria	long term storage costs disposal costs	proliferation concern stakeholders interaction transparency
Plutonium management	burn-up parity (MOX vs.UO_2) immobilisation, dirty-MOX spent MOX fuel reprocessing	parity (MOX vs. UO_2) asset or liability?	proliferation risk MOX-use perceived as extension of NE
Advanced fuel cycles	readiness and maturity	parity (w/existing systems) share R&D costs	training for next generation technical personnel international co-operation

TABLE 2. AN EXAMPLE OF BACK END FUEL CYCLE OPTIONS, REQUIREMENTS AND BRIEF EVALUATION

Option	Requirement (Tech. + Inst.)	Partial Investigation	Overall Judgement	Economic + Acceptance
DD	ISFS facility conditioning transport repository	yes not yet, but can be solved not yet, but can be solved not yet, but must be solved	yes	This is very country specific This can be considered as a final tool for decision making for decision revision or change
Rep	ISFS facility reprocessing plant or contract interim storage conditioning + transport repository	yes yes under development not yet, but can be solved not yet, but must be solved	yes	
Rep + Tm	same as Rep, with additional separation and transmutation facilities or contract Transmutation may help lighten the burden for repository	yes	yes	
Wait and see	ISFS R&D toward final selection	yes yes	yes	
Regional	ISFS conditional + transport repository	yes not yet, but can be solved not yet, but must be solved	today: Not subject of change	

DD: direct disposal.
Rep: reprocessing.
Rep+Tm: reprocessing +transmutation.

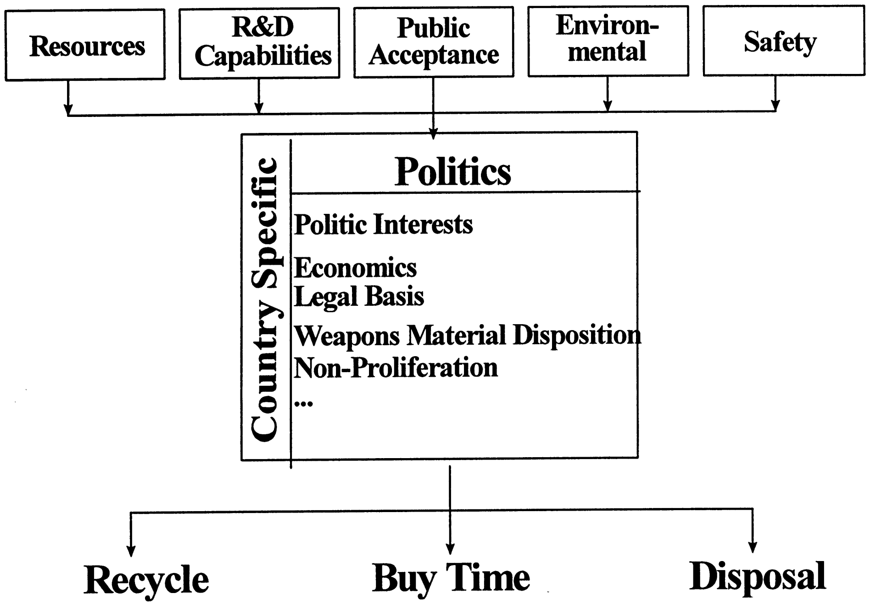

FIG. 1. A diagram depicting the factors determining the back end fuel-cycle strategy.

Options for disposition of separated plutonium are limited. Of these, the direct utilization of plutonium as fuel in nuclear reactors (the irradiation option) is currently pursued by several countries. Other disposition alternatives include immobilization, or "dirty-MOX". However, there is no easy way to introduce the same level of self-protecting radiation as that provided by the irradiation option.

Separated plutonium presents a perception of proliferation risk. The challenge to the back end fuel cycle is to provide a thorough assessment of such risk and to ensure that separated plutonium is properly managed, especially in countries which have decided to phase out their respective nuclear power programmes.

(5) Advanced fuel cycle

Several countries are interested in R&D efforts on advanced and innovative technologies for future nuclear systems. The goal is to ensure that new concepts would consist of unique features and special attributes to enhance proliferation-resistance, promote resource conservation, preserve environmental integrity, improve safety, and reduce waste generation. Many innovative technologies proposed so far were focused on the development of reactors and/or sub-critical systems aided by an accelerator. The challenge is to pursue advanced fuel cycle technology which could help solve the back end fuel cycle issues.

(6) Recommendations

The back end fuel cycle issues are complex, involving many aspects, e.g., technological, economic, institutional, as well as political. This was reflected in many presentations by the participants, especially when they described their countries' back end fuel cycle options and factors affecting their strategies. Discussions on issues and factors were carried out in a Panel

Session. The Scientific Secretary provided a list of factors from the technical, economical, and institutional aspects to help facilitate the discussion. It is shown in Table 1.

An example of back end fuel cycle options, with requirements and brief evaluation for each option is shown in Table 2. Options indicated are: direct disposal, reprocessing, reprocessing and transmutation, wait and see, and regional. Requirements common to all include interim spent fuel storage (ISFS facility), transportation, and repository. Some options require additional processing, and R&D efforts. Important factors identified are economics and public acceptance.

Among the many factors discussed, the central one, according to the meeting Chair, is political. A diagram depicting the factors determining the back end fuel cycle strategy is shown in Figure 1. Here, factors governing the resource requirement, R&D capability, public acceptance, environmental, and safety are fed into a country specific political system which includes the political interests, economics, arms controls and non-proliferation, etc. And ultimately, the decision for selecting a country's back end fuel cycle strategy (direct disposal, reprocessing and recycling, or wait-and-see) will be based on its specific politic system.

Finally, several consensus viewpoints were made by the meeting Rapporteur. These are:

- Back end nuclear fuel cycle issues are unique to each Member State concerned,
- Socio-economic factors are importance,
- Acquisition of future nuclear systems can not be made without considering resolution of back end fuel cycle issues,
- "Wait and see" may not be a viable option, as valuable time could be lost to engage R&D activities and other regional/international collaboration,
- R&D efforts and constructive collaboration with countries having common concerns and interest should reduce the uncertainty in the back end nuclear fuel cycle.

Several future works in the area of back end strategy were also suggested, these are:

- "Mapping" Member States' (MS) back end fuel cycle conditions,
- Strength and weakness of a specific back end nuclear fuel cycle strategy should be readily available, and
- Assessing viable international/regional co-operation.

FOCUS: NUCLEAR FUEL CYCLE

Critical review of uranium resources and production capability to 2020*

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Abstract. Even with a modest forecast of nuclear power growth for the next 25 years, it is expected that the world uranium requirements will increase. This analysis indicates uranium mine production will continue to be the primary supply of requirements through 2020. Secondary supplies, such as low enriched uranium blended from highly enriched uranium, reprocessing of spent fuel would have to make-up the remaining balance, although the contribution of US and Russian strategic stockpiles is not well known at this time.

1. INTRODUCTION

Following a period in which world uranium production fell to only about half of annual reactor requirements, world output increased in both 1995 and 1996. Excess inventories held by Western utilities, which built up in the period to 1985, have been gradually liquidated. See Fig. 1. Although inventories remain substantial, the increase in spot uranium prices during 1995-1996 was a sign that inventories are much closer to desired levels.

While expectations of nuclear power growth during the next 25 years are modest, most forecasters expect some further increase in world uranium requirements. Since utility inventories are nearing desired levels, inventory draw down will diminish as a significant supply source. Uranium from the conversion of military weapons and stockpiles is expected to increase in importance as a supply source. It is anticipated, however, that uranium production will continue as the predominant fuel source. Therefore, the question arises as to the adequacy of both uranium reserves and production capacity to match demand on a timely basis.

2. SCOPE OF STUDY

The objective of this study is to evaluate uranium supply and demand relationships on an annual basis through the year 2020. The following steps were taken in completing the study:

- Establish annual worldwide reactor demand expressed in metric tonnes of uranium
- metal (t U).
- Identify all sources of uranium potentially available to fill reactor demand.
- Determine the most likely contribution that each potential source will make toward
- satisfying demand.
- Assess the adequacy of projected supply and broadly define market prices required to ensure supply availability.
- Define what actions must be taken to meet uranium requirements:

  - new mine-mill development }
    } (to a schedule)
  - additional blending capacity for HEU }

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* This paper is based on IAEA-TECDOC-1033, Critical Reviewof Uranium Resources and Production Capability through 2020.

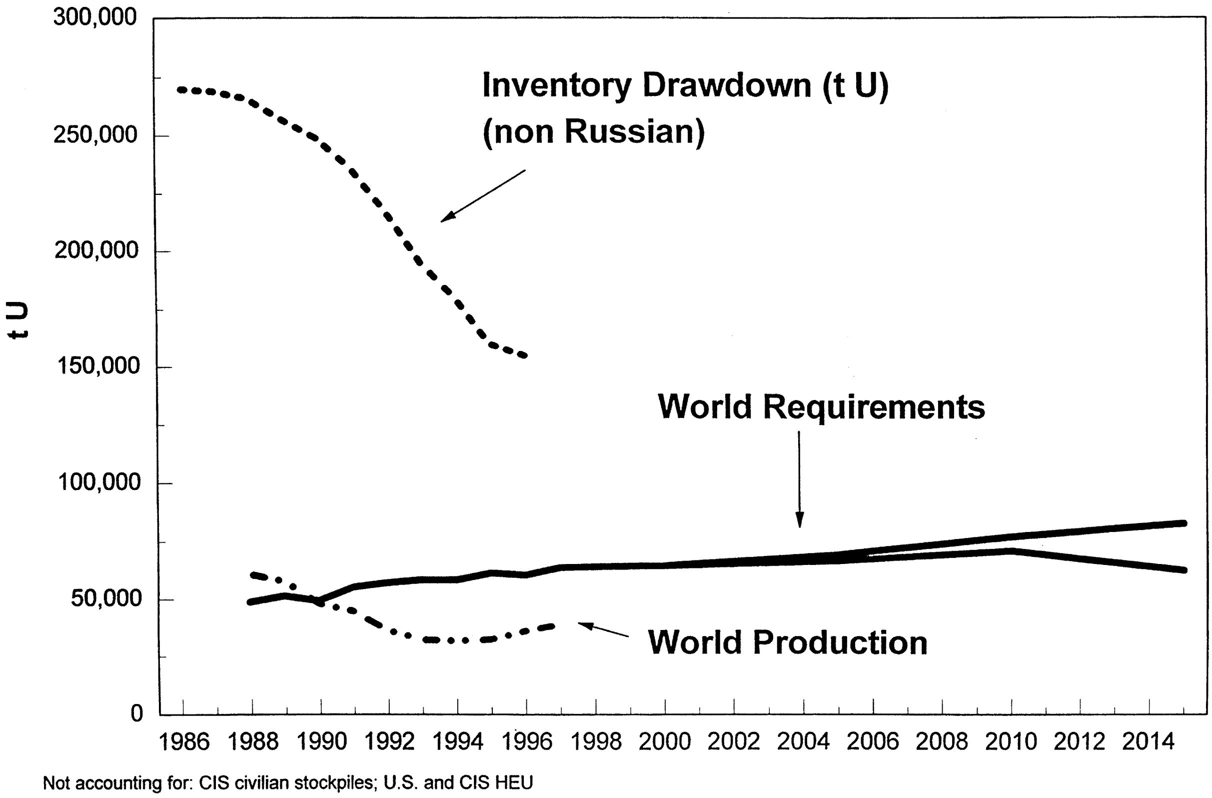

FIG. 1. Production, requirements and the inventory in the world uranium market.

3. METHODOLOGY AND ASSUMPTIONS

To focus attention on primary uranium production, the largest contributor to supply, relatively simple assumptions are made for other potential supply sources. There are ranges of possibilities for future reactor requirements on the demand side of the equation and for further inventory draw down and the entry of ex-military High Enriched Uranium (HEU) on the supply side. However, a single case of uranium requirements is adopted, together with 2 levels of market entry of ex-military HEU. The 2 levels are believed to bound the likely range of supply from the 500 t of Russian HEU plus sales from the US Department of Energy (USDOE) stockpile. The High HEU case is equivalent to the schedule for delivery of Low Enriched Uranium (LEU) from 500 t HEU purchased by the US from Russia; and a Low HEU case, equivalent to the rate at which, under US law, the Russian origin LEU may be sold in the US market, together with 20 000 t natural uranium equivalent from the USDOE stockpile.

3.1. Demand

Worldwide reactor uranium requirements (demand) have been the subject of several recent studies. To establish the requirements projection for this study, forecasts of annual reactor requirements from 9 projections shown in Fig. 2 were reviewed. They include: the High and Low cases from the 1995 Red Book [1\, the Reference (i.e. mid-range) scenario from The Uranium Institute (UI) report Global Nuclear Fuel Market Report-Supply and Demand" [2], the U.S. Department of Energy's Reference (i.e. low) and High cases from the "Nuclear Power Generation and Fuel Cycle Report 1996" [3]. NUKEM's uranium requirements projection through 2005 [4] and TradeTech's projection through 2010 [5] are also included. The projections through 2020 identified as "Int. Sym. High and Low ", are Shani and Deroubaix [6] high and low cases from a study of long term nuclear fuel cycle requirements.

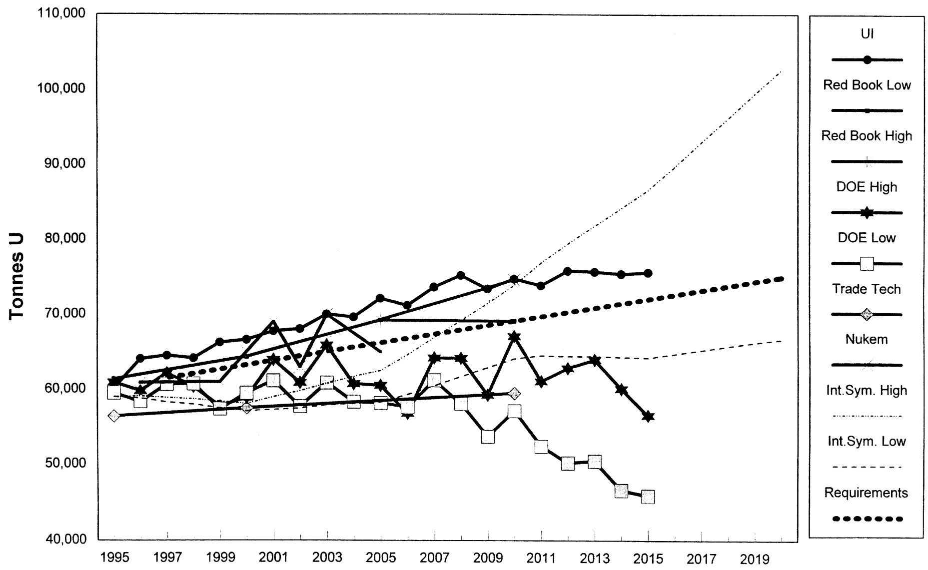

FIG. 2. Projections of world uranium requirements.

There is a reasonable consensus in the trend of the projections of reactor requirements through about 2003. The projections then diverge to 2010. After 2010 the high level of uncertainty regarding the continuing utilization of nuclear energy is reflected by the wide range of projected uranium requirements. The low trend is clearly shown by the decreasing requirements in the USDOE Reference case. In this projection requirements decrease by 30 per cent from a high of 64 800 t U in 2003, to a low of 45 600 t U in 2015. This contrasts with the UI's Reference projection indicating 76 300 t U will be required in 2015. In Shani and Deroubaix's longer term study, uranium requirements in 2020 range from a low of approximately 67 000 t U, to a high of 103 000 t U.

Several recent forecasts of reactor demand consider more than one scenario. For this report a single case has been adopted for uranium requirements. An estimated best-fit straight line was drawn through the projections defining the approximate middle of the demand range. The straight-line, identified as "Requirements" in Fig. 2, is the demand projection used in this study. Reactor demand is projected to increase steadily by nearly 600 tonnes per year from 61 500 t U in 1997 to 75 000 tonnes in 2020. This represents a rate of growth of close to one per cent per annum, unremarkable by any standards, and reflecting the strong likelihood that new reactor startups will be few in number. The cumulative requirements over this period are about 1638 000 t U. This is a substantial amount of uranium, equivalent to about 85 per cent of total world production through 1996 [7].

3.2. Supply

Supply sources expected to be available to satisfy reactor requirements include:

- utility and producer inventory (Western and Russian Federation);
- 500 t HEU from Russian nuclear weapons, plus 20 000 t U natural equivalent from the USDOE stockpile;
- mixed oxide fuel (MOX) and reprocessed uranium (REPU); and
- newly mined and processed natural uranium.

Estimates were made of the availability of inventory, HEU, MOX and REPU on an annual basis. The sum of these estimates was subtracted from annual reactor demand to determine the amounts of newly produced natural uranium necessary to meet the remaining unfilled requirements. Following is a brief description of assumptions used to establish the annual availability of each component of total supply.

3.2.1. Inventory

Two types of inventory are addressed in this analysis: excess Western inventory and estimated inventory held by the Russian Federation. The majority of non-Russian inventories are held by utilities for security of supply reasons. There are also smaller amounts owned by producers, uranium traders and the US Department of Energy. Discretionary utility inventory at the beginning of 1997 (inventory held by utilities in excess of preferred or mandated levels) is estimated to total about 50 000 t U. In this study the last discretionary inventory (designated "Invent. W" on figures) is projected to be sold in the year 2000.

Inventory of natural and/or low enriched uranium held by the Russian Federation at the beginning of 1997 is estimated to total about 30 000 t U. This inventory (identified as "Invent. CIS" in this report) is projected to be gradually drawn down through 2004.

The projected schedules for draw down of the utility and Russian (CIS) inventory, in t U, are as follows:

Year	Invent. W	Invent. CIS
1997	17 000	4 000
1998	14 000	6 000
1999	11 000	5 000
2000	8 000	5 000
2001	0	4 000
2002	0	3 000
2003	0	2 000
2004	0	1 000

No reliable estimate of inventory held by producers/suppliers is available; and therefore no provision was made for this potential source of supply.

3.2.2. Russian HEU plus USDOE stockpile material transferred to USEC

This supply source consists of HEU from the Russian Federation, together with US Department of Energy (USDOE) stockpile material transferred to the United States Enrichment Corporation (USEC). The Russian HEU consists of 500 t warhead material under sales agreement with the US Government. The quota for US sales of the LEU blended from the 500 t HEU ranges: from 769 t U (2 million pounds U_30_8) in 1998, to 5 000 t U (13 million pounds U_3O_8) in 2004, reaching 7692 t U (20 million pounds U_3O_8) in 2009. An additional amount equivalent to 5 384 t U may be sold from 2001 to 2005 at the rate of 1 154 t U/annum.

Sale of the USDOE stockpile material (including up to 20 000 t U natural equivalent of HEU, LEU and natural uranium) for use in the USA, following transfer to USEC, is provided for in the law authorizing privatization of USEC [8]. Following transfer to USEC, the USDOE stockpile material is to be sold at the rate not exceeding 1 538 t U/annum from 1998 to 2010.

Probably no other supply source is surrounded by more uncertainty than HEU held by the Russian Federation. Politics, economics and technology will all play a role in determining the availability of uranium from Russian HEU. A total of 500 tonnes HEU, equivalent to 153 000 t U natural, is scheduled for delivery. The High HEU case for this report assumes that the HEU delivery schedule agreed upon in 1996 will be adhered to over the period. A total of 18 tonnes was delivered prior to 1997, the initial year of this report. The annual delivery schedule for the remaining 482 tonnes is as follows:

Year	t HEU	t U Equivalent*
1997	18	5 700
1998	24	7 644
1999 to 2012	30/year	9 555/year

*The HEU is to be blended with 1.5wt% LEU in Russia to produce 4.4wt% LEU for use in commercial reactors. The blending of 1.5% LEU with 94% HEU to obtain 4.4% product (the assay of the material to be imported into the US) requires 30.85 kg U of LEU for each kilogram of HEU. To process 10 tonnes of HEU requires 308.5 tonnes U of 1 LEU and produces 318.5 tonnes U of 4.4% product.

In the high HEU case of this report, it has been assumed that all of the uranium derived from the 500 tonnes of Russian HEU will be available on the world market in the year it is delivered to the US. There are, however, political and technical uncertainties as to whether the proposed delivery schedule can be maintained. In the Low HEU case, it has been assumed that the uranium derived from the Russian HEU will be available on the world market at the rate mandated for sale in the USA, according to US Law [9]. It is also assumed that the USDOE stockpile transferred to USEC will also be available for sale at the maximum rate mandated by the same US Law. The delivery of HEU under the High and Low cases is shown in Fig. 3. The potential for additional sales of uranium derived from US weapons HEU or stockpiles is not directly addressed in this study. However, this potential mitigates the uncertainty surrounding the Russian HEU delivery schedule.

It should be noted that both the Low and High HEU cases provide for market entry by 2020 of amounts of uranium in excess of the 173 000 t U derived from both the 500 t of Russian weapons HEU under purchase agreement with the USA, and the 20 000 t U from the USDOE stockpile. In the Low and High HEU cases respectively, the total amount of natural U equivalent projected to enter the market by 2020 is 181 000 and 219 150 t. The Low and High cases exceed the 173 000 t U of Russian plus US HEU equivalent by 8 000 and 48 150 t U, respectively. In the Low HEU case market entry of the Russian and USDOE equivalent will end in 2019, while in the High HEU case it would end in 2015. The excess amounts provide for market entry of either Russian or US weapons and/or military stockpile material not otherwise provided for in this analysis. The possibility that additional amounts of such material may be made available to the market is believed to be a realistic possibility over the more than 20 year term of this analysis.

3.2.3. Mixed oxide utilization and reprocessed uranium

Assuming that countries maintain existing policies regarding the reprocessing of spent nuclear fuel versus opting for direct disposal, the future market for these options will be limited. Reprocessed uranium and plutonium for MOX fuel is an important component of the supply in only a limited number of countries. The contribution of MOX and REPU is not expected to exceed about 6 per cent of annual requirements through 2020. The rate of utilization of MOX and REPU is shown in Fig. 4.

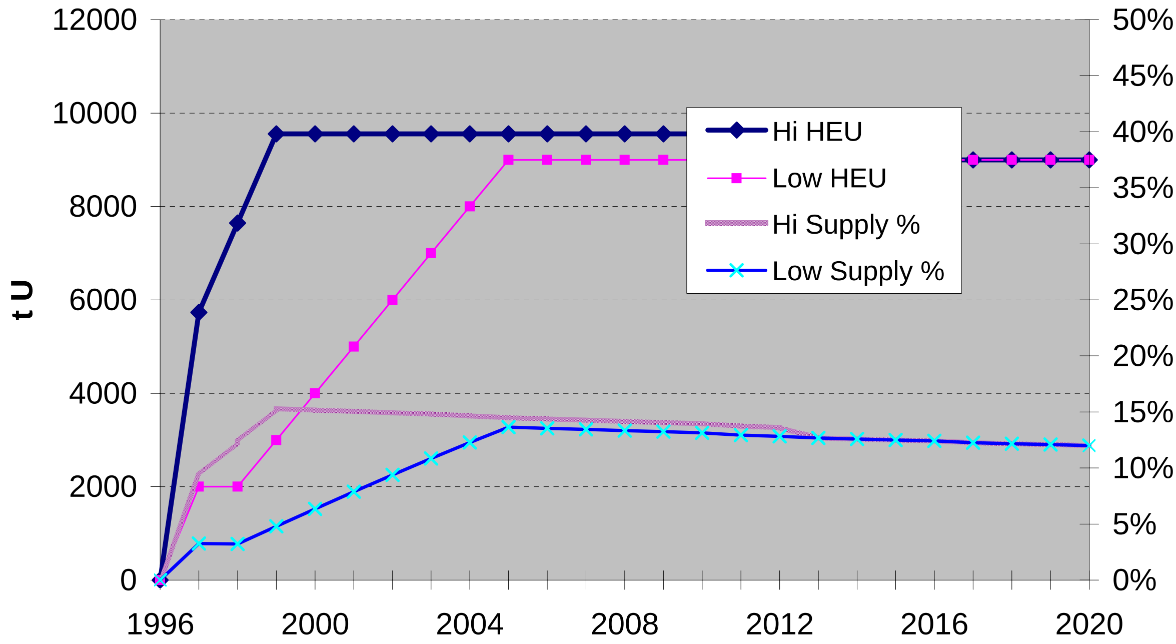

FIG. 3. Delivery of U (nat. equiv.) In high & low HEU cases.

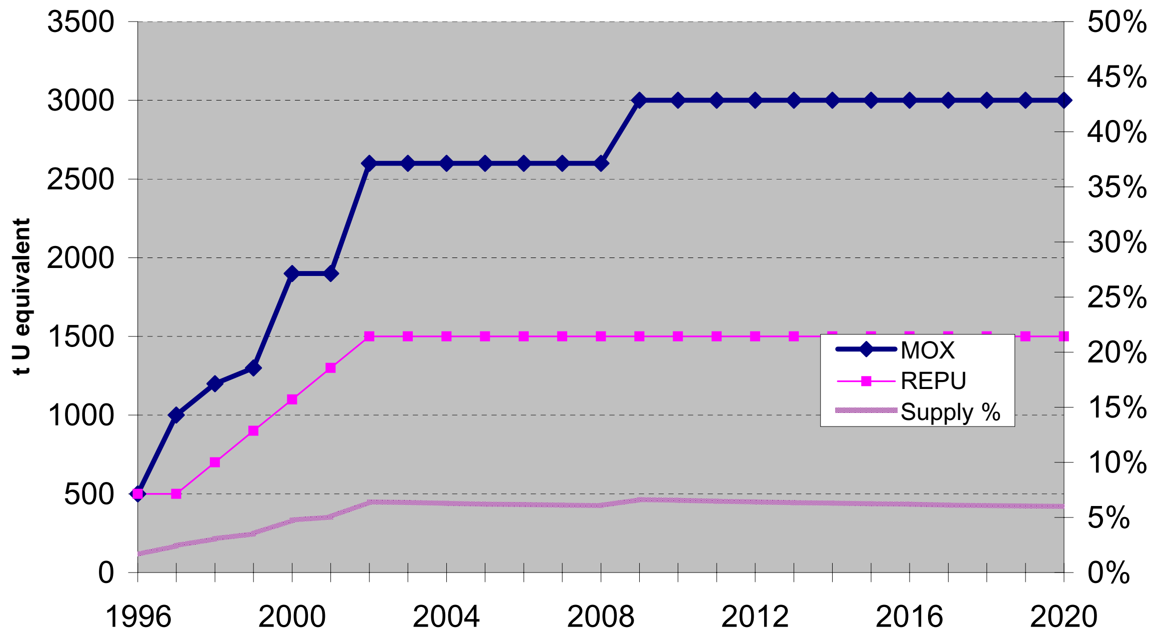

FIG. 4. MOX & REPU use by year.

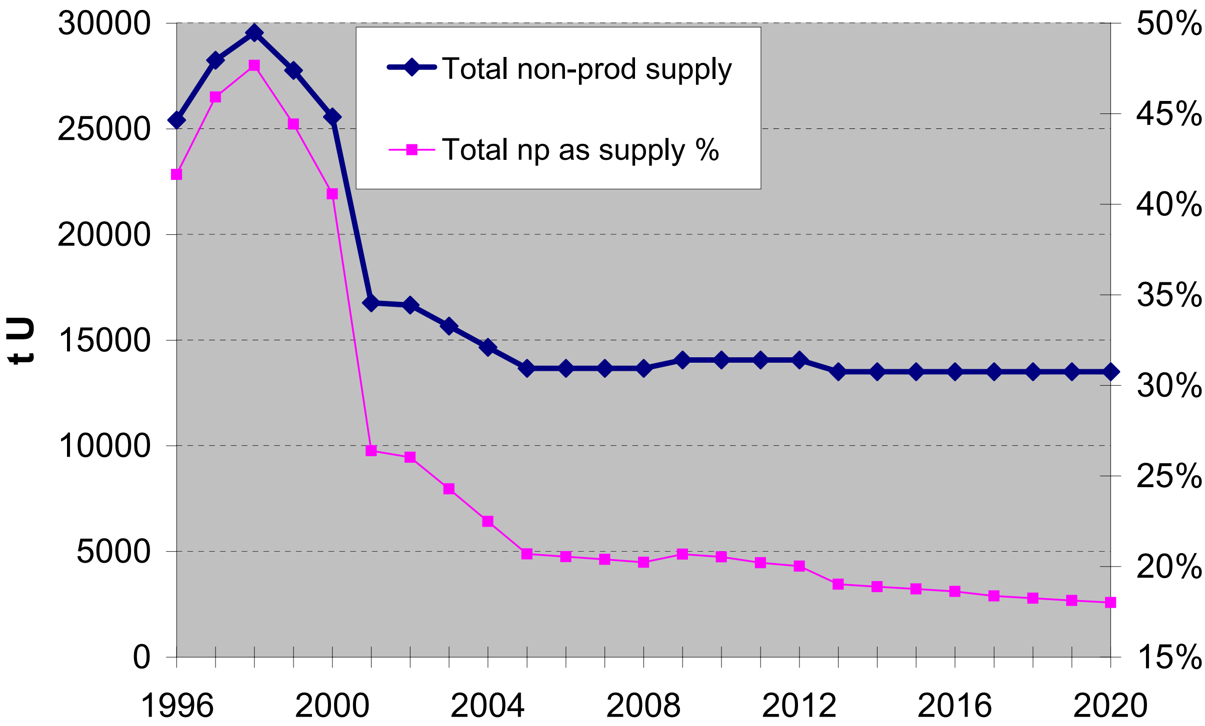

FIG. 5. Non-production U supply and share (high HEU case).

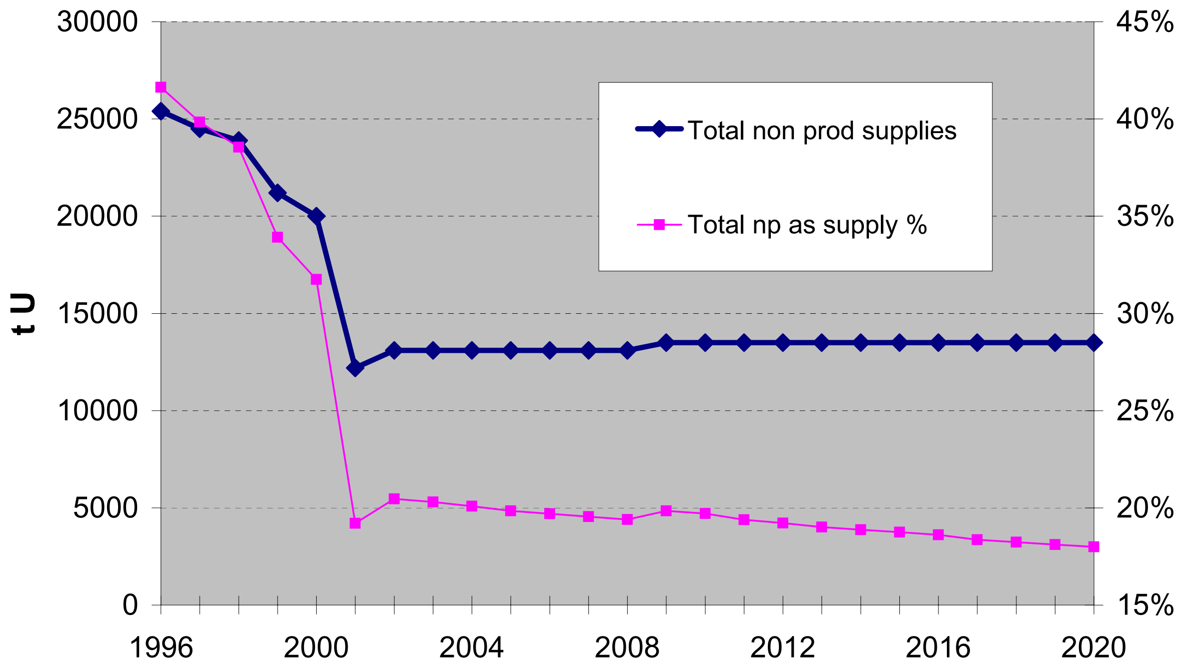

FIG. 6. Non-production U supply and share (low HEU case).

In summary, the total non-production supplies (inventory, LEU from HEU, MOX and REPU) for the High and Low HEU cases are shown in Figs 5 and 6. The requirements not filled by these non-production sources will have to be met from production of natural uranium.

3.2.4. Natural uranium

Potential sources of natural uranium were subdivided into three categories:

1. Supply from the Commonwealth of Independent States (CIS);
2. National programmes in which production is dedicated to domestic nuclear power programmes ("Captive Production");
3. All "Other" uranium production centres not included in the first two categories.

Following are assumptions made as to the availability of supply from each of these sources.

3.2.4.1. CIS Supply

The projected production schedule for the CIS producers (t U) follows:

	1997	1998	1999	2000	2001
Kazakhstan	1 700	2 485	3 525	3 525	3 525
Russian Federation	2 500	2 500	2 500	2 500	2 500
Ukraine	385	385	385	385	770
Uzbekistan	1 700	2 130	2 560	2 560	3 000
TOTAL	6 285	7 500	8 970	8 970	9 795

This schedule projects a 55 per cent increase in CIS uranium production between 1997 and 2001. Projected capacity levels for the year-2001 are used throughout the remainder of the study (i.e. through 2020). In 2001 the CIS production equals about 15.5 per cent of requirements. It then decreases gradually to about 13 per cent of requirements in 2020. Over the period 1997 to 2020 CIS production meets 14 per cent of requirements.

3.2.4.2. Captive (national) programmes

Several countries have small uranium production programmes dedicated to meeting domestic nuclear power programme requirements. While several of these programmes have high production costs, they are maintained either because of their importance to local economies or for reasons of national security. For this study, the production schedule for the Captive programmes is balanced with reactor demand in the following countries:

Argentina	Pakistan
Brazil	Romania
India	Spain

Also included in the Captive programmes category are the production industries of:

France	(projected to produce through 1999)
The Czech Republic	(projected to produce through 2003)
Hungary	(production scheduled to terminate in 1997)
Portugal	(projected to continue production through 2020)

Over the period of this study Captive production equals about 5 per cent of requirements.

3.2.4.3. "Other" natural uranium production

Total projected annual availability of CIS and Captive programme material was subtracted from total natural uranium requirements to determine uranium demand supplied from *"Other"* sources (i.e., non-CIS and non-captive sources). Figure 7 shows the distribution of production between CIS, Captive and *"Other"* producers. ***Projections of annual production were then made for each "Other" project, based on announced plans of these existing or potential producers***. Using NAC International's *Uranium Supply Analysis (USA) System* (Appendix 1), an initial analysis was completed based on the assumption that the lowest cost producers will fill uranium requirements for the *"Other"* category. Higher cost capacity not required to meet a given year's production is assumed to be deferred to subsequent years when, and if, it becomes cost competitive.

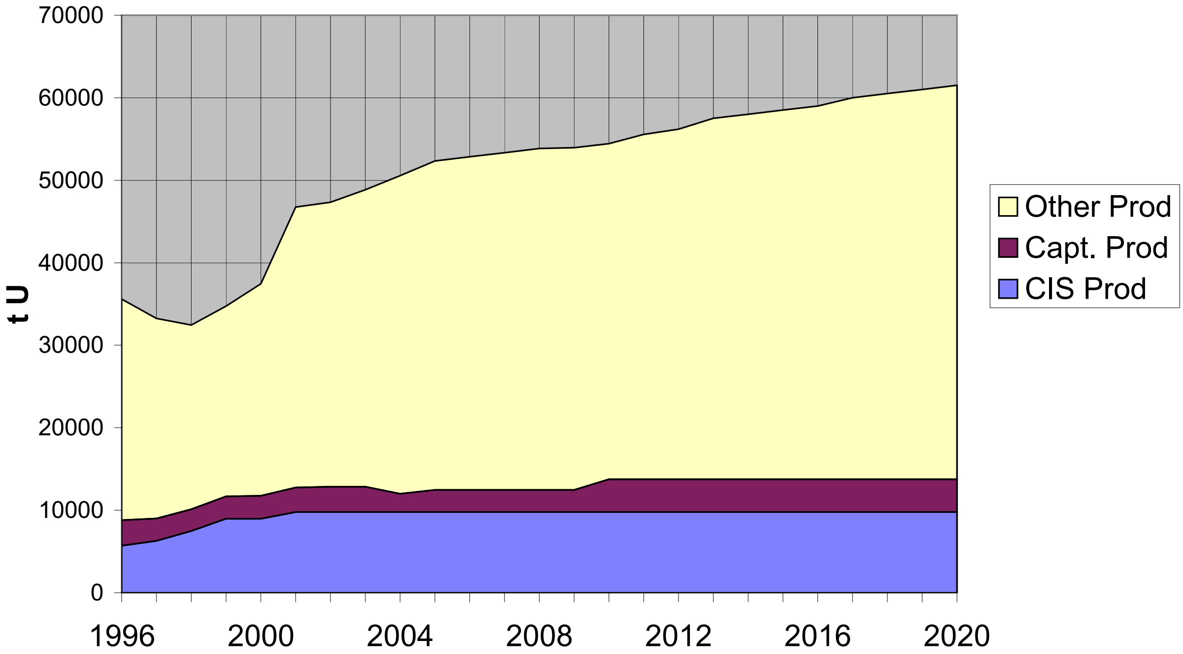

FIG. 7. Uranium production forecast (high HEU case).

For this study the forward production cost is used. The production cost of all production centres are classified in 3 categories:

Low Cost < \$33.80/kgU (\$13/lbU_3O_8)
Medium Cost \$33.80 to \$52.00/kgU (\$13 to \$20/lbU_3O_8)
High Cost > \$52.00/kgU (\$20/lbU_3O_8).

The market balancing routine of the *USA System* provides a rigorous evaluation based on competitive market theory in which demand is filled by the lowest cost producers. In reality, however, there are exceptions to this theory in which higher cost operations continue production for contractual, political or social reasons. In the real world, therefore, these higher cost producers displace lower cost projects, which are then forced to delay startup to later years. Modifications were made to results from the USA *System* market balancing routine to accommodate these special situations. In addition to project deferrals, adjustments were also made in output from lower cost production centres already in operation to accommodate output from the higher cost producers.

TABLE 1. CAPACITY AND STATUS OF CURRENT AND PROPOSED PRODUCTION CENTRES

	1 Jan. 97 [10] Capacity	Maximum Capacity (Tonnes U)*	Status (if known)
Australia	5000		
Ranger/Jabiluka		6 000	Operating
Olympic Dam		3 750	Operating
Kintyre		1 500	Permitting
Koongarra		1 000	
Beverley		600	Preliminary
Yeelirrie		2 120	Feasibility Study
Canada	12950		
Key Lake/McArthur River		6 920	Operating
Rabbit Lake		4 620	Operating
Cluff Lake		1 200	Operating
McClean Lake/Cigar Lake		9 230	Under Construction
Dawn Lake		770	Exploration
Kiggavik		1 400	Exploration
China & Mongolia	890	1 730	
Gabon	587		
Okelobondo/Mounana		630	Operating
Namibia	3000		
Rössing		3 850	Operating
Niger	3800		
Akouta		1 970	Operating
Arlit		1 000	Operating
South Africa Gold & Copper	1900		Operating
By-product		1 320	
United States	4230		
Highland		580	Operating
Crow Butte		380	Operating
Kingsville Dome/Rosita		770	Operating
Christensen Ranch		770	Operating
Smith Ranch		1 150	Operating (1997)
Gas Hills		960	Permitting
Church Rock/Crownpoint		770	Permitting
Uncle Sam & New Wales		960	Operating (Uncle Sam)
Reno Creek/Dewey Burdock		380	Permitting
Alta Mesa		380	Permitting
Jackpot/Sweetwater		1 540	Stand-by
Ticaboo		380	Permitting
TOTAL	**32357**	**58630**	

* Maximum capacity expected to be achieved during study period.

Results of the analysis indicate production from current operations and proposed projects with well established reserves is adequate to supply demand for "*Other*" natural uranium through the year 2012. Table 1 lists current and proposed production centres expected to contribute to the supply of *Other* natural uranium.

In addition to the projects with well established reserves (Reasonably Assured Resources, RAR) listed in Table 1, exploration programmes in Australia, Canada and the United States have identified resources in the Estimated Additional Resources-Category 1 (EAR-1) that are expected to be developed in the future. Contributions from these less well known resources are expected to be needed to fill demand from some projects beginning in the year 2013. Exploration programmes are ongoing in these three countries, the results of which are expected to increase confidence in these less well known resources. These delineation and evaluation programmes will have to continue for the EAR-1 to become available to meet future requirements.

In this analysis *"Other " production* meets 57 and 59 per cent respectively, of requirements in the Low and High HEU cases. The main differences between these 2 cases occurs over the period through 2004. In the Low case, *Other* production increases from 28 000 t U in 1997 to 38 500 t U by 2001. It then gradually increases to 47 700 t U in 2020. For comparison, in the High HEU case *"Other"* production increases from 24 000 t U in 1997, to 34 000 t U in 2001. It then increases to 39 900 t U by 2005, and then gradually expands to 47 700 t U in 2020.

4. ANALYSIS

Three major concerns were addressed in this study:

Analysis of these issues are discussed in this section.

4.1. Adequacy of uranium resources

As previously noted, production from projects with well-defined reserves is adequate to fulfill requirements through the year-2012. Beyond that time, lower confidence resources will be required to fill demand. The main question to be answered is the source of the lowest cost resources most likely to be produced for the market.

Based on extensive exploration programmes completed to date, Australia, Canada and the United States are considered the most likely sources of the lowest cost resources among those countries contributing to the "Other" natural uranium category. Accordingly it was assumed that new projects in these countries will be the source of additional production after 2010. The Athabasca Basin is considered the most likely source of additional low- to medium-cost resources in Canada. The Northern Territory, South Australia and Western Australia are all considered to have excellent potential to host low- to medium-cost resources. Sandstone deposits amenable to in situ leach extraction are considered to be the most likely source of additional low- to medium-cost resources in the United States.

Figures 8 and 9 show the projected distribution of production by country from 1997 through 2020 for the High and Low cases. Canada is expected to be the dominant producer throughout the study period. Canada's production is expected to peak at 20 400 t U in the year-2002 (54% of requirements in the "Other" natural U category) of the Low HEU case when McArthur River and Cigar Lake reach capacity. Canada's total production capacity is greater than that shown in the year-2000. It has, however, been adjusted downward along with other low-cost production to accommodate continued output of high-cost producers elsewhere in the world. Canada's share of the world market is expected to decline to about 38% of the "Other" natural uranium category in 2020 as reserves are depleted and as low-cost production capacity increases elsewhere in the world.

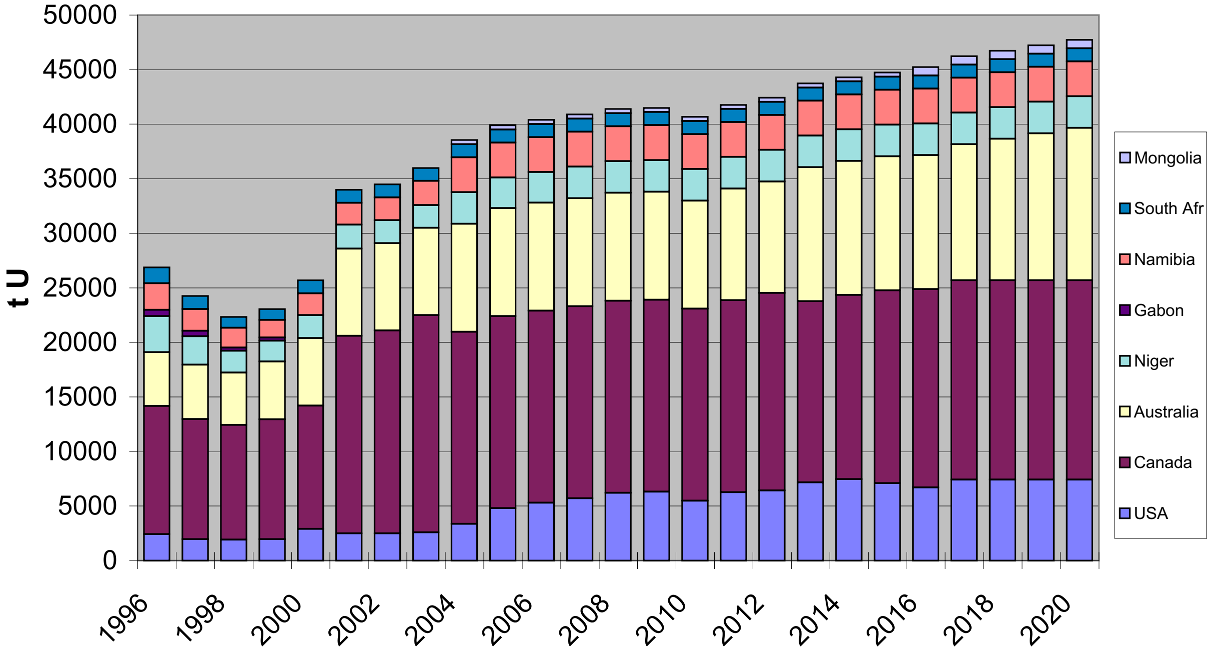

FIG. 8. Detailed production forecast (high HEU case).

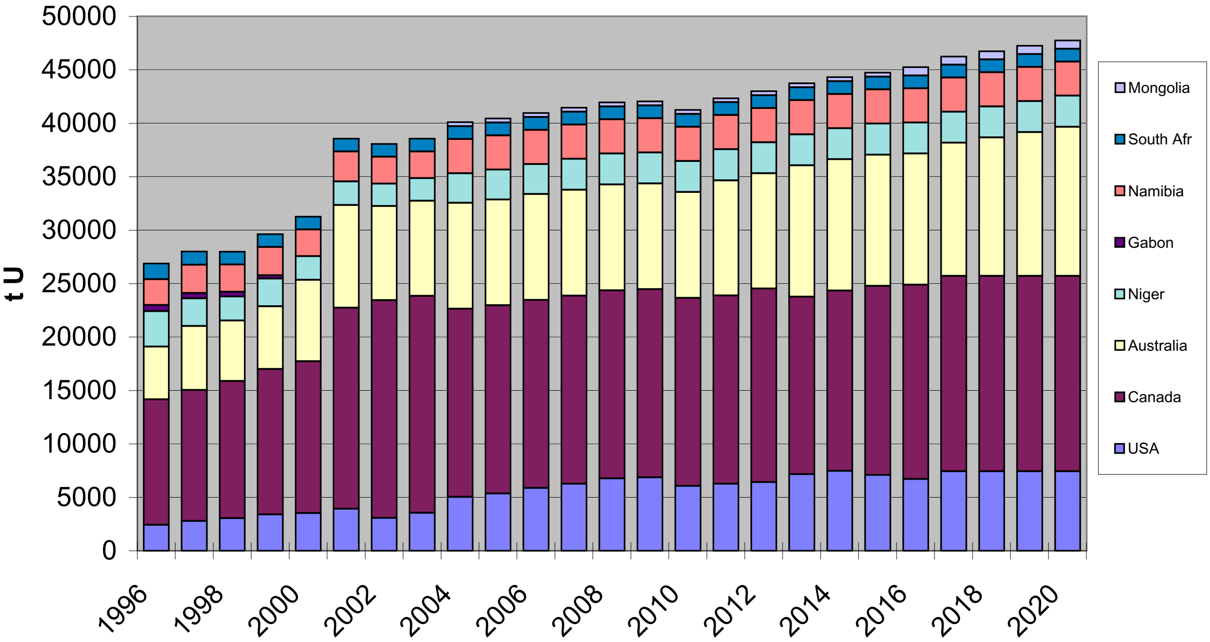

FIG. 9. Detailed production forecast (low HEU case).

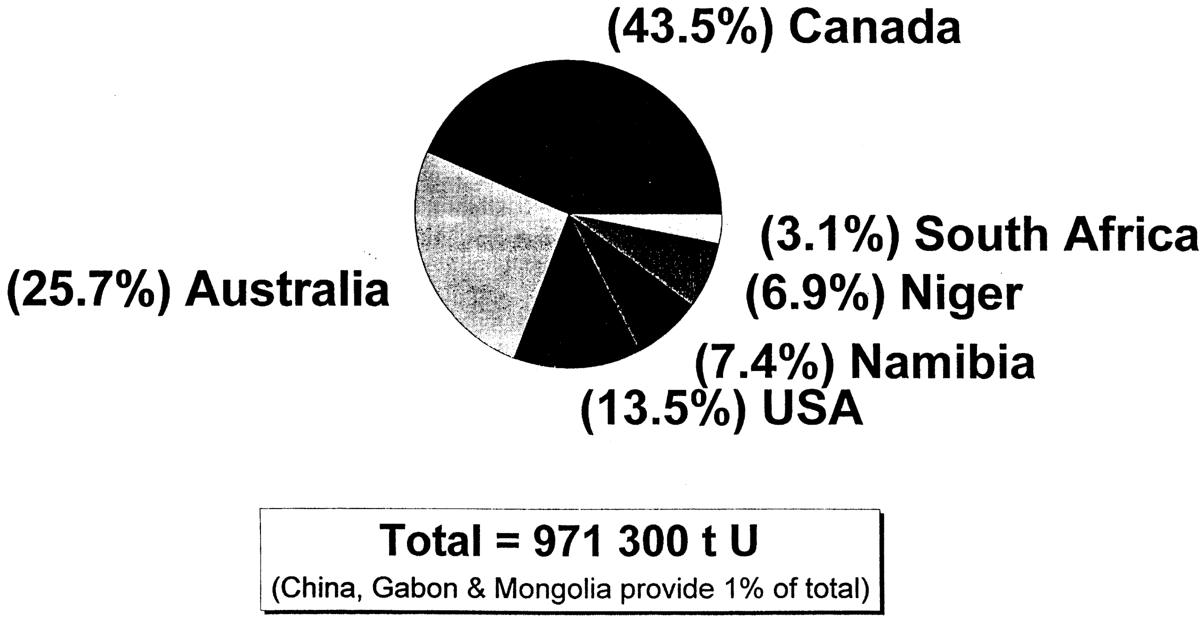

FIG. 10. "Other" production: 1997–2020.

Production in Australia and the United States is expected to increase from about 21 % and 10 % of " Other " natural uranium in 1997 Low HEU case, respectively, to about 29 % and 16 % in the latter part of the study period. Cumulative output from Niger and Namibia is expected to meet between 5 and 10% of demand through 2020. A summary of production from the major producing countries is given in Fig. 10.

4.2. Adequacy of supply capability

Uranium: As reported in the Red Book, the world annual production capability on 1 January 1997 was 43 000 t U [11].This is comprised of 8050, 2600 and 32 350 t U/annum, respectively of CIS, Captive and "Other" production capability.

Uranium production in 1996 was 36 195 t U. This represents a world production capability utilization of 84%. Production capability utilization is defined as: production divided by available production capacity. Production was distributed: 6275, 2440, and 27 450 t U, respectively in the CIS, Captive and "Other" groups. In 1996 capability utilization was: 78, 93 and 85 per cent, respectively for the CIS, Captive and "Other" production groups.

Based on historical performance, 85 per cent is about the maximum sustainable utilization level achieved by the uranium industry.

In the high HEU case total uranium production will change little through 2000. If, as planned, CIS production expands by 42 per cent, "Other" production could decrease by nearly 2000 t U, or a temporary production surplus could develop. In contrast, in the Low HEU case world production will increase to 43 000 t U by 2000, or by 19 per cent. In principal this requires no production capability increase over the capability reported in the 1997 Red Book. It should be noted that CIS production is projected to increase 55 per cent by 2001. However, achieving this increase will require substantial growth from an industry segment where production has fallen in every year since information on production first became available in 1992 [12].

Estimated production for both the High and Low HEU cases is about 52 500 t U in 2005. This is an increase of over 16 000 t U, or about 44 per cent, from 1996. To produce this amount, a capability increase of between 22 and 26 per cent is required from the 1 January 1997 existing capability of 43 000 t U. The lower value is valid if 100 per cent capacity utilization level were achieved. However, a 26 per cent increase will be required if 85 per cent capacity utilization level is met. Under this schedule only 7 years remain to plan, license, construct and bring these projects into production. Additional capacity will be required to produce about 61 500 t U/year by 2020, as well as to replace capacity that closes because of resource depletion (See Fig. 11).

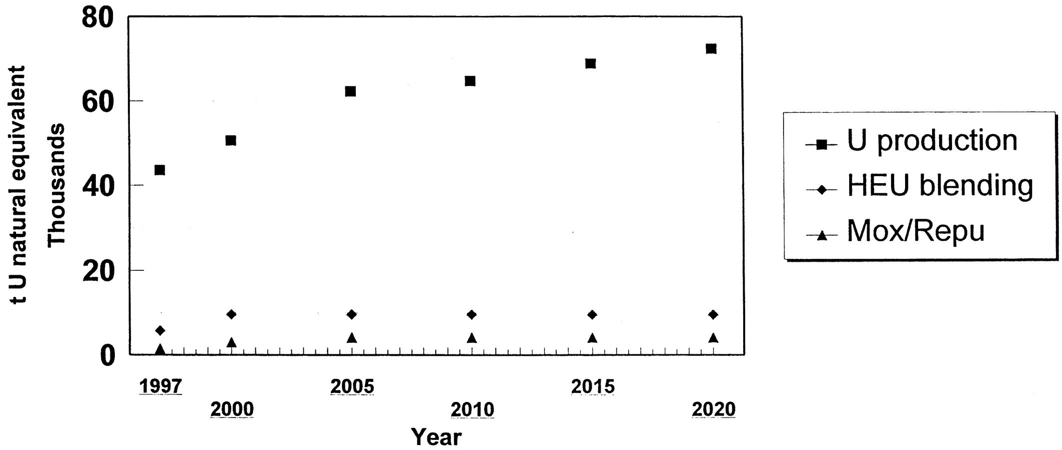

FIG. 11. Required capacity for U production, HEU blending & Mox/Repu fabrication (low HEU).

Installing new capacity requires substantial capital investment. It was recently reported the initial capital investment, in US$/t U annual capacity for high grade unconformity and in situ leach projects, ranges from $44 000 to $66 000 and $30 000 to $44 000, respectively [13]. In comparison the last conventional uranium mill built in the US (in the early 1980s) at Ticaboo, Utah, required a capital investment of about $100 000/t U annual. Mine development costs added substantially to the capital cost of the project. Based on these estimates the capital costs to expand production capability by 1000 t U/year could range from $30 million to over $100 million. An expansion of 16 000 t U/year would cost between $480 million and over $1 600 million.

HEU blending: It was reported in November 1996 that the Russian Federation planned to add a new facility at Tomsk-7 to increase its capability for blending HEU to LEU from 12 t/year to 18 t/year [14]. Russia will have to implement this capacity increase to meet both the High and Low HEU cases of this report. Assuming this is the total blending capacity of the Russia Federation, it is apparent additional blending capacity will have to be put into service to achieve the planned deliveries from 24 t HEU in 1998 and 30 t HEU in 2000 and subsequent years. It therefore appears that substantial additional blending capacity must be added if Russia is to meet either the High or Low HEU cases of this study. The cost of these facilities is not known.

MOX and REPU: The contribution for MOX and REPU is modest, increasing from about 1500 t U (equivalent) in 1996, to 3000 and 4100 t U, respectively in 2000 and 2005. This can only be achieved if planned expansion of facilities for recovering and fabricating these materials are developed and put into production.

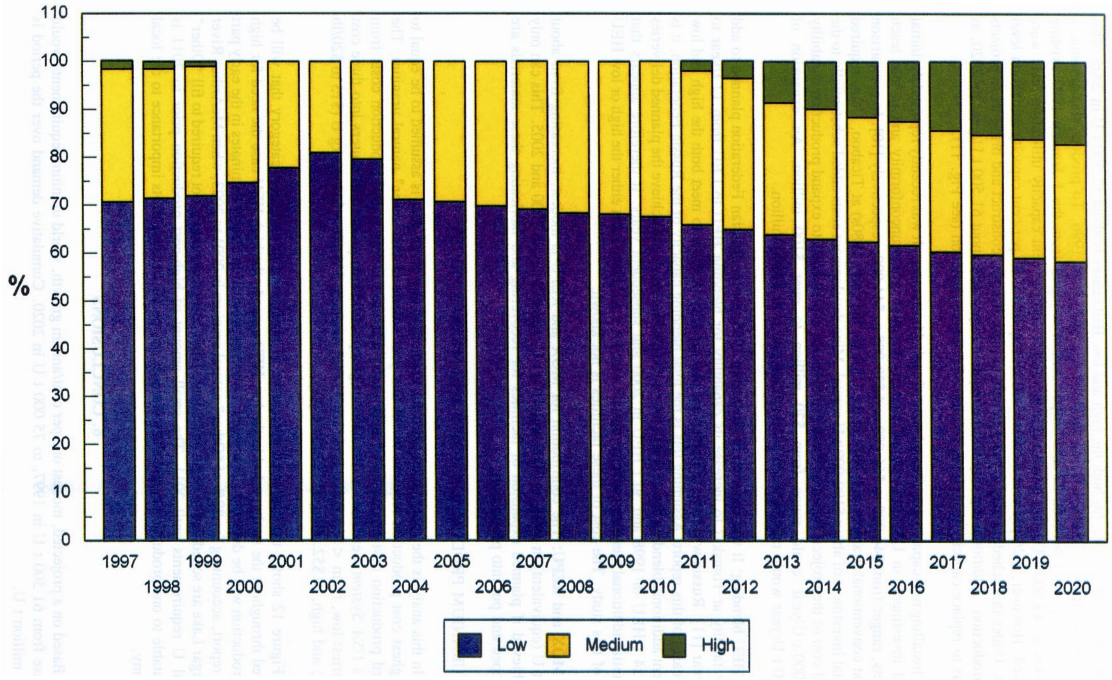

FIG. 12. Distribution of production by cost category (low HEU case).

4.3. Uranium production costs and market prices

In this analysis the spot market price for uranium in any given year is assumed to be equal to the highest cost production required to fill that year's demand for "Other" natural uranium. The forward production cost (exclusive of sunk costs) was used in this study. Production costs from NAC's *USA System* were used as the basis to group worldwide production centers into three cost categories: Low, from <$33.80/kg U ($13/lb U_3O_8); Medium, $33.80 to $52.00/kg U ($13 to $20/lb U_3O_8); and High, >$52.00/kg U ($20/lb U_3O_8).

Figure 12 shows the estimated percentage of production in each cost category that will be required throughout the study period in the Low HEU case. In the High HEU case the need for high cost production will be deferred a few years. Low-cost production clearly dominates in the early part of the report, accounting for 80% of "Other" natural uranium in 2001 to 2003, when McArthur River and Cigar Lake are scheduled to begin operations. High-cost production is not required to fill "Other" natural U requirements until 2011. The small amount of high-cost production prior to 2011 is attributable to one production centre that continues to operate because of its importance to the local economy.

5. CONCLUSIONS

Based on a projected, modest one per cent/annum growth, world uranium requirements would increase from 61 500 t U in 1997, to 75 000 t U in 2020. Cumulative demand over the period is 1.638 million t U.

Production of 36 195 t U in 1996 met about 60 per cent of world requirements, with most of balance coming from inventory. This source, which has been supplying an average of about 22 000 t U/annum since 1992, is coming to an end. The remaining excess inventory is estimated to be 80 000 t U; comprised of 50 000 t U held by utilities, and 30 000 t U by the Russian Federation.

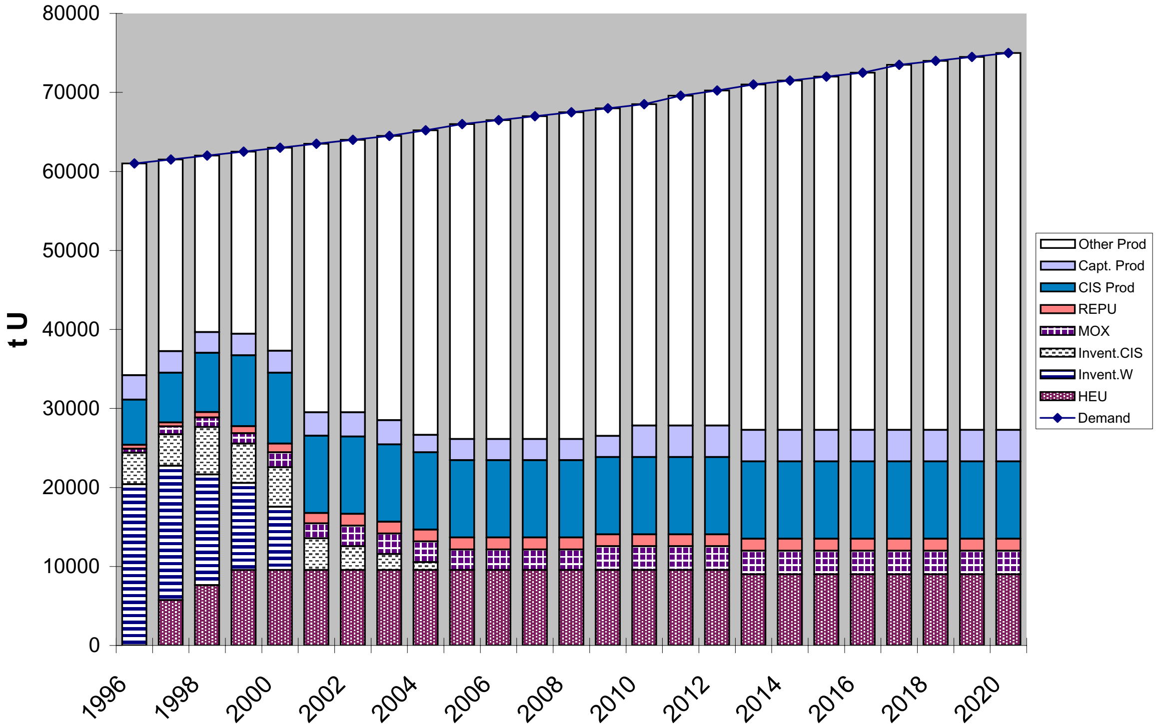

FIG. 13. World uranium supply & demand (high HEU case).

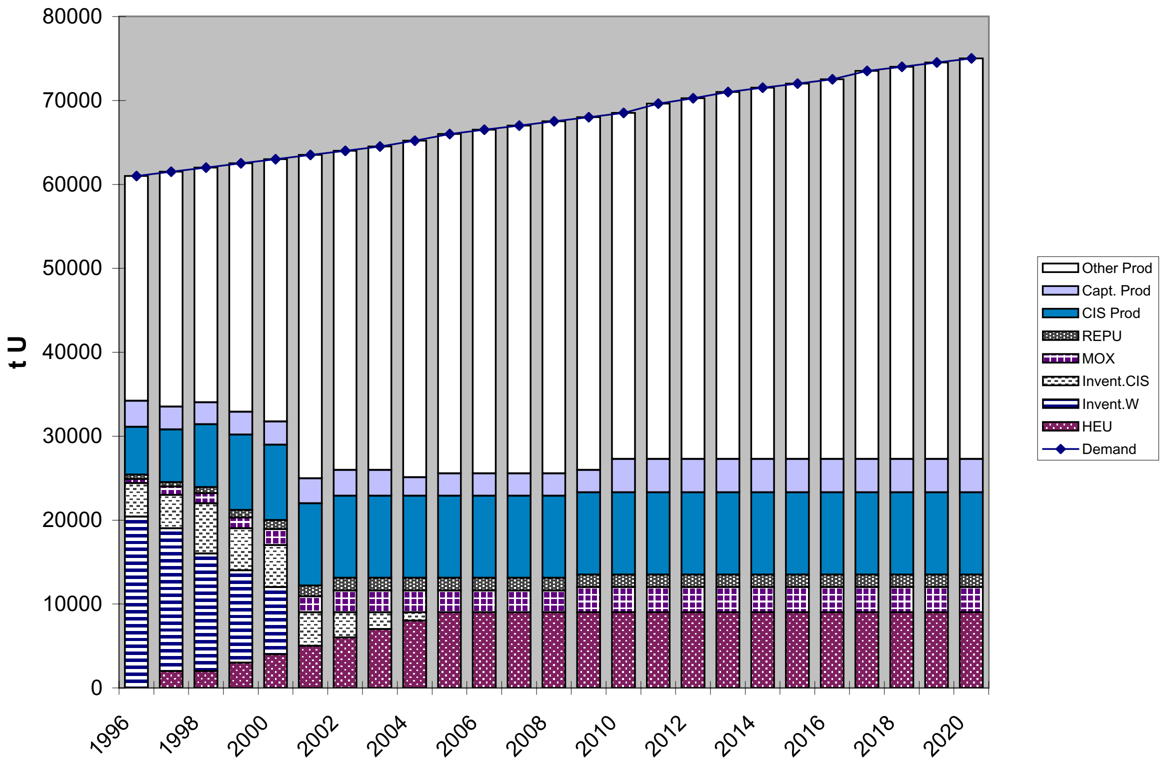

FIG. 14. World uranium supply & demand (low HEU case).

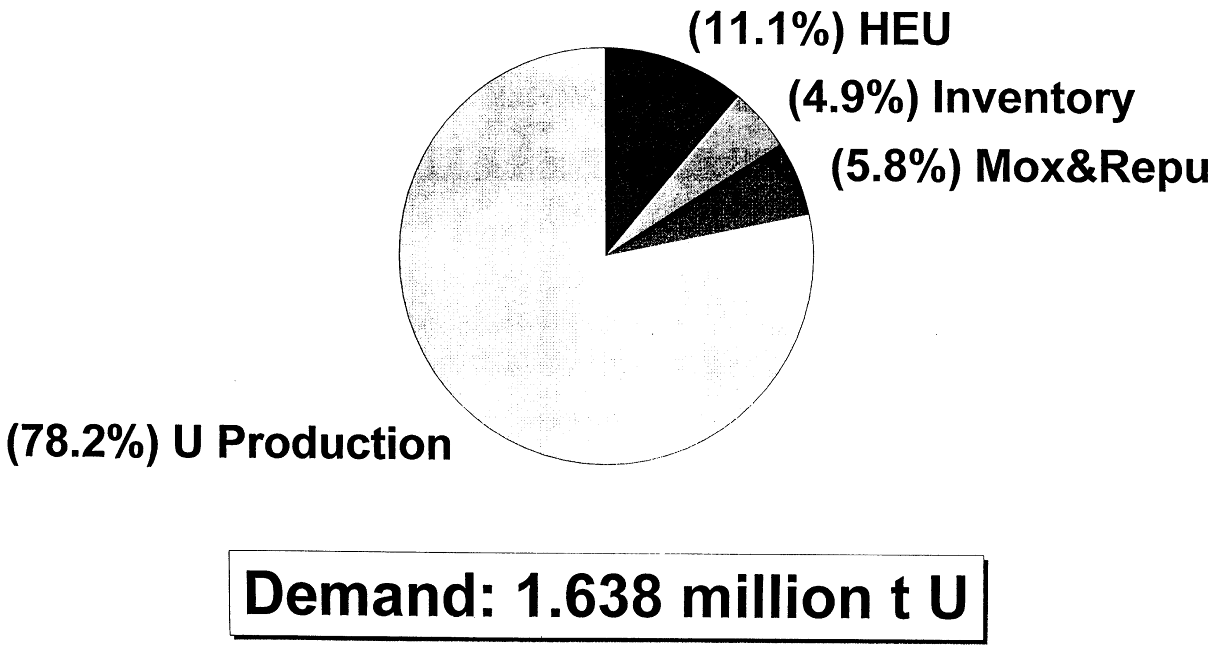

FIG. 15. Uranium supply to 2020 (low HEU).

With the end of excess inventory in sight, uranium supplies from other sources will have to increase to meet requirements. What supply sources are available to meet requirements through 2020?

This analysis indicates uranium mine production will continue to be the primary supply meeting 76 to 78 per cent of cumulative requirements through 2020. Alternative sources supplying the balance, in order of relative importance, are: 1) low enriched uranium (LEU) blended from highly enriched uranium (HEU) weapons (11 to 13 %), reprocessing of spent nuclear fuel (6 %), and excess inventory (5 %). This information is summarized in Figs 13, 14 and 15. The contribution of US government and other Russian strategic stockpiles is not known at this time. However, the potential for supplying levels ranging from these sources (i.e. 28 000 (1.7%) and 68 150 (4% of requirements) are accommodated in this analysis.

5.1. Adequacy of known uranium resources

There are adequate Known Reasonably Assured Resources (RAR) and Estimated Additional Resources-Category 1 (EAR-1), producible in the low (≤\$33.80/kg U or ≤\$13.00/lb U_3O_8) and medium (≤\$52.00/kg U or ≤\$20.00/lb U_3O_8) cost categories, to meet requirements through about 2010. After 2010 it will become necessary to start producing from the high cost (≤\$52.00/kg U or ≥\$20.00/lb U_3O_8) category. It will also be necessary to continue exploration (i.e. delineation and evaluation) of the EAR-1 resources to move them into the RAR class. New discoveries elsewhere in the world could obviously change the production distribution shown in Figs. 8 and 9. While the distribution may change, it is important to note that there are adequate known resources worldwide to satisfy demand through the year-2020.

Assuming equilibrium conditions between the supply sources and demand, the spot price would not be expected to exceed \$52/kgU (\$20/lb U_3O_8) until at least the year 2011. Of course this basic assumption may not hold during periods of market disequilibrium or disruption, which could result in higher prices. As indicated in Figure 11, high cost production is never expected to fill more than about 17% of the "Other" natural uranium requirements through 2020. And, there is a very real likelihood that additional low- to medium-cost discoveries will be made that could displace some, if not most, of the high-cost production required in the middle to latter part of the study period. Introduction of additional low priced uranium to the market could also lower the price.

5.2. Adequacy of production capability

As of 1 January 1997 annual world uranium production capacity was 43 000 t U. The Russian capability for blending HEU to LEU, either installed, or being installed, was 18 t HEU/annum (or 5733 t U natural equivalent). The capacity of facilities for fabricating mixed oxide (MOX) and reprocessed uranium (REPU) fuel was about 1500 t U (natural equivalent). To meet the increasing amounts of supply projected for each of these sources it is necessary to increase their respective capabilities.

Assuming 100 per cent capacity utilization, it will be necessary to increase the uranium production capacity in both the High and Low HEU cases by about 9300, 11 700 and 18 500 t U/annum, respectively in 2005, 2010 and 2020. The capability will have to be increased by 10 900, 13 400, and 21 800 t U/annum, respectively in 2005, 2010 and 2020, if an 85 per cent capacity utilization is achieved. This is equivalent to an increase of 25, 31 and 50 per cent, respectively.

For the Russian Federation to meet the planned delivery of LEU blended from 500 t HEU warhead material, the blending capability will have to be increased. The required increase is from 18 t HEU/annum in 1997, to 24 t HEU before 1998 and 30 t HEU before 1999.

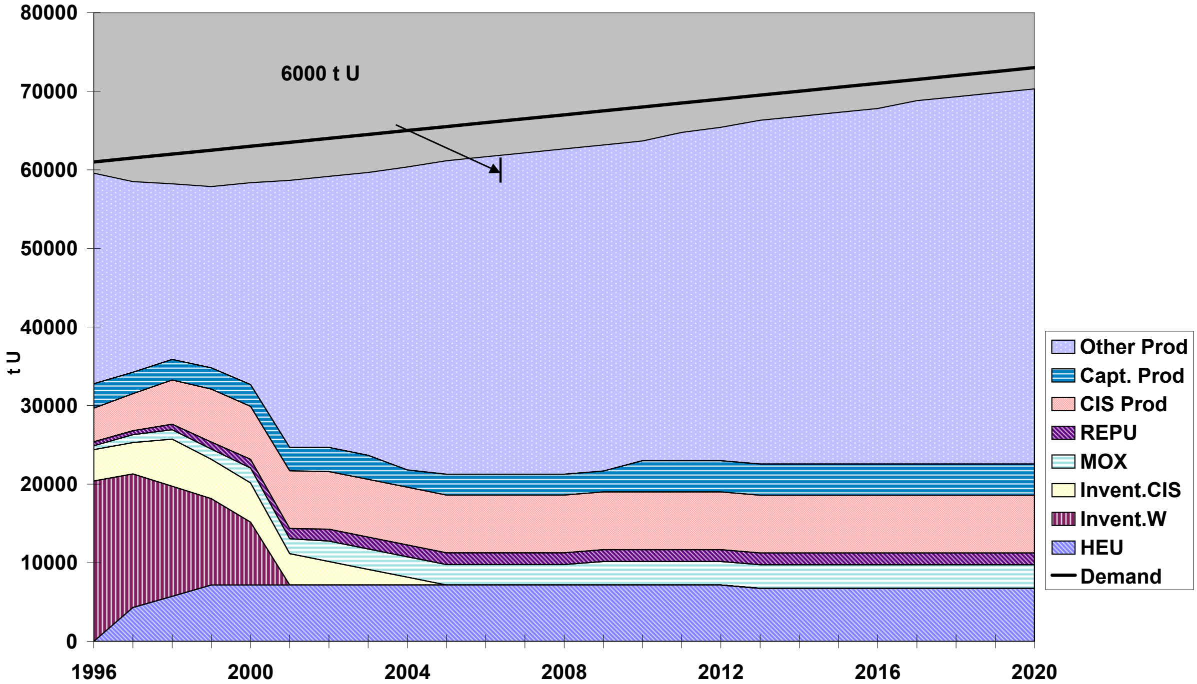

FIG. 16. U supply shortfall scenario (high HEU case) (HEU: -25%; CIS Prod: -25%).

5.3. Recommended required actions

There may be significant market instability unless all of these supply sources are developed according to the projected schedules. Delay in the development of any of the sources would result in a market shortfall leading to price increases. The greatest impact would result from delays in expanding existing and putting new uranium production centres into operation. Shortfalls in the delivery of LEU blended from HEU could have smaller, but still significant impacts. Figure 16 shows an example of an hypothetical uranium supply shortfall where Russian HEU and CIS production both supply uranium at 25 per cent below planned levels. The failure of any supply component to operate at planned levels could result in a shortfall. It is expected that such a shortfall would not persist for an extended period. A shortfall would most probably lead to increased market prices which would be expected to stimulate additional production.

This is a period of major change and uncertainty for uranium supply and demand. In the short term there is a need for significant amounts of new production. In the longer term, particularly after 2010, the level of uncertainty surrounding uranium requirements is greater. Unforeseen events will likely impact the long term supply/demand balance. Under these conditions we must expect the unexpected.

APPENDIX

The *Uranium Supply Analysis System* (USA System) is an interactive computer system developed by NAC International for modeling technical and financial information in the uranium production industry. The System includes two main components:

(1) A comprehensive data base of technical and financial information on uranium production centres throughout the world; and
(2) Interactive programmes for analyzing a broad spectrum of uranium industry supply and demand issues.

The cornerstone of the USA System is a data base that includes technical and production cost information on approximately 130 of the world's operating, planned and potential uranium production centres. Planned production centres are those facilities which are either under development or those with announced development plans and/or production schedules. Potential producers include uranium deposits with proven and/or probable reserves (RAR and/or EAR-1). Production schedules for potential projects are based on technical feasibility - what is the shortest possible time frame that a project could begin operations considering practical permitting and construction schedules. Also included in the potential category are speculative or undiscovered resources that have that have been identified in highly favorable geological provinces such as the Athabasca Basin in Saskatchewan, Canada, and the Arnham Land region, Northern Territory, Australia. However, no speculative resources were included in the present IAEA study.

Reserves/resources in the USA System total 3 354 180 t U. Distribution of this total by country or region is as follows:

	1 000 t U
Australia	410
Canada	420
Central and Western Africa	290
Kazakhstan	470
China and Mongolia	90
Namibia	110
Russia	340
South Africa	190
Ukraine	60
United States	350
Uzbekistan	230
Other	390

Technical parameters stored in the database for each project include: reserves; average grade; production capacity; and mill recovery factor. For practical reasons, the USA System is limited to a 30-year life. Capital and operating costs are input into the System according to production activity. Capital cost line item categories include: mine development; mill construction; and infrastructure. Operating cost line item categories include: mining; haulage; milling; production and severance taxes; royalty; and environmental monitoring and reporting. Production costs are calculated on a forward cost and full cost basis, both with and without a rate of return (ROR). RORs vary depending on project status. Operating projects are presumed to have lower risk and are assigned a 10% ROR. Projects under development and potential projects carry increased risk, and are assigned required rates of return of 12% and 15 %, respectively. Amortization of capital costs is based on a units of production schedule.

The USA System includes primary uranium and uranium produced as a by-product of gold, phosphate and copper production. Uranium derived from weapons-grade highly enriched uranium (HEU) from Russia and the United States is also included in the System. The reserve/resource total includes 172 000 tonnes HEU. Modeling routines in the USA System are interactive. Therefore, the schedule for entry of HEU-derived U308 into the market can be varied to evaluate its impact on supply-demand relationships and market prices.

The market analysis model in the USA System has been used in the present IAEA analysis to help identify the supply sources that will most likely fill demand through the year 2020. This model provides a rigorous analysis based on the assumption that annual demand will be filled by the lowest cost producers. The lowest cost producer operating at full capacity will fill the first increment of demand. Remaining demand is filled by progressively higher cost producers until is annual demand is filled. Production from higher cost projects is deferred until it is cost competitive. Modifications have, however been made to the market analysis model results to accommodate higher cost operations that continue production because of contracting obligations and/or social and political responsibilities.

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Back-end of the nuclear fuel cycle

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Abstract. Current strategies of the back-end nuclear fuel cycles are: (1) direct-disposal of spent fuel (Open Cycle), and (2) reprocessing of the spent fuel and recycling of the recovered nuclear materials (Closed Cycle). The selection of these strategies is country-specific, and factors affecting selection of strategy are identified and discussed in this paper.

1. INTRODUCTION

The generation of nuclear energy in nuclear reactors produces spent fuel (SF). The spent fuel discharged from the reactor is stored in wet pools at reactor sites. When the pool is filled, or when the reactor is at the end of its operating life, the spent fuel would be removed, stored in At-Reactor (AR) storage on site, or transported to away-from-reactor (AFR) storage facility, pending on decisions of final disposal in a geologic repository. This strategy for managing the back-end nuclear fuel cycle is generally called "Direct-Disposal, or Open Cycle". Many countries manage their back-end nuclear materials in this manner, including those which have not made the final decision for the disposition of spent fuel. For them (with a "Wait-and-See" option), spent fuel is kept in wet/dry storage facilities on an interim basis.

The spent fuel could be reprocessed to separate plutonium and uranium from other highly radioactive materials. The separated plutonium could be fabricated as MOX fuel and recycled back to the reactor to produce nuclear energy. This back-end fuel-cycle strategy (so-called "Reprocessing and Recycling", or "Closed Cycle") were selected in the 70s' by several countries primarily because of resource conservation. The strategy also provided the utilities an outlet for their spent fuel (e.g., to the reprocessors). Now, when the separated plutonium and reprocessed high-level radioactive wastes (HLWs) are returned, the utilities would have to deal with the utilization and disposition of these materials.

For the Closed Cycle, the separated plutonium can be utilized as MOX fuel in reactors, or safely and securely stored until decision for its final disposition is made. Beside plutonium, storage for the separated reprocessed uranium is required until its recycle to the reactor is economically viable. Also, the vitrified HLW needs to be stored until final geologic repository is available.

These back-end fuel-cycle strategies are different, due mainly to the management of different nuclear materials arising from these strategies (e.g., either SF, or separated fissionable nuclear materials, and HLWs). In this paper, we attempt to closely examine each strategy to identify important factors determining the strategy and affecting the current and future nuclear systems.

2. BACK-END FUEL CYCLE STRATEGIES

Current strategies of the back-end nuclear fuel cycles are: (1) direct-disposal of spent fuel (Open Cycle), and (2) reprocessing of the spent fuel and recycling of the recovered nuclear materials (Closed Cycle). One may include a third option (3), called "Wait and See" as decision for final disposition of spent fuel has not been made.

These strategies consist of common components, they are:

- Spent fuel management (for all 3 strategies)
- Management of separated fissionable materials (for the Closed Cycle)
- Geologic repositories (for all 3 strategies)

Spent Fuel Management

The back-end fuel cycle begins with spent fuel discharged from the nuclear reactors. As of 1998, the total amount of power-reactor spent fuel discharged world-wide is about 220,000 t HM. Of this, about 75,000 t HM were reprocessed, and the remaining 145,000 t HM is stored.

Figure 1 shows schematically an Open Back-End Cycle. Spent fuel is managed at each stage of the Cycle with specific considerations and controls. For examples: when spent fuel is stored at reactor pools during reactor operation, the consideration is to ensure adequate storage capacity is provided so that potential loss of full core reserve (FCR), a requirement for safe reactor operation, would not occur. Assurance of control for such consideration would be provided by the plant management. When the power plant is at the end of its operating life, or when the plant site needs to be decommissioned and decontaminated, special provisions must also be provided by plant management that the spent fuel is transported to AFR facilities for interim storage.

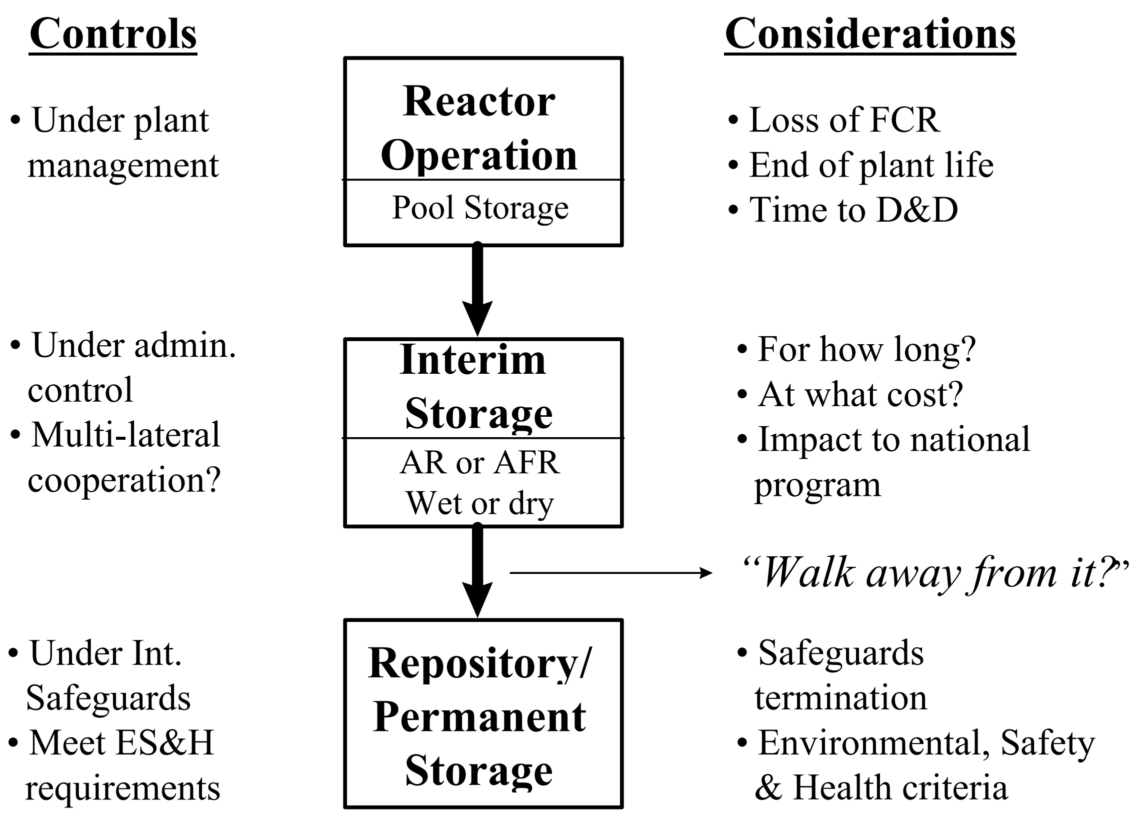

FIG. 1. Spent fuel management.

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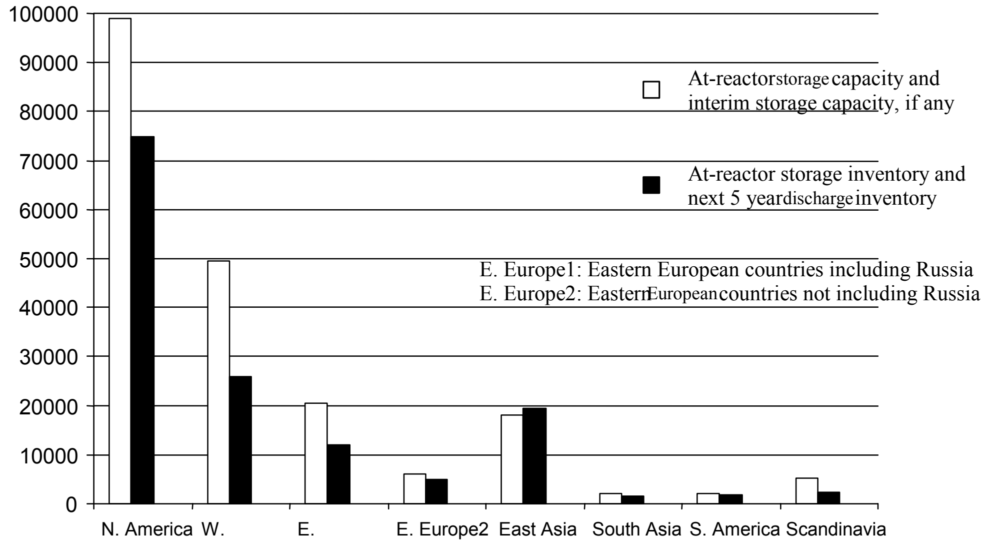

FIG. 2. Spent fuel capacity and inventory by regions.

There are an increasing number of nuclear utilities whose SF inventories may exceed their planned storage capacities. Additional AR and/or AFR storage capacities would be needed. The timing of such need is different for each utility, and/or each country if the utility is state-owned. Figure 2 shows a comparison of spent fuel inventory and planned storage capacity, on a regional basis. The datum is based on Year 1998, and a 5 year additional spent fuel discharged is added to the inventory for the comparison to ensure that the utility and/or country would have at least 5-year time period to prepare for such need.

Technologies for spent fuel storage, both wet and dry are well-developed and commercially available. Considerations at this stage of spent fuel management are: interim storage for how long? at what costs? and what are the potential impacts to national programs if multilateral arrangement is pursued?

For an Open Cycle, spent fuel is to be disposed of in a geologic repository. Two features in spent fuel demand specific considerations and controls. These are: (1) spent fuel contains special nuclear fissionable materials, requiring institutional controls and international safeguards, (2) radioactivity in the spent fuel is generally higher than that of the original uranium ore, requiring a long-time decay or engineered containment to provide a safety assurance. For as long as these two considerations remain unresolved, one cannot simply “walks away” from a spent-fuel repository. As a result, the Open Cycle may remain open and spent fuel management may not be completed.

Management of Separated Fissionable Materials

Figure 3 shows schematically a Reprocessing and Recycling Cycle (Closed Cycle). It is different from that of Figure 1 by an additional stage, e.g., spent fuel reprocessing. There is considerable experience in the civil reprocessing of spent fuel on an industrial scale in some countries. France is successfully operating reprocessing plants for oxide fuels. It has already reprocessed more than 13,000 t HM in its La Hague plants, while the United Kingdom (UK)’s Thorp plant has reprocessed about 1,500 t HM of AGR and LWR spent fuels. France and UK have also reprocessed about 60,000 t HM of gas-cooled fuel at the UP1 and B205 plants respectively. Russia's RT-1 plant has a capacity of 400 t HM/y and to date some 4,000 t HM of WWER fuel has been reprocessed. Reprocessing experience in India and Japan is equally relevant although their installed plant capacities are not as large. Japan is building a 800 t HM plant at Rokkasho-mura with completion expected in 2005.

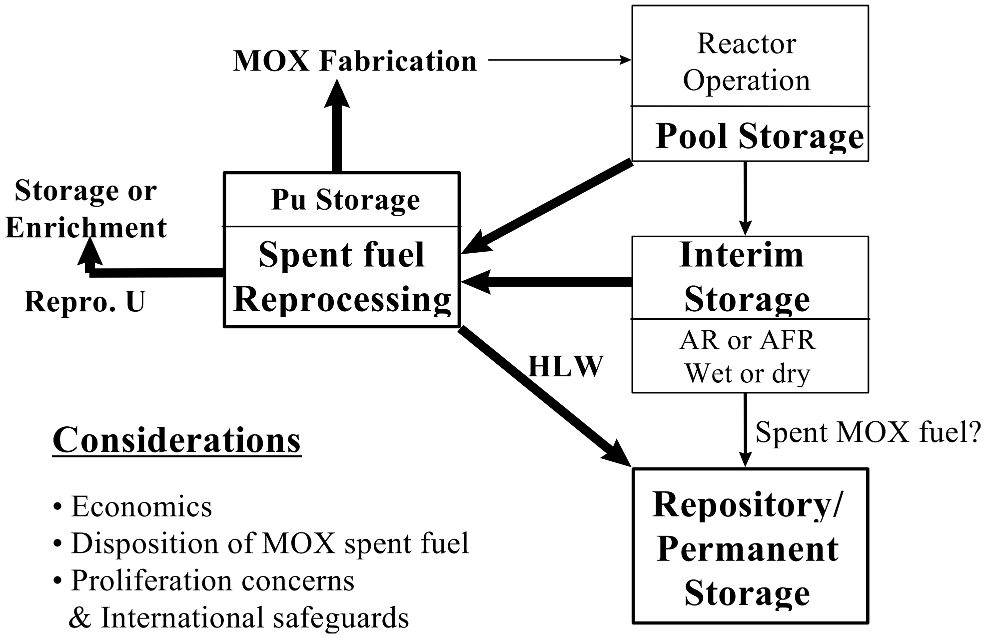

FIG. 3. Reprocessing and recycling.

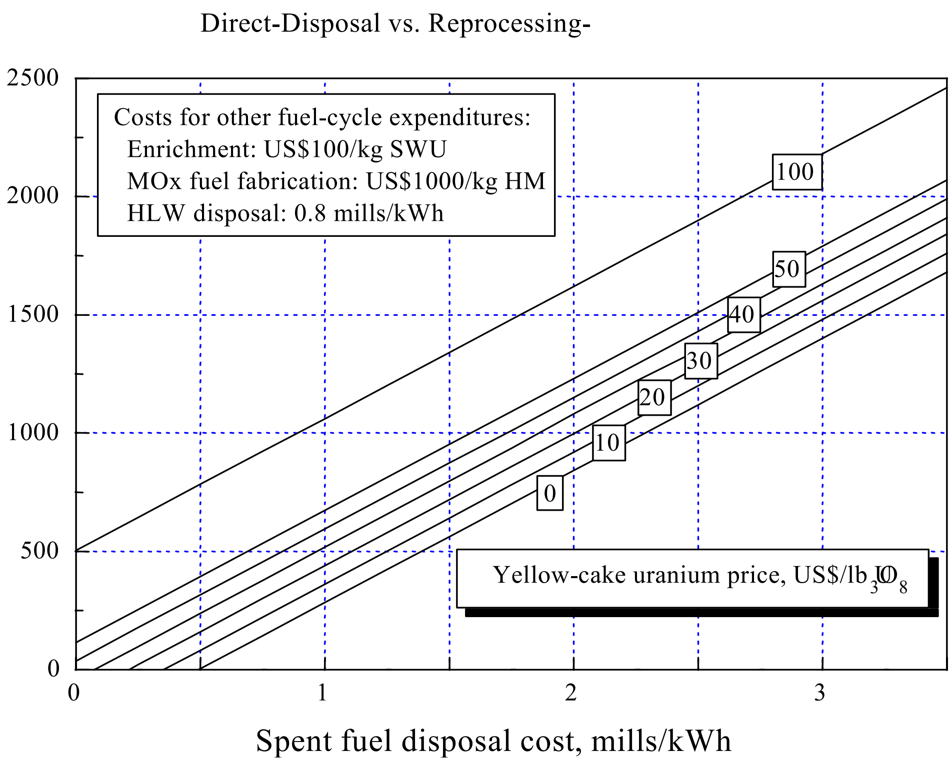

FIG. 4. Break-even fuel reprocessing cost (US$/kg HM).

Spent fuel reprocessing is a costly expenditure. Figure 4 indicated that for a fuel reprocessing cost of $1000 per kg HM and a spent fuel direct-disposal cost of 1 mill per kWh (which is the fee paid by the US utilities to the US Department of Energy (DOE) to dispose of their spent fuel), the unit price for natural uranium would have to be as high as $80 per pound (more than $200/kgU), a price almost 7 times higher than the current spot price.

- *Mono-recycling of separated plutonium in LWRs*

The separated plutonium can be fabricated as MOX fuel and recycled back to the reactor to produce nuclear energy. As of 1998, there are 40 nuclear power reactors in Belgium, France, Germany, Japan and Switzerland licensed to use MOX. Of these, 32 LWRs are loaded partially (~>30% core) with MOX fuel.

MOX fuels are currently used as replacement fuels in LWRs. They are in the reactor core partially replacing the UO_2 fuel. The MOX fuel assemblies (FA) design is basically the same as that of the UO_2 FAs, thereby preserving the thermal-mechanical integrity of the reactor. The plutonium contents (total or fissile) and the burn-up for the MOX FAs are limited such that when they are loaded into the core, they would not compromise the safety margins established as the licensing bases for the reactor. Table 1 shows the experience with MOX-use in LWRs[4]. It includes the licensing limits, expressed in terms of maximum MOX loading in the core, and the maximum concentration of plutonium in the MOX fuel.

TABLE I. EXPERIENCE WITH MOX-USE IN THERMAL REACTORS

Country	**Operating reactors**	**Reactor licensed to use MOX**	**"Moxified" reactors**	**First MOX loading date**[1]	**Licensing limits, %** Max in-core	Max Pu_t conc
Belgium	7	2	2	1995	33	7
France	57	20	17	1987	30	5.3
Germany	21	11	10	1972	50	--
Switzerland	5	4	3	1984	40	--
Japan	52	3	0	pending	33	13

[1]from Booklet: "Cogema: Reprocess to recycle," Feb. 1999.

The discharged spent MOX fuel assemblies are currently not reprocessed. This is because of the low economic incentive for recycling the plutonium from spent MOX fuel. Also, there is a limit to the number of recycles in LWRs because multiple-recycling degrades the fissile plutonium content to a level below that required to maintain the reactivity of the core. The discharged MOX spent fuel would require interim and/or long-term storage. It may eventually be disposed of in a geologic repository.

The global separated plutonium inventory will continue to grow, due to an imbalance between its production and utilization. On separated civil plutonium alone, the total at the end of 1998 was about 200 tonne. It will be in excess of 250 tonne by the end of the decade. Figure 5 predicts the future trends of the global separated civil plutonium inventory.

- *Storage of Reprocessed Uranium and Vitrified HLW*

The separated reprocessed uranium in the Closed Cycle is needed to be stored as currently its recycle to the reactor is not yet economically viable, due to the relatively low price of natural uranium. Also, storage for the vitrified HLW is needed until the final geologic repository is available. A HLW repository do not contain the large quantity of spent nuclear materials and do not require the same decay time for radioactivity to reach to the uranium ore level as in a spent fuel repository. It may have an advantage from the standpoint of acquiring for public acceptance. However, if spent MOX fuel is to be disposed of in a HLW repository, same constraints for a spent-UO_2-fuel repository will be applied. Also, the repository may have to deal with additional considerations because of the higher radioactivity and heat content in the spent MOX fuel.

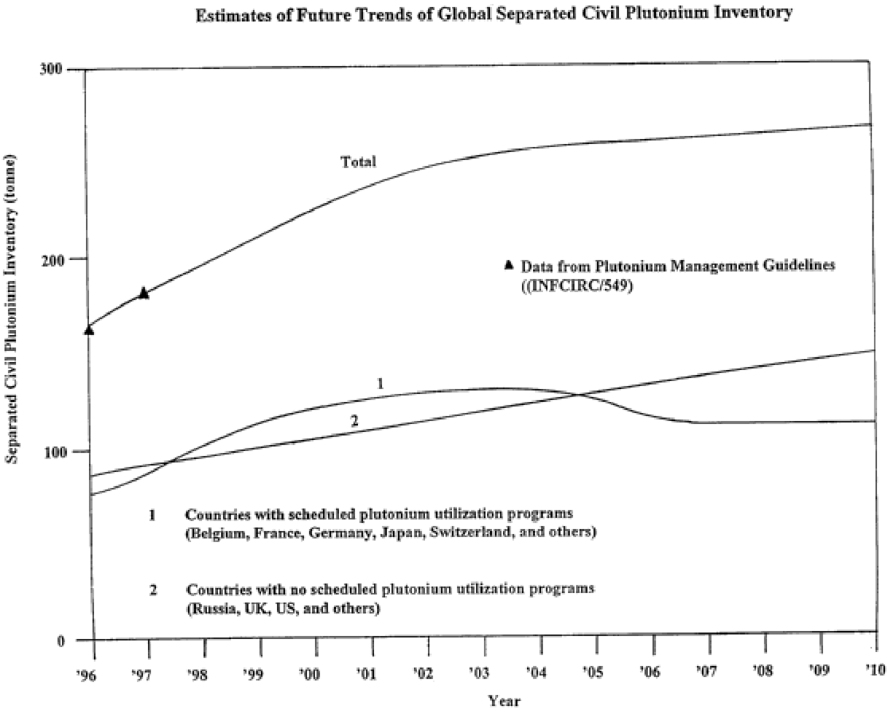

FIG. 5. Estimates of future trends of global separated civil plutonium.

Geologic Repositories

Regardless of back-end fuel-cycle strategies, spent fuel (either MOX or UO_2) and high-level waste (HLW) ultimately would have to be disposed of in a geologic repository. Several countries have embarked on their respective national repository programs, at specific or demonstration sites, e.g., the US, Sweden and Germany, etc. The developed repository technology is site-specific, e.g., the US selected a site in Yucca Mountain with an oxidizing medium, while Sweden's demonstration site in Granite and Germany's in Salt are both in reducing environment. The timing for a repository is country-dependent, e.g., a country with a nuclear phase-out program may need to have a geologic repository sooner than those operating a continuous nuclear program, because of political sentiment. The challenge for repository development is institutional and political, e.g., how to overcome the NIMBY (not-in-my-back-yard)-mentality and obtain public and stakeholders' acceptance and support for a repository site, especially the local public and governments.

For countries with small nuclear power programs and therefore relatively small amounts of spent fuel and radioactive wastes, and for countries with dense population and small geographic areas, consideration of regional and multilateral co-operative arrangements for repository development may be attractive. These countries may have limited potentials to develop their own systems for the back-end fuel cycles. Furthermore, it may not be in the interest of the international community that repositories are spread out all over the world which may constitute a proliferation risk. However, the challenge is again institutional and political, e.g., how to ensure that an attempt for a regional co-operative framework would not jeopardize individual country's national repository program.

3. FACTORS DETERMINING THE BACK-END FUEL-CYCLE STRATEGY

As countries select their back-end fuel-cycle strategy most suitable for their respective nuclear programs, factors determining such strategies will be country-specific. As a result, it would be difficult to quantify one set or even a few sets of factors which will be universal for all countries. However, it may be do-able to identify some important factors and qualitatively categorize them into such groupings as: economical, technical, political, and institutional, for options available in each strategic components evaluated in the previous section. Tables 2, 3 and 4 provided examples of how this can be done for components: (1) spent fuel management, (2) management of separated civil plutonium, and (3) geologic repositories, respectively.

TABLE II. SPENT FUEL MANAGEMENT
Responsible Party: Generators (Private/State Utilities)

Factors Determining options	Options			
		Prolonged storage		
	Reprocessing[1]	AR[2]	AFR[3]	AFR (Multi-national)
Economics				
Life-cycle costs	$R	$X	$Y	$Z
Technical				
Safety: cladding material	rapid deterioration	cladding interity over storage time		
Political:				
National Policy	Y or N[4]	NA[5]	Y or N	NA
Suppliers' consent right	Y or N	NA	NA	Y or N
Institutional				
Contract based	Y	NA	Y	Y
Environmental laws	Y	Y	Y	Y

[1] Include transportation, reprocessing, Pu use/store, storage of rep. U and HLW.
[2] At-reactor storage.
[3] Away-from-reactor storage.

Tables 2 to 4 identify life-cycle costs for options available in these strategic components are important **economic factors**. For the option chosen, specific life-cycle costs associated with the option are needed as inputs to the decision makers.

Technical factors would be different for different options: for example, the need to reprocess the spent fuel may be based strictly on safety ground: that claddings of some spent fuel in wet pools could be deteriorated in such a fast rate that would prohibit any prolonged-storage options. For geologic repository development, R&D efforts are needed to reduce the uncertainty for long-term performance of repository, as well as to meet the licensing requirements.

TABLE III. MANAGEMENT OF SEPARATED CIVIL PLUTONIUM
Responsible Parties: Generators (Private/State Utilities)
Governments (States Holding Stocks)

Factors Determining options	**Options**			
		Prolonged storage		
	Prolonged storage	**MOX[1] in reactors**	**Immobilizations[2]**	**Dirty MOX[3]**
Economics	$R	$X	$Y	$Z
Life-cycle costs				
Technical				
R&D	N		Y	some
Licensing	N		Y	Y
Political				
National Policy	Y or N[4]	Y or N	Y or N	Y or N
Non-proliferation	Y	Y	Y	Y
Institutional				
Contract based	NA[5]	Y or N	Y or N	Yor N
Environmental laws	Y	Y	Y	Y

1. PuO_2-UO_2 fuel.
2. Plutonium is immobilized in ceramic matrix with HLW.
3. A "quick & dirty" fabrication of MOX fuel, to be disposed of with spent UO_2 fuel.
4. Yes or NO.
5. Not Applicable.

TABLE IV. GEOLOGIC REPOSITORY
Responsible Parties: Governments (States Holding Spent Fuel and/or HLW)

Factors Determining options	**Options**	
	National Repository	**Multinational repository**
Economics	$X	$Y
Life-cycle costs		
Technical:		
R&D	Y[1]	Y
Licensing	Y	Y
Political		
National Policy	Y	Y for host countries
International support	NA[2]	Y
Institutional:		
Contract based	NA	Y
Environmental laws	Y	Y
Stakeholders' interetsts	Y	Y

1. Yes.
2. Not applicable.

There are many **political factors** which could affect the back-end fuel-cycle strategies. Only those which are relevant to national policy and supports for the strategic components, and those requiring international co-operation and supports are suggested here. Acquiring the necessary political supports for these components is essential to the successful outcome of the strategy.

Institutional factors include many aspects. Some are legalistic and based on contracts among bilateral or multilateral parties. Different countries may have different environmental laws governing each strategic components, e.g., the Russian Federation currently has environmental laws prohibiting the imports of other countries' spent fuel and/or radioactive wastes. There are also international laws governing the transportation of nuclear materials and/or wastes in international sea-lanes and waters. For geologic repository development, local as well as national stakeholders' interests are needed to be satisfied before such development can be proceded.

4. SUMMARY

In this paper, we briefly evaluated each component of the back-end fuel-cycle strategies and attempted to identify relevant and important factors affecting these strategies. The aim is to provide background materials for the discussion of topical sessions in the Technical Committee Meeting on "**Factors Determining the Long Term Back-End Nuclear Fuel Cycle Strategy and Future Nuclear Systems**." It is recognized that factors determining these back-end fuel-cycle strategies are country-specific. The quantification of the identified factors should be evaluated and provided by individual country selecting the most relevant strategy for its current and future nuclear systems.

SPENT FUEL MANAGEMENT

The back-end of the nuclear fuel cycle: The Argentine view

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Abstract. The strategy of the back-end nuclear fuel cycle for Argentina is presented in this paper. Although one may consider Argentina’s current policy on spent fuel management as “wait and see”, Argentina is continuously directing her efforts towards the development of technology and human resources for the future demand of the closure of the nuclear fuel cycle.

1. INTRODUCTION

Argentine nuclear programme

The Nuclear Program started formally in Argentina with the creation, the 31st of May 1950, of the National Atomic Energy Commission (CNEA), and was always focused on the peaceful applications of nuclear energy

Nuclear power

Argentina has two Nuclear Power Plants (NPP) in operation and one under construction providing 12% of the national electricity production. The first NPP, Atucha-1 (CAN-1), located 100 km away from Buenos Aires, the capital city, started operation in 1974. It is a 350 Mw Pressurize Heavy Water Reactor (PHWR) type with Pressure Vessel, the Natural Uranium Fuel Elements are placed in coolant channels, Heavy Water is employed as a moderator and for refrigeration. The second NPP, Embalse (CNE), a 600 Mw CANDU type plant, located in the province of Cordoba at the central part of the Country, was connected to the grid in 1984. A third plant, Atucha-2 (CAN-2) , a 700 Mw PHWR similar to Atucha-1, is 80 % completed but its construction was interrupted , for economical reasons, due to the deregulation and privatization process in the power sector currently underway in Argentina.

Previous ambitious nuclear deployment plans have been delayed due to more economic exploitation of abundant gas reserves. Nuclear energy, however, remains a valid option for the main political groups recognizing its strategic and ecological importance and is expected to expand in the next decades, accompanying world's growing consciousness of harm climate changes by continued combustion of fossil fuels. The front end of the Fuel Cycle includes seven Uranium Mines along the Argentinean territory, one U02 conversion plant, and a factory for Fuel elements production, including manufacturing of Zircalloy tubes, Uranium pellets, and assembly of fuel bundles for both NPPS. A 200 Ton/year Heavy Water Plant is in operation in the province of Neuquen, in the southern part of the Country. An Uranium Enrichment Plant was designed and built in order to produce slightly enriched fuel elements for the NPPS, as well as enriched fuel for the research and radioisotope production reactors operating in the Country and built for export.

Medical and industrial applications

Radioisotopes for medicine and industry are produced either in the RA-3 a 5 Mw reactor or in a production cyclotron, hot cells and laboratories complete the production and research facilities. Argentina is one of the mayor producers of Cobalt 60, which is obtained from the control rods of the Embalse NPP and is encapsulated for export or for the radiotherapy equipment locally manufactured. Nuclear medicine, employing radioisotopes for diagnosis and treatment, is applied in hospitals and medical centers along the country.

Industrial applications include sealed source for gammagraphy and gauge meters, sterilization and food treatment facilities, a plant for sewage sludge treatment as well as various applications in agriculture and farming.

Research development

The nuclear project is supported by a Research and Development program that takes place mainly in four atomic Centers as well as in several Universities and research institutes . Major R&D programs include innovative reactors, advanced fuel elements, materials, radiochemistry, radioisotopes, waste management, health physics, nuclear safety, etc. Eight research reactors and several laboratories including hot cells and glove boxes facilities support these researches.

2. ACTIVITIES IN THE BACK END OF THE FUEL CYCLE

2.1 Spent fuel management

Argentine policy regarding SF is to consider them as an asset and not as a waste, because of their fissile remnant. The present day scale of NPP park does not justify the deploying of an industrial reprocessing plant. In this sense, we may consider the Argentine policy regarding this issue to what is called "wait and see". But this must not be taken as a passive delay in decision making. Meanwhile we carry on with research and development programs in this fields in order to maintain and provide the necessary technical and human resources to be ready when times arrives.

2.1.1. Atucha NPP spent fuel elements

The NPP CAN-1 began operating in 1974. As we have already mentioned this is a PHWR that uses natural uranium as a fuel. During the first years it operated with a burnup of 6500 Mwd/t and rapidly reached 7000 Mwd/t. The average discharge rate with natural uranium for this burnup is of 1.4 SF/day. The arising spent fuels were stored in the original decaying water pool. Some years later towards the end of the 70's the need to increase the storage capacity of the existing pool was foreseen. This was achieved by using the "double-tier " technique which duplicated the existing capacity. At the same time the possibility of storing these long spent fuels (5.5 m) under the concept of dry storage was considered , a very new concept at that moment. Finally the chosen option was not to innovate and to increase the storage capacity by means of adding a second pool to the existing one. If natural uranium would have been continued to be employed , the present day storage capacity would only be enough to store the SF until 2003. Since 1995 experimentation began in Atucha-1 NPP by using slightly enriched uranium (SEU) as a fuel. The first trials with 0.85% enriched uranium were successful and this lead to implement a series of stages in order to operate with SEU in homogeneous nucleus. It is expected to achieve this goal by mid-2000. The use of SEU, besides achieving a burnup of 11.000 Mwd/t also produces a decrease of SF discharge, from 1.4 to 0.7 SF/day. With this 50% decrease in the arisings and a partial re-racking of the pools, it is expected that all the fuel arising until the EOL will be stored.

2.1.2. Embalse NPP spent fuels

La NPP-EMBALSE began to operate in 1982. It is a PHWR CANDU 600 type, that uses natural uranium as fuel. It produces an average discharge of 13SF/ day. It has a decaying pool that initially had a storage capacity of 44688 SF (30% of the arising until EOL). At the beginning of the 90's the Argentine energetic situation as already mentioned, did not favour a rapid growth of the nuclear park that would justify reprocessing and facing the need to have more storage capacity the decision was taken to search for an economical and safe alternative. This alternative is the dry storage of SF under the concept of silos (ASECQ), an On-Site dry storage system. After 5 or 6 years of cooling in the water pool the SF are loaded in basket under water. Then each basket containing 60 is transferred to a cell where they are dried, after that the basket is covered with a hood and welded hermetically. Afterward it is carried in a transport flask to the dry storage field and loaded into a silo. The silos are cylindrical metal container (tubes) imbedded in concrete that works as a biological shielding. Each silo loads 9 baskets and once completed it becomes hermetic by locking it with welded lid. It is important to remark that the possibility of using SEU as fuel also in this NPP is being studied at present. This would decrease the discharge of SF and the number of silos to be built.

2.1.3. Production and experimental reactors spent fuel

The reactor RA-3 is a pool type reactor used for the production of radio isotopes for medical, industrial use and for testing materials. It began operating using fuel with highly enriched uranium (95%). It worked in this way until 1989 Later on the reactor RA-3 was converted to work with uranium 20% enriched (LEU). The rate of discharge of SF is variable and may reach 20 SF / year as a maximum. Up to now 227 MTR type is the total amount stored of spent fuels. The spent fuels discharged by the RA-3 are stored in small pool next to the reactor and they are rapidly transported to the facility DCMFI which is an AFR on-site storage for MTR spent fuels. This facility consists of a grid of vertical underground channels of SS AISI 316 (2m long and 0.144 m diameter). The whole system is filled with processed and controlled water. Each channel can accommodate 2 SF. The total capacity of the facility is for 396 SF MTR type. The storage facility has a water demineralization system with closed circuit for recirculation, a single track bridge crane and a shielded transportation cask for fuel elements and control rods . In order to remove impurities from the water ion exchange resins are used.

2.2. Reprocessing

The first Argentine reprocessing plant operated successfully in 1969 processing MTR type fuel elements belonging to the core of the RA-1 research reactor It was recuperated 12Kg of enriched uranium 12% and 450mg of plutonium. The uranium was reused to elaborate new MTR fuel elements for the reactor RA-3. A larger facility, with a capacity of 50Kg/day of oxide fuel, is 80% completed but its construction has been interrupted for economic reasons. This plant includes a mechanical head end for the chop-leach process and three decontamination cycles for uranium and plutonium by solvent extraction with TBP. Solid liquid and gaseous storage and treatment facilities are also provided. Research and development work is going on both wet and dry technology. The IMPUREX one cycle solvent extraction process is being developed for its application to the TANDEM fuel cycle. By this synergetic fuel cycle, codecontaminated uranium and plutonium from spent LWR fuel is re-elaborated and fed into HWRS. A laboratory scale facility equipped with mini-mixer-settlers is being installed in the LFR hot cells to test the IMPUREX flow sheet. Experiments are performed on pyroelectrochemical reprocessing including chlorination of oxide fuels and recovery of actinides by electrolysis in fused salts. Partition and transmutation is also subject of research and development. Simulated high level waste is being chlorinated and partitioned by pyro electrochemical means.

2.3. MOX fuel development

Since 1973, when a laboratory conceived for safe manipulation of a few hundred grams of plutonium was built, Argentina was involved in the small scale development of MOX technology. The plutonium laboratory consists of a glove box facility featuring the necessary equipment to prepare MOX fuel rods for experimental irradiations and to carry out studies on preparative processes development and chemical and physical characterization. Irradiation of first prototypes Of $(U,Pu)O_2$ fuels prepared in Argentina began in 1986 in the Petten High Flux Reactor in Holland. Post Irradiation Examinations were performed in KFK Laboratory in Germany and in the Joint Research Laboratory in Petten. In the period 1991-1995, development of new laboratory methods of co-conversion of uranium and plutonium were carried out: reverse strike co- precipitation of $ADU\text{-}Pu(OH)_4$ and direct denitration using microwaves. The reverse strike process produced pellets with a high sintered density, excellent microhomogeneity and good solubility in nitric acid. The microwave direct denitration process was optimized with uranium alone and the conditions to obtain high density pellets, with good microstructure, without using a milling step, have been developed. At present, new experiments are being carried out to improve the reverse strike co-precipitation and direct microwave denitration processes.

2.4. High level waste repository

Although a final decision, regarding reprocessing or direct disposal, will be taken in the future, nevertheless independently of this decision, a final repository for HLW will be needed. The plan is that the spent fuel elements, or the vitrified waste arising from reprocessing process, will be finally disposed off in a deep geological repository. The major problem at present, regarding this option, is the adverse public opinion which impedes the necessary studies to select the siting for the repository. It is expected that a strong communication program, together with negotiations with local government regarding royalties and additional benefits for communities, will allow the selection and characterization of appropriate site by the year 2030. Between 2030 and 2040 a deep geological laboratory will be erected on the selected site, to gather the necessary information for the engineering stage, and in the year 2045 the construction of the HLW repository will start. This is tentative and depends strongly on technical, political and social issues.

3. CONCLUSION

Argentina has an important record on the development of back end of the fuel cycle technology and continues directing her efforts towards the development of the technology and human resources required in case the future demands the closure of the fuel cycle.

The strategy of the long-term back-end nuclear fuel cycle in the Czech Republic

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Abstract. The present status of the strategy of the long-term back-end nuclear fuel cycle in the Czech Republic is briefly outlined in this paper. This strategy is based on the once-through option in the use of the nuclear fuel with subsequent interim storage of the spent fuel and its final disposal as a declared high level waste. However, other technologies for the management of the back-end of the nuclear fuel cycle are not excluded at all. Besides the first already existing and the second interim spent fuel storage facility being sited at Dukovany Nuclear Power Plant, an interim spent fuel storage facility at Temelín Nuclear Power Plant is also under the siting process. To cover the total storing needs a central spent nuclear fuel interim storage facility at Skalka in the Czech-Moravian Highlands is also under consideration. These facilities are or will be equipped with dry-storage containers of cask-type placed in the concrete building and cooled by natural air ventilation. Since 1993 there is a joint effort of several governmental organisations and institutions and private companies to study the scientific, technical and economical possibilities of the construction of the deep geological repository for spent nuclear fuel disposal. A horizontal repository facility with vertical access was selected and a reference project has been accepted. A time horizon for construction in about the year of 2035 was scheduled. The necessary legal and administrative basis of the spent fuel and radioactive waste management was laid down by the law No. 18/1997 (Atomic Act) passed in 1997. This basic law with its implementing regulations fully reflects the internationally accepted principles of the provision of nuclear safety and radiation protection in this respect and it also strongly supports the policy and strategy of the back-end of the nuclear fuel cycle.

1. INTRODUCTION

The first nuclear power reactor on the territory of the Czech Republic was commissioned in 1985 at Dukovany Nuclear Power Plant (NPP). In 1987 when the last 4th unit of this power plant was put into operation, the nuclear share of the electricity generation reached more than 20% and still remains on the same level. The total net output of the four 440 MWe power WWER-type nuclear reactors in the last year was 1 648 MWe. After operating the Temelín NPP with two 1 000 MWe power WWER-type nuclear reactors, which is expected by the very beginning of the first decade of the next millennium, the nuclear share of electricity production in the country will exceed 40%, which means about 2 500 MWe. The electricity production by nuclear reactors is safe and highly environmentally friendly, it generates small amount of wastes in easy controllable volume. Nevertheless, it is not possible to develop nuclear energy programme safely without taking care of the spent nuclear fuel, which represents of about 95% of the total radioactivity at the NPPs.

The interim storage of the spent fuel is relatively simple. This technology has been proved in different alternatives (i.e., at the reactor or away from the reactor in the dozens of cases. The reality that there is not hurry with the construction of final repositories worldwide has many reasons. One main of them is the fact that spent fuel cannot be considered to be common radioactive waste as it may serve as a raw material for new nuclear fuel in the future. There is a lot of solutions to manage this problem suitably. The possibility to defer the final decision, in spite of the meaning of the opponents of the nuclear energy, is a great advantage from the point of view of the safe and environmentally sound management of the back-end of the nuclear fuel cycle.

The present status of the strategy of the long-term back-end nuclear fuel cycle in the Czech Republic is briefly outlined in this paper. This strategy is based on the once-through option in the use of the nuclear fuel with subsequent interim storage of the spent fuel, first, by wet storage at the reactor, following by dry storage far from the reactor and its final disposal as a declared high level waste. A brief outline of the present management of spent fuel and radioactive wastes generated in nuclear power reactors is also given and the necessary legal framework for the ensuring of nuclear and radiation safety of this management is dealt with.

2. SPENT NUCLEAR FUEL MANAGEMENT

Spent fuel arising

Spent fuel arising of Dukovany NPP represents the main source of spent nuclear fuel in the Czech Republic. After year 2000 Temelín NPP will be the other significant source. The total amount of spent fuel to be generated both by Dukovany NPP and Temelín NPP during 30-year operation, estimated to 1500 tons for Dukovany NPP respective 1350 tons for Temelín NPP. Thus the total spent fuel arisings from Czech NPPs will be about 2850 tons.

In September 1999 there were 2342 spent fuel assemblies in the pools of Dukovany NPP and 33 loaded CASTOR-440/84 casks with 2772 spent fuel assemblies.

The fuel cycle back-end concept

- **WWER-440 at Dukovany:** after discharging from reactors, spending from 5 to 6 years cooling period in NPPs at reactor pools, this spent fuel is stored in an interim spent fuel storage facility (ISFSF) Dukovany, which operation started in December 1995. The capacity of ISFSF Dukovany (dry storage, utilizing dual use CASTOR-440/84 casks) was originally limited to 600 metric ton of heavy metal by the political decision of former Czech government. Its storage capacity could cover spent fuel arising from the **operation of Dukovany NPP only until 2005.**

- **WWER-1000 at Temelín**: commissioning of the first unit is officially announced for the year 2000. At reactor pools capacity will be sufficient for about nine year's operation. Approximately in the year 2010, ÈEZ (Czech Electric Power Company) will **have to utilize storage for WWER 1000 spent fuel either in the territory of NPP Temelín** or at the Central ISFSF at the Skalka site. The utility (ÈEZ) management in the year 2000 will make the decision which concept (at Temelín or Skalka sites) will be chosen. The technology of the new (dry cask) interim storage has been already selected, the vendors of the casks not yet.

New storage capacity will be needed after the year 2005 for spent fuel from WWER-440 and after 2010 for WWER-1000. The ČEZ concept of fuel cycle back-end still considers disposal of spent fuel after 2030.

Interim spent fuel storage facility (ISFSF) at Dukovany

ISFSF consists of a light storage building with dual - transport and storage CASTOR - 440/84 casks inside the building with natural air venting. The casks are filled with helium inert gas.

Storage was commissioned in January 1997. Each cask contains 84 spent fuel assemblies (10 tons) and weighs 120 tons. The total capacity of the ISFSF is approximately 600 tons of heavy metal (60 casks).

Recent situation and further plans in the spent fuel management

In March 1997 the Czech government cancelled some parts of the previous Government Resolution No. 213 concerning Dukovany 600 tons storage limit. Following this decision it recommended to prepare next storage capacity at Dukovany and Temelín NPPs sites as priority sites and at Skalka (underground alternative, near the uranium mines Dolní Rožínka in Czech-Moravian Highlands) as stand by site.

A number of negotiations has passed and public discussions, where difficult discussions about the new project have been held. All the participants in the approval procedure (investor, municipal authorities, citizens' groups, and ecological groups) tried to put through their own views and interests. Following the existing Czech legislative principles the Environmental Impact Assessment under the supervision of the Ministry of Environment and the siting approval issued by the State Office for Nuclear Safety (SÚJB) are necessary preconditions for siting permit, which is issued by the local construction authority.

The public hearing is an obligatory part of the environmental impact assessment (EIA) preparation process. On 8th June 1999 there were held public hearing connected with Dukovany storage facility and a week later the other for Skalka storage facility. A new phenomenon occurred. While the local movement against the storage construction was weaker than in previous years, there were active participants from neighbouring Austria expressing their comments and reservations. The majority of reservations were connected with the storage technology specifications. Especially detailed descriptions and licensees for storage casks declared to be utilised have been claimed. Based on the public hearing results there is an intensive exchange of opinions between ČEZ and the Ministry of Environment focused to find a way how to finalise the EIA process. At present the Ministry of the Environment of the Czech Republic, that has to deliver the opinion of the EIA study, has interrupted the procedure and asks for the storage cask specification.

Siting safety analysis reports (SAR) of both Dukovany and Skalka were submitted to the State Office for Nuclear Safety (SÚJB), together with the request for siting approvals, in July 1998. In August 1998, after first evaluations SÚJB requested for some amendments and corrections in the submitted documentation. In January 1999 SARs submitted to the SÚJB were revised. In spite of the fact that respective SAR (content of them is defined by the Atomic Act) were positively evaluated by the Office experts, the Office was obliged to interrupt its licensing process and is waiting for the EIA final results.

Future steps in the spent fuel management

- site permit for both Dukovany and Skalka site in 1999 (or 2000) and following decision of ČEZ to stop or continue in preparation of the Skalka central facility (CF),
- final bidding process for Dukovany interim facility (IF) and/or Skalka CF in 2000,
- construction permit for Dukovany IF and/or Skalka CF in 2000 or 2001,
- construction of the Dukovany IF or Skalka CF in 2002-2004,
- commissioning of the Dukovany IF or Skalka CF in 2004-2005,
- preparatory works for Temelín IF (when Skalka CF is not under construction) in 2002-2008,

- commissioning of the Temelín IF in 2011 (when Skalka CF is not under operation),
- final disposal – after 2030.

3. RADIOACTIVE WASTE MANAGEMENT

Radioactive waste arisings

Similarly as in Central and Eastern European countries, in the Czech Republic the pressurised water WWER-type reactors designed in the former Soviet Union annually generate the following amounts of operational liquid and solid radioactive wastes estimated per one 440 MWe:

- evaporator liquid bottom concentrates 300 m^3
- spent ion exchange resins 22 m^3
- low-level solid wastes 100 m^3
- intermediate-level solid wastes 30 m^3

A main objective in the waste management is to minimise the volume of the radioactive waste. The following processes are considered as radioactive waste management: collection, segregation, treatment, conditioning, storage, transport and disposal. By the disposal (elimination) of the radioactive waste is understood their deposit into repository and/or their storage until spontaneous radioactive decay decreases their activity so that they cease to be radioactive waste. There is no stringent categorisation of the radioactive waste according to the activity. The radioactive wastes are collected in the place of their origin and are segregated according to their physical and chemical properties. They are separated into gaseous, liquid and solid wastes. The liquid wastes are divided into aqueous solutions and organic solutions. The solid wastes are further segregated into combustible and incombustible. The incombustible is separated to compressible and incompressible.

A radioactive waste solidification facility at Dukovany NPP was commissioned in 1996. The liquid concentrates are mixed with bitumen and conditioned into 200-l barrel packages. The volume reduction factor of the waste is about 5 and the average salinity of the bituminous product is 40%. This process is more or less continual. The output of a bituminization facility is about is around 8 barrels per 24 hrs. The cementation technology can also be used. The solid wastes are also treated and conditioned in this on-site facility. The prevailing conditioning technology for solid wastes is the compaction and supercompaction carried out discontinuously. The combustion technology was not introduced until now, as the responsible authority has not granted the license for the operation of the incinerator plant.

Radioactive waste storage

The overall approach to the Soviet design WWER-type reactor powered NPPs was to solve the waste management in the decommissioning stage of these plants. The consequence of this philosophy was the postponement of the considerations about the storage, treatment, conditioning and disposal facilities for the radioactive waste management. Now, several thousands of radioactive liquid are stored in steel tanks on-site of the Dukovany NPP. There are also several hundreds tons of spent sorbents in the storage facility. The liquid wastes are continuously treated by evaporation. The treatment and solidification technology is based on bituminization.

Radioactive waste disposal into the surface repository

For the disposal of radioactive wastes coming from the nuclear power industry the radioactive waste repository at Dukovany NPP was commissioned in 1995 and operated by the NPP and is a property of ČEZ. This so-called regional waste repository is a special near surface (on the ground) facility for the disposal of conditioned operational low- and intermediate-level radioactive waste packages in 200 l zinc-coated steel barrels. Liquid wastes are evaporated for reduction in volume and finally solidified into bitumen. Solid wastes are supercompacted to a minimum volume and placed into the same type of barrels. The repository has 112 disposal basins (compartments) each with capacity of 1500 barrels and is suitable to accept maximum 2.5×10^{12} Bq of β,γ emitting radionuclides. The total capacity of the repository is 55 450 m^3 that means 130 000 barrels. A part of the repository is reserved for non-standard wastes. Until the end of 1998 less than 5 basins were filled up. This repository will also serve for the disposal of the same type of operational radioactive wastes from Temelín NPP. The repository is also intended to accept the decommissioning low- and intermediate-level radioactive wastes from both NPPs in the future.

According to the Atomic Act, this repository will also be transferred from the possession of ČEZ into the possession of the State and operated by the governmental Radioactive Waste Repository Agency until the beginning of the year 2000. On this account, by then the Agency is obliged to apply for the license for this operation granted by the Office according to the law. The transfer of the repository shall be accomplished by a contract between the agency and the former owner of the repository. Disposal of radioactive wastes will also be proceed on a contract basis between the operator of the repository and the individual waste generators.

Radioactive waste disposal into the deep geological repository

A deep geological repository is intended for the disposal of high-level radioactive wastes including the spent nuclear fuel. Since 1993 there is a joint effort of several governmental organisations and institutions and private companies to study the scientific, technical and economical possibilities and the public acceptance of the construction of the deep geological depository. The Nuclear Research Institute has been the programme co-ordinator until 1998. The overall repository programme is divided into 3 main areas:

- development and technical activities,
- survey and verification (testing) of localities,
- demonstration of the safety of disposal systems, study of engineering and geological barriers and materials.

By the year 1998 the co-ordination of the programme of deep geological repository was transferred to the Radioactive Waste Repository Agency (RAWRA). The first, conceptual part of this programme was accomplished and a so-called reference project has been accepted in 1999. The reference project was focused on the

- project of the repository in a hypothetical locality with granitic rock,
- identification and evaluation of the environmental impact of the repository,
- demonstration of the nuclear and radiation safety of the projected repository,
- estimation of the cost and lifetime cycle of the repository,
- demonstration of the feasibility of the deep repository in the Czech Republic.

In 1998 the following projects were included additionally into the deep repository programme:

- site selection,
- activity programme on the testing locality,
- regime measurement on the testing locality,
- programme of testing and experiments in an underground laboratory,
- study of natural analogues,
- programme of experiments for the study of engineering barriers.

A concept of the horizontal repository facility with vertical access was selected, in which eight possible sites in granite geological formation of Melechov massif were taken into consideration. A time horizon for construction of the repository in about the year of 2035 was scheduled.

4. WASTE MANAGEMENT USING TRANSMUTATION TECHNOLOGY

The management of long-lived radioactive waste resulting from the operation of nuclear power reactors, i.e. spent fuel, by fast neutron transmutation (so-called incineration) technology is an alternative to burning it in dedicated nuclear facilities. This technology coupled to a partitioning technology is seen as a complementary approach, which could moderate but not eliminate the disposal problem. Research activities in this respect are conducted in the framework of the consortium of the Czech Technical University Prague, Škoda Nuclear Machinery Plzeň, Institute of Nuclear Physics of the Czech Academy of Science Prague and Nuclear Research Institute Řež.

5. REGULATORY FRAMEWORK OF THE SPENT FUEL AND RADIOACTIVE WASTE MANAGEMENT

Legal and administrative basis

The necessary legal and administrative basis of the spent fuel and radioactive waste management was laid down by the law No. 18/1997 (Atomic Act) passed in 1997. This basic law with its implementing regulations fully reflects the internationally accepted principles of the provision of nuclear safety and radiation protection in this respect and it also strongly supports the policy and strategy of the back-end of the nuclear fuel cycle. This law entrusted the State Office for Nuclear Safety (hereinafter Office)), as a regulatory authority, with the State administration and supervision of all practices resulting in the exposure of workers, public and environment to ionising radiation, including spent fuel and radioactive waste management. Radioactive waste management is not subject to the law No. 125/1997 about wastes (Waste Act) and is solely under the jurisdiction of the Atomic Act. Specification of the radioactive waste management is laid down in the implementing SÚJB decree No. 184/1997.

Basic terms

The law defines the activities related to nuclear energy utilisation to which belong, among other, the management of nuclear materials, including spent nuclear fuel and radioactive waste. Installations as stores, repositories of these items, with the exception of materials containing natural radionuclides exclusively, are defined as nuclear facilities and both their siting and decommissioning is obliged to EIA according to law No. 244/1992 (Environmental Act). Installations for radioactive waste storage are considered to be nuclear facilities if the total activity of these wastes exceeds a limit determined by the implementing regulation (SÚJB decree No. 184/1997). Radioactive waste is defined as a waste material, items or equipment for which no further use is foreseen by their owner with a radionuclide content or surface contamination exceeding limits permitting their release into the environment. According to law radioactive wastes may be temporarily stored or permanently disposed in suitable storage installations or repositories, respectively. Repositories can be either near surface or underground facilities.

Responsibility of the licensee

The responsibility of the safe radioactive waste management rests on the waste generator. A licence granted by the Office is required for individual activities with connection of the use of nuclear energy and ionising radiation, including management and of spent nuclear fuel and radioactive wastes. For siting, construction, commissioning, operation, reconstruction and decommissioning of nuclear installations a licence from the Office is also inevitable. Transportation and import or export of radioactive wastes is liable to licence granted by the Office. The licence from the Office for these activities does not substitute licences or permissions, which are required by other governmental organisations according to special laws or regulations.

Licence process

For all activities in which management of spent nuclear fuel or radioactive wastes are involved, the licensee shall prove that this management is safe and financially is fully covered. Within its administrative procedures, the Office proceeds independently to the procedures of any other administrative body. The applicant is the single part in the process. The licence concurrently expresses an approval required by a special law No. 50/1976 about land-use planning and construction rules (Construction Act). The licensee is obliged to reduce the generation of radioactive wastes and spent nuclear fuel to the minimum necessary level. The licensee of a nuclear installation is obliged to ensure the reserve fund so that financial resources are available, in necessary amount and time, for the needs of preparation and realisation of decommissioning, in accordance with the proposed decommissioning option approved by the Office. Packing assemblies (containers) for transportation or storage of nuclear materials and radionuclide radiation sources may only be produced and used on the basis of the licence issued by the Office. The careful type approval process precedes the licensing procedure. An importer of containers is obliged to import only types approved by the Office.

Until the generator (or the Office) declares spent or irradiated nuclear fuel radioactive waste, its management, apart from requirements ensuing from other provisions of the law, is subject to the same requirements as valid to radioactive wastes. The owner of spent or irradiated fuel is obliged to manage the radioactive waste in such a manner that a possibility of its further conditioning for disposal is not aggravated. The owner of radioactive waste, or any other natural or juristic person (corporate body), acting on behalf of the owner, shall bear all costs of this management from its generation to its disposal, including monitoring of radioactive waste repositories after their closure and the necessary research and development activities.

Responsibility of the State

The Atomic Act charged the State with the liability for safe disposal of all radioactive wastes and spent fuel. The State guarantees, under conditions determined by the law, safe disposal of all radioactive waste, including monitoring and supervision of repositories after their closure. For this purpose the State has established the Radioactive Waste Repository Agency as a responsible organisation in this respect. The Agency's activities encompass, among others, preparation, construction, commissioning, operation and closure of radioactive waste repositories and monitoring of their impact on the environment; radioactive waste management; conditioning of spent or irradiated nuclear fuel into a form suitable for its disposal or further utilisation; provision for and co-ordination of research and development in the field of radioactive waste management; provision of services in the field of radioactive waste management; and monitoring of reserves of licensees for decommissioning of their installations.

The law also introduced a nuclear account for financing both the activities associated with the radioactive waste disposal and with the activity of the Radioactive Waste Repository Agency. Ministry of Finance manages the nuclear account. Waste generators are required to provide financial resources, accounted as costs, to cover expenses for these activities. The financial resources are accumulated on the nuclear account in the form of levies. The amount of levies is determined on the basis of the estimated costs of activities provided by the Agency. The government establishes the amount, method of levies to the nuclear account and details of nuclear account management. On the date the Agency accepts radioactive wastes from their generators, the wastes pass into the ownership of the state. The Authority and the generator endorse, in a written form, the acceptance of the radioactive wastes.

6. CONCLUSIONS

There is a still increasing public awareness of the importance of proper control of the safety and environmental impact of energy production and use. A crucial element of minimising this impact in the production and use of nuclear energy is the safe management and minimisation of all waste arisings from this production. The spent fuel management option determines the environmental impact of this radioactive by-product. In the open or once-through fuel cycle option the full amount of the spent fuel is committed to disposal in an underground geological repository, and all long-lived radionuclides contained in the spent fuel have the theoretical possibility to be dispersed in the natural environment on geological time scales. Therefore, siting a radioactive waste repository refers to the responsible process of selecting a suitable location that must take into consideration various technical and societal factors, including nation-wide public acceptance. The technical factors shall ensure that the risk of the dispersion of radionuclides from the repository into the environment could be maintained on a level as low as reasonable achievable.

We are convinced that our strategy fully reflects the world-wide accepted basic principles of the spent fuel and waste management coming from the UNESCO Declaration on the responsibilities of the present generations towards future generations, as well as the UN Conference on environment and development, i.e. the polluter-pays principle, and the precautionary principle. Moreover, the approach of the Czech Republic to the back-end of the nuclear fuel cycle is flexible, and is open to the other alternatives, including reprocessing and/or transmutation technology of the spent fuel. Both political and economical circumstances strongly support the presented approach to the spent fuel management. It seems that the public acceptance is sufficiently positive towards the presented nuclear fuel cycle strategy in our country.

Spent fuel management perspectives in the Russian Federation

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Abstract. Today nuclear power plants in the Russian Federation Produce about 800 tHM of spent nuclear fuel (SNF) per year. About 100 tHM/year of Russian origin spent fuel is being reprocessed at the PA Mayak RT-1 plant. Most part of the spent fuel is being accumulated in wet storage facilities both at-reactor and away-from-reactor types. Due to economic difficulties construction of the new reprocessing plant near Krasnoyarsk (so called RT-2 plant) was delayed for at least 15 years. Near–term storage of the accumulated spent fuel is now the only option considered for the RBMK spent fuel and part of the VVER-1000 spent fuel. Dry storage of the above spent fuel both in casks and in vaults is considered as a preferable storage option for near-term future. First dry storages will be in operation before the year 2005. Besides part of the VVER-1000 spent fuel will be reprocessed at PA Mayak site after PT-1 reprocessing plant modernization. Increased amounts of submarine spent fuel resulted from disarmament process also will be reprocessed at the plant.Both current and delayed reprocessing will be the main spent fuel management option for the Russia in years to come. This will be accompanied by R&D activities to optimize and simplify chemical reprocessing and waste management operations to improve economics of the reprocessing activity. Partitioning of the waste streams, conditioning and long-term storage of immobilized minor actinides will be the most important part of the work to be performed in laboratory and semi-industrial scale. Future deployment of fast neutron reactors will facilitate both separated plutonium and MA utilization in a form of reactor fuel.

1. ACCUMULATION OF SPENT NUCLEAR FUEL IN RUSSIA

Three types of spent nuclear fuel primarily are to be taken in consideration:

- RBMK spent nuclear fuel;
- VVER-1000 spent nuclear fuel;
- VVER-400 spent nuclear fuel.

In addition BN-600 spent fuel, submarine and research reactor fuel are of concern.

TABLE I. ACCUMULATION OF RUSSIAN ORIGIN SPEND NUCLEAR FUEL

Type of spent nuclear fuel	No of reactors in operation (Russian/Foreign)	Spent fuel discharge MT HM/year (Russian/Foreign)	Spent fuel accumulated in Russia, MT HM (1998)
RBMK	11/3	~ 550/n.a.	7800
VVER-1000	7 / 13	~ 150 / ~280	Over 2500
VVER-440	6 / 18	~ 80 / ~240	-

- All RBMK spent fuel is now stored at reactor sites in wet AR and AFR storage facilities. Estimated amount of RBMK fuel to be discharged during reactors life-time is around 22500 MT HM.
- VVER-1000 spent fuel is stored up to 5 years at reactor pools and after that period is transported to RT- 2 plant wet storage facility (Krasnoyarsk). The RT-2 storage capacity is 6000 MT HM.
- VVER-440 spent fuel from Russian, Ukrainian and some European countries is reprocessed at RT-1 plant (South Ural). Most part of the European fuel is stored at NPP sites. Starting from 1978 about 2400 MT HM was reprocessed at RT-1 plant. Currently PT-1 plant also reprocess uranium spent fuel from BN-600 reactor and from submarine/icebreaker reactors.

2. OPTIONS FOR SPENT FUEL MANAGEMENT IN RUSSIA

Taking into account operation of NPPs in Russia the near-term perspective (up to the year 2010) will be connected with the following problems:

- Current storage capacities for RBMK and VVER-1000 spent fuel is being exhausted;
- Only a few new VVER-1000 units will be deployed, while 4 VVER-440 units will be at the end of operation life;
- PT-1 plant reprocessing capacity is being used on less then 30%;
- Increased amount of submarine spent fuel will be released during disarmament process. This fuel will be stored and to be reprocessed.
- Waste management system of the RT-1 reprocessing plant need to be improved to meet regulations and requirements for final disposal and shipment of the immobilized radioactive wastes.

To meet these challenges Minatom experts are now evaluating different possibilities. Draft of the Minatom's Concept for Spent Fuel Management includes possible delay in reprocessing of accumulated VVER-1000 spent fuel for at least 20 years and dry storage option for both VVER-1000 and RBMK fuel.

Delay in reprocessing will allow MINATOM:

- to accumulate funds for future investments in construction of reprocessing and final disposal facilities;
- to develop and test in laboratory & semi-industrial scale advanced reprocessing technology (including dry technology) with improved waste management scheme for long -lived radionuclides.

Meanwhile in years to come there is need for investments to refurbish PT-1 plant with the primary aim to shift reprocessing activity from mainly VVER-440 fuel to VVER-1000 fuel with the reprocessing rate of ~300 MT/year of and optionally to reprocess PWR-type fuel. This will help to improve economics of the plant and PA "Mayak" and accumulate the necessary investments.

As a part of RT-1 plant refurbishment construction of 2 new ceramic melters for vitrification of liquid waste is under way. The new facility will start operation early 2000 year. Additional bench-scale facility with cold crucible melter will be in operation for the immobilization of long-lived waste and wastes with complicated chemical composition.

Future spent fuel management in Russia include:

- refurbishment of RT-1 plant for reprocessing VVER-440 fuel and part of discharged VVER-1000 spent fuel, increase in reprocessing rate for the submarine spent fuel (currently ~ 10 MT/year);
- delay in reprocessing of the already accumulated VVER-1000 spent fuel;
- long-term dry storage option for the accumulated VVER-1000 and RBMK fuel;
- geologic storage/disposal of some amount of spent fuel which can not be reprocessed.

Preliminary results of feasibility study show that construction cost for the dry storage facility at RT-2 site close to 500 million USD for ~30 000 MT HM capacity. The cost estimates include construction of transport containers and concrete vaults for dry storage of ~22 500 MT of RBMK fuel (in halves of subassemblies) and ~8 000 MT of VVER-1000 fuel.

Final decision for processing or final disposal of RBMK spent fuel will depend on such factors as:

- economic considerations (storage, reprocessing and/or disposal cost),
- long-lived waste management option including feasibility of the environmentally safe long-term storage / final disposal of spent fuel;
- development of Russian nuclear legislation/regulation;
- public and local authorities involvement

One of important issues is interconnection of spent fuel management program with future steady development or decrease in the nuclear power in the Russian Federation.

Taking into account Russian experience in spent fuel storage and reprocessing, current intermediate wet storage and future dry storage availability, there are plans for receiving foreign fuel (up to 10 000 MT) for its long-term storage on Russian territory with future optional reprocessing or return back. These options might include reactor burning or incapsulation of minor actinides and some of the long-lived nuclides from this fuel. We believe that new developments in quality of spent fuel service will facilitate construction of new NPPs in countries which have no possibilities for final disposal of long-lived wastes.

3. RECOVERED NUCLEAR MATERIALS AND RADIOACTIVE WASTE MANAGEMENT

Reprocessed uranium from RT-1 plant is currently used for fresh fuel manufacture. Amount of recovered and stored civil plutonium is 30,5 MT (mid of 1999). This material was considered as future fuel for BN-type reactors - construction of the first BN-800 unit is scheduled for the year 2010 at Beloyarskaya NPP. Reactor core for BN-800 was designed for the RT-1 reprocessed plutonium utilization.

Additionally to the civil Pu stock up to 50 MT of excess weapon-origin plutonium have to be utilized in Russian reactors according to bilateral Russia-US initiatives. Minatom consider W-Pu utilization in commercial reactors as a first step to future industrial-scale use of Pu in a form of reactor fuel.

Well known problem of long-lived radionuclides such as minor actinides (MA), Tc-99 and I-129 also is under consideration.

Different extraction technologies for partitioning of radionuclides are considered and tested for use at RT-1 and future RT-2 plants.

As an example cobalt dicarbollide extraction technology was already tested in pilot scale for Cs and Sr removal from RT-1 waste streams. Later on MA oxide concentrate was obtained from the raffinate of this process.

Electrochemical reprocessing of fast reactor fuel coupled with vibropacked MOX technology is under development and bench-scale testing in RIAR (Dimitrovgrad). This technology can also be used for development of proliferation-resistant closed nuclear fuel cycle.

Final stage of any fuel cycle connected with waste disposition. Major part of low- level waste is now stored on site at NPPs or fuel cycle facilities. Near surface disposal for solid and solidified waste, as well as deep well injection of short-lived liquid wastes are currently in use in Russia.

For geological disposal of the radioactive waste and unreprocessed fuel several sites are considered. The potential candidates are:

- PO “MAYAK” site (porfirite) for vitrified wastes from RT-1 operation;
- Nijnekanskiy rock massif (granite) for future RT-2 plant immobilized wastes and/or spent fuel;
- Novaya Zemlya island (permafrost);
- East Siberia, Bilibino NPP (permafrost) for spent fuel and NPP operational waste.

Final selection will depend on such factors as transportation routes for waste delivery to the disposal site, nature of the waste and the isolation properties of host rock.

Russian nuclear power and industry for nuclear fuel cycle now are in a transition period. From former socialist economy with the governmental funds available for the nuclear facilities construction and operation today Russian nuclear industry shifts to the new market economy. This transition includes new “rules of the game”, new legislation and regulation procedures, changes in business relations between enterprises and companies. All that mean that necessary steps are to be taken in current complicated situation for future development of Russian nuclear power and fuel cycle industry.

Activity of VVER-440 spent fuel for long period

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Abstract. In this paper activity, neutron and gamma source strength of 1kg VVER-440 spent fuel (enrichment 3.6% burnup 40 MWD/kg U) for long period (up to 100 000 years) published. In the first period (up to 100 years) dominate activity from fission products. After 100 years main sources are actinides (mainly Pu). Gamma is produced first 100 years from fission products and later from actinides. Neutrons (a, n) and spontaneous fission) are from various isotopes Pu, Am and Cm.

1. INTRODUCTION

For safety store of spent fuel is necessary to know its activity, neutron and gamma source strength. In Slovakia current typical VVER-440 spend fuel has enrichment 3.6% U235 and burnup 40 MWd/kg U.

Results

It was the code system SCALE 4.3 used. All source strengths were calculated with ORIGENS were calculated sources (activity photons and neutrons) for period 10-100 000 years, which are important for store. The basic unit is 1 kg U. Results are in Tables I-IV and Figs 1-5 shown.

Activity

Up to 100 years dominate activity from fission products, later from actinides (in period 250-1000 years Am, later Pu). From fission products have main activity Cs137 (first 250 years), Sm151 (around 500 years) and Tc99 (for period 1 000-100 000 years).

Gamma

Up to 100 years are main sources of photons fission products, later actinides. Around 25 000 years approximately 30% photons are from light elements. Normalized spectrum is in Fig.1 shown.

Neutrons

Main source of neutron is spontaneous fission, only for Am241 is more strength (a, n) reaction. In the first period (up to 50 years) are more than 90% neutrons from Cm244. For period 100-10 000 years dominate, Pu240, Am241 and Cm246 (each 20-45%). After 25 000 years are more than 85% neutrons from Pu242. Normalized spectrum is in Fig. 2 shown.

2. CONCLUSION

In the period up to 100 years neutrons and photons are from fission products (more than 80%). After 250 years are more than 80% from actinides. For long period is necessary or to safety save actinides or destroy them by transmutation.

REFERENCE

[1] SCALE 4.4, Oak Ridge National Laboratory, 1999.

TABLE I. ACTIVITY [Bq] OF 1 KG SPENT FUEL (3.6 %, 40 MWd/kg)

time	total		FP		actinides					
year					total		Pu		Am	
10	1.91E+13	100%	1.43E+13	75%	4.81E+12	25%	4.51E+12	24%	1.02E+11	1 %
25	1.19E+13	100%	9.32E+12	78%	2.56E+12	21%	2.28E+12	19%	1.72E+11	1 %
50	6.16E+12	100%	5.11E+12	83%	1.05E+12	17%	7.92E+11	13%	2.13E+11	3%
100	1.97E+12	100%	1.56E+12	79%	4.03E+11	20%	1.83E+11	9%	2.14E+11	11%
250	2.88E+11	100%	4.85E+10	17%	2.38E+11	83%	6.59E+10	23%	1.70E+11	59%
500	1.60E+11	100%	1.39E+09	1 %	1.58E+11	99%	4.18E+10	26%	1.14E+11	71%
1000	9.18E+10	100%	8.21E+08	1 %	9.03E+10	98%	3.68E+10	40%	5.18E+10	56%
2500	4.18E+10	100%	8.07E+08	2%	4.03E+10	97%	3.30E+10	79%	5.81E+09	14%
5000	3.17E+10	100%	7.99E+08	3%	3.03E+10	96%	2.79E+10	88%	1.08E+09	3%
10000	2.34E+10	100%	7.88E+08	3%	2.21E+10	94%	2.05E+10	88%	6.25E+08	3%
25000	1.17E+10	100%	7.55E+08	6%	1.06E+10	91%	9.88E+09	84%	1.53E+08	1 %
50000	5.67E+09	100%	6.99E+08	12%	4.81E+09	85%	4.18E+09	74%	1.47E+07	0%
75000	3.48E+09	100%	6.51E+08	19%	2.75E+09	79%	2.05E+09	59%	1.41E+06	0%
100000	2.51E+09	100%	1 6.07E+08	24%	1.85E+09	74%	1.04E+09	41%	1.37E+05	0%

TABLE II. ACTIVITY [Bq] OF 1KG SPEND FUEL (ONLY IMPORTANT FISSION PRODUCTS) (3.6%, 40 MWd/kg)

Year	Sr90		Zr93		Tc99		Pd107		Sn126	
10	2.65E+12	19%	5.44E+07	0%	6.14E+08	0%	5.96E+06	0%	2.72E+07	0%
25	1.83E+12	20%	5.44E+07	0%	6.14E+08	0%	5.96E+06	0%	2.72E+07	0%
50	9.92E+11	19%	5.44E+07	0%	6.14E+08	0%	5.96E+06	0%	2.72E+07	0%
100	2.89E+11	19%	5.44E+07	0%	6.10E+08	0%	5.96E+06	0%	2.72E+07	0%
250	7.18E+09	15%	5.44E+07	0%	6.10E+08	1%	5.96E+06	0%	2.71E+07	0%
500	1.52E+07	1%	5.44E+07	4%	6.10E+08	44%	5.96E+06	0%	2.71E+07	2%
1000	6.84E+01	0%	5.44E+07	7%	6.10E+08	74%	5.96E+06	1%	2.70E+07	3%
2500	6.22E-15	0%	5.44E+07	7%	6.07E+08	75%	5.96E+06	1%	2.67E+07	3%
5000	0	0%	5.40E+07	7%	6.03E+08	75%	5.96E+06	1%	2.62E+07	3%
10000	0	0%	5.40E+07	7%	5.92E+08	75%	5.96E+06	1%	2.53E+07	3%
25000	0	0%	5.37E+07	7%	5.62E+08	75%	5.96E+06	1%	2.28E+07	3%
50000	0	0%	5.33E+07	8%	5.18E+08	74%	5.92E+06	1%	1.92E+07	3%
75000	0	0%	5.25E+07	8%	4.77E+08	73%	5.92E+06	1%	1.62E+07	2%
100000	0	0%	5.18E+07	9%	4.40E+08	73%	5.88E+06	1%	1.36E+07	2%

year	I129		Cs135		Cs137		Sm151	
10	1.53E+06	0%	2.34E+07	0%	4.00E+12	28%	1.98E+10	0%
25	1.53E+06	0%	2.34E+07	0%	2.82E+12	30%	1.76E+10	0%
50	1.53E+06	0%	2.34E+07	0%	1.58E+12	31%	1.45E+10	0%
100	1.53E+06	0%	2.34E+07	0%	4.99E+11	32%	9.92E+09	1%
250	1.53E+06	0%	2.34E+07	0%	1.56E+10	32%	3.12E+09	6%
500	1.53E+06	0%	2.34E+07	2%	4.85E+07	3%	4.55E+08	33%
1000	1.53E+06	0%	2.34E+07	3%	4.63E+02	0%	9.66E+06	1%
2500	1.53E+06	0%	2.34E+07	3%	4.11E-13	0%	9.29E+01	0%
5000	1.53E+06	0%	2.33E+07	3%	0	0%	4.03E-07	0%
10000	1.53E+06	0%	2.33E+07	3%	0	0%	0	0%
25000	1.53E+06	0%	2.32E+07	3%	0	0%	0	0%
50000	1.53E+06	0%	2.31E+07	3%	0	0%	0	0%
75000	1.53E+06	0%	2.29E+07	4%	0	0%	0	0%
100000	1.52E+06	0%	2.27E+07	4%	0	0%	0	0%

TABLE III. GAMMA SOURCE STRENGTH [photons/s] of 1kg Spend Fuel (3.6%, 40MWd/kg)

year	total		LE		FP		actinides	
10	9.12E+12	100%	6.45E+09	0%	8.98E+12	99%	1.30E+11	1%
25	5.74E+12	100%	3.69E+09	0%	5.58E+12	97%	1.59E+11	3%
50	3.19E+12	100%	2.29E+09	0%	3.02E+12	95%	1.70E+11	5%
100	1.08E+12	100%	1.77E+09	0%	9.17E+11	85%	1.58E+11	15%
250	1.48E+11	100%	1.68E+09	1%	2.65E+10	18%	1.20E+11	81%
500	8.24E+10	100%	1.67E+09	2%	2.63E+08	0%	8.05E+10	98%
1000	4.20E+10	100%	1.64E+09	4%	1.86E+08	0%	4.02E+10	96%
2500	1.18E+10	100%	1.55E+09	13%	1.84E+08	2%	1.01E+10	85%
5000	7.50E+09	100%	1.43E+09	19%	1.82E+08	2%	5.89E+09	79%
10000	5.26E+09	100%	1.20E+09	23%	1.77E+08	3%	3.88E+09	74%
25000	2.30E+09	100%	7.22E+08	31%	1.62E+08	7%	1.42E+09	62%
50000	1.10E+09	100%	3.09E+08	28%	1.41E+08	13%	6.48E+08	59%
75000	7.88E+08	100%	1.33E+08	17%	1.23E+08	16%	5.32E+08	68%
100000	6.78E+08	100%	5.78E+07	9%	1.08e+08	16%	5.12E+08	76%

TABLE IV. NEUTRON SOURCE STRENGTH [n/s] of 1kg SPENT FUEL (3.6%, 40MWd/kg)

year	total		Pu240		Pu242		Am241		Cm244		Cm246	
10	E+05	100%	3.22E+03	0%	1.32E+03	0%	2.30E+03	0%	6.46E+05	97%	5.30E+03	1%
25	E+05	100%	3.24E+03	1%	1.32E+03	0%	3.94E+03	1%	3.64E+05	95%	5.29E+03	1%
50	E+05	100%	2.34E+03	2%	1.32E+03	1%	4.89E+03	3%	1.40E+05	88%	5.27E+03	3%
100	E+04	100%	2.34E+03	9%	1.32E+03	3%	4.93E+03	13%	2.06E+04	54%	5.23E+03	14%
250	E+04	100%	2.34E+03	22%	1.32E+03	9%	3.91E+03	26%	6.58E+01	0%	5.12E+03	35%
500	E+04	100%	2.34E+03	25%	1.31E+03	10%	2.62E+03	21%	4.57E-03	0%	4.93E+03	39%
1000	E+04	100%	2.34E+03	28%	1.31E+03	13%	1.18E+03	11%	0	0%	4.58E+03	44%
2500	E+03	100%	2.34E+03	32%	1.31E+03	16%	1.07E+02	1%	0	0%	3.68E+03	46%
5000	E+03	100%	2.33E+03	32%	1.30E+03	21%	2.34E-00	0%	0	0%	2.55E+03	41%
10000	E+03	100%	2.33E+03	29%	1.29E+03	32%	2.62E-01	0%	0	0%	1.23E+03	31%
25000	E+03	100%	2.32E+02	13%	1.26E+03	68%	7.70E-02	0%	0	0%	1.36E+02	7%
50000	E+03	100%	2.31E+01	1%	1.20E+03	87%	1.00E-02	0%	0	0%	3.49E-00	0%
75000	E+03	100%	2.29E-00	0%	1.15E+03	91%	1.30E-03	0%	0	0%	8.96E-02	0%
100000	E+03	100%	2.27E-02	0%	1.09E+03	91%	1.70E-04	0%	0	0%	2.30E-03	0%

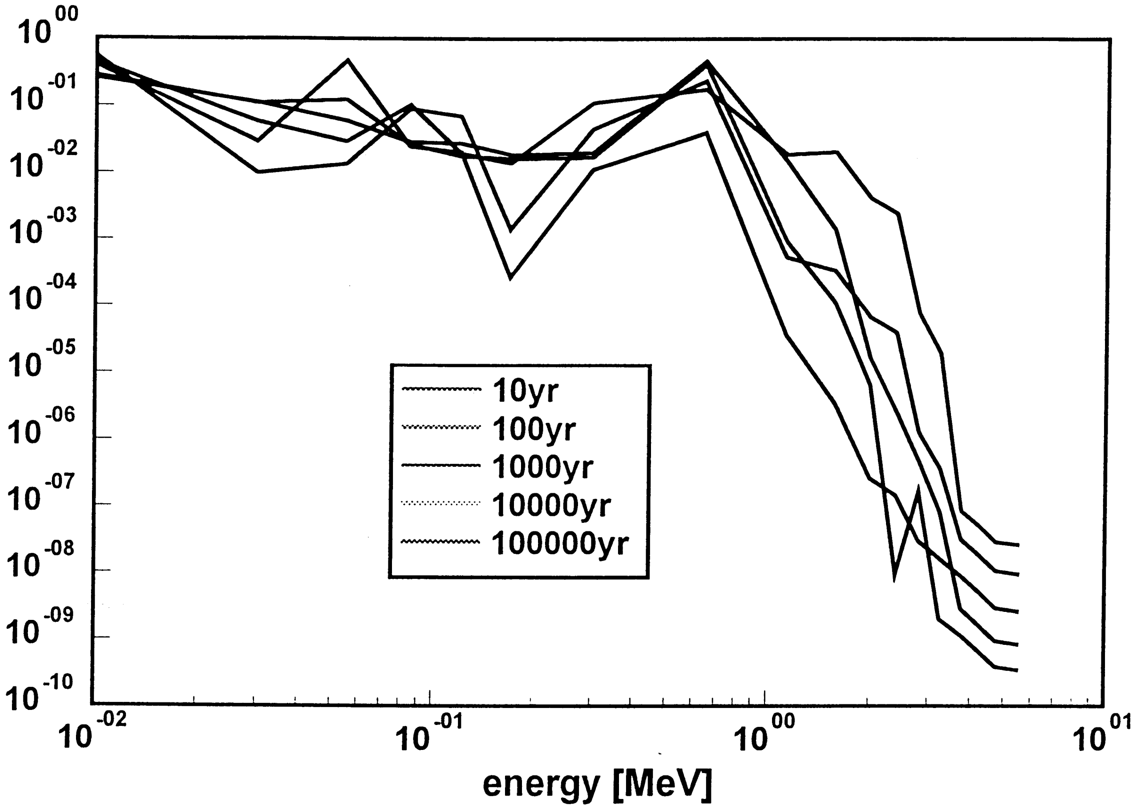

FIG. 1. Gamma spectrum.

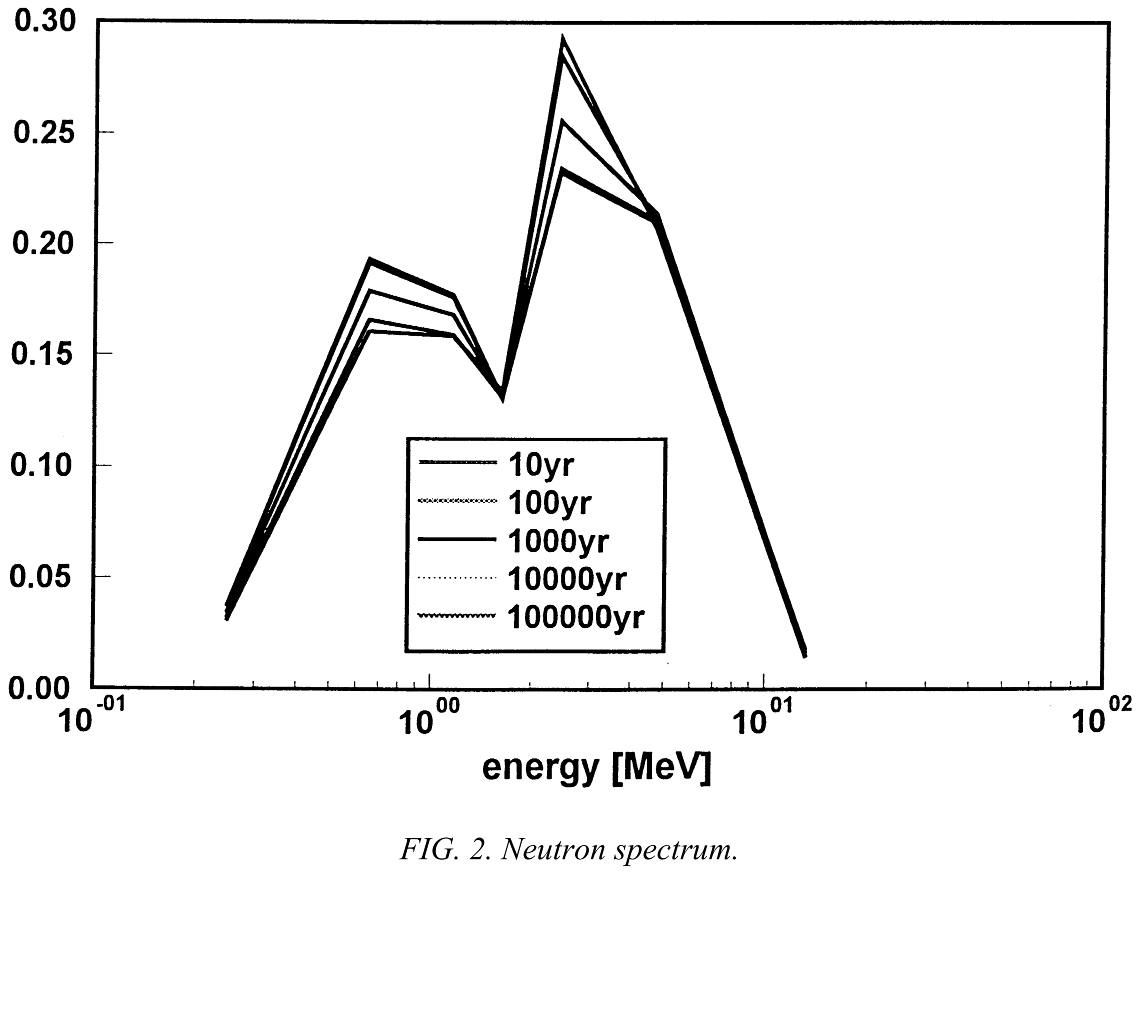

FIG. 2. Neutron spectrum.

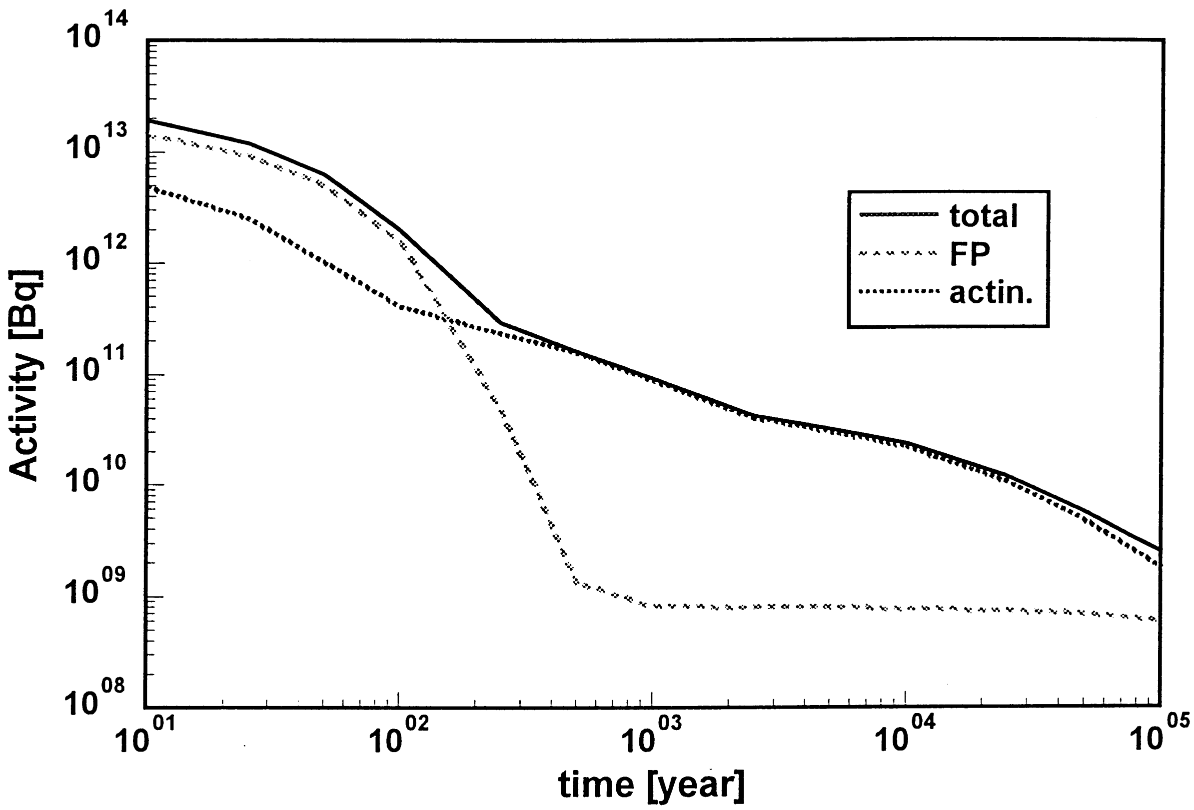

FIG 3. Activity [Bq] of 1kg spend fuel (3.6%, 40 MWd/kg).

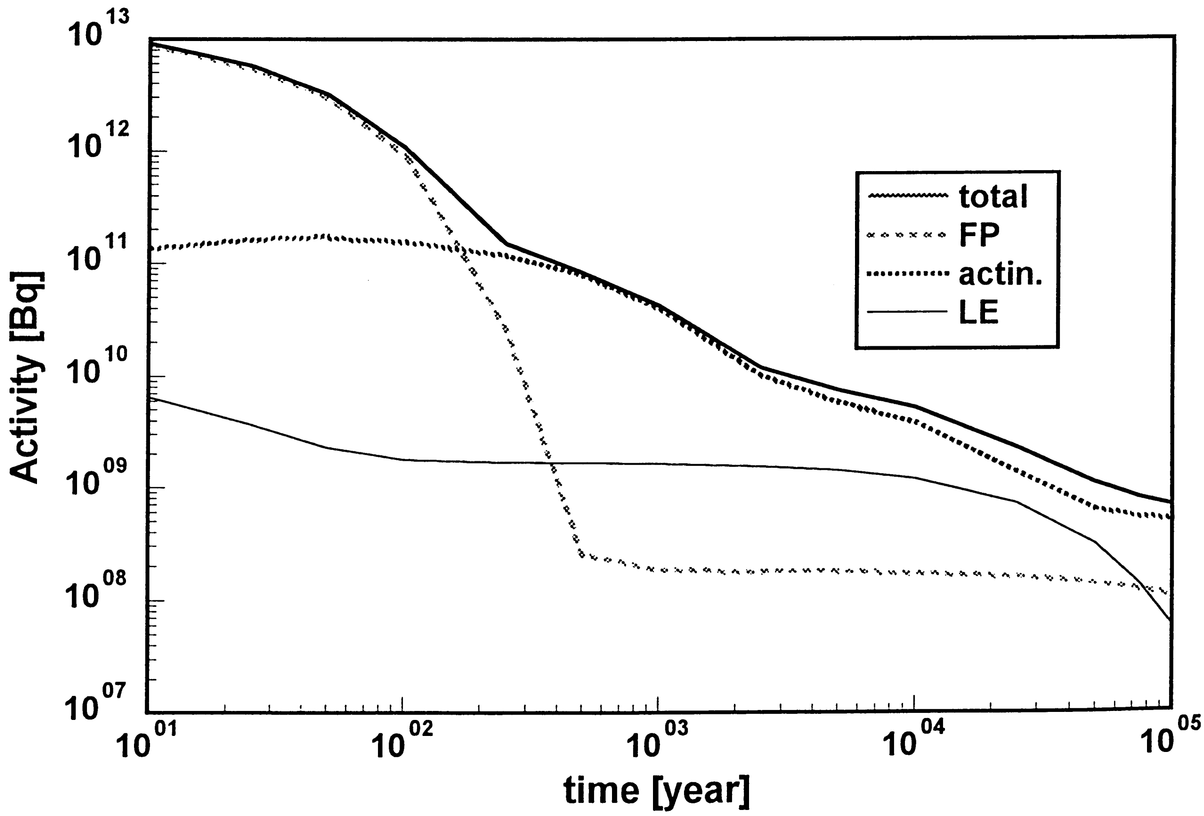

FIG. 4. Gamma source strength [phons/s] of 1 kg spend fuel (3.6%, 40 MWd/kg).

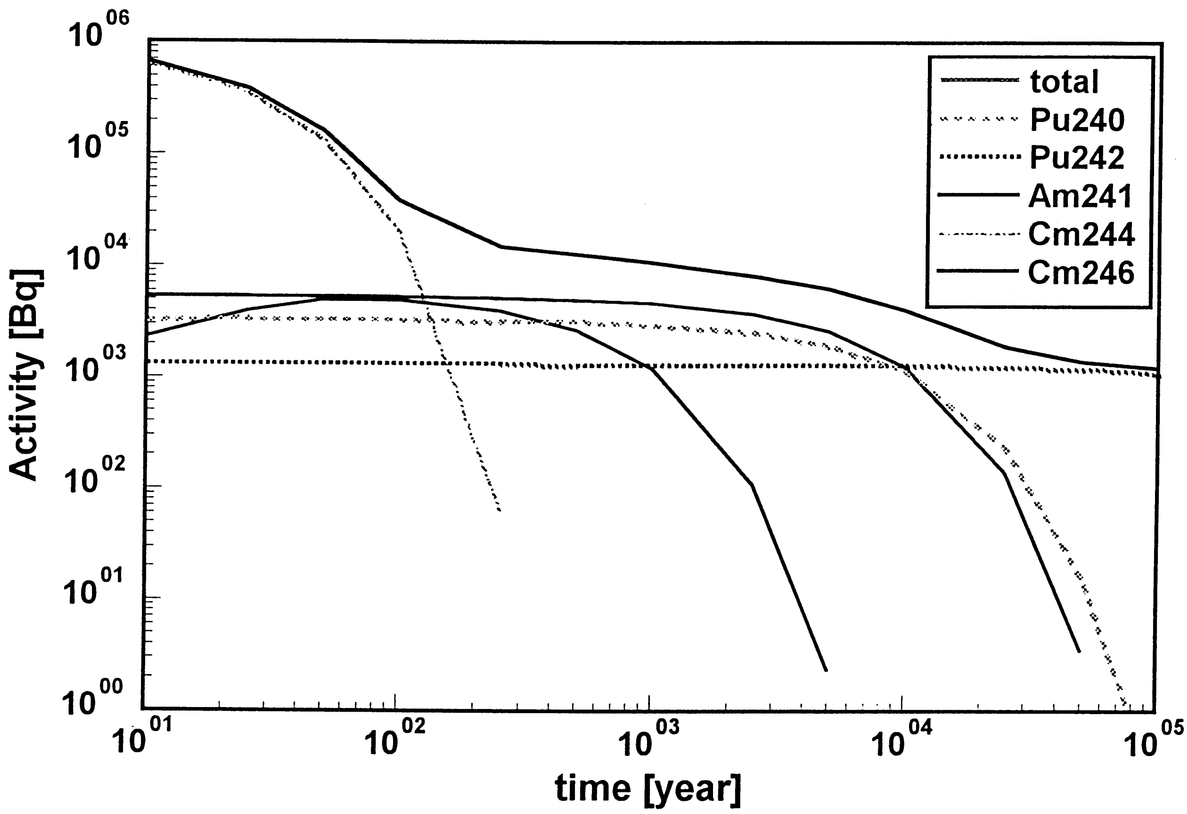

FIG. 5. Neutron source strength [n/s] of 1 kg spent fuel (3.6%, 40 MWd/kg).

Spent fuel management in Spain

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Abstract. The spent fuel management strategy in Spain is presented. The strategy includes temporary solutions and plans for final disposal. The need for R&D including partitioning and transmutation, as well as the financial constraints are also addressed.

1. INTRODUCTION

On July 31st 1999, the Spanish Government approved the 5th General Radioactive Waste Management Plan (GRWP), which sets the new strategies with regards to the Spent Fuel and other High Level Wastes.

The GRWP has been drawn in accordance with the contents of article 4 of the Royal Decree 1622/1984 of July 4th, authorising the constitution of the Spanish Radioactive Waste Management Company, ENRESA. GRWP includes a revision of all the necessary activities and technical solutions applicable throughout the period of activity of the radioactive wastes and covers an updated economic-financial study of the costs of such activities.

As it is known, there are 9 LWR units in operation with a total output of over 7.6 GW. In addition a Natural Uranium, Graphite moderated and Gas cooled, reactor (NUGG) of French design was in operation from early 70's to 1989.

The GRWP forecast that some 6 750tU from LWR will have to be managed, having considered, for planning purposes, that the expected operation life will be 40 years, being the open cycle the primary back-end management system, even through it does not exclude the possibility of a closed cycle. It also considers the necessity of managing conditioned radioactive wastes that will have to return back to Spain, following the reprocessing of the NUGG fuel in France.

As of December 31st 1998, the tonnage already stored at the different NPP's was 2 249t U.

The degree of occupation of the storage pools varies, at that date, from 32% (Vandellos II) to 69% (Trillo). These relatively low percentages of occupation, with some NPP's commissioned in late sixties and early seventies (José Cabrera and Santa María de Garoña) is due to the fact that in all 9 LWR units reracking of the fuel ponds was undertaken. In such a circumstances, only Trillo NPP will need additional storage capacity during the first decade of 21st Century, while the others will require new storage capacity beyond 2010 to up 2022. Special case are the two oldest units where reracking has allowed not to saturate their ponds before the 40 years period expected life.

Table I resumes the spent fuel ponds situation as of December 31st 1998.

2. TEMPORARY SOLUTIONS

In view of the above, a distinction may be made between two phases in analysing possible management alternatives: an initial phase, which would cover up to the year 2010 during which a specific problem would be the case of Trillo NPP and a second phase, from that year onwards, which would include in addition to the tonnage in excess of ponds capacities, the expected return of reprocessing wastes and also the management of the fuel of the first LWR plants to be decommissioned.

TABLE I

Unit	t U	Degree of occupation (%)	Forecast date of saturation
José Cabrera	55	43	---
Sta. María de Garoña	229	58	---
Almaraz 1	318	42	2020
Almaraz 2	314	41	2022
Ascó 1	297	51	2013
Ascó 2	258	44	2016
Cofrentes	364	50	2014
Vandellós 2	210	32	2021
Trillo	204	69	2003
TOTAL	**2249**		

a) Phase 1

A specific temporary storage facility is to be built at Trillo NPP site, which will house the spent fuel in dual purposed metallic casks, which should be available by 2002. This facility has already been designed and the cask has been licensed by the Spanish Authorities.

b) Phase 2

The strategy for this phase consists of having available a centralised temporary storage facility by the year 2010, in order to provide a solution to the problem of the vitrified wastes to be returned. This installation will also be required to store wastes other than spent fuel and HLW which cannot be stored at Shallow Land Low Level disposal facility of El Cabril, as well as, the spent fuel itself as the storage capacity of the LWR's ponds decreases or their dismantling is addressed.

It is considered essential that decisions be taken as regards the location of this Centralised Temporary Storage facility with sufficient time to guaranty its start up in the year 2010. Bringing forward this date would provide a better capacity to respond to any eventuality that might occur in the near future.

Although this strategy is considered to be basic in might be complemented with the construction of individual temporary storage facilities at certain of the NPP's or with another centralised facility serving various such plants.

An alternative to the above management of spent fuel would be to send it abroad for reprocessing, with the disadvantage that this would be economically very costly and that there would be a need for subsequent management of wastes. Consequently a CTS would still be necessary.

3. FINAL DISPOSAL

The strategy adopted to date for the final management of SF and HLW has been based on ensuring the availability of the scientific and technological know-how and capacity required for final disposal in deep geological formations (DGD).

The work performed has led to the following:

- The identification of a large number of zones in the national geography which, from the geological point of view, might be valid pending "in situ" confirmation.

- Significant progress in the generic and specific designs of the disposal systems in each geological medium studied (granite, salt and clays), as well as, in the development and preliminary applications of tools and methodologies required for the assessment of long term behaviour and safety.

- Partial development, through the R+D plans, of basic technologies for site characterisation and modelling of the most relevant processes taking place in the different confining barriers, for applications in safety assessment.

In view of generalised delay affecting the programmes in other countries, the uncertainties regarding definitive solutions and the availability of temporary solutions, decisions regarding a final solution will be postponed until year 2010.

This will allow studies to be made on Separation and Transmutation with the aim of having two lines of progress, DGP and S+P, in such away that is will be possible to provide the Government with the information required for performance, by the year 2010. All the above will be in keeping close relationships with the international programmes, and more particularly of than of those of the European Union, and the initiatives of the different countries channelled through NEA and IAEA.

This new approach needs to reorientate past activities characterised by the postponement of definitive decisions until next decade. These future courses of action will be orientated on the basis of the following considerations:

- The activities relating to the focusing and solution of specific sites are to be suspended. The work will be limited to maintaining the existing know-how and to ensuring its value.

- The safety assessment capabilities should be maintained in the future through exercises incorporating the experimental data and model of the research groups.

Research and Development

Research plays an important part on the waste management actions. Different R+D plans have included partial development of the technologies required, the geophysical, hydrogeological and hydrogeochemical of the geological barrier site.

The 4th R+D Plan, which starts in 1999 and will last up 2003, had to revise its goals and adequate is achievement in accordance with the new strategies settled above, participating actively in the 5th EU Framework Programme.

In that a sense, regarding S.F. the goal of this Plan is to deep in the following fields:

- Basic Technologies:

  It is intended the follow up through dedicated specific research groups the characterisation of fuel and actinide retention, as well as the radionuclide behaviour in the biosphere.

- Partitioning and Transmutation:

  ENRESA will initiate a R+D programme in close collaboration with CIEMAT (National Research Institution on Nuclear Energy). This research programme should be closely linked to those of other European countries, and will deal specially on Hydro and Pyrometalurgical partitioning as well as studies on Accelerator Driven Systems.

- Geological Disposal:

  Natural and artificial confinement will be followed.

4. ECONOMIC AND FINANCIAL ASPECTS

The financing of the costs of the spent fuel management by the entire electricity industry is included within a percentage fee based on the billing of electricity sales. This levy covers all ENRESA activities related to NPP's.

For the purpose of drawing up the GRWP and performing the corresponding economic calculations, it is necessary to establish a series of hypothesis, any variation to which affect the results obtained.

The main hypothesis are:

- Installed power 7.6 GWe.
- Average operating value at 100% output: 7000 hours/year.
- Main economical data:
  - Inflation rate 2%.
  - Discount rate 2,5%.
  - Average increase in electricity demand: 3%.

The cost estimate of all the activities included in the GRWP will be 1.63 TPts'99 (10 G Euro), 57% of which will be dedicated to Spent Fuel and High Level Waste Management.

Taking into account the schedule for construction, operation and dismantling of the projected facilities that would last up to 2070, and the expected life (for economical purpose) of the NPP's the average quota would be 0,8% of the electricity bill up to year 2028, equivalent to 0,464 Ptas'99/kW nuclear.

Overview on spent fuel management strategies

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Abstract. This paper presents an overview on spent fuel management strategies which range from reprocessing to interim storage in a centralised facility followed by final disposal in a repository. In either case, more spent fuel storage capacity (wet or dry, at-reactor or away-from-reactor, national or regional) is required as spent fuel is continuously accumulated while most countries prefer to defer their decision to choose between these two strategies.

1. INTRODUCTION

The management of spent nuclear fuel is among the most pressing issues to be addressed in the nuclear fuel cycle. Because technologies, needs and circumstances vary from country to country, there is no single, standardized approach to spent-fuel management. Two options exist - an open, once-through cycle with direct disposal of the spent fuel and a closed cycle with reprocessing of the spent fuel, recycling of reprocessed plutonium and uranium in new mixed oxide fuels and disposal of the radioactive waste. The selection of a spent fuel strategy is a complex procedure in which many factors have to be weighed, including political, economic and safeguards issues as well as protection of the environment.

Delays in the implementation of the fuel reprocessing option in some countries, the complete abandonment of this option in other countries and delays in the availability of final spent fuel disposal in almost all countries has led to increasingly long periods of interim spent fuel storage. This “wait and see” approach gives more time and freedom to evaluate the available options and to select the most suitable technology. The problem of spent fuel management has therefore increased in importance for many countries.

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

2. SITUATION

2.1. Challenges

It is noted that there continues to be worldwide growth in the generation of electric power using nuclear energy as its source. It is further noted that the rate of growth of nuclear energy generation has essentially levelled in Europe and North America while it has increased significantly in Asia. Although these trends have some impact on spent fuel management, including storage, the worldwide spent fuel production rate continues at about 10 500t HM/yr.

Many at-reactor (AR) storage pools have been used to full capacity in recent years. This threatens the routine operation of the power plant, in some cases. Due to limited pool capacity away-from-reactor (AFR) storage is necessary to maintain operation.

The trend to higher fuel burnup, and consequently higher enrichment of the fresh fuel, and the use of plutonium in mixed oxide fuel, leads to other spent fuel characteristics (i.e. higher decay heat and flatter downward curve over time). This demands a longer storage period than for the present spent fuel with burnup lower than 40 GWd/t.

In many Member States, the lack of final repositories and the deferral of the decision will lead automatically to long storage periods even of uncertain duration. The lifetime of existing storage facilities will be extended and new facilities for long-term storage have to be built. The design of new facilities has to take into account not only the fuel behaviour during long-term storage but also the behaviour of the materials, equipment and installation.

With respect to operating experience, spent fuel can be safely stored for long periods of time. Some spent fuel has now been stored for over 30 years. There is a scientific and technical consensus that the present technologies of spent fuel storage give adequate protection to population and the environment.

2.2. Spent fuel arising

About 35% of the spent fuel arising came from each of the two regions West Europe and North and South America and 15% of each of the two regions East Europe and Asia and Africa. Figure 1 shows the current and projected regional spent fuel arising.

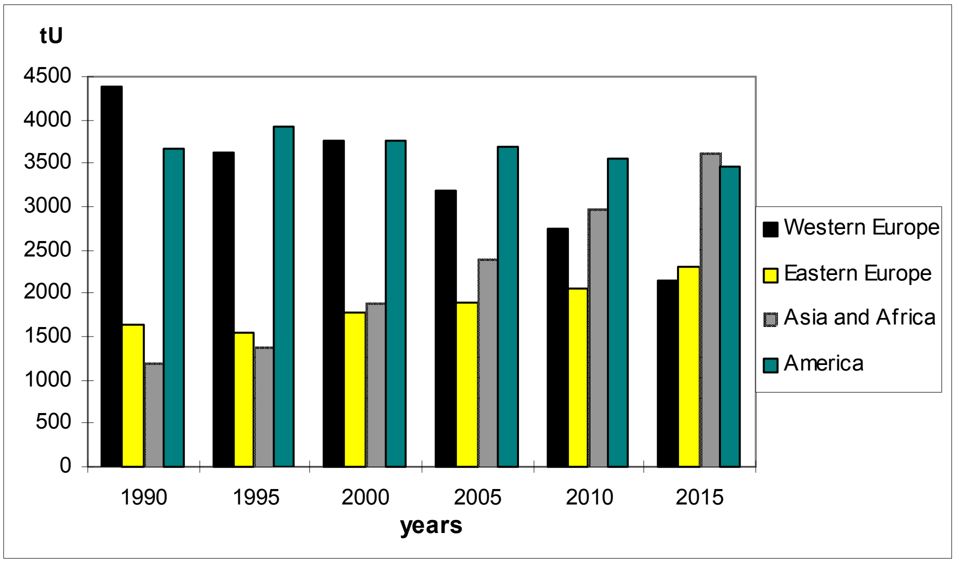

FIG. 1. Annual spent fuel arising in world regions.

The total amount of spent fuel accumulated worldwide is about 220 000 t HM. About 75 000t HM of this fuel were reprocessed. The remaining 145 000 t HM of spent fuel is presently being stored in at-reactor (AR) and away-from-reactor (AFR) storage. Over 70% of this amount is stored in at reactor pools, the remainder in away from reactor wet and dry storage facilities.

Projections indicate that the cumulative amount generated in the world by the year 2010 may surpass 340 000 t HM and 395 000 t HM by the year 2015. In 2010 about 225 000 t HM of spent fuel has to be stored, in 2015 more than 260 000 t HM. Of this total amount in 2015, the amount in West Europe will remain about the same (because of reprocessing spent fuel) and will four fold in Asia and Africa.

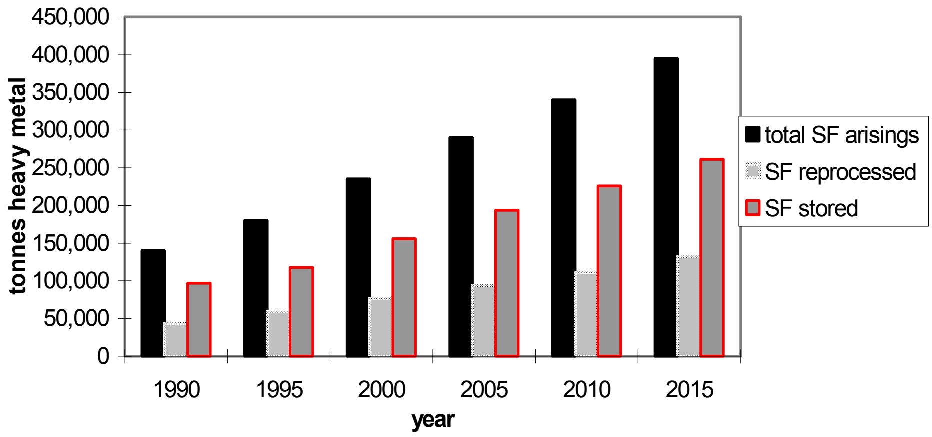

FIG. 2. World spent fuel arising and amounts of spent fuel reprocessed and stored.

On a regional basis, the picture looks different than the annual spent fuel arising. About 50% is stored in North and South America, there is no reprocessing, 25% in West Europe and the remaining part in East Europe and Asia and Africa.

2.3. Spent fuel storage capacity

Nearly all countries operating nuclear power plants have increased their original AR storage capacity by reracking the spent fuel pools with high density racks and by implementing burnup credit. Several countries commissioned additional AFR storage facilities. The AR spent fuel storage pools are of the wet type, as were also most of the AFR spent fuel storage facilities built in the past. Most of the newly built and future planned facilities are dry storage type facilities. Dry facilities involve storage of spent fuel in a gaseous environment, such as an inert gas or air, in casks, vaults or silos. A cask is a massive shielding container which may or may not be transportable. Vaults consist of above or below ground reinforced concrete buildings containing arrays of storage cavities suitable for containment of one or more fuel units. Silos consist of a steel container as containment for the fuel and an outer concrete shielding.

TABLE I. STATUS OF SPENT FUEL STORAGE CAPACITY IN WORLD REGIONS [kt HM]

	In operation				Under construction		
Regions	at NPP	Wet	Dry	Total	Wet	Dry	Total
West Europe	26.1	31.7	9.2	67.0		0.8	0.8
Asia & Africa	20.0	1.9	0.7	22.6	0.7	0.8	1.5
East Europe	14.3	19.6	0.8	34.7	0.8	1.6	2.4
North & South America	94.9	1.8	10.0	106.7		6.8	6.8
World	155.3	55.0	20.7	231.0	1.5	10.0	11.5

Various types of wet and dry storage facilities are in operation. The current world storage capacity is about 231 000 t HM, of which 46% is situated in North and South America, 30% in West Europe, 15% in East Europe and the remaining part in Asia and Africa (Table I). The storage capacity of facilities under construction world wide is about 12 000 t HM.

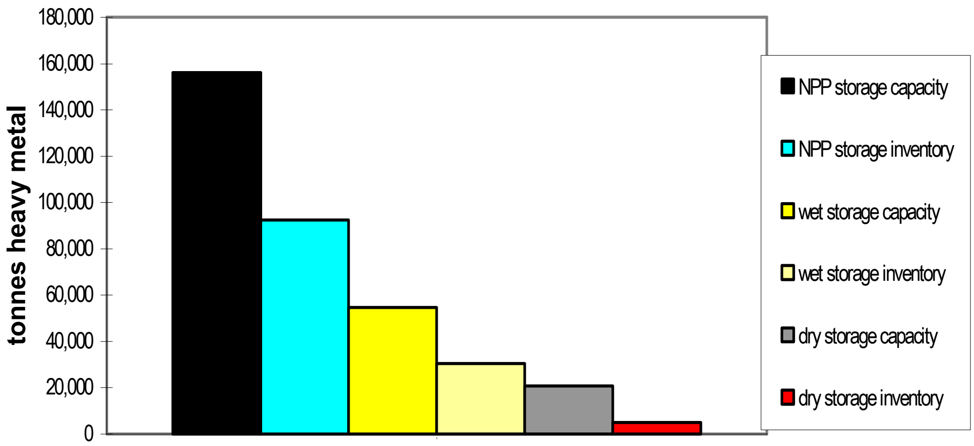

FIG. 3. Comparison of capacities and inventories of different types of spent fuel storage.

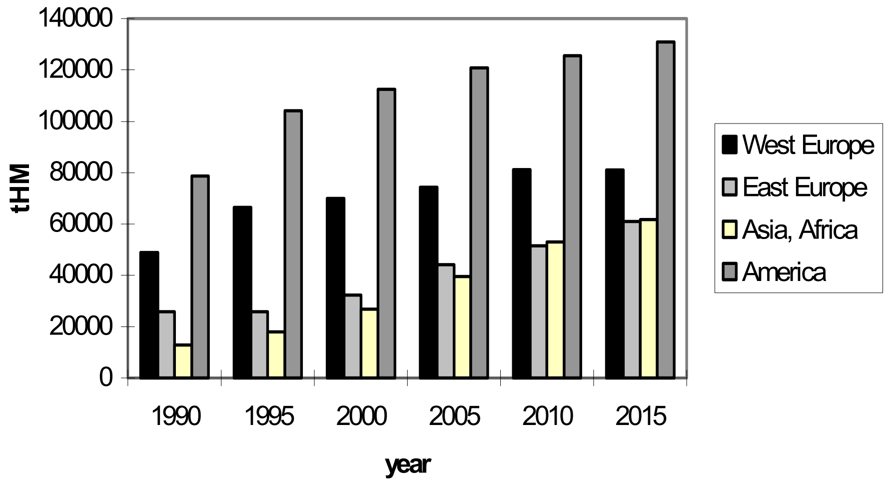

FIG. 4. Spent fuel storage capacity in world regions.

2.4 Balance of spent fuel arising and storage capacity

Figure 4 shows the projected spent fuel storage capacity.

At the beginning of 98, the spent fuel storage capacity world-wide exceeded the amount of spent fuel to be stored by about 100 000 t HM. All types of storage facilities had excess capacity available. Fig. 3 compares the capacities of the various storage types with their current inventories.

On a world basis, the projected spent fuel arising will overflow the existing storage facilities and those under construction by around 2010, if no additional capacity is provided by that time. The average values imply no problems. However a more detailed investigation is necessary to note specific shortage in storage capacity.

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 like in Armenia. In other cases, additional storage capacity has to be installed timely to replace wet storage facilities which can not be refurbished, as is the case in Chernobyl

If there will be a delay in reprocessing, more spent fuel needs to be stored. This situation would demand more storage capacity. Further postponement of the decision related to the development of final repositories will also lead to a higher demand in spent fuel storage capacity.

3. SPENT FUEL MANAGEMENT STRATEGIES

The spent fuel management strategies range from the clear strategy to reprocess as much fuel as technical reasonable and close the fuel cycle, like in France, to the clear strategy of the open fuel cycle, to store the spent fuel in a centralized facility and dispose of in a final repository, like in Sweden. All other countries range more or less in between, most of them choose the wait and see strategy with long-term interim storage to postpone a final decision.

Besides France reprocessing facilities operate in the UK, Russia, India and Japan and are under construction in China and Japan.

The first final repository is operating in the USA, the WIPP facility, the most advanced investigation of a final repository in Germany is under question now. In the USA the Yucca Mountain facility is under investigation and Sweden is steadily proceeding with its preparation for closing the fuel cycle.

In Germany the Pilot Conditioning Plant, the first plant of its kind world-wide is ready for operation and awaiting its license.

All other countries are far from final disposal and will have to store their fuel for long periods of time.

4. FUTURE PERSPECTIVES

The world annual spent fuel arising, now about 10 500 tHM, will increase during the next years to about 11 000 tHM in 2000 and about 11 500 tHM in 2010. As in the past, less than one third of the spent fuel will be reprocessed and this mainly in Europe. Thus the storage capacity has to be increased accordingly. Some initiatives can be noticed for developing multinational spent fuel storage facilities even by private companies.

Several countries with a small nuclear power programme or only research reactors face the serious problem of extended interim storage and disposal of their spent nuclear fuel. The high specific costs for the construction of away-from-reactor extended interim storage facilities and/or geological repositories for the relatively small amounts of spent fuel accumulated in such countries is obviously not reasonable and, therefore, from an economical point of view, access to a regional/multinational interim storage facility and/or repository for their fuel would be an ideal solution.

It is interesting to note that *de facto* Regional/Multinational Spent Fuel Storage Facilities (RSFSFs) exist in several countries. The word “regional” is used in the broad sense of the word that is a geographical area covering more than one country.

In Western Europe, Eurochemic was one of the most significant early projects for a multinational arrangement. COGEMA and UKAEA-BNFL are effective involved in interim storage of spent power reactor fuel from a number of countries while awaiting reprocessing.

Research reactor fuel of US origin from all over the world is at present stored in wet interim storage pools at the Receiving Basin for Off-Site Fuels (RBOF) facility at DOE's Savannah River Site. Proliferation concerns weighed heavily in this case.

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

However, there are still various problems to solve as to find operators of such facilities with governmental support and to convince countries of proliferation concern to participate.

The time is ripe for serious discussion of such regional facilities and to begin planning for the day when neither take-back programmes nor the reprocessing option might be available.

5. CONCLUSIONS

The duration of interim storage becomes longer than earlier anticipated and the storage facilities will have to be capable of receiving spent fuel also from advanced fuel cycle practices (i.e. high burnup and MOX spent fuel) in the future.

The handling and storage of spent fuel is a mature technology and can meet the stringent safety requirements applicable in the different countries.

Wet storage remains dominant, even as the use of dry storage concepts increases. Wet storage is essential for cooling newly-discharged fuel, and will continue to be the method of storage used in connection with reprocessing.

Dry storage is being used increasingly, as more long-term storage of spent nuclear fuel is done. Dry storage may prove to be a cost-effective activity that can easily accommodate multipurpose systems (e.g., storage/transport, storage/transport/disposal).

More spent fuel storage capacity is required because most countries deferred their decision to choose between the open and closed fuel cycle;

The first geological repositories for the final disposal of spent fuel are not expected to be operational before the year 2010 anywhere in the world and many countries did not yet start investigations. Thus, the use of interim storage will be the primary spent fuel management option for the next decades in many of the IAEA Member States.

The "wait-and-see" option chosen by many countries and the use of high burnup and MOX fuel with higher residual heat and longer heat decay times imply a longer interim storage period before final disposal.

For long-term storage the dry storage technology has advantages if the fuel is stored in an inert atmosphere.

6. IAEA ACTIVITIES

The main IAEA activities in the field of spent fuel management for power reactors are covered in the following tasks:

- Long-term storage of spent fuel including advanced, high burnup and MOX fuel
- Requirements for extremely long-term storage facilities
- Implementation of burnup credit in spent fuel management systems
- Technologies and safety aspects of regional spent fuel storage facilities
- Remote technology in spent fuel management
- Selection criteria for away from reactor storage facilities
- Spent fuel treatment

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WASTE MANAGEMENT/REPOSITORY

Overview of the United States' nuclear waste repository programme

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Abstract. Regardless of the future of civilian or defense-based nuclear materials, the United States will be responsible for a vast array of these materials for generations to come. The cornerstone programme for the disposal of waste materials is the Yucca Mountain Programme. Based on the Nuclear Waste Policy Act of 1982, as amended in 1987, it has been the United States' policy to develop a geological repository for the permanent disposal of radioactive waste materials. This presentation will discuss the process and strategy leading to the present and will include the scientific and management activities required to support the recent Viability Assessment. Also to be discussed are the timeline and milestones leading to the opening of the repository. The focus will be on the scientific and engineering studies required for a successful Site Recommendation, and then for a similarly successful License Application. Both of these activities will require considerable management efforts in addressing legal and regulatory issues. Finally, the presentation will discuss projections for the future operation of the facility, including emplacement projections, coupled with the required locations of nuclear materials. Additional scientific research and engineering studies will also be conducted to determine the longer-term viability of the facility, which is designed, by policy, for permanent storage. Retrievability is currently not an option, although access to the facility will be maintained for several decades. The focus of the discussion will be on the scientific and engineering advances made on understanding the natural systems for preventing migration of radionuclides, coupled with new developments in engineered systems in areas such as cask cladding, drip shields, and related materials engineering developments. The coupling of engineered and natural systems is designed to offer safety factors that are several orders of magnitude greater than what is estimated to be necessary.

1. INTRODUCTION

Currently 104 nuclear power facilities (reactors) provide approximately 20% of the electricity produced in the United States. These reactors contribute between 1800 and 2200 metric tons of uranium (MTU) annually to the accumulating amount of spent nuclear fuel (SNF), estimated to be approximately 41,000 MTU at the end of 1999. It is projected that by the year 2040, if all reactors continue to operate, the inventory of SNF will have increased to 85,000 MTU. The Nuclear Waste Policy Act of 1982 (Public Law 97-425) established the Office of Civilian Radioactive Waste Management (OCRWM) within the Department of Energy (DOE) to develop and safely manage a Federal system for disposing of the Nation's SNF and high-level radioactive waste.

2. USDOE OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT (OCRWM)

Funding

OCRWM continues to be funded through appropriations from the Nuclear Waste Fund, which is financed through a 1.0 mil per kilowatt hour fee imposed on the utilities for electric power generated and sold by nuclear power facilities. Contributions are as much as approximately \$650 million per year. At the end of fiscal year 1999 (FY 99), the fund had received a total of approximately \$13.7 billion, including investment earnings, and expended approximately \$5.3 billion. In addition, during FY 99 OCRWM also earned approximately \$500 million in defense revenue from the Office of Environmental Management and Office of Energy's Naval Nuclear Propulsion Program, custodians of the DOE inventory of high-level radioactive waste and spent nuclear fuel. The total accrued defense revenue at the end of FY 99 is approximately $2.6 billion.

Program Strategy

Funds appropriated by Congress for FY 99 continued to be used toward objectives that maintained the momentum toward a national decision on the geologic disposal option: 1) completion of the Final Environmental Impact Statement next year (FY 01), 2) continuing the necessary work towards a decision on whether to recommend the Yucca Mountain site to the President in 2001, if the site is suitable for a repository, and 3) submit a license application for construction authorization to the Nuclear Regulatory Commission (NRC) in 2002, if the site is approved by the President and Congress. As discussed below, implementation of program requirements to reflect this strategy continued throughout 1999 and into 2000.

Repository Regulatory Framework

In 1999, the regulatory framework for evaluating the suitability of the Yucca Mountain site moved closer to final form. On November 30, 1999, the Department published a proposed revision to its repository siting guidelines. The proposed revised guidelines reflect a shift away from a generic approach that could apply to any site and that focused on individual technical criteria, to a site-specific approach that relies on an overall systems evaluation of the expected performance of a repository at Yucca Mountain. This same approach had been taken by the NRC in the proposed repository licensing regulations it published on February 22, 1999. The Department's siting guidelines must lead to selection of a site that can satisfy the NRC regulation and receive a license. On August 27, 1999, the Environmental Protection Agency (EPA) published proposed radiation protection standards for a repository at Yucca Mountain. The NRC regulation must implement the EPA standards. The NRC has announced that it will, if necessary, revise its regulation once the EPA standard is finalized.

Repository Design

In 1999 work was focused on preparing for a Secretarial determination on site recommendation. It is expected that a Site Recommendation Consideration Report will be released in Fiscal Year 2001. In 1999 the work to develop the Site Recommendation Consideration Report built on the *Viability Assessment of a Repository at Yucca Mountain* that the Secretary released in December 1998. A major accomplishment in 1999 was the adoption of repository design enhancements for the total system performance assessment that will support the Secretarial determination on site recommendation. Some of the new features had been recommended by the Nuclear Waste Technical Review Board. To be more flexible in meeting changing expectations and to simplify the approach to licensing the repository, the Program adopted an approach for a "cooler" repository that employs long term ventilation to remove heat from the waste packages. The benefits of this approach include reducing the uncertainties associated with predicting the effects of heat on the natural system along with being able to utilize a single emplacement drift that is representative of the entire repository. This concept allows for repository closure after 50 years or to remain open for as long as 300 years, depending on the results obtained from performance confirmation monitoring and the views of future generations.

Another important task that will support the Site Recommendation Consideration Report was continued development of system description documents for major repository subsystems related to safety, such as the materials handling system. These documents specify requirements for repository subsystems and describe the resulting design.

Draft Environmental Impact Statement

On August 6, 1999, the DOE Draft Environmental Impact Statement (DEIS) for a Geologic Repository on Spent Nuclear Fuel and High-Level Radioactive Waste was issued for public comment. The DEIS provides information on the potential environmental impacts that could result from the proposed action to construct, operate and monitor, and eventually close a deep underground repository at Yucca Mountain, in Nye County, Nevada. The DEIS also analyses an alternative to the proposed action: a no-action alternative. The DEIS further analyzes the potential impacts of transporting spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site from 77 sites across the United States. The analysis also includes the use of active institutional controls (controlled access, inspection, maintenance, etc.). DOE has held 21 public hearings on the DEIS. The public comment period on the DEIS closed on February 28, 2000. DOE is in the process of reviewing the input and will prepare a Final Environmental Impact Statement in FY 01.

Scientific Investigations Of Yucca Mountain

The Project activities associated with scientific and engineering investigations were focused on the remaining key uncertainties about the Yucca Mountain site. Those uncertainties, discussed in the Viability Assessment, include: the presence and movement of water through the repository block; the effects of water movement on waste package degradation; and the effects of heat from the decay of radioactive materials inside the waste packages on the site's geologic and hydrologic behavior. The Program's main thrust in 1999 (and further planned for most of the year 2000) has been to ensure that sufficient data has been obtained to support the Site Recommendation (SR). The current goal is to obtain 80 % qualified data required for the SR, and to further obtain 100 % qualified data by the License Application.

Construction in the underground Exploratory Studies Facility progressed significantly in the last year. It is expected that work associated with construction of the cross drift alcoves and niches will continue towards completion over the next few years.

The Drift Scale Heater Test in the Thermal Testing Facility is continuing, and conservative analyses of measured and predicted temperatures in the rock mass surrounding the heated drift indicate generally good agreement between measured and predicted values. It is planned that, subsequent to heater operation over the next year, a four year cool down evaluation cycle will be initiated.

C-Well tracer testing of the Prow Pass interval in the saturated zone below the level of the proposed repository is being continued to better characterize the flow, dilution, and sorption potential in the uppermost hydrogeologic unit in the saturated zone immediately downgradient from the potential repository.

Scientists continued working at the Busted Butte where they are studying tracer movement, fluid flow, and transport behavior in a distal extension of the rock of the Calico Hills Formation, which lies between the repository horizon and the water table. These tests are yielding information that is being used to evaluate how far and how fast key radionuclides may move in the non-welded rock below the repository in the unsaturated zone.

It is expected that, in the year 2000, the project activities will transition from scientific investigations to data synthesis, model validation, repository and waste package design, and safety analysis.

Waste Acceptance, Storage And Transportation

With funding for this Project at less than 1 percent of our FY 99 budget, work remained curtailed. We continued to manage the contracts we executed with utilities under the Nuclear Waste Policy Act and to gather the data about their spent fuel inventories that are required for waste acceptance.

Under these contracts, DOE was to start accepting spent nuclear fuel from utilities in 1998. With no Federal facility available to receive the material, utilities continued to pursue litigation to seek relief from hardships they allege as a consequence of the Department's inability to accept waste. In an effort to resolve this dispute, in a March 1999 testimony before Congress, the Secretary proposed that the Department take title to utilities' spent nuclear fuel and manage it at their sites. Analysis of this option is under way.

In October 1999, a successful demonstration of a prototype for a dry transfer system for spent nuclear fuel, which Congress had directed DOE to develop cooperatively with the nuclear utility industry, was concluded. We expect the NRC to approve our Topical Safety Analysis Report on the dry transfer system by April 2000.

Accelerator Transmutation of Waste (ATW)

The U.S. Congress directed the DOE, through the Fiscal Year 1999 Energy and Water Appropriations Act, to evaluate the accelerator transmutation of waste (ATW), and by the end of the Fiscal Year 1999 to prepare a roadmap for development of this technology. DOE developed an ATW roadmap, including among other tasks: 1) identification of the technical issues that must be resolved, 2) a proposed schedule, and cost estimate for such a program, including an estimate of the capital and operational life-cycle costs to treat spent fuel, 3) proposed collaborative efforts with other countries and other programs developing ATW technologies, 4) identification of the institutional challenges, and 5) an assessment of the impact that ATW technology could have on the civilian spent nuclear fuel program along with identification of development areas that could benefit other ongoing programs. A major conclusion of the study is that a repository is an essential element of the nuclear fuel cycle with or without ATW deployment. A summary of the major recommendations resulting from development of the roadmap includes, but is not limited to: 1) during an initial 6 year trade and systems studies period, science-based R&D should address the key technology system element issues identified during roadmap preparation; and 2) work planned and implemented during this period should be accomplished through robust international collaboration.

3. USDOE OFFICE OF ENVIRONMENTAL MANAGEMENT

Overview

The primary mission of the DOE Environmental Management (EM) program is to reduce health and safety risks from contamination and waste at the sites associated with the development of nuclear weapons and nuclear power systems. This mission is realized through the following program areas: waste management; environmental restoration; nuclear material and facility stabilization; science and technology; pollution prevention, and public accountability. The overall program is responsible for the storage and treatment (both short and long-term) and disposal nuclear and chemical wastes generated during more than 50 years of nuclear weapons production and nuclear research.

The EM program currently has responsibility for cleanup of over 113 geographic sites in over 30 states and one territory -- over 2 million acres. EM continues to work toward the goal of cleaning up as many of its sites by the year 2006. As of the beginning of FY 00, cleanup had been completed at 69 of these sites, leaving 44 to be completed. Three DOE sites -- Rocky Flats, Colorado; Fernald, Ohio; and Miamisburg, Ohio – are pilot sites for accelerated closure.

The number of sites and facilities managed under the EM program has grown as projects have been transferred from other DOE programs (such as Defense Programs and Nuclear Energy). The program now manages several hundred high-level radioactive waste tanks and thousands of contaminated buildings that remain to be deactivated and decommissioned. The volume of waste managed by DOE is enormous – 36 million cubic meters, containing about one billion curies of radioactivity

In addition to managing the existing legacy of wastes, materials, and contaminated sites and structures, the prevention of further waste generation and pollution is a major goal of EM and of the Department as a whole. In 1996, and again in 1999, DOE issued aggressive pollution prevention goals in order to reduce generation of hazardous, radioactive, and sanitary wastes by at least 80 percent by 2010 or earlier (using 1993 as a baseline).

Key aspects of the EM program are summarized below.

Waste Management

An important part of the EM mission is to protect people and the environment from the hazards of Departmental waste by providing an effective and efficient system that minimizes, stores, treats, and disposes of waste as soon as possible. Currently, waste management facilities store and manage more than 700 000 cubic meters of radioactive waste and a wide variety of hazardous chemical wastes at more than 40 sites nationwide. Some 80 percent of the radioactive waste is also mixed with hazardous chemicals. Much of this waste has been stored at DOE sites for up to 50 years. These wastes include high-level radioactive waste, transuranic waste, and low-level waste. Highlights of progress in managing these wastes are provided below.

High-Level Waste

The focus of Environmental Management's (EM’s) activities for managing high-level waste is on storage and treatment so that it can be disposed in a geologic repository. (Disposal of this waste, along with spent nuclear fuel, is the responsibility of the Department of Energy's (DOE's) OCRWM -- covered in the first section of this paper.) DOE currently manages about 345 000 cubic meters of high-level waste generated from the reprocessing of spent nuclear fuel at four DOE sites. Most of this inventory is in the form of sludge, liquids, salts, and calcine. The strategy for preparing this waste for ultimate disposal is to vitrify the waste into glass logs. Treatment began in 1996 at vitrification facilities at the Savannah River Site in South Carolina and the West Valley Demonstration Project in New York, two of the four sites that store significant quantities of high-level waste. To date, about 1 000 canisters have been produced (less than 10% of the total number of canisters to be produced at the four sites over their life-cycle). Vitrification at West Valley is nearing completion. Design is underway for a new vitrification facility at the Hanford Site and early planning is underway at the Idaho National Engineering and Environmental Laboratory. Once vitrified, the canisters of high-level waste will remain in storage at the sites that generated the waste until a repository is available.

Transuranic Waste

DOE is currently managing more than 100 000 cubic meters of transuranic waste, the bulk of which is at six major sites. The strategy for managing transuranic waste is to dispose of the waste in a geologic repository built in salt deposits. DOE selected this type of disposal for transuranic waste because of the geologic stability of the salt formations, which will safely and permanently isolate transuranic waste for thousands of years. The Waste Isolation Pilot Plant (WIPP) is a series of chambers carved into salt beds 645 meters (2 150 feet) underground, located about 30 miles east of Carlsbad, New Mexico. After two decades of development, WIPP opened for disposal operations in March 1999. During the first six months of operation, WIPP received 32 shipments (containing 276 cubic meters) of transuranic waste from three sites — Los Alamos National Laboratory, Rocky Flats, and Idaho National Engineering and Environmental Laboratory (INEEL). The WIPP disposal capacity is about 175 000 cubic meters, and it will take about 30 years to fill.

More recently (October 1999), the State of New Mexico issued a permit that will allow for mixed transuranic waste (transuranic waste containing hazardous constituents regulated under Resource Conservation and Recovery Act regulations) to be received at WIPP providing the waste meets the WIPP acceptance criteria. Some mixed transuranic waste will require treatment prior to shipment to WIPP. Therefore, the EM program is working toward providing treatment capacity for this waste. New treatment facilities are planned, with the first new facility — the Advanced Mixed Waste Treatment Project at INEEL — scheduled for operation in 2003.

Low-Level and Mixed Low-Level Waste

Approximately 1 million cubic meters of low-level waste and 176 000 cubic meters of mixed low-level waste will require disposal over the next twenty years. Currently, DOE has six low-level waste disposal sites. Waste generators without an on-site low-level waste disposal facility ship waste to one of the operating sites for disposal and in some instances to commercial facilities when practical and economical. In February 2000, DOE decided to continue disposal at these sites for wastes generated onsite, while allowing other waste generators to ship their wastes to the disposal facilities located at the Nevada Test Site, NV, and Hanford Site, WA. In addition, DOE decided to dispose of its mixed low-level waste at Nevada Test Site and Hanford Site, where facilities have already been constructed but to date used only for on-site generated waste.

Privatization of Traditional DOE Functions

EM has taken steps to transfer functions traditionally performed by the Department's management and operating contractors to private companies that will provide the service on a competitive, fixed-price basis. EM is working to privatize the Hanford Tank Waste Remediation System -- the largest single project at the Department - to reduce the technical and cost-performance burden on the Department. Other privatization projects include capabilities to store spent nuclear fuel; to treat transuranic and mixed waste at the INEEL; to treat transuranic waste at Oak Ridge, TN; and to transport remote-handled transuranic waste to WIPP.

Environmental Restoration

Another key part of the EM mission is environmental restoration -- the remediation and management of contaminated environmental media (e.g., soil, groundwater, sediments) and the decommissioning of facilities and structures at some 113 geographic sites in order to protect human health and the environment from existing risks and provide for future beneficial reuse of restored land and facilities. As of the start of FY 00, 69 of the 113 geographic sites have been cleaned up. Two more will be cleaned up in FY 00 and three more in FY 01. At each geographic site, there may be numerous individual waste sites, referred to as "release sites," and contaminated facilities whose cleanup ultimately leads to completion of an entire geographic site.

Cleanup progress at environmental restoration sites takes the form of "remedial actions," which are actions taken to identify and contain or remove soil and groundwater contamination to prevent it from spreading, to decommission and dismantle facilities, and to clean up contaminated structures. Decommissioning operations range from small cleanup activities involving portions of buildings to complete structural dismantlement.

Since 1989, EM completed cleanup actions at almost half (about 4 300) of the individual release sites (out of a total inventory of 9,700 release sites), and decommissioned about 15% (558) of the facilities that need to be decommissioned (over 3 300). In addition, EM is continuing multiple environmental restoration activities and groundwater effort at all major EM sites.

After completing cleanup, the EM program will maintain a presence at most sites to monitor, maintain and provide information on any contained residual contamination. These activities are designed to maintain long-term protection of human health and the environment. Such long-term stewardship will include passive or active institutional controls and, often, treatment of contaminated groundwater over a long period of time. The extent of long-term stewardship required at a site will depend on the desired end-state to be reached at that particular site.

Nuclear Materials And Facility Stabilization

Stabilizing, monitoring, and maintaining the large quantity of nuclear material and spent fuel left over from Cold War weapons activities is one of the most urgent tasks in the EM program. DOE must stabilize these materials and fuel (i.e., produce a safer chemical and/or physical form of the material) to reduce the level of potential risks, such as exposure to radiation, contamination of people and the environment, and critical events. Stabilization activities have been prioritized so that the most serious risks are addressed first.

Nuclear materials will be stabilized at the Plutonium Finishing Plant at Richland and in several facilities at the Rocky Flats Environmental Technology Site and the Savannah River Site. DOE's spent nuclear fuel, and the foreign research reactor spent fuel that the U.S. is accepting, will be treated where necessary, packaged suitably for final disposal, and placed in dry interim storage pending disposal in a geologic repository. As nuclear material and spent fuel are placed in more stable forms, the physical plant (buildings, production systems, machinery, and utilities) where the materials had been stored can be deactivated.

Milestones have been established for the stabilization of some nuclear materials by the year 2002, including various forms of plutonium, uranium, special isotopes, and spent nuclear fuel. Based on the current inventory of materials and facilities in the program, it is projected that the stabilization mission will be complete by 2010.

Throughout FY 99, the EM program reduced environmental risks by stabilizing nuclear and other materials and spent nuclear fuels at the Savannah River Site, plutonium residues and plutonium metals and oxides at Rocky Flats, and materials at several other sites. These materials are located in spent-fuel storage pools, reactor basins, reprocessing canyons, and various facilities once used for processing materials for nuclear weapons. All plutonium pit shipments from Rocky Flats to Pantex Plant were completed in FY 99, and starting in FY 00, the metal/oxide containers will be shipped from Rocky Flats to the Savannah River Site. Plutonium stabilization activities at Hanford Site's Plutonium Finishing Plant were restarted in FY 99.

The Department currently stores and manages spent nuclear fuel resulting from DOE missions and from domestic and foreign research reactors. This spent fuel, approximately 2 500 metric tons of heavy metal, is currently stored in facilities at four DOE sites – Hanford, INEEL, Savannah River Site, and West Valley Demonstration Project. A geologic disposal facility is not expected to be ready to accept Department-owned spent nuclear fuel before 2015. New dry storage facilities are being developed at Hanford and INEEL to provide long-term storage of spent nuclear fuel and allow aging facilities to be decommissioned.

The EM program continues to play a key role in implementing U.S. nuclear weapons nonproliferation policy regarding foreign research reactor spent nuclear fuel. Under this policy, the United States will accept, over a 13-year period, up to approximately 20 metric tons of research reactor spent nuclear fuel from 41 countries. Only spent nuclear fuel containing uranium enriched in the United States falls under this policy. DOE has completed 14 shipments of spent nuclear fuel from foreign research reactors from 23 countries. Twelve shipments have been received at the Savannah River Site in South Carolina. Two shipments to INEEL have been completed, including the first cross-country shipment of foreign research reactor spent nuclear fuel from Savannah River Site in August 1999.

Science and Technology

EM develops and deploys innovative environmental cleanup technologies that reduce cost, resolve currently intractable problems, and/or are more protective of workers and the environment. Technology development activities are organized in five major focus areas: (1) Mixed Waste; (2) Radioactive Tank Waste; (3) Subsurface Contaminants; (4) Deactivation and Decommissioning; and (5) Nuclear Materials. Crosscutting activities are conducted in support of these focus areas, such as robotics, efficient chemical separations; characterization, sensors, and monitors; industry and university programs; and technology integration.

The success of the EM science and technology program is currently measured by several factors: (1) the number of innovative technology systems demonstrated that meet the performance-specification-based needs identified by Site Technology Coordination Groups; (2) number of innovative technology systems ready for implementation with cost and engineering performance data; and (3) number of deployments of innovative technologies in cleanup activities. In FY 99 alone, 27 innovative technologies were demonstrated to meet identified needs; 129 innovative technology deployments were accomplished at DOE sites; and 40 were made ready for implementation.

Pollution Prevention

DOE's respect for the environment has lead to an aggressive pollution prevention program, which focuses on reducing or eliminating the creation of pollutants or waste at the source. In 1996, DOE outlined specific goals for reducing waste generation and the use and release of toxic chemicals and for increasing recycling and the purchase of environmentally preferable products. These goals require the complex-wide reduction of routine operations' low-level radioactive, mixed, and hazardous wastes by 50 percent, and routine operations' sanitary waste by 33 percent, compared to the 1993 baseline. Within two years, DOE sites implemented over 1,800 pollution prevention projects and were able to avoid more than 390,000 cubic meters of waste, for a reported cost savings/avoidance of $405 million. Beginning with FY 99, DOE set an aggressive goal for a 10 percent annual reduction in waste generation from cleanup/stabilization activities. Sites are recognized for their achievements in pollution prevention through annual awards. Below are some examples of site successes:

- Los Alamos National Laboratory recycled lead and steel material from an accelerator facility, which was earlier thought to be low-level mixed waste due to its origin and lead content. After surveys determined the material was not activated, it was able to be recycled, reducing low-level mixed waste by about 338 cubic meters at a reported cost savings of over $25 million.

- INEEL replaced a hazardous nitric acid cleaning process with an environmentally friendly high pressure water cleaning system, thus eliminating nitrogen oxides emissions and nitric acid safety concerns and reducing hazardous waste by six metric tons, for a reported cost savings of $1 million.

- Oak Ridge personnel conducted radiological surveys at facilities during the deactivation process in order to segregate free-releasable items from activated and contaminated ones and was able to release 515 tons of material. This segregation activity reduced low-level waste by over 460 cubic meters, for reported cost savings of $2.3 million.

Public Accountability

The EM program maintains a close working relationship with its various stakeholder communities. The Environmental Management Advisory Board, consisting of individuals representing federal and local environmental agencies, corporations, universities, and other organizations, provides advice as an unofficial "board of directors."

The DOE has established Advisory boards at each site with an EM activity to give the public a forum to express its concerns and recommendations. The boards are composed of local citizens, including representatives from local governments, Indian Tribes, environmental and civic groups, labor organizations, universities, waste management and environmental restoration firms, and other interest groups. Board members recommend options to resolve difficult issues facing the EM program, including site-specific cleanup criteria, risk assessment, land use, priority setting, management effectiveness, cost-benefit analyses, and strategies for site waste management and disposal facilities.

4. U.S. NUCLEAR REGULATORY COMMISSION

High-Level Radioactive Waste

Regulatory Development Activities

As directed by the Energy Policy Act of 1992, the Environmental Protection Agency (EPA) contracted with the U.S. National Academy of Science (NAS) to conduct a study and provide recommendations to the EPA on the appropriate technical basis for Yucca Mountain standards. Although the NAS could consider a range of issues, its recommendations were to address:

1) whether a standard based on doses to individuals is reasonable;
2) whether post-closure oversight and active institutional controls can effectively ensure that exposures to individuals will be maintained within acceptable limits; and
3) whether scientifically-supportable probability estimates of human intrusion into a repository over 10,000 years can be made.

The NAS issued its report on August 1, 1995 and EPA issued a proposed rule, 10 CFR Part 197 on August 27, 1999 to set environmental standards for Yucca Mountain. In a November 2, 1999 letter to EPA, the Nuclear Regulatory Commission (NRC) provided comments on the Notice of Proposed Rulemaking, Environmental Radiation Protection Standards for Yucca Mountain, Nevada (64 FR 46976).

On February 22, 1999, NRC published for public comment a proposed regulation for disposal of high-level radioactive waste in a proposed geologic repository at Yucca Mountain, Nevada. The proposed regulation is available from the Nuclear Regulatory Commission's (NRC's) interactive rulemaking web site at http://www.nrc.gov/NRC/rule/html. The NRC proposed rulemaking for 10 CFR Part 63 contains risk-informed, performance-based criteria for both pre-closure operations and post-closure performance of the proposed geologic repository for high-level waste at Yucca Mountain, Nevada. NRC's standard is consistent with recommendations of the National Academy of Sciences and with national and international recommendations for radiation protection standards. Although NRC's draft identifies a standard on which to judge the post-closure performance of the proposed repository, the draft recognizes the need for NRC's final rule to be consistent with the final EPA standards when they are promulgated. The public comment period for the proposed rulemaking ended on June 30, 1999 (64 FR 24092), and the staff, in preparing the draft final rule, has carefully reviewed and considered more than 700 discrete comments enclosed in about 100 individual letters filed during the public comment period. The final rule is expected to be published in FY 00.

Repository Program Activities

Under the provisions of the Nuclear Waste Policy Act of 1982 as amended, DOE is required to do several things in the near future which impact NRC. Among them are: (1) Development of siting guidelines for a geologic repository at Yucca Mountain. NRC will be called upon to review and concur in the DOE siting guidelines. (2) As required under the Nuclear Waste Policy Act of 1982, NRC is currently preparing for its review of the Department of Energy's (DOE's) Site Recommendation Consideration Report to provide preliminary comments, concerning the extent to which the at-depth site characterization analysis and the waste form proposal for such site seem to be sufficient for inclusion in any [license] application, (3) If Yucca Mountain appears to be suitable as a site for a geologic repository, DOE is to submit a License Application to NRC in 2002. NRC then has the responsibility of issuing a final decision regarding issuance of a construction authorization within 3 years with a provision for a 1-year extension.

NRC's entire program of prelicensing consultation is focused on identifying issues early and providing DOE, prior to license application, the guidance needed to resolve the issues so that DOE can submit a complete and high-quality application that NRC can review and make its decision within this statutory time frame. NRC has several other repository program activities underway and they include the following:

Viability Assessment

Although not an explicit regulatory requirement, NRC reviewed DOE's Viability Assessment in order to provide DOE, Congress, and the public with NRC's views on progress made in development of a complete license application and to identify potential licensing vulnerabilities that could either preclude or pose a major risk to licensing. NRC completed its review of the DOE Viability Assessment and provided comments to DOE in July 1999.

Review of DOE Draft Environmental Impact Statement (DEIS) for Yucca Mountain

The NRC staff has completed the review of DOE's DEIS for Yucca Mountain and comments have been provided to the Commission. NRC will provide DOE with comments on the DEIS before the February 28, 2000 end of the public comment period. In accordance with the Nuclear Waste Policy Act of 1982, NRC's comments on the DEIS are to be included as part of any site recommendation of Yucca Mountain.

Total System Performance Assessment

NRC staff has continued development of a Total Performance Assessment Code for use in assessing performance of Yucca Mountain. Version 3.2 was issued and peer review was completed this year.

White Paper on Model Validation

In March 1999, staff from the NRC and Swedish Nuclear Power Inspectorate (SKI) prepared a White Paper on a model validation strategy that can be implemented in a regulatory environment. The document, designated NUREG-1636, should not be viewed as, and is not intended to be formal guidance or as a staff position on this matter. Rather, based on a review of the literature and previous experience in this area, this White Paper presents the views of members of the two organizations regarding how, and to what degree, validation might be accomplished in the models used to estimate the performance of HLW repositories. However, the two organizations may move jointly or individually to develop formal guidance or a staff position on this matter, at a later date.

Development of a Standard Review Plan for Yucca Mountain

NRC staff is working on development of a standard review plan for a license application for Yucca Mountain. The review plan is building on insights derived during formulation of key technical issues (KTIs) and Issue Resolution Status Reports (IRSR).

Low-Level Radioactive Waste

Currently the regulation of low-level radioactive waste disposal is being administered by the States under the NRC Agreement State Program. The principal on-going NRC activity is finalization of a Branch Technical Position on a performance assessment methodology for low-level radioactive waste disposal facilities. In 1997, the Branch Technical Position, designated NUREG-1573, was circulated for public comment. At present, the staff is responding to public comments and finalization of the Branch Technical Position is scheduled in calendar year 2000.

Nuclear Facilities Decommissioning

NRC has regulatory and oversight activities for decommissioning which involves safely removing a facility from service and reducing residual radioactivity to a level that permits the property to be released. This action is to be taken by a licensee before termination of the license. Some power reactor licensees have recently decided to shut down their facilities prematurely, before the expiration of the current operating licenses (e.g., Haddam Neck, Maine Yankee, Zion, etc.). These unexpected shutdowns have resulted in additional staff efforts in the areas of decommissioning inspections and in the licensing area to process license amendments and exemptions reducing regulatory requirements to correspond to the reduced risk posed by the permanently shutdown plants. In some cases, non-licensed facilities may also be required to reduce or stabilize contamination before sites are released. This activity comprises NRC's integrated regulation of the decontamination and decommissioning of facilities and sites associated with NRC-licensed activities, including associated research, rulemaking efforts, and the technical interface with the EPA to resolve issues of mutual interest.

In 1997 NRC published a final rule on Radiological Criteria for License Termination (10 CFR Part 20 Subpart E (the License Termination Rule) and currently there are several projects underway to provide guidance on implementation of the rule. They include the following:

Standard Review Plan

NRC is developing a standard review plan for use by NRC staff in reviewing and evaluating plans and information submitted by licensees under the provisions of the License Termination Rule. The Standard Review Plan is to be completed by the end of calendar year 2000.

Regulatory Guide on Demonstrating Compliance with the Radiological Criteria for License Termination (Draft Regulatory Guide 4006)

The regulatory guide is intended for use by licensees to comply with the License Termination Rule. It addresses the release from regulatory control of buildings and soil, but does not pertain to the release of contaminated equipment. It contains sections on dose modeling, methods for conducting final status surveys, ALARA analysis, and license termination under restricted conditions. The public comment period for the guide ended in November 1999 and the current schedule is for revision of the draft Guide based on the comments in calendar year 2000.

Multi-Agency Radiological Laboratory Analytical Protocols (MARLAP) Manual

The draft manual of radiological laboratory analytical protocols has been developed as multi-agency guidance for project managers and radioanalytical laboratories. Participants in the draft manual's development include the NRC, the EPA, the U.S. Department of Defense (DoD), the DOE, the National Institute of Standards and Technology (NIST), the U.S. Geological Survey (USGS), and the U.S. Food and Drug Administration (FDA), the Commonwealth of Kentucky, and the State of California The draft manual uses a performance-based approach and will support a wide range of data collection activities including decommissioning of nuclear facilities, remedial and removal actions, characterization and cleanups, compliance demonstration, environmental monitoring, and waste management activities. The draft manual is undergoing internal review by the participants with a planned completion in May 2000.

DandD Screen Model

The DanD Screen software is intended to be used as a screening tool for implementation of the technical dose criteria contained in NRC's License Termination Rule (10 CFR Part 20

Subpart E). DandD uses a generic approach in defining radiation exposure scenarios for residual radioactive contamination inside buildings, in soils and in ground water. To provide useful and defensible screening level calculations, the NRC has developed 'reasonably conservative' scenarios, pathway models, and parameter values, and has implemented this in DandD Screen. In the case of DandD, 'reasonably conservative' means that the calculated doses are much more likely to be overestimates of the actual dose rather than accurate estimates or underestimates, but at the same time are not necessarily worst case estimates. As a result, the scenarios and models implemented in DandD Screen are relatively simple. Currently a probabilistic version of DandD is under development and completion is scheduled in calendar year 2000.

RESRAD

The RESRAD (**Res**idual **Rad**ioactive Material) computer code was developed by Argonne National Laboratory for DOE to assist in DOE's decontamination and decommissioning of sites. As part of the development of implementation guidance supporting the License Termination Rule and development of a Standard Review Plan for License Termination, NRC contracted with Argonne National Laboratory to incorporate probability distributions of input parameters and a driver for conducting site-specific probabilistic radiation dose analysis into DOE's version of RESRAD and RESRAD-BUILD. These modules will contain user-friendly features based on a specially designed graphic-user interface. The modules will be tailored to running the RESRAD and RESRAD-BUILD codes to perform site-specific probabilistic dose assessments for supporting decontamination and decommissioning of radioactively contaminated sites due to NRC licensed operations in the past. The codes will be finalized for NRC regulatory applications by November 2000.

Joint NRC-EPA Sewage Sludge Radiological Study

Disposal of radioactive material into sewage systems and re-concentration of radioactivity in sewage sludge and incinerator ash became an issue in the 1980s with the discovery of elevated levels of radioactive materials in sewage sludge and incinerator ash at several sewage treatment plants. To address this problem, the NRC changed its regulations in 1991 to further restrict the discharge of radioactive material to sewage systems by NRC and Agreement State licensees. In response to a recommendation of the U.S. General Accounting Office, the NRC and the EPA are sponsoring a study of radionuclide levels in sewage sludge and ash from sewage treatment plants. The study is coordinated by a Federal interagency working group (the Interagency Steering Committee on Radiation Standards, ISCORS). The objectives of this joint NRC/EPA sewage sludge and ash study are to: (1) obtain data on the levels of radioactive material in sludge and ash at sewage treatment plants from across the country; (2) estimate the extent to which radioactive contamination comes from either NRC and State licensees or from naturally-occurring radioactivity; and (3) support potential rulemaking by NRC or EPA, if appropriate. The full study will involve sampling sludge and incinerator ash at up to 300 sewage treatment plants. The methods to be used for the full study were tested at nine sewage treatment sites. The survey and guidance document for sewage treatment plant operators is scheduled for completion in 2001.

Uranium Recovery Programs

NRC efforts for the uranium recovery program are governed by the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978. UMTRCA established two programs to protect public health and the environment: Title I and Title II. The Title I program established a joint federal/state funded program for remedial action at abandoned mill tailings sites, with ultimate Federal ownership under license from NRC. Under Title I NRC must evaluate the Department of Energy's (DOE's) designs and concur that DOE's actions meet standards set by the Environmental Protection Agency. For Title I, all surface remedial action was completed in FY99 and only reviews for DOE's groundwater remedial action program remain.

The Title II program deals with sites under license to the NRC or Agreement States. Under Title II the NRC has the authority to control radiological and non-radiological hazards and ensure that sites (NRC and Agreement State licensed) meet all applicable standards and requirements during operations and before termination of the license. The staff reviews Title II licensee plans for operation, reclamation, decommissioning, and ground-water corrective action; license applications and renewals; license conditions changes; and annual surety up-dates. The staff also prepares environmental assessments for certain licensing actions.

5. U.S. ENVIRONMENTAL PROTECTION AGENCY

WIPP

The Environmental Protection Agency (EPA) certified in May 1998 that the Waste Isolation Pilot Plant (WIPP) will comply with EPA’s regulations for disposal of transuranic radioactive waste. Two conditions of the certification call for EPA to inspect and approve DOE transuranic waste sites around the country before they may ship waste to the WIPP for disposal. Specifically, EPA must verify that the characterization of waste at the sites, including the application of quality assurance to waste characterization, complies with EPA’s regulations and the express commitments contained in DOE’s compliance application for the WIPP. Since certification, EPA has approved three sites to ship waste: Los Alamos National Laboratory, Idaho National Engineering and Environmental Laboratory, and the Rocky Flats Environmental Technology Site. As of February 2000, DOE has sent a total of 44 shipments to the WIPP from these sites. In addition to site inspections, EPA has a number of other ongoing regulatory responsibilities. The Agency monitors changes to the WIPP program in order to verify that the activities described in the compliance application still apply. If DOE alters the program significantly, EPA will initiate an informal rulemaking to consider the changes. Also, EPA must recertify every 5 years whether the WIPP continues to comply with EPA regulations. The first recertification will be completed in 2004. EPA’s authority to regulate the WIPP comes from the 1992 WIPP Land Withdrawal Act (as amended). General disposal standards for transuranic waste are codified at 40 CFR Part 191, and specific performance criteria for the WIPP are codified at 40 CFR Part 194. The conditions of the initial WIPP certification are codified as Appendix A to 40 CFR Part 194.

Yucca Mountain

The Energy Policy Act of 1992 gave EPA the authority to set site-specific environmental standards for Yucca Mountain, Nevada -- the potential deep geologic repository for spent nuclear fuel and high-level radioactive waste. The Act also directed EPA to contract with the National Academy of Sciences (NAS) to provide technical input into the standards. EPA received the report from NAS in 1995. The EPA held public meetings and received public comments on that report. The Agency has since been considering technical issues. In 1999, the Agency proposed the standards as 40 CFR Part 197, opened a public comment period, and held public hearings in four locations. Approximately 800 individual comments were received. The EPA plans to issue the final standards in 2000.

Low Activity Mixed Waste

As part of a larger set of EPA initiatives designed to ensure the safe disposal of all radioactive wastes, EPA is considering generally applicable environmental standards for the disposal of commercial low-activity mixed waste. Various facilities, including medical, educational, industrial, and nuclear power plants, generate commercial mixed waste in small amounts. For these facilities, current disposal options are unavailable or prohibitively expensive. EPA is working with the NRC, the licensing agency, to define conditions for safe disposal of some mixed waste-hazardous waste containing extremely low concentrations of radioactivity-in facilities originally designed for non-radioactive hazardous waste.

Technologically Enhanced, Naturally Occurring Radioactive Material (TENORM)

EPA is investigating sources of technologically enhanced, naturally occurring radioactive materials (TENORM) and the associated risks from exposure to these materials. In general terms, TENORM is material containing radionuclides that are present naturally in rocks, soils, water, and minerals and that have become concentrated and/or exposed to the accessible environment as a result of human activities such as manufacturing, water treatment, or mining operations. EPA recently compiled existing data on TENORM associated with the copper mining industry in the southwestern United States (EPA Report 402-R-99-002). The Agency is currently developing a report on uranium mining, which will address the volumes, types, and locations of TENORM associated with the industry, as well as risks from reasonable exposure scenarios. EPA is also supporting the U.S. DOE's participation in the NEA Working Group on Environmental Restoration of World Uranium Production Facilities.

R&D activities for spent fuel and high-level waste management in Korea

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Abstract. This paper focuses on the R&D activities for spent fuel and high-level waste management in the Republic of Korea (ROK) emphasizing on the proliferation resistant aspect. As the DUPIC fuel contains uranium, plutonium, and the highly radioactive fission products, it would be more resistance to proliferation than MOX fuel produced from Purex reprocessing.

1. STATUS OF NUCLEAR POWER GENERATION AND SPENT FUEL MANAGEMENT

In Korea, 16 commercial reactors (12 PWRs and 4 CANDUs) are currently in operation, two of which started commercial operation just this year, representing 13.7 GWe of nuclear generation capacity. In 1998, nuclear power generation occupied 41.7% of the total electricity generation. As of June 1998, the accumulated spent fuel accounts for 3 365 ton (1 888 ton PWR and 1 477 ton CANDU). The cumulative amount is expected to reach 11 000 ton by 2010. As for back-end fuel cycle in Korea, the government has not decided a definite policy yet, while waiting and seeing. However, as we have very rare energy resources in Korea, it is regarded that spent fuels will be also one of semi-domestic energy resources in the future.

2. THE DUPIC FUEL CYCLE

It is fortunate for us that we have two types of commercial reactors, that is, PWR and CANDU in Korea, because it allows us to develop the new fuel cycle, DUPIC (**D**irect **U**se of spent **P**WR fuel **I**n **C**ANDU reactors). This fuel cycle would make it possible for the PWR spent fuel to be reused as a CANDU fuel without utilizing conventional reprocessing. PWR spent fuel still contains a considerable amount of fissile material that can be more burned up in CANDU reactors. The fuel is to be manufactured by using a thermal/mechanical process only without employing a wet chemical process. This process is composed of disassembly of PWR spent fuel assemblies, fuel rod cutting, oxidative declading, oxidation and reduction, pelletization, sintering, and rod fabrication. Accordingly, manufactured DUPIC fuel contains uranium, plutonium, and most of the fission products in it, while a part of the fission products is removed as gases in the oxidation and sintering steps. Therefore, the fuel materials in whole fabrication process retain high radioactivity, maintaining proliferation-resistant characteristics like spent fuel. However, such reuse is limited to CANDU reactors only, and it requires a special consideration for handling due to its high radioactivity.

The technology development of DUPIC is currently being focused on the reactor physics assessment, manufacturing of prototypical fuel elements, fuel performance test, and also on the safeguards. KAERI, LANL, AECL, and IAEA are now actively involved in this project in the areas of process development and analysis, safeguards technology evaluation, and safeguards system establishment. According to the present R&D plan, several DUPIC fuel rods will be manufactured and then loaded in the HANARO research reactor for irradiation test by 2002. Recently, simulated DUPIC fuel pellets were fabricated using enriched UO_2 with ZrO_2 and M_0O_3 etc., and then irradiated for two months in the Hanaro research reactor. This test was carried out to estimate in-core performance of DUPIC fuel, and also to improve the performance evaluation codes.

3. FACTORS CHARACTERIZING FACTORS BACK END FUEL CYCLE

Though a lot of factors should be considered to evaluate various back end fuel cycles, several key factors in general can be extracted as follows: proliferation-resistance, economy. technological merits, usefulness of final product, and advantages in waste disposal, etc. The technological merits here includes various advantages in process technology such as technological maturity, process simplicity, safety on criticality, stability against radiation, reduction in secondary waste generation, and so on. A certain weighting parameter can be given to each factor in Table I in order to evaluate fuel cycles in more detail. However, it will heavily depend on the back-end fuel cycle policy or nuclear technology situation of the corresponding country.

4. COMPARISON OF NOTABLE NUCLEAR FUEL CYCLES

A few promising fuel cycles including DUPIC has been analyzed on the basis of the key factors described above as shown in Table I. As shown in this table, the PUREX process has the most superiority in the aspect of product usefulness because both highly pure uranium and plutonium are obtained in this process as final products, which can give a flexibility in the manufacturing of fuel. Various kinds of fuels can be made with the single component of uranium or plutonium, and a mixture of uranium, plutonium, and/or other components. It is of great advantage that various types of nuclear fuels can be manufactured with the final products complying with the reactor type or fuel requirements. However, it is known that reprocessing by the PUREX process is currently neither economical nor proliferation-resistant compared with the direct disposal of spent fuels.

As in DUPIC fuel, however, it still contains uranium, plutonium and highly radioactive fission product, it will be possible to exclude the risk of proliferation. Though an economic evaluation has not been made in depth yet, KAERI has done a preliminary evaluation for the economy of DUPIC fuel cycle to reveal a positive sign in the whole fuel cycle cost. The only drawback currently being pointed out is the high radioactivity of the fuel, which would work as an obstacle in handling of the fuel.

TABLE I. COMPARISON OF FUEL CYCLES

Factor	DUPIC	PUREX	PYRO-ELECTRO
Final product			
• Economy			
• Proliferation-resistance	✧		
• Technological merits			
- technological maturity		✧	
- process simplicity	✧		✧
- safety on criticality	✧		✧
- stability against radiation	✧		✧
- secondary waste	✧		
• Product usefulness		✧	
• Advantages in waste disposal	✧		✶

✧ favorable.
✶ favorable when used as partitioning & transmutation cycle.

The pyroprocess for treating metal fuel was already developed in 1980's in ANL as a part of IFR fuel cycle. Recently, the scope of pyroprocess has been further extended to oxide and nitride fuels in Japan and Russia. It is expected that the pyroprocess may be more economical rather than the PUREX process because of its simplicity in process and equipment. In addition, the mixture of transuranium elements can be obtained as a final product in this process, which enables that a transuranium fuel for nuclear transmutation can be made through this process. When certain weighing parameters are given to the factors in Table I, the evaluation results will depend on the values of parameters. Let's take an example of one extreme case. If a certain country places an exclusive weighting factor of 1.0 only on the factor of proliferation-resistance, then the DUPIC fuel cycle will turn out to be the best choice.

5. R&D FOR HIGH-LEVEL WASTE DISPOSAL

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

Performance Assessment and Disposal System Development

- Concept development of a reference disposal system and its optimization,
- Development of an integrated performance assessment code including unit models for nuclide transport both in near-field and far-field barriers, and
- 1,-,Development of safety assessment technologies, and
- 'Study on the geo-mechanical characteristics of rock masses around a repository, such as hydraulic and mechanical couplings

Geo-environmental Science

- Delineation of unstable tectonic zones throughout the geological history,
- Characterization of groundwater flow in different geomorphic conditions,
- Evaluation of deep groundwater chemistry, and
- Establishment of site characterization methods

Engineered Barriers

- Characterization of potential domestic buffer materials,
- Development of a reference buffer material,
- Development of a disposal container for HLW, and
- Long-term behavior of waste forms and container materials under repository conditions

Radionuclide Behavior in the Underground Environment

- Sorption experiments of long-lived nuclides on single minerals under various solution conditions,
- Experimental and mechanistic sorption modeling, and
- Fracture migration experiments in artificially and naturally fractured rocks

International Collaborations

KAERI has been participating in projects and meetings organized by OECD/NEA, for example, the ASARR (Analogue Study at the Alligator River Region) project, which has been managed by Australian Nuclear Science and Technology Organization (ANSTO). KAERI has also been participating in two of the IAEA's CRP programs, titled "The extrapolation of short-term observations to time periods for isolation of long lived radioactive wastes" and "Biosphere modeling and assessment methods". In addition, for the development of the Korean disposal concept and the safety assessment, KAERI has been collaborating with foreign organizations such as AEA Tech. (U.K), AECL (Canada), SKB (Sweden) and SNL (USA).

6. CONCLUDING REMARKS

Even though a completely different fuel cycle technology may be born in the future, it will not be able to stand without regarding the factors described in this study. Proliferation-resistance, economy, technological merits, usefulness of final product, advantages in waste disposal, and so on will become the important factors for the evaluation of back end fuel cycles. A certain weighting parameter can also be given to each key factor in Table I in order to evaluate the fuel cycles more in depth. In Korea, the greatest priority for the choice of back end fuel cycle will be placed on the proliferation-resistance among the various factors in Table I, and thus the DUPIC cycle is being taken into account as a prominent candidate fuel cycle.

Spent fuel management in Japan – Facts and prospects

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Abstract. This paper discusses recent developments and future issues related to spent fuel management in Japan. With increasing pressure of spent fuel discharge from the power plants in operation and, in contrast, uncertainties in their processing and management services, spent fuel storage in short and medium terms has been receiving the highest priority in nuclear policy discussions in Japan. While small-scale interim storage devices, as well as capacity expansion (re-racking, etc.) and shared uses of existing devices, are introduced at number of power stations, large scale AFR (away from reactor) "Storage of Recycle Fuel Resources" is expected to come in a medium and long-run. Commercial operation of "Storage of Recycle Fuel Resources" is allowed its way, as the bill of amendment to the law for regulation of nuclear power reactors and other nuclear-related activities has passed in the Diet. In the meantime, the Atomic Energy Commission has launched working group discussions for revision of "The Long-term Program of Research, Development and Utilization of Nuclear Energy" to be completed in 2000. This revision is hoped to set up a stage of national debate of nuclear policy, which might lead to fill conceptual gaps between bodies promoting nuclear development and general public. The author's attempt to illustrate the role of storage in spent fuel management is also presented from a theoretical point of view.

1. INTRODUCTION

With years of nuclear operation, increasing pressure of spent fuel arising is receiving a serious attention in Japan in order for those plants keep their operation without causing an overflow of built-in storage pools. There are other key constraints with spent fuel management, such as the delay of Japan's first reprocessing plant construction, which was reported lately. In a more general context, several accidents in the nuclear facilities, such as sodium leak at Monju fast reactor, asphalt explosion at PNC Tokai reprocessing plant and the latest criticality accident at JCO conversion plant, have made the public more and more suspicious and distrustful of nuclear establishment in general. Although the causes and measures with the last accident is still under investigation, we should recognize such conceptual gaps between the nuclear establishments and the general public as an important factor to influence nuclear development pathways.

This paper discusses about the present situation and future prospects of spent fuel management in Japan. An attempt to describe optimal conditions of spent fuel management is formulated as an analytic framework of spent fuel management, with a special emphasis on factors and their trade-off relations influencing choices of spent fuel storage.

2. CURRENT STATUS OF SPENT FUEL MANAGEMENT IN JAPAN

Table I shows the present status of spent fuel accumulation at the end of 1996 and 1997, respectively. A few notes are found in the table. Japan's nuclear power generation, with a total capacity of 45 GWe with 52 reactor units (see Figure 1), discharges about 900 MTU (metric ton of uranium) of spent fuel per year. This spent fuel arising primarily accumulates in the reactor pools built-in those power reactor units. As discharged spent fuels have accumulated in those reactor pools, some nuclear power stations are forced to supplement storage capacity in order to avoid overflow of the reactor pools. At Fukushima Daiichi Nuclear Power Station (NPS, hereafter) of Tokyo Electric Power Company (Tokyo EPCo), a

1,120MTU water pool storage facility has been implemented in 1997, as well as a dry metal cask storage capability. These facilities are shown in Photo 1 and also reflected in Table I as an increase of storage capacity. Several other stations are also found with additions of storage capacity by re-racking of storage pools.

TABLE I. SPENT NUCLEAR FUEL STORE AT REACTORS IN JAPAN

As of end 1996

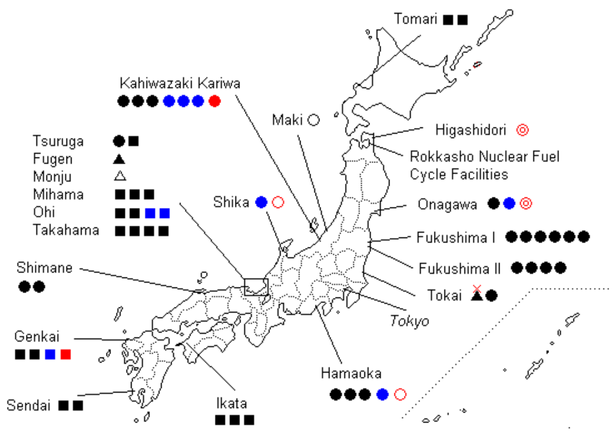
Utility Company	NPS	Loading in Core	Fuel per Batch	SF in Store	Storage Capacity
Hokkaido	Tomari	100	30	160	420
Tohoku	Onagawa	160	40	80	370
Tokyo	Fukushima-1	580	150	740	780
	Fukushima-2	520	140	940	1110
	Kashiwazaki-Kariwa	810	210	770	1420
Chubu	Hamaoka	420	110	420	820
Hokuriku	Shika	60	20	40	100
Kansai	Mihama	160	50	180	300
	Takahama	290	100	540	1100
	Ohi	360	120	370	840
Chugoku	Shimane	170	40	180	390
Shikoku	Ikata	170	60	200	470
Kyushu	Genkai	190	60	180	520
	Sendai	140	50	460	570
JAPCo	Tsuruga	140	40	330	450
	Tokai-2	130	30	170	260
Total		4390	1250	5750	9920

As of end 1997

Utility Company	NPS	Loading in Core	Fuel per Batch	SF in Store	Storage Capacity
Hokkaido	Tomari	100	30	170(+10)	420
Tohoku	Onagawa	160	40	100(+20)	370
Tokyo	Fukushima-1	580	150	790(+50)	1950(+1170)
	Fukushima-2	520	140	1000(+60)	1220(+110)
	Kashiwazaki Kariwa *1	960(+150)	250(+40)	910(+140)	1640(+220)
Chubu	Hamaoka	420	110	500(+80)	860(+40)
Hokuriku	Shika	60	20	20(-20)	100
Kansai	Mihama	160	50	170(-10)	300
	Takahama	290	100	620(+80)	1100
	Ohi	360	120	440(+70)	840
Chugoku	Shimane	170	40	200(+20)	390
Shikoku	Ikata	170	60	240(+40)	470
Kyushu	Genkai *2	270(+80)	100(+40)	240(+60)	1060(+540)
	Sendai	140	50	470(+10)	570
JAPCo	Tsuruga	140	40	360(+30)	450
	Tokai-2	130	30	170(+-0)	260
Total		4630(+240)	1330(+80)	6400(+650)	12000(+2080)

Sources: MITI (1996,1997, 1998)
Notes: *1 7th Unit of Kashiwazaki-Kariwa NPP started commercial operation on July 2nd, 1997.
*2 4th Unit of Genkai NPP started commercial operation on July 25th, 1997.

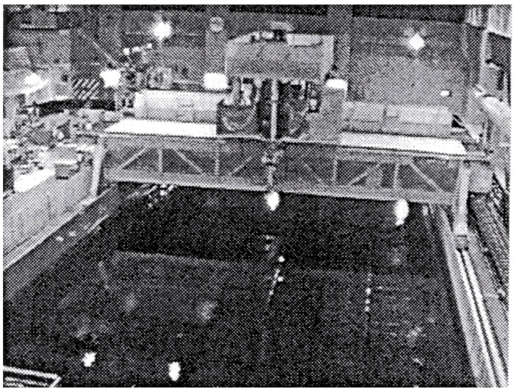
Nuclear Power Stations in Japan 1991-1998



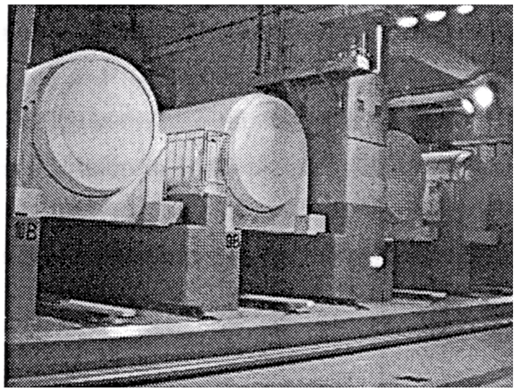
Reactor Type	In Operation	Under Construction	Planned	Shutdown
PWR	■	◇	□	
BWR	●	◎	○	
GCR, FBR	▲	△		✗

Total Capacity	In Operation		Under Construction		Planned		Total	
	Units	GWe	Units	GWe	Units	GWe	Units	GWe
Sep. 1991	41	32	11	12	3	2	55	36
Dec. 1996	51	43	4	3	2	2	57	48
Dec. 1998	52	44.9	2	1.83	3	3.56	56	50.3

Fig. 1. Nuclear power stations in Japan with illustration of changes 1991-1998.



(a) Water pool storage.



(b) Dry metal cask storage.

Photo 1. AR storage devices at Fukushima Daiichi NPS (source: TEPCo.).

TABLE II. STORAGE CAPACITY ENHANCEMENT MEASURES TO BE IMPLEMENTED IN NPSS IN JAPAN (Source: Ohnishi (1998)) (as of 1998/2)

EPCo.	NPS	Date of Application	Date of Permission	Description	Completion
1. Facilities Under					
EPCo.	NPS	Date of Application	Date of Permission	Description	Completion
Kansai	Ohi	1994/10/11	1995/4/6	Shared use of exisiting storage capacities	FY1997
Kyusyu	Genkai	1995/6/20	1996/3/12	Shared use of existing storage capacities	FY1997
Shikoku	Ikata	1995/12/6	1996/7/10	Shared use of exisiting storage capacities	FY1998
Touhoku	Onagawa	1997/2/18	1997/8/28	Shared use of exisiting storage capacities	FY1999
2. Facilities with Application					
Kansai	Ohi	1997/8/1		Additional Shared use of exisiting storage capacities	FY2001
JAPCo.	Tsuruga	1997/8/1		Re-racking	FY1999-2000
Kyusyu	Sendai	1997/9/17		Re-racking	FY1999-2000
JAPCo.	Tokai-2	1997/9/17		Metal cask	FY2001
Chugoku	Shimane	1997/11/25		Re-racking Shared use of exisiting storage capacities	FY2002
Kansai	Mihama	1998/2/3		Re-racking Shared use of exisiting storage capacities	FY2001
Chubu	Hamaoka	1998/2/19		Additional racks. Shared use of existing strorage capacities	FY2002 FY2000

Such "At Reactor" (AR) storage enhancement is planned also at number of NPSs, as listed in Table II, according to their risks of overflow of existing storage pools. Many of them are shared uses of existing storage capacities, which are none of increase of capacity actually. Several others may include re-racking, additional pools and dry casks stores.

3. FUTURE PROSPECTS OF SPENT FUEL MANAGEMENT

3.1. Discharge of Spent Nuclear Fuel and Potential Demand for Storage

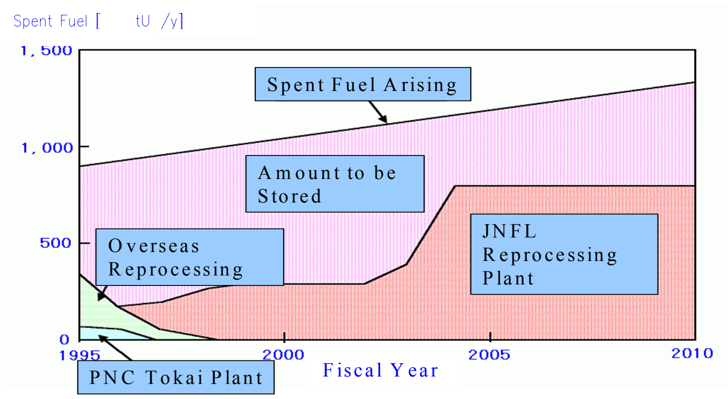
Future prospects of spent fuel management are influenced largely by the following factors:

1. The spent fuel reprocessing plant under construction at Rokkasho-mura of Aomori Prefecture; when it starts its operation and how successfully.
2. One-time full-core discharge upon decommissioning of reactor units foreseen from 2010, while built-in storage pool is also dismantled at a certain stage of decommissioning.

Storage demand all over Japan may depend heavily on these two factors; annual amount of discharge and processing. Even if the Rokkasho reprocessing plant is successfully operated at its capacity of 800MTU/year, it cannot receive the whole discharge of Japan's NPSs every year, neither the past discharges. As an official view of the future prospect, Table III and Figure 2 have been presented by the government.

TABLE III. THE OFFICIAL PROSPECT OF SPENT NUCLEAR FUEL DISCHARGE FROM JAPANESE NPSS [TU] (Source: Nuclear Pocket Book 1997 edition, p.186)

Fisical Year	Annual Discharge	Cumulative Discharge
1996	900	12 000
2000	1 000	16 000
2005	1 200	22 000
2010	1 300	28 000



Assumptions:

Fiscal Year	1995	2000	2010
Nuclear Power Generation Capacity [GWe]	4150	4500	7000
Annual Discharge of Spent Fuel	900 – 1300 tU/y		
PNC Tokai Reprocessing Plant	Receipt of spent fuel ends in 1996.		
JNFL Reprocessing Plant	Receipt of spent fuel is started in 1997. Operation starts at 800tU/y in 2003.		

Fig. 2. Prospect of spent fuel arising and management. (souce: MITI).

JNFL (Japan Nuclear Fuel Limited) made an announcement in April 1999 that the construction work of the Rokkasho reprocessing plant is further delayed, so that the plant will start its operation in July 2005. The measures listed in Table II seem capable to absorb influences of this delay, but it should be noted that there will remain risks of overflow of reactor pools if further delay or any changes would occur.

In the long run, it is obvious that large-scale storage devices are needed. This is particularly true after 2010 when the first commercial LWR plant expires its 40-years lifetime. After then, series of LWR plants would be shut down, which mean on one hand a large amount of one-time discharge of spent fuel, and at the same time loss of storage capacity of the reactor pools

Under this circumstance, the Sub-committee for Nuclear Energy of the Council for Comprehensive Energy Policy, an advisory body to Ministry for International Trade and Industry (MITI), has proposed to develop AFR (Away From Reactor) storage of "Recycle Fuel Resources" of spent nuclear fuel in its interim report published in June 1998 (Sub-Committee for Nuclear Energy, 1998). As a follow-up of the report, MITI submitted a bill to amend the nuclear regulation law to the parliament, which has been passed the Diet and will be enforced in June 2000. In the bill, it is proposed that any private ventures, besides power utilities or existing nuclear fuel cycle service providers, are eligible to apply a license to do the storage. Based on the existing amount of spent fuel stocks and projections to the future, it is expected that the Japanese nuclear industry should prepare a storage capacity at around 10 000 to 15 000 MTU in the near future, e.g. by 2030.

Special attentions should be given to the plutonium utilization in LWR plants. Table IV specifies current plans to use MOx fuel in LWR plants. The spent MOx fuel to be discharged will have to be stored, as the Rokkasho plant is not licensed for the type of spent fuel with higher generation of heat and radiation.

TABLE IV. NUMBERS OF LWR UNITS WITH PLANS TO USE MIXED-OXIDE (MOX) FUEL IN JAPAN (Source: White Paper of Nuclear Power, 1998 edition)

EPCo	until 2000	early 2000s	until 2010
Tokyo	1 in 1999 (*1) 1 in 2000 (*2) [2]	1 [3]	0-1 [3-4]
Kansai	1 in 1999 (*3) 1 in 2000 (*4) [2]	[2]	1-2 [3-4]
Chubu		1 [1]	[1]
Kyusyu		1 [1]	[1]
JAPCo		2 [2]	[2]
Hokkaido, Touhoku, Hokuriku, Chugoku, Shikoku			1 for each EPCo [5]
EPDC			1 (*5) [1]
Total	4 [4]	5 [9]	7-9 [16-18]

Notes: Numbers in [ ] are cumulative.
*1: Unit No.3, Fukushima-1 NPS.
*2: Unit No.3, Kashiwazaki-Kariwa NPS.
*3: Unit No.4, Takahama NPS.
*4: Unit No.3, Takahama NPS.
*5: Ohma NPS of EPDC (Electric Power Development Corp.) is designed with a MOx fuel core, planned to commence in 2007.

3.2. Economics of Spent Fuel Storage

Figure 3 shows an engineering-economic comparison among storage technology options available and adjusted in Japanese circumstances. Although the water pool is mature with plenty of experiences with existing reactor pools, its economics may suffer from high capital investments as well as high O&M costs due to requirements of forced circulation and quality control of cooling water.

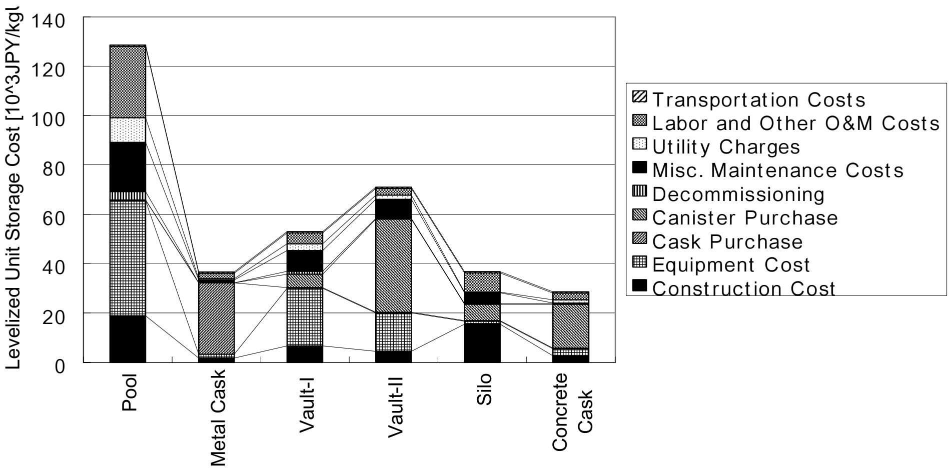


Fig. 3. Comparison of levelized unit storage costs (source: Saegusa (1998)).

The metal cask has been receiving highest priority in implementing storage facilities in a short and medium run, with its superb economics compared to water pool. With longer perspective, research is ongoing for the other dry storage technologies, aiming at better economic performances. Key issues of research include:

- Long-term integrity of concrete materials,
- Long-term integrity of canisters, and
- Safety standards in O&M, especially unloading/loading for transportation.

4. ROLE OF STORAGE: A THEORETICAL VIEW

Storage has an important role to secure time for research and development. The following is an attempt of mathematical formulation to capture cost-benefit relations of storage and R&D in the strategic analysis of spent fuel management (Nagano, 1998a, 1998b, 1999.)

4.1. Fundamental Formulation and Solution

Suppose that a unit (1 MTU) of spent fuel is discharged from a reactor plant, which is to be stored until it will be reprocessed or disposed. From the reprocessing, corresponding amount of Pu will be recovered, which will then be fabricated as mixed oxide (MOX) fuel and reloaded to the reactor or another. The problem to be addressed here is to optimize the duration of storage to maximize total utility function, i.e.;

$$\begin{aligned} TU = & - f_1(x) - e^{-rx} \cdot f_r \cdot (1-i_r)^x - e^{-rx} \cdot f_2(y) \cdot (1-i_2)^x \\ & - e^{-r(x+y)} \cdot f_m \cdot (1-i_m)^{(x+y)} - e^{-r(x+y)} \cdot f_3(z) \cdot (1-i_3)^{(x+y)} + e^{-rT} \cdot U \to \max . \end{aligned} \quad (1)$$

$$s.t. \quad T = x + y + z \quad (2)$$

where,

TU : Total utility at net present value at the year of the spent fuel discharge,
$f_1(x)$: Cost of spent fuel storage for x years,
f_r : Cost of reprocessing,
i_r : Rate of reprocessing cost reduction due to 1 year addition of R&D,
$f_2(y)$: Cost of storage of the corresponding amount of Pu for y years,
i_2 : Rate of Pu storage cost reduction due to 1 year addition of R&D,
f_m : Cost of MOx fuel fabrication with the corresponding amount of Pu,

i_m : Rate of MOx fabrication cost reduction due to 1 year addition of R&D,
$f_3(z)$: Cost of storage of the corresponding amount of MOx fuel for z years,
i_3 : Rate of MOx fuel storage cost reduction due to 1 year addition of R&D,
U : Utility obtained from the Mox fuel burning at T years from the discharge of the originated spent fuel,
r : Discount rate.

At this moment, the utility of Pu burning (either by FBR or light water reactor (LWR)) is not very clear, as implied by major countries' withdrawal from FBR development. If U is assumed as zero for simplification, then the original utility maximization problem turns to the total cost minimization. For another simplification, let various improvement rates i_x equal to i uniformly. Then, the formula (1) turns to formula (3).

$$\begin{aligned} TU = &- f_1(x) - e^{-(r+i)x} \cdot f_r - e^{-(r+i)x} \cdot f_2(y) - e^{-(r+i)(x+y)} \cdot f_m \\ &- e^{-(r+i)(x+y)} \cdot f_3(z) + e^{-rT} \cdot U \rightarrow \max. \end{aligned} \tag{3}$$

The assumption of uniform rate of technology improvement is translated as increase of discount rate from r to (r+i) superficially. However, the nature of technology improvement is not merely a function of time spent for R&D but indeed also influenced by the experiences accumulated throughout research, development and commercialization. This is one of the largest issues, among all in this report, that needs further refinement. Now to solve to problem, the *laglange* coefficient λ is introduced.

$$I = TU - \lambda(T - x - y - z) \rightarrow \max. \tag{4}$$

Essentially, the optimality condition of this problem is derived as follows:

$$f_1'(x) = e^{-(r+i)x} \cdot f_2'(y) + (r+i)e^{-(r+i)x} \cdot (f_r + f_2(y)) \tag{5}$$

If one can assume that Pu storage is always too costly, then;

$$f_1'(x) = (r+i)e^{-(r+i)x} \cdot f_r \tag{6}$$

The equations (5) or (6) is the fundamental form of condition to determine the optimal duration of each of the storage options, which makes the following two equal;

- the increase of cost of storage due to 1 year prolongation of storage duration, and
- the decrease of the net present value of total cost of all processes after the storage due to 1 year delay caused by prolonged storage.

The latter factor consists of a change of net present value due to 1 year discounting and improvement resulted from R&D efforts taken during the storage duration. Note that the improvement in this notation should be defined in a broad sense, so that those reduction of institutional and transaction costs, such as improved public awareness and acceptance, more efficient and appropriate planning into the future, should be recognized as parts of those technology improvement. Also note that the above formulation does apply also for the case of direct disposal, simply replacing suffix r for reprocessing with d for disposal.

4.2. A Numerical Example

Based on the published cost data (OECD/NEA, 1989 and 1994) shown in Table V, the author tried to solve the original problem numerically. The result is shown in Figure 4. If discount rate *r* (added with the uniform rate of technology improvement *i*) equals to zero, there is no reason to postpone, and the optimal strategy is to skip storage and go immediately to the next step, either reprocessing or disposal. In the cases of positive discount rate and technology improvement, optimal storage duration could be obtained which minimizes the total system cost.

TABLE V. COST DATA USED FOR THE NUMERICAL ILLUSTRATION

Item	Data	Source
SF Storage	(51+5/ yr) [$/ kgHM]	OECD/ NEA(1994)*
Reprocessing	720 [$/ kgU]	OECD/ NEA(1994)
Pu Storage	1 [$/ gPu/ yr]	OECD/ NEA(1989)
MOx Fabrication	800 [$/ kgHM]	OECD/ NEA(1989)
MOx Storage	(omitted)	
(SF Disposal)	610 [$/ kgHM]	OECD/ NEA(1994)

* Adopted from OECD/ NEA(1985) and converted.

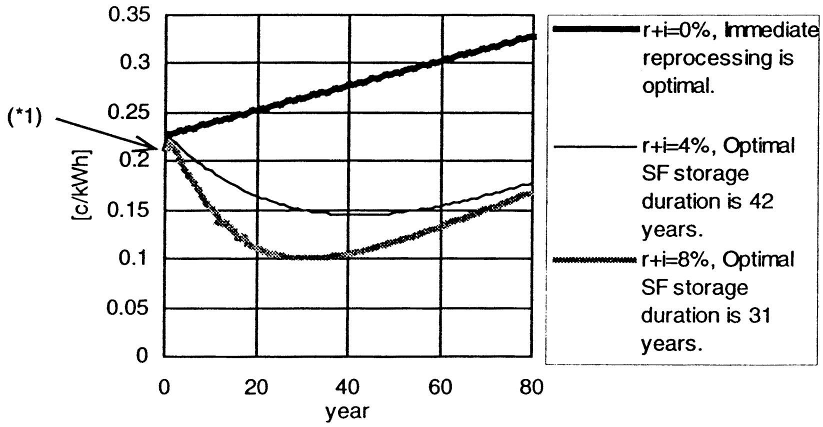


FIG.. 4: An example of optimal spent fuel storage duration based on the problem definition and the cost data (Source: Nagano, 1998b).

(Note) The values plotted here is the total cost as the objective function defined as neglecting *U* in the formula (3) divided by the kWh the fuel under consideration had generated.

*1: When t=0, the value is 0.2137.

It should be noted that these characteristics are highly dependent to the functional form how storage duration influences to the storage cost. In the numerical example shown in Figure 3, the net present value of storage cost is assumed to be linear with duration, which means the marginal increment with 1 year prolongation of storage is kept constant. In this case, as present value of the benefit of postponing reprocessing and other steps is large at t=0 and then declining in proportion to e^{-rt}, an optimal duration can be found. If the storage cost function is proportional to e^{-rt}, no optimal duration is found and the optimal strategy is determined simply by *U* whether immediate recycle or unlimited storage.

Under more realistic circumstances, the storage cost function may not be continuous against storage duration, as replacement of large equipment or even the whole facility may cause a large cost addition occasionally. If such a marginal storage cost increment is larger than the benefit of postponement, such time point with a large marginal storage cost increment is a chance to stop the storage and proceed to the subsequent processes.

Moreover, if there is realized a technological breakthrough of reprocessing and other processes, the cost reduction through it might be large enough to justify immediate termination of storage and move to those subsequent processes.

5. CONCLUDING REMARK

Although the policy debate on back-end has made remarkable progresses during the last couple of years, there might still be a need for re-definition of an overall picture, on which the public may really be able to decide whether and how the nuclear system operation has to be done. In such a comprehensive picture, a strategic use of spent fuel storage will for sure play an important role to enable advanced technology R&D while keeping reliable operation of the nuclear energy system.

Japan Nuclear Cycle Development Institute (JNC), reorganized from the PNC, issued a final report on May 26, 1999 on the accident of its reprocessing plant in Tokai-mura, wishing its sooner restart. The plant stopped due to an explosion accident in March 1997, and the report summarized the detailed investigation of the accident, proposed measures to prevent similar occurrences of accidents and reassurance of degree of safety and management of the plant. However, although the plant has been designated to supply plutonium for Monju, a prototype FBR, which has been also suspended since the sodium leak accident in December 1995. There is nothing announced until now when Monju is planned to restart. On September 30, a criticality accident happened at JCO Tokai Works, when JNC planned to submit the application of re-opening the Tokai plant but forced to retain until the situation is settled. Accidents may make up chains, but accountability of policy does not.

The Atomic Energy Commission, knowing necessity to accommodate recent changes fully into the national policy, has launched working group discussions for revision of “The Long-term Program of Research, Development and Utilization of Nuclear Energy” to be completed in 2000. This “Long-term Program” has provided the overall policy framework for research, development and utilization of nuclear energy. This revision must not be a mere re-decoration of the older version published in 1994, but is hoped to set up a stage of national debate of nuclear development policy, which might lead to fill conceptual gaps between bodies promoting nuclear development and general public.

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Back-end nuclear fuel cycle strategy: The approaches in Ukraine

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Abstract. Ukraine has 14 nuclear units in operation and 4 units more under construction. Now in Ukraine a share of installed nuclear capacity in total installed capacity is essential and it is planned to increase it further. In this connection a spent nuclear fuel management in Ukraine for the current period and future is becoming important in a nuclear fuel cycle. A current situation in relation to the spent nuclear fuel management in Ukraine is described in the paper. It is reviewed: legislative basis for a spent nuclear fuel management strategy; an assessment for a spent fuel growth; the national possibilities for the spent fuel management; an organization chart for a spent nuclear fuel management, etc. Some factors that can determine a "long-term spent fuel management strategy in Ukraine are in the conclusion.

1. INTRODUCTION

Practically, Ukraine is located in geographical centre of Europe, bordering with Russia. Belarus, Poland, Slovakia, Hungary and Romania. Covering an area of 603.7 thousand square kilometres, Ukraine is the second largest European country after Russia by territory. There are about 50 million people in Ukraine. The population density is 85 people per a square kilometre in average.

At present Ukrainian power plants have total installed capacity of 53.9 GW(e), including 12.8 GW(e) of nuclear capacity (23.8 % of the total one).

It must be noted that in Ukraine the nuclear electricity production share ill total electricity production is constantly increasing (*Fig. 1*). Taking into account a current economical crisis in Ukraine the nuclear power engineering carries important stabilizing role in a society life.

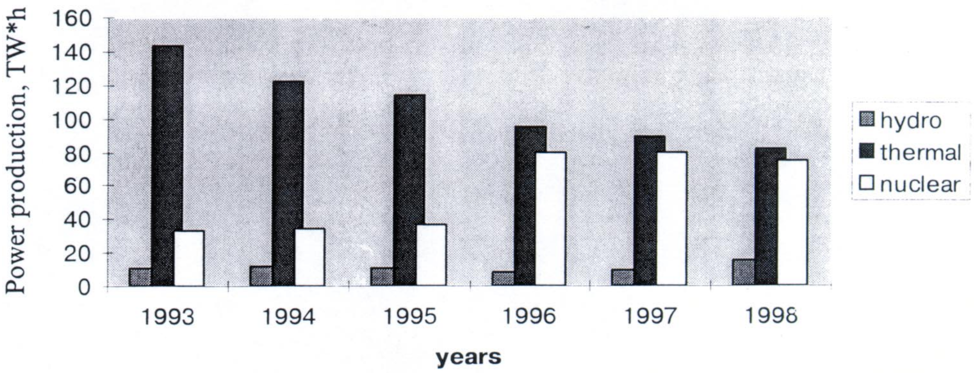


FIG. 1. Electric power production in Ukraine.

In 1998 the total electricity production was 173.0 TW*h (3% decreasing against year 1997), including

- thermal electricity: 47.3%,
- nuclear electricity: 43.5%,
- hydro electricity: 9.2%.

In Ukraine energy policy is determined by a set of the laws and its major priorities - by the National Power Programme until the Year 2010. According to the National Programme (the total electricity generation will reach 280 TW*h by the year 2010 (43 % growth compared to 1995), including 95.7 TW*h of the nuclear electricity generation.

By the prognosis of the Ukrainian National Academy of Sciences, in perspective the Ukrainian power industry's needs will be satisfied by coal (50%), uranium (40 %) and other sources (10 %).

2. NUCLEAR POWER IN UKRAINE: SOME FEATURES

Ukrainian nuclear power industry was set up in the middle of 70s. The first nuclear power unit was commissioned in 1977. In average a unit per year was put into operation till 1990. Since 1977 there were commissioned 16 nuclear power units in Ukraine. The Chernobyl NPP's unit 4 was destroyed in 1986 as a result of Chernobyl disaster. 2 units more were shut down in 1991 and 1996 on Chernobyl NPP.

The Figure 2 depicts the nuclear power growth during last two decades in Ukraine.

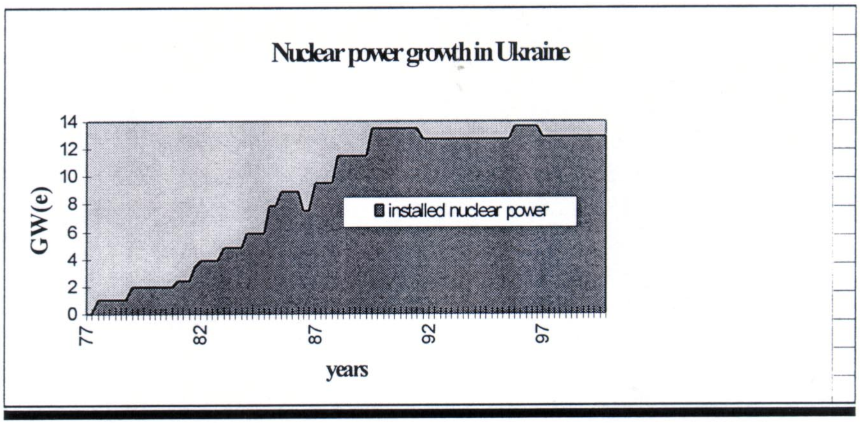


FIG 2. Nuclear power growth in Ukraine.

At present there are 14 nuclear power units in operation in Ukraine (1 unit more was commissioned in 1995) including:

- 1 RBMK-1000 unit (Chernobyl NPP). RBMK - channel-type high-power reactor of 1000 MW(e),
- 2 VVER-440 units (Rivneh NPP, mods. 213). VVER model 213 is water-moderated/cooled reactor of the second generation with an improved containment and effective emergency system;
- 11 VVER-1000 units (6 - Zaporizhia NPP, mods. V-320; 3 - South-Ukrainian NPP. mods. V-302, V-338 and V-320; 1 - Khmelnitsky NPP, mod. V-320; 1 - Rivneh NPP, mod. V-320). VVER model 320 is water-moderated/coo led reactor of the second generation with a full reinforce concrete containment and emergency core cooling.

Four more VVER-1000 units are under construction. Two of them are completed about 80-85 % and other two units - 30-35%.

3. SPENT NUCLEAR FUEL IN UKRAINE

Reactor operation modes/running spent fuel status. The VVER reactors allow refuelling only during the reactor shutdowns. Any VVER reactor is shut once a year for fuel reloading and planned maintenance, if its operation is normal. As for the RBMK reactor operation it allows continuous on-load refuelling.

Currently in Ukraine it is about generated:

- 11.2 Mg of spent fuel (SF) per a year by a VVER-440 reactor;
- 20 Mg of spent fuel per a year by a VVER- 1000 reactor;
- 44 Mg of spent fuel per a year by a RBMK- 1000 reactor.

The annual spent fuel currently generated by all Ukrainian nuclear power units amounts to 290 Mg of heavy metal (HM).

Since the start of their operation all Ukrainian nuclear units have generated 4 932 Mg of spent fuel in total. Nowadays (as of 01.01.99) 3403 Mg of spent fuel is stored on the Ukrainian NPPs' sites.

Some characteristics of actual Ukrainian NPPs' spent fuel are depicted in the following table.

Criteria/Parameter		Measure	VVER-440	VVER-1000	RBMK-1000
HM weight per a fuel assembly		kg	$\leq$120	429 (2-year cycle) 402 (3-year cycle) 435 (3-year cycle, advanced fuel)	115
Enrichment	initial	wt. %	1.6 - 4.4	1.6 - 4.4	1.8 - 2.4
	residual		1.1 - 1.5	0.8 - 1.3	0.5 - 0.6
Burnup (operating)	max.	MWd/kg U	46-47 (for 4.4% fuel)	48.8 (for 3.6+4.4;4% fuel)	21 (for 2.4% fuel
	average		40-42 (for 4.4% fuel)	41 (for 3.6+4.4;4% fuel	
Cooling time before transport		years	$\geq$ 3	$\geq$ 3	$\geq$ 1.5
Pu (fissile) contents after fuel cycle lifetime		kg/tU	$\leq$ 8.0	$\leq$ 8.0	$\leq$ 5.5
Residual heat of discharged fuel after	1 year	kW/SFA	1.2	6.0	0.6
	3 years		0.3	1.7	0.17

3.1. The former spent fuel management scheme [1,2]/original capacities for SF storage

Till 1991 Ukraine was a part of the Soviet Union as one of its republics. A spent fuel management that has been developed historically in the USSR covered the Ukrainian NPPs.

In the former USSR a spent nuclear fuel management concept, concerning the commercial nuclear power units, was based on "closed" nuclear fuel cycle. Generally the management concept was as follows: spent fuel of the commercial VVER reactors was reprocessed to extract the fissile materials - residual U and newly generated Pu. Extracted uranium was used to produce RBNIK fresh fuel. In it's turn the RBMK spent fuel wasn't Supposed to be reprocessed in consequence with commercial inexpediency of this option. It was planned to be stored for a long period on the NPP's sites.

According to this concept the VVER spent fuel was required to be reprocessed shortly after its unloading. Therefore, VVER spent fuel storage capacities (at-reactor (AR) cooling pools) were designed to hold spent fuel of 3-4 refuelling only.

Following capacities for spent fuel storage are defined at original designs of AR cooling pools:

- a RBMK- 1000 AR pool - 1,704 cells. In addition, "on-site" wet storage facility with capacity of 17,520 cells for all Chernobyl RBMK-1000 units was constructed in 1986. It is worth to note that planned SF quantity per whole design lifetime of the CHNPP (120 reactor*years) would be more 45,900 spent fuel assemblies (SFAs);
- a VVER-440 AR pool - 729 cells. Annual design spent fuel discharge is 120 SFAs (3-year fuel cycle). No "on-site" SF storage facility was designed;
- a VVER-1000 AR pool - 392 cells. Annual design spent fuel discharge is 54-55 SFAs (3-year fuel cycle). No "on-site" SF storage facility was designed.

As for the Ukrainian NPPs participation in that concept scheme it was as follows:

- 100% of generated VVER spent fuel had to be transported to Russia (Cheliabinsk and Krasnojarsk) for reprocessing (VVER-440) and current technological storage and further reprocessing (VVER-1000). After SF reprocessing both radioactive waste management and extracted fissile materials management were the responsibility of the reprocessing plants, including long-term storage and disposal of;
- 100% of the RBMK-1000 spent fuel stayed on the NPP's site for a long-term storage.

Thus, till 1991 Ukrainian NPPs had neither "short-term" nor "long-term" own spent fuel management strategy, as far as they were one of the components of the former USSR's nuclear fuel cycle. From standpoint of the USSR nuclear political basis that position was understandable and explainable. But the situation sharply changed after 1991.

3.1. Present spent fuel management in Ukraine

In the end of 1991 the USSR was disintegrated and Ukraine became an independent state. About 36 % of installed civilian nuclear power capacities of the former USSR (13.48 GW(e) of total installed 37 GW(e) as of 1991) stayed on Ukraine's territory, but there was practically no infrastructure (legislation, state management, science, designing, manufacture etc.) to provide the necessary support to the nuclear power industry in Ukraine. Moreover, according to newly developed Russian Federation's legislation, spent fuel import for reprocessing into Russia was restricted in the starting of 90s, although, as for Ukraine, afterwards its spent fuel export to Russia was renewed (since 1995) on the business contract basis. Now it is important to note that: (1) in accordance with the current contract agreements the reprocessing products (high level radioactive waste/HLW) have to come back in Ukraine and (2) after 14LW restitution the waste management is a responsibility of Ukraine. As for fissile reprocessing product Ukraine keeps the owner's rights in relation to that one.

As a result the VVER spent fuel export from Ukraine for reprocessing was rather limited. During of 1992-98 Ukraine exported for reprocessing not more 55% of generated VVER spent fuel.

Legislative basis. Till 1992 national legislation for nuclear energy utilization was practically absent in Ukraine. In 1995 Ukrainian Parliament (Supreme Council) approved the Concept for state nuclear safety regulation and nuclear power management in Ukraine. On the Concept basis two fundamental Ukrainian national nuclear energy laws were prepared and adopted in 1995. The first one is the Law for nuclear energy utilization and radiation safety and the second one is the Law for radioactive waste management. Generally the approaches to spent fuel management are declared by the Law for radioactive waste management in its article 17 named "Storage and disposal of radioactive waste":

If spent nuclear fuel needn't be reprocessed it has to be stored into the special spent ,fuel storage facilities with a multibarrier safety system, a safeguard system and a technical rigging to remove spent fuel out of the storage facilities.

One more requirement of the above-mentioned Law concerns a disposal of spent fuel as radioactive waste is:

Long-lived radioactive waste has to be disposed into the deep stable geological formations provided that the radioactive waste was transferred to a solidified only, explosion-proof, fire-safe, nuclear-safe form.

In a framework of a development of these legislative norms the National Power Programme until the Year 2010 declared further steps for spent fuel management strategy. These steps are as follows: (1) to use "dry" spent fuel storage technologies in Ukraine, (2) to construct on-site "dry" spent fuel storage facilities on each NPP's site and, further, (3) to construct a centralized "dry" spent fuel storage facility.

Now the detailed Programme for NPP spent fuel management till year 2010 [3]. Based on above-mentioned steps, has been developed and is studied by the Cabinet of Ministers.

As for a spent fuel disposal, the approved Compound Programme for radioactive waste (RW) management defines this option. Some planned activities till year 2005 are described in the Compound Programme. Concerning SF disposal they are as follows:

- to develop a programme for RW disposal in deep geological formations;
- to develop a legislative/regulation basis concerning RW management for disposal and central geological RW disposal facility construction;
- to develop a Central geological storage/disposal facility concept;
- to develop an underground geological experimental laboratory concept;
- to determine the geological criteria and to develop the methodical recommendations to carry out the compound ecological/geological investigations concerning site determination for central geological RW disposal facility construction;
- to determine some areas and geological formations in Chomobyl exclusion zone that is the most perspective for central geological RW disposal facility construction;
- to choose some perspective sites into determined areas and geological formations, etc.

Organizational chart for spent fuel management in Ukraine. After some organizational changes in state management sphere in 1997-98 the fields of responsibilities in nuclear fuel cycle concerning spent fuel management were separated in such manner:

Ministry of Energy/State Department for Nuclear Power - the authorized state body for nuclear power utilization. Its responsibilities, in particular, are:

- planning, development, assurance and realization of the state programmes for nuclear fuel cycle, in particular the state programmes for spent fuel management covering the management till SF transferring for its disposal;
- foundation of the operating bodies to organize a fuel cycle facilities operation including spent fuel storage facilities operation;

- organizing of development and implementation of the branch regulations, rules and standards etc.;

"Energoatom" company - the organization that integrates all Ukrainian NPPs and is responsible for implementation of the state spent fuel programmes. It is planned the "Energoatom" company will be a license holder in some future to operate the nuclear power units and the spent fuel storage facilities;

NPPs - now they are the nuclear operators holding the temporary permissions to operate the nuclear power units.

Ministry and Emergencies and affairs of population of protection from the consequences of Chernobyl catastrophe (Ministry of Emergencies) - the authorized state body for radioactive waste management except of radioactive waste arising from the facilities subordinated to the Ministry of Energy. In relation to back-end nuclear fuel cycle the Ministry of Emergencies is responsible for SF management and disposal afterwards SF transferring for its disposal.

Ministry for Environmental Protection and **Nuclear Safety/State Nuclear Regulatory Administration -** the authorized state body for radiation and nuclear safety regulation and licensing of activities in the nuclear energy utilization areas.

Ministry for Environmental Protection and Nuclear Safety/State Nuclear Inspectorate - the authorized state body for nuclear safety supervision.

Organizational chart for spent fuel management is depicted in Figure 3.

As it is followed from an above information there are three authorized state bodies in Ukraine to provide SF management for back-end fuel cycle. Except of the regulatory body (Ministry for Environment Protection and Nuclear Safety) with its responsibility covering all spent fuel management fields on the whole, including SF disposal, two another ministries have different fields of responsibility concerning spent fuel management:

- Ministry of Energy is responsible for state SF management until transferring of spent fuel for its disposal;

- Ministry of Emergencies is responsible for all options concerning SF management for disposal (site selection, storage facility construction, SF disposal, R&D activities etc.).

Such fields of responsibility for both ministries are defined by a President's Decree.

Present and prospective (till 2010) spent fuel amounts. As of the beginning of 1992 there was 2 205 Mg of spent fuel (HM) in total in Ukraine. Since 1992 till 1999 about 2 166 Mg of spent fuel was generated by Ukrainian NPPS, including:

- VVER-1000 spent fuel -1458Mg;
- VVER-440 spent fuel - 168 Mg;
- PBMK-1000 spent fuel - 540 Mg.

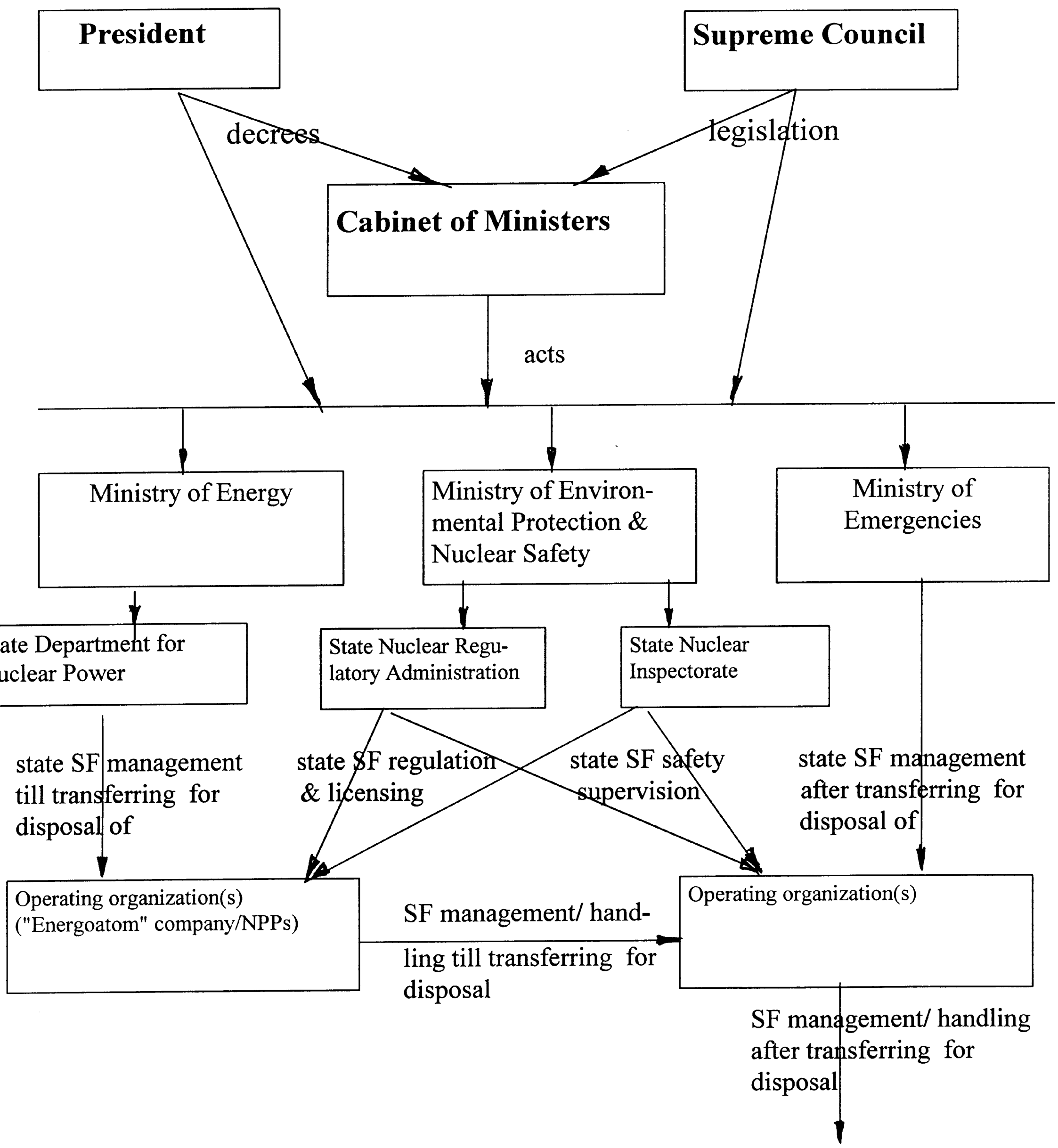


FIG. 3. Organizational chart for spent fuel management in Ukraine.

Taking into account the NPPs' very limited at-reactor storage capacities a share of VVER spent fuel was transported to the reprocessing plants in Russia. 993 Mg of spent fuel was transported in Russia for its reprocessing during the same period (since 1992 till 1999). Thus, 3 378 Mg of SF stayed in Ukraine as of the beginning of 1999. Nuclear spent fuel amount growth in Ukraine, as a prognosis one so in fact one, is depicted in Figure 4. The prognosis growth was estimated provided that the annual design refuelling is as follows: 54 SFAs per a VVER-1000 unit, 120 SFAs per a VVER-440 unit.

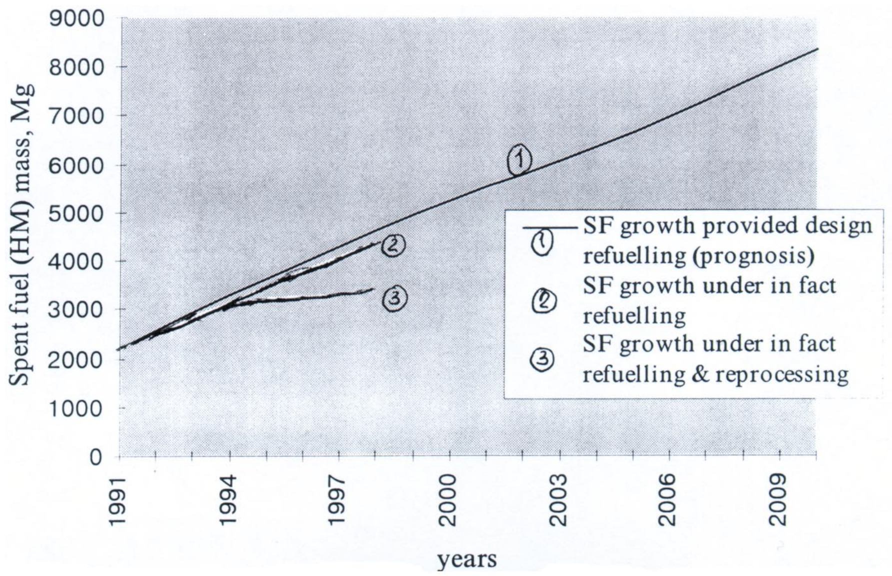
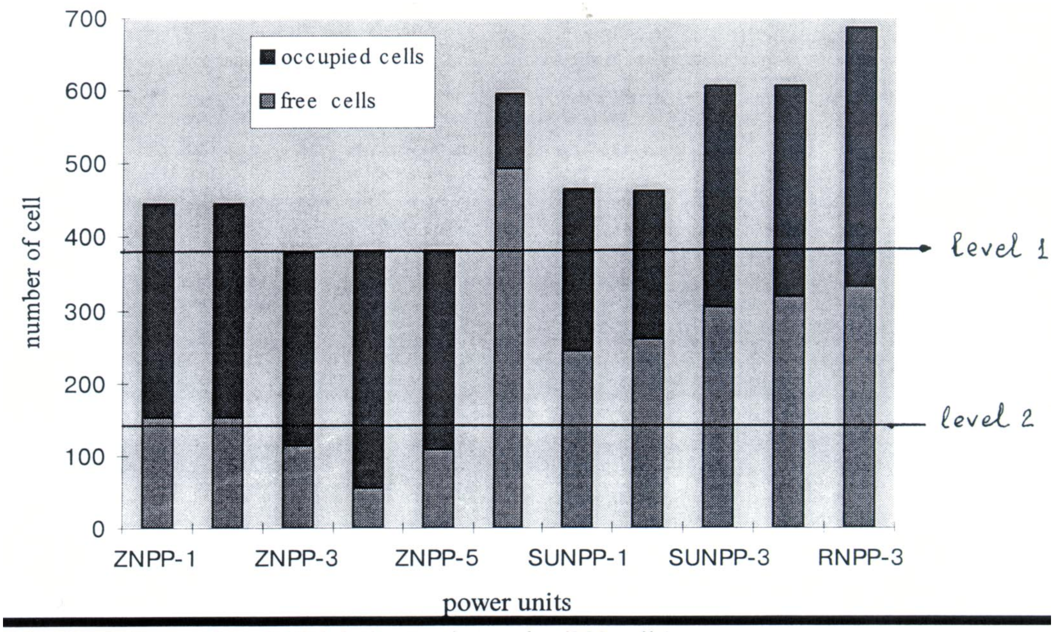


FIG. 4. Spent fuel growth in Ukraine: prognosis and in fact.



Remarks: 1). *level 1* – initial design pool capacity (392 cells);
2). *level 2* – constantly free pool capacity according to safety standards (163 cells);
3). column tops – actual pool capacity.

FIG. 5. VVER-1000 at reactor pools fullness with SFAs.

Capacities for spent fuel storage. Till nowadays in Ukraine the spent fuel storage capacities are restricted with the AR pools only. As it has been noted above the initial design AR pool capacities were designed to store spent fuel of 3-4 annual design refuelling not more.

In the first half of the 90s there were re-racked the VVER at-reactor pools in Ukrainian nuclear power units, except some Zaporizhia and Rivne NPPs' units, and pool capacities have been enlarged from 1.4 to 1.7 times in such manner. In spite of the pool capacities enlarging the free storage capacities in the at-reactor pools have being diminished a year by year as spent fuel has been reprocessed in more few amounts then it has been generated. As example, the VVER-1000 at-reactor pools fullness is depicted in Figure 5.

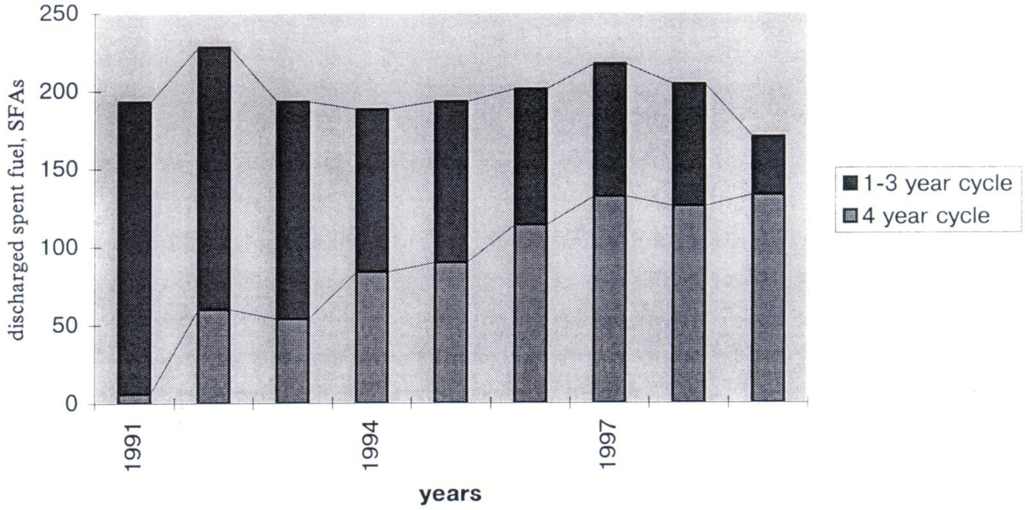


FIG. 6. Share of 4-year cycle fuel in VVER-440 discharged spent fuel (in total).

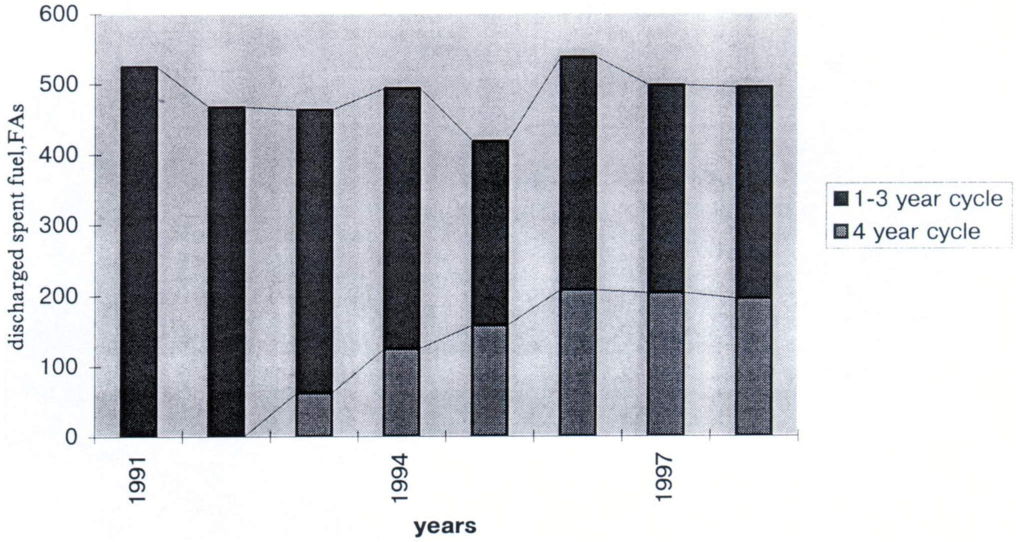


FIG. 7. Share of 4-year cycle fuel in VVER-1000 discharged spent fuel (in total).

To provide some additional capacities to store spent fuel the possibilities are considered to put the "dry" spent fuel storage technologies in practice in Ukraine.

The "dry" storage facility project for Zaporizhia NPP based on the Sierra Nuclear Corp.'s VSC-24 cask design has been begun in 1994. This project is aimed to store the whole spent fuel to be generated during Zaporizhia NPP lifetime. Now the facility's "start" stage (3 casks*24 SFAS) is ready to be put into operation. It is expected a regulatory body's license to put the facility's "start" stage into operation will be issued by the end of 1999.

Other "dry" storage facility project has been begun this year for Chomobyl NPP. The modular spent fuel storage facility will be constructed near Chomobyl NPP site. The facility is based on the NUHOMS technology by Pacific Nuclear (USA) and Framatome ATEA (France). It can contain 25 000 RBNIK spent fuel assemblies (entire SF account for Chomobyl NPP's real lifetime) and store spent fuel safely up to 100 years. After putting into operation of the "dry" facility, the on-site "wet" storage facility will be decommissioned.

Application in practice of prolonged operating fuel cycles and advanced fuel to reduce the spent fuel amounts. Now the three-year operating fuel cycle is used as a basic one for Ukrainian NPPs with the VVER power units. During the last 5-6 years in connection with some operating reasons the NPP operators have begun to utilize some share of a makeup fuel for prolonged (four-year) operating fuel cycle. At present the fuel shares to be used in four-year fuel cycle has approximately amounted to 30% (VVER-1000) and 70% (VVER-440) of discharged spent fuel (*Fig. 4. 6-7*). Application of prolonged operating fuel cycles in practice has permitted to reduce annually discharged spent fuel by 10-12% in average in comparison with design basis.

Besides that, it is planned that Ukrainian NPPs will be transferred to 4-year operating fuel cycle as a basic one by 2007-2008 and advanced nuclear fuel will be applied for such cycle (advanced nuclear fuel is U-Gd fuel with average design basis bumup 46-47 MW*d/kgU and fuel mass 435-440 kgU per assembly). In this case it can expect the annual spent fuel production will be reduced by 15-16% in comparison with design basis.

Further activities plan. Apart from the fact that Ukrainian nuclear legislation gives a wide range for selection of the spent fuel management options (reprocessing, storage or disposal of SF as radioactive waste), a real range is restricted with the country's individual technical and economical possibilities in its nuclear fuel cycle.

The nuclear fuel cycle strategy to be developed in Ukraine for next decade is as follows: (1) further VVER power units operation, (2) development of own fabrication of some nuclear fuel components, (3) a selection of new type nuclear power reactor(s) for future, (4) "dry" spent fuel storage technologies development.

Thus, in Ukraine the short-term back-end fuel cycle strategy is determined. It is based on interim spent fuel storage.

To confirm a safety conditions of "dry" interim storage of VVER spent fuel, in-advance investigations of spent fuel behaviour under long-term "dry" storage conditions are started this year in Russian Research Institute of Atomic Reactors by business contract of Ukrainian "Energoatom" company. As a result of the investigations the main safety criteria has be determined.

Some VVER spent fuel share will be reprocessed, but at present the reprocessing option isn't very attractive for Ukraine, firstly because the growing prices for reprocessing services, secondly because there isn't a necessity for Ukrainian nuclear power industry to utilize the SF reprocessing fissile products. It is expected that the spent fuel share for reprocessing will be reduced in further taking into account the reprocessing service prices.

On the other hand the main requirements for spent fuel disposal are determined too and some activities are planned to provide disposal of Nevertheless, indefinite period lies between interim storage termination (years 2040-50) and disposal of spent fuel. There are too much uncertainties nowadays for Ukraine to select a long-term spent fuel management strategy identically. Most probably in Ukraine the long-term spent fuel management strategy will be chosen during next decade or later.

4. CONCLUSION

In Ukraine the spent nuclear fuel amounts are considerable now and it is increasing further.

As for RBMK spent fuel, the methods to utilize it as power resource are unknown now. Most probably from present standpoint it will be disposed directly.

On its turn the VVER spent fuel includes essential quantities of the fissile products. Under certain conditions such kind of spent fuel can be considered as valuable power resource and it is utilized for further nuclear fuel cycle.

It will be depend on much factors, whether the conditions arise for Ukraine.

As for Ukraine the factors can be divided in 3 groups conditionally: global, external and internal.

It can be attributed to

(1) the global factors:

- world-wide status in power consumption;
- trends in price policy concerning energy carriers;
- accessibility of traditional energy carriers;
- ecological problems.

(2) external factors:

- successful development of new generations of nuclear power reactors to utilize MOX fuel or spent fuel directly;
- high safety under nuclear energy receipt based on spent fuel utilization;
- accessibility of new technologies concerning spent fuel utilization for countries with limited economic resources.

(3) internal factors:

- limited traditional power resources and uranium ore supplies (if there are) and costs of their timing;
- population's positive relations to development of nuclear power engineering;
- long-term country's policy concerning nuclear power engineering;
- country's economic possibilities to buy the new technologies for spent fuel utilization;
- country's scientific and technical potential to provide the nuclear fuel cycle.

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Safeguards policy and strategies: An IAEA perspective for spent fuel in geological repositories

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Abstract. Safeguards for nuclear materials in geologic repositories have to be continued even after the repository has been backfilled and sealed. The nuclear materials disposed in a geologic repository may pose a higher and long-term proliferation risk because the inventory is many times the "significant quantity" needed safeguards. The safeguards measures must be flexible enough to respond to the changing development of technology and changing need for current as well as future generations. Change in social, economic, environmental and other scenarios might demand recovery of nuclear and other materials from the repository sometime in the future.

1. INTRODUCTION

Geological repositories are at present being designed and will be built and operated early next century. Geological media for direct disposal may include clay, crystalline rock, salt or tuff. The operational life of a repository is expected to be 20 - 70 years before it is finally closed, completely back-filled and sealed. The spent fuel in a typically filled repository is expected to contain 2 000-200 000 tons of uranium and 20-2 000 tons of plutonium. In addition, a repository might also contain a wide range of radioactive waste some of which will not qualify for termination of safeguards. Excess fissile material resulting from the dismantling of nuclear weapons may be disposed by being placed in geological repositories as well. Thus, geological repositories will eventually have a higher and long-term proliferation risk.

Most of the nuclear material disposed in this way will be subject to international safeguards pursuant to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) requiring "the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection." However, any safeguards measures should "be consistent with prudent management practices required for economic and safe conduct of nuclear activities."

Geological repositories for radioactive material are designed to provide long-term isolation from the human environment by means of a system of barriers both natural and man-made. It is important that once a repository has been closed and sealed, it should be ensured that it is not disturbed in a way which could impair its safety barriers. For repositories containing spent nuclear fuel (and possibly also for those containing high level wastes) safeguards are to be continued in order to prevent possible diversion of nuclear materials. A possible issue is about the nature of the safeguards needed for repositories and, in particular, whether those measures would disturb the passive safety features of a repository. It is, therefore, imperative that safeguards measures take into account safety considerations. The prescribed measures should meet the safeguards objectives effectively and efficiently. When and if, anomalies are detected the International Atomic Energy Agency (IAEA) should be able to determine whether the anomalies are due to a cause that needs attention and follow up.

2. VERIFICATION OBJECTIVE

The objective of safeguards for nuclear material in geological repositories is to assure with a high degree of confidence that the material to be disposed is as declared, is emplaced in the repository and remains within the repository. The specific objectives for each phase of the disposal process are:

- Operating repository

Provide credible assurance that the quantity of nuclear material declared to be transferred into a repository is transferred into the repository and any undeclared removal of nuclear material would be detected.

- Closed repository

Provide credible assurance that an undeclared breaching of the integrity of a repository is detected and continuity-of-knowledge of the nuclear material is not lost because of a safeguards system failure.

3. INTERNATIONAL COOPERATION

The IAEA took the lead in initiating an integrated approach to identify safeguards issues and to develop safeguards measures with active participation of Member States. Advisory Group and Consultants' Meetings have been organised since 1988 at germane intervals. These meetings were instrumental in formulating policy and technical guidelines [1,2,3,4,5,6]. In addition, a programme for the Development of Safeguards for Final Disposal of Spent Fuel in Geological Repositories (SAGOR)[7,8] was launched to foster technological advancement. A departmental policy paper for application of safeguards has been adopted in order to provide sufficient guidance to identify safeguards measures beginning with the planning and design phase, in particular: to indicate the standards to be used for the application of safeguards, to ensure that these standards are integrated into the repository design and to permit adherence to these standards during the construction and operation of the repository in order to establish an effective and efficient safeguards system.

The mission of SAGOR, a multinational Member State support programme, is to ensure that the safeguards systems developed for the final disposal of spent fuel effectively meet the objectives of IAEA safeguards, optimise IAEA resources, and make best use of existing technologies while still meeting the objectives of safety and environmental protection[9].

4. HIGHLIGHTS OF THE POLICY

Spent fuel disposed in geological repositories is subject to safeguards in accordance with the applicable safeguards agreements. Safeguards for such material are to be maintained after the repository has been back-filled and sealed and for as long as the safeguards agreement remains in force. The safeguards applied should provide credible assurance of non-diversion and absence of any undeclared activities.

The safeguards systems must meet rigorous system specifications and standards in order to function for a very long period with minimum or no service, perhaps in a rugged environment and preferably in unattended mode. Since emplaced spent fuel cannot be re-verified, sufficient redundancy, diversity and robustness should be incorporated into the safeguards system and adequate maintenance measures be applied to avoid system failure and ensure continuity-of-knowledge. The safeguards systems for a repository will be based on: verification of design information during design, construction and operation; verification of receipts and flow that no nuclear material is removed by any declared or undeclared access routes; and maintenance of continuity-of-knowledge on the nuclear material content.

5. INTEGRATED SAFEGUARDS VERIFICATION SYSTEM (ISVS)

An ISVS will be applied to verify transfer, flows and inventory of the spent fuel disposal containers and to maintain continuity-of-knowledge on the nuclear material. It should be comprised of elements of containment and surveillance (C/S), monitoring and non-destructive assay (NDA) systems, as well as design information verification (DIV), geophysical, environmental and radiological systems, as applicable. An ISVS should have the capability of functioning, as far as is practicable, in automated, remote control and remote data transmission modes. An ISVS should have high system reliability and capability to detect component failures and to notify the IAEA in a timely manner of such failures, preferably by remote transmission.

6. DESIGN INFORMATION VERIFICATION (DIV)

DIV constitutes an important safeguards measure during the pre-operational and operational phases. DIV should confirm the design of the geological repository and detect any undeclared modifications and activities, both in the repository and in its vicinity. The IAEA should verify that the excavation areas are as declared and that there are no undeclared excavations. As the repository design will change during excavation, for example to adapt to geological findings, the application of DIV must be a flexible, ongoing process. During the operational phase, the IAEA should also provide assurance of the absence of undeclared underground reprocessing and an assurance of no undeclared operational capability underground which could mask the substitution between containers.

Once the repository is closed and sealed, safeguards should consist of suitable surface monitoring measures to provide assurance of ‘no access’ to nuclear material, e.g., visual observation through photographic techniques or video-recording, remote surveillance including optical, satellite, geophysical and environmental techniques. These measures should be adapted to site specific requirements. Upon request by the IAEA, the State should provide access to any building or to any location at the geological repository site or to any location outside a geological repository site which the IAEA considers might be functionally related to the geological repository. Arrangements should be made with the State for advance notification to the IAEA of any a) intention to access the sealed geological repository after final closure; b) intention to retrieve the spent fuel from the geological repository; c) intention to retrieve any other material from the geological repository; and d) tunnelling, mining or blasting activities in the vicinity of the repository.

7. SAGOR PROGRAMME

The SAGOR programme, established at IAEA in September 1991 has been subscribed to by eight Member States, viz., Belgium, Canada, Finland, France, Hungary, Sweden, UK and USA. Germany who had earlier initiated a separate programme participated as an observer. A Technical Co-ordination Committee (TCC) was formed in March 1994 to ensure an integrated approach and to foster technological advancement towards the development of a comprehensive safeguards system.

8. INTEGRATED PROGRAMME

An integrated task programme was adopted for detailed analysis of geological repository scenarios with the sub tasks: describe a model facility, identify diversion path and detection point, identify events and conditions requiring DIV and examination procedures, evaluate IAEA use of operators’ safeguards, safety and process system outputs, identify potential application of geophysical techniques, evaluate NDA techniques for spent fuel verification and radiation monitoring, evaluate C/S techniques and integrated verification systems for spent fuel monitoring, determine guidelines for acceptable safeguards approaches, design safeguards approach, evaluate candidate approach, select safeguards approach and identify R&D needs towards developing the complete system.

Scenario 1: Operating Repository

The safeguards strategy is based primarily on maintenance of continuity-of-knowledge. The recommended safeguards approach is to use item accounting supported by a reliable and comprehensive monitoring system above ground to verify, inter alias, the flow of full casks and over packs. DIV is recommended as the primary safeguards measure underground. DIV would include geophysical methods.

Inaccessibility of emplaced canisters for verification implies a very high degree of reliability to be incorporated in the safeguards system. This will be achieved by the use of intrinsically reliable systems with multiple redundancies. The monitoring system will be comprised of an integrated system of motion and radiation detectors, optical surveillance, seals and NDA, as appropriate. These will be designed for independent operation and remote monitoring to minimise the presence of inspectors at the site. These should give inspectors confidence that the received full casks are transferred to the underground area without tampering. All potential access points to the repository would be rigorously monitored to ensure that no undeclared items leave or enter the underground area.

DIV would be periodically implemented to provide, among others, assurance that the physical structure and operations of the repository are in full accord with the details provided in the State’s official reports to the IAEA. Geophysical methods would be used as a means of supporting DIV activities as well as an element of surveillance. Additional safeguards measures were considered for the underground material balance area if measures in the recommended approach prove impractical or ineffective in specific cases.

A major goal of future development is integrating individual safeguards components into an integrated safeguards verification system(s) that will collect, store and process the information, and, if required, transmit it to IAEA Headquarters. Important areas for future research and development include the following:

- safeguards application of seismic techniques;
- ground penetrating radar technology for routine use in verifying repository environment;
- automated data review and interpretation for geophysical methods;
- unique identifiers for canisters and casks; and
- Environmental monitoring to detect undeclared underground processing activity.

Scenario 2: Closed Repository

After closure of a repository diversions can only take place by excavation. Therefore, the diversion paths consist of (i) excavations of the original shafts and tunnels, (ii) excavations of new shafts and tunnels, and (iii) excavations from other mines, tunnels, or caves. The best locations for strategic point measurements are (i) the entrance of the original shaft, (ii) the surrounding area on the surface above the repository, and (iii) adjacent mines, tunnels, or caves.

The proposed safeguards approach consists of safeguards measures implemented at the detection points. For the closed repository, the suggested safeguards measures include (i) unannounced random visual inspection by an inspector with possible application of geophysical techniques, (ii) satellite or aerial monitoring, and (iii) seismic monitoring. All these measures can be employed at each detection point with varying degrees of sensitivity. Visual inspection would be most effective at the original shaft or at adjacent mines or caves. It would be less useful for surface inspections over the larger surrounding area above and near the repository. However, satellite monitoring would be most useful for surface area inspections. Seismic monitoring could be both passive and active. Passive seismic monitoring could be used on a continuous basis to detect seismic signals created by excavation activities. Other geophysical monitoring techniques could be used during on-site inspections to detect the results of excavation activities. Remote transmission of data could be expected to reduce costs and provide for early detection of undeclared activities. Additional safeguards measures applicable to detection of diversion from a closed repository include environmental sampling and information analysis.

The proposed safeguards procedures and technologies show promise but need further development effort to demonstrate their effectiveness. Research and development for geophysical monitoring systems and satellite surveillance are needed to increase their sensitivity and to determine their powers of detection of undeclared excavation activities. Applicable technologies are expected to evolve significantly over the time until repositories will be closed. The final safeguards approach must take into account these technological advances, the evolution of the strengthened safeguards system, and synergies with other arms control monitoring systems.

9. FUTURE WORK

It was clear that elements of the Strengthened Safeguards System have a direct role to play in the application of safeguards to the final disposal of spent fuel in a geologic repository. Nevertheless, future research and developments are needed for:

- NDA of spent fuel,
- Reliability of safeguards equipment,
- Design of integrated, remotely interrogated verification systems,
- Confirmation of the identity and integrity of disposal casks,
- Satellite imagery,
- Environmental sampling, and
- Conceptual work on safeguards for a closed repository.

10. CONCLUSION

Safeguards for nuclear material in geological repositories have to be continued even after the repository has been back-filled and sealed. The effective application of safeguards must assure continuity-of-knowledge that the nuclear material in the repository has not been diverted for an unknown purpose. The nuclear material disposed in a geological repository may eventually have a higher and long-term proliferation risk because the inventory is substantially larger. However, the safeguards measures must be flexible enough to respond to the changing development of technology and changing need for current as well as future generations. Change in social, economic, environmental and other scenarios might demand recovery of nuclear and other material from the repository sometime in the future. Current development effort will have to be in tune with other factors to optimise safeguards effectiveness, such as:

- The different geological media - with consequent differences in excavation difficulties and scope for use of geophysical techniques;
- Technical concepts, for example, repository layout, depth, potential for retrievability, time period for which the repository will be kept open;
- Advancement in safeguards technology, i.e. procedures, equipment;
- Advancement in mining technology which may make disposal of spent fuel more easily accessible;
- Incentives for recovery of spent fuel material for energy production or other purpose;
- Socio-political factors (e.g. regarding the other institutional controls, the importance of safeguards in the distant future is not known);
- The interpretation of the principle of radioactive waste management that waste shall be managed in a way that will not impose undue burdens on future generations.

Detailed threat analysis, diversion strategies and safeguards approaches are need to be examined for each specific repository. The primary assumption on which the threats and diversion strategies for geological repositories are based is that spent fuel will be disposed only as verified nuclear material on which continuity-of-knowledge has been maintained.

Close co-operation between the IAEA and international community is the key to effective and efficient safeguards for such a complicated facility. In addition, participation of experts as well as advancement in other disciplines, namely safety, waste management, environmental protection, whose understanding of safeguards needs are indispensable, will have a significant role in geological repository affairs.

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PLUTONIUM MANAGEMENT

An alternative plutonium disposition method

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Abstract. This paper provides a feasibility study on vitrification of plutonium with high active waste concentrate, and fabrication of MOX fuel rods for direct final disposal. These are potential alternatives to the direct use of MOX fuel in a reactor.

1. INTRODUCTION

Some 26 to 29 tons of separated plutonium from German spent fuel are still stored as PuO_2-powder. The Oeko-Institute investigated options for the separated plutonium handling in a project done for the city of Hamburg.

2. ESSENTIAL REQUIREMENTS FOR INVESTIGATION OF DIFFERENT METHODS

The different methods must fulfil a range of requirements in order to be feasible. Each different method must not only propose a valid solution for the final disposal of plutonium but also satisfy every aspect of the feasibility requirements. It is also important that the necessary safeguard measures to determine protection against proliferation are carried out at each stage; the potential for final disposal is of particular importance.

Achieving a state suitable for final disposal of plutonium is a fundamental aspect to be considered whilst assessing methods proposed for the handling of plutonium.

Methods proposed for the handling of plutonium have limited use if they only offer a temporary solution and not a clear definitive solution for its final disposal. The solutions offered must therefore lead to the safe and final disposal of plutonium. One complete method can comprise stages from different methods, after completing these stages, however, the plutonium must be in a form suitable for final disposal and in keeping with the guidelines.

It is imperative that each stage of a proposed method be either technically feasible, or - due to similarities with methods already in use - can be easily and quickly put into practice. To apply a method already in use and to adapt it for the purpose of handling plutonium should then ensure the technical feasibility of the method. The further development of a current method, when adapted for the handling of plutonium, should (according to experience) present no particular problems to the technical feasibility of a method. To revert back to methods no longer used in nuclear engineering would mean that the requirements are difficult to fulfil; which is disadvantageous if the technical feasibility could not given for one possible method. Safety is an important part of technical feasibility because every stage of a method must fulfil the safety requirements of nuclear engineering. Technical feasibility also includes the authorisation of the plant, since the different technical aspects must be checked, assessed and decided upon.

The question of time must also be taken into consideration:

- How long would the method take, if all available plutonium were to be handled?
- When would the necessary plants be available?
- Where other material is required (HAWC for vitrification for example): is the right amount of the material available at the right time?

The question of economic viability is one aspect which can be of good use during the selection of a method. “Safety comes before economy” is a saying that must be respected in nuclear engineering. It is therefore helpful to have an economic assessment of one or more methods. It is in any case necessary to have at least one method assessed for the handling of the excess separated plutonium, because the current situation (storage as separated PuO_2) cannot continue; the economical criteria may not therefore lead to the exclusion of all methods.

For every method proposed for the handling of plutonium the safety requirements of nuclear engineering must be fulfilled at each stage of the procedure. Included is the question, whether the safety of a method is seen as guaranteed or whether there is still uncertainty. The detailed safety checks are carried out in corresponding licence procedure.

Where certain stages of the method are already being carried out in existing plants, the official safety standards are first approved when the danger potential of the material to be handled is identical to that of a material already approved. The concentration of fissile material (criticality safety) also plays a part here. The official regulations used in approved plants are sufficient when dealing with materials with not only the same radiological danger potential but also the same radiation protection and emission into the environment.

The relevant safety aspects are above all:

- The criticality safety must be guaranteed for every step of the procedure. For every stage before final disposal this is completed by criticality safe design of components, storage and transport containers as well as the usual method (e.g.: limitation of material amount, limitation of the volume of components for handling fissile material). The criticality safety of final dispose requires checks over a longer period of time.

- Necessary transportation must be carried out in approved transport containers that are designed to cope with criticality safety in the case of an emergency. The requirements result from the internationally applied criteria for nuclear transportation. This also means that certain materials should not be transported- in connection with the handling of plutonium, plutonium nitrate solution and HAWC solution. The transportation of liquid solutions of these substances would bring added risks, which for previous transportations of separated fuels from German nuclear power stations was never and should be avoided.

The methods for the handling of plutonium entail many stages. The last step is always the final disposal of the remaining plutonium. The final disposal “suitability” arises through the combination of final storage and the plutonium-bearing product.

For this reason special attention should be paid to the necessary requirements. These are already present due to the general demands for final storage and its long-term safety. The releasing of plutonium into the biosphere must be prevented as far as possible so that even in view of the extremely long timespan no danger could emerge. This requirement is a special case of the general requirement made on final disposal storage sites–keeping the stored radio active product away from the biosphere. One part of the requirement is the correct choice of a suitable storage site with regard to situation and geology. This report doesn’t discuss this aspect of the requirements in detail.

There exist other requirements regarding the form of the plutonium to be disposed of permanently. Of particular relevance here is the avoidance of recriticality during final storage. This comes from the possibility that the fissile isotope of plutonium (and uraninum-235) through a series of events in the final disposal site could result in a configuration, through which a nuclear chain reaction is possible. Therefore, through different specific measures the possibility of a chain reaction must be avoided.

The disposal container must always be criticality safe before it is brought into final storage because the criticality safety is already checked and approved during the previous stages: interim storage, transport, conditioning for final storage handling. With regard to changes of moderation trough the addition of water, the criticality safety must be guaranteed during the phase of early storage.

Concerning the long-term stay in final storage, only additional mechanisms are important, which could lead to the long-term change of concentration and accumulation of fissile material. Such mechanisms can either result from a segregation of the fuel and subsequent concentration of the fission isotopes, or through specific changes in the composition of the isotopes due to radioactive decay. In order to avoid these effects to ensure that no recriticality occurs, additional measures are required. Such measures could include geometrically designed composition of the plutonium storage containers, which are seen as stable enough under final storage conditions. The additional methods could also include thinning of fissile fuel through material which behaves chemically similar (uranium-238). Other neutron-absorbing materials like uranium-238 would also come into consideration, the chemical similarity must, however, be guaranteed.

Due to its suitability for weapons plutonium especially must be carefully observed in order to avoid proliferation (misuse of civil plutonium for military purposes), or at least to discover it early on. For this control the so-called safeguard measures are used. Inspectors from international organisations (IAEA/Euratom) control the civil nuclear plants using certain methods. The methods must be adapted to the prevailing technical processes in the observed plant and to the proliferation resistance of each handled material containing plutonium. The isotopic composition is not considered because of plutonium’s fundamental suitability to weapons on examination of proliferation resistance.

Safeguards are relatively easy as long as countable items containing fissile materials are handled. For fuel, this is done determined by measuring the fissile material content. As long as this element is not dismantled it can be surveyed easily during handling, storage or transportation, because only the methods of containment and surveillance have to be applied. Sealed containers are, in view of safeguards, similar countable “items” as fuel. Interim storage sites are, therefore, relatively easy to survey as during construction safeguard aspects were considered.

One main weakness of the current safeguard concept is exposed through all the so-called “bulk-handling facilities”. These are plants where plutonium (and other fission products) are handled in great amounts in their free-flowing separated form; for example, MOX fuel factories and reprocessing plants. As long as the plutonium is packed and kept together, in fuel assemblies for example, it can be accurately counted using the above methods. If, however, material containing plutonium appears in a non countable form (powder, solution, pellets, abrasive dust and other scrap material) accuracy is not guaranteed during counting - up to 1% inaccuracy is unavoidable with every check carried out. The calculated material balance differs then from stock-take to stock-take. This can lead to considerable amounts of material unaccounted for (MUF). Should there appear a MUF greater than zero, the safeguard organisations are faced with the problem of not being able to distinguish between a real proliferation and a statistical inaccuracy. The bigger the inventory handled in a plant, the greater the inaccuracy. This in turn means that the absolute amount of MUF which can be tolerated is greater- due to this inability to distinguish between real proliferation and inaccuracy.

Due to the fundamental problems of balancing as a measurement technique, the proliferation risk of free-flowing plutonium is high. It is therefore preferable to avoid storing plutonium in such forms and to opt for a speedier conversion into countable items (Itemisation).

The proliferation resistance of handled plutonium can only be understood as gradual criterion, since only a more or less incomplete protection against the abuse of plutonium for military purposes can be achieved. In the US debate, the term ”spent fuel standard” was introduced for the assessment of suggestions made for the handling of plutonium after nuclear disarmament. It states that new plutonium should be converted into a form, like plutonium in spent fuel, which achieves protection against access. The radiological barrier – which is built up over a long period of time from the fission products in the spent fuel – is therefore a deciding factor in the attainment of the “spent fuel standard”. Furthermore, technical and economical factors also have to be taken into consideration. It makes no sense, therefore, to go to great expense and convert the plutonium into a form which claims to be safer than spent fuel element (the form in which most of the plutonium world-wide appears and will be finally disposed of), since there is no safer form. It seems appropriate then to use the “spent fuel standard” as a yardstick for proliferation resistance for reactor plutonium.

Two important facts are to be noted here:

- Firstly, the “spent fuel standard” would only be a safe reference if reprocessing technology were not available. A safe proliferation resistance would then only be achieved on termination of separating the plutonium from fuel and through the main prohibition of operating plants designed for this purpose.
- Secondly, proliferation resistance standard must not be interpreted too narrow or too formal. Storage forms that do not have certain properties of spent fuel can have a much better proliferation resistance than metallic or oxidic plutonium in its pure forms and could be judged as resistant enough.

Even today there is still no operated site in the world suitable as a final storage site for highly radioactive, heat-producing waste, although planning and preparation is being undertaken in many states that use nuclear power. The developments to date show that, according to their policy, heat-producing waste these states are only prepared to build one final storage for highly active waste and not several storages. There are different technical and financial reasons for this:

The final storage must be planned in such a way as to ensure that all highly active heat-producing waste can be stored. Types of waste are as follows:

- Glass containers from the vitrification of highly active fission products (from reprocessing).
- “Normal” spent fuel from the usual plants (direct final disposal)
- Special fuel; for example, spent MOX fuel, fuel from specially built reactors.
- Scrap containing fissile products from fuel production
- Storage containers for final storage of plutonium.

The above forms of waste, with the exception of the glass containers, show such a high proportion of plutonium that termination of safeguards is not possible. To investigate the nuclear energy programmes of certain states ascertains that the majority of states have already opted for the direct final storage in earlier years and the final storage of spent fuel must be done in any case – examples are Spain, Sweden and Canada. Germany and Belgium, on the other hand, turn to final storage of spent fuel after having reprocessed over a longer period. Even in those few countries using reprocessing in general – there are in practice again special fuels which must be sent for final storage without having been reprocessed.

All in all it means that the proposed final storage sites in each country for highly radioactive waste either contain only material with considerable plutonium content (in the case of direct final storage strategy) or at least partly contain material with considerable plutonium content (in the case of the reprocessing strategy). Under the specific German conditions it is to be expected that approximately 6 000-12 000 tons of spent fuel for direct final disposal will have to be disposed, depending on the shut-down of the nuclear power stations. As a result, at least 60 to 120 tons of plutonium will be contained in German storage sites for highly active heat producing wastes.

The safeguards are necessary for practically all of the final storage sites being built internationally, in order to guard against proliferation of nuclear fuel. This requirement is the result of the fear that future generations could use the final storage sites as a plutonium mine and use the stored plutonium for military purposes.

3. RESULTS OF THE INVESTIGATIONS OF THE DIFFERENT OPTIONS FOR FURTHER HANDLING OF PLUTONIUM

One option for the further use of plutonium is by using MOX fuel elements, which for various reasons are usable with just a portion of recovered plutonium.

The disadvantage of MOX utilisation, besides the technical risks, is also far higher costs in the manufacture as for uranium fuel elements. Also with regard to the spent MOX fuel elements, as compared to the spent uranium elements, there are more requirements to be met.

With regard to the alternatives for the utilisation of MOX fuel elements for the purpose of handling the recovered plutonium, there have already been discussions for some years now.

One of such debates took place in the US regarding 50 tonnes of weapons' plutonium that became surplus after disarmament. The decision-making body at the time designated that a portion of the weapons' plutonium was to be vitrified and another was intended for MOX fuel elements, but possibly even to go for vitrification option for the whole amount.

The vitrification variation that is most popular in the US is the production of plutonium-charged titanium ceramic, which are mounted in high-grade steel containers that are then filled with borosilicate glass, adding HAWC (highly radioactive fissile solution) from military reprocessing.

This produces a product that by its high radiation and chemicophysical shape is considerably better protected in terms of the danger of proliferation than pure plutonium.

In respect to the German situation, the procedure has been adapted in this report, to ensure quick feasibility. The usual mixed-oxide ceramic forms the basis for the ceramic. The product should be the same as glass blocks fabricated from German HAWC (highly radioactive fissile solution). For this method, large-scale technical experience and existing plant can be utilised.

In this report the "direct vitrification of plutonium with fission products" is investigated as a further method of vitrification, where plutonium is vitrified along with HAWC (highly radioactive fissile solution). This obviates the production of mixed-oxide ceramic, thereby steeply reducing the costs of the procedure. Vitrification, using this method can however only be undertaken in a vitrification plant, which would need to be constructed from scratch.

Two other researched alternatives are to be found in the area of the storage rod method. This technique was developed by the Oeko-Institut eV in 1992 in a detailed report.

In the storage rod method, storage rods are manufactured, which in terms of their storage-relevant characteristics correspond to MOX fuel rods, but are designated for direct permanent disposal.

Since there will be no deployment in a reactor, the specifications associated with reactor deployment can be disregarded (internal pressure, level of demand on dimensional stability) when manufacturing the storage rods.

Thereby clear simplifications of the manufacturing process are resultant, which can trickle down as cost reductions.

The storage rod method with storage elements comprises the following discrete steps:

- The manufacturing of mixed-oxide ceramic, storage rods and storage elements, analogous to the conventional procedure in existing MOX fuel element factories
- The transportation of storage elements and corresponding to the transportation of fresh MOX fuel elements to the nuclear power plant
- The mix loading of storage elements and spent fuel elements in a transportation and storage container
- Mixed interim storage of storage elements and spent fuel elements
- Mixed permanent disposal of storage elements and spent fuel elements.

The second variation of the storage rod method is different in that the mixing with spent fuel takes place by exchanging individual rods in the spent fuel elements with storage rods in order to produce mixed elements. This would take place in the cooling pond for spent fuel at a nuclear power plant. The other steps in the production and storage processes remain identical to the other variations.

In both cases the spent fuel guarantees protection via its high radiation. The storage rod method is based in both variations on techniques of which Europe has expansive large-scale technical experience. Only existing plants are needed for the treatment of the plutonium.

The capability for permanent storage is guaranteed in all of the four methods that have been examined. This was revealed through the comparison with other materials that would inevitably have to be housed in a permanent storage facility. In respect to the evidence that needs to be supplied for long-term safety and critical safety as well as for safeguard measures there are no additional requirements since the same type of evidence would have to be provided. The future permanent storage facility certainly needs to be equipped in such a way that vitrified plutonium or storage rods can be permanently stored.

TABLE I. RESULTS FOR THE DIFFERENT OPTIONS

	Can-in-canister	**Direct vitrification**	**Storage fuel (fuel assemblies)**	**Storage fuel (rods)**
Technical experience for the steps before final disposal	exists	major part exists	exists	exists
Demand of new facilities	none	new vitrification plant	none	none
Need of new licences	modifications only	new licences	modifications only	modifications only
Conditioned amount of Pu-tot in kg/y	ca. 900 – 1 800	ca. 4 000	ca. 3 300-7 000	ca. 3 300-7 000
Conditioned total amount of Pu-tot in kg	ca. 4 000–1 6000	ca. 3 000-9 000	no upper limit	no upper limit
Costs in DM/kg	ca. 36 000-115 000	Ca. 23 000-25.000	ca. 42 000-139 000	ca. 43 000-142 000
Product suitable for final disposal	yes	yes	yes	Yes

TABLE II. COSTS OF THE STORAGE ELEMENT OPTION AS A FUNCTION OF THE TOTAL AMOUNT OF Pu THAT IS PROCESSED

Total amount of Pu processed	**Costs in DM/kg Pu-tot**	
	5% fissile material	**10% fissile material**
1 000 kg	88 700 – 139 400	44 000 – 68 500
25 000 kg	87 300 – 138 100	42 100 – 66 600

Fabrication and use of MOX fuel assemblies: 67 000 – 86 000 DM/kg Pu-tot

The storage rod method with the fabrication of storage elements has displayed the greatest advantages of all the researched methods for permanent storage of plutonium and should be given preference in the continued handling of reactor plutonium.

There is large-scale technological experience available for all the necessary processing steps and all these steps can be carried out in existing plants - which are also accessible to German clients.

The storage rod method leads to a product capable of permanent storage and therefore cannot be seen only as a interim solution. Through the manufacturing of storage rods and elements a figure of at least 3.3 to 7 tonnes can be processed, meaning that by this method, the present plutonium is expected to transformed into a form capable of permanent storage within the shortest time frame in comparison to other methods. A further advantage of the storage rod procedure is that the entire German plutonium could be handled in this way as the number of available spent fuel elements is sufficient for mixing for interim and permanent storage. In comparison to MOX utilisation, this can result in clear economic advantages; the costs are dependent on the achievable amount of fissile materials in the fabricated ceramic.

In comparison to the fabrication of storage elements, the storage rod method with rod exchange entails the additional procedure of exchanging the storage rods for fuel rods in spent fuel elements. Due to a large number of rods to be exchanged, there is a greater risk of rod damage and also a higher collective dosage for personnel. The storage rod method with rod exchange should therefore only be used if, on the basis of far-reaching "physical protection" (because of a more intensive mixture), it is considered to be absolutely necessary.

The realisation for this and for the storage rod method with fabrication of storage elements can be achieved by resorting to existing plants and technical procedures available.

The can in canister method of plutonium vitrification displays fundamental disadvantages in comparison to the storage rod method insofar as technical conformity may be required in a foreign vitrification plant. Using this method would require that an appropriate agreement would need to be reached with the foreign company. With the desired agreement in place the feasibility may be possible without massive additional expenditure.

The direct vitrification of plutonium with fission products should only be considered if the construction of a new vitrification plant was already a necessity. The demand for such a newly planned plant is not foreseeable in Europe's future. If a new plant were to be constructed, direct vitrification would be economically attractive.

All in all, priority should be given to the implementation of the storage rod method with the fabrication of storage elements and thereby provide a practicable opportunity to reduce the existing mountain of recovered (separated) German plutonium.

The main results are summarised in Table I. In Table II costs of the plutonium handling by storage element fabrication and final disposal are given and compared to the costs of fabrication and use of MOX.

The nuclear future; prospects for reprocessing and mixed oxide nuclear fuel; why use MOX in civil reactors?

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Abstract. There are many answer to the question "Why us MOX in civil reactors?" The most likely one is because plutonium is an energy source and MOX is used when it is economic to do so. Other incentives include the reduction of global separated plutonium stocks and the subsequent potential reduction of proliferation risk.

1. INTRODUCTION

There are reasons to use MOX- fuel in civil reactors as there are reasons not to use it and the question of using it can be a question of principle for a specific energy- and nuclear power system in a particular region of the world or a practical question for a utility operating a nuclear power plant. Timing of MOX-use can be crucial: At times utilities may have to use MOX-fuel even if it is apparently more expensive than fuel made from enriched uranium at a given point in time and there are other times, when the same utility may avoid using MOX-fuels irrespective of cost considerations.

Most civil reactors are fuelled with fuel assemblies made from natural uranium oxide or uranium oxide that is slightly enriched in fissile uranium isotope U-235. In mixed oxide fuel most of the U-235 is replaced with plutonium, resulting in a mixture of uranium and plutonium oxide, hence "mixed oxide fuel" or MOX-Fuel.

Plutonium is generated during the use of uranium fuel in the reactor and it is partly consumed as it produces for example about 40 % of the energy generated in a light water reactor (LWR), the most common type of civil power reactor currently in use. After 3 -6 years a fuel assembly is replaced in the reactor by a fresh fuel assembly. The spent fuel assemblies from power reactors are the source of most of the plutonium in existence today, about 1 400 t. OECD and IAEA have estimated, that about 180 t of this is in the form of separated plutonium oxide, most of it currently stored at the reprocessing plants in France and the UK. The remaining 1 220 t are stored in the form of spent fuel in many spent fuel storage facilities in more than 30 countries around the world.

Plutonium -in the first place- is an energy resource contributing today 40 to 50 percent of the energy produced from nuclear power.

Plutonium is also a material used in nuclear weapons. Although for weapons purposes this is normally produced in purpose-built "production reactors", plutonium from any source is considered by some countries as "weapons-usable material". In fact, a few years ago, the US Department of Energy (DOE) announced, that it had built and tested a nuclear weapon made from plutonium separated from spent commercial nuclear fuel.

Plutonium therefore being a material that can also be used for nuclear weapons is a dual use material, a resource to be separated from spent power reactor fuel and recycled and/or weapons material, extracted from production reactor fuel to build nuclear weapons. It is this duality, which makes the use of plutonium in the civil nuclear fuel cycle a politically controversial issue. In civil reactors plutonium is used in the form of MOX-fuel. The question then is: "Why use MOX in civil reactors"?

2. USE MOX WHEN IT IS ECONOMIC TO DO SO

In power reactors MOX fuel replaces fuel made from natural or slightly enriched uranium. The relative cost of MOX-fuel -compared to that of uranium fuel is normally assessed by comparing the cost of procurement of an MOX-fuel assembly with that of an uranium fuel assembly. This comparison appears straight-forward to make, as the plutonium in a MOX-fuel assembly simply replaces most of the uranium 235, that is otherwise present in a uranium fuel assembly. The comparison of the cost of mining or buying uranium, enriching it to the desired level of uranium 235 (normally 4 to 5 % in weight) and fabricating it into a fuel assembly to the cost of blending plutonium oxide with depleted or natural uranium oxide and fabricating the MOX-fuel assembly gives a direct comparison of the relative procurement cost, which results in MOX-fuel being cheaper or more expensive to procure than uranium fuel.

Historically both situations have occurred: The procurement cost of MOX-fuel has been lower than that of uranium fuel. Currently this comparison favours uranium-fuel over MOX-fuel, as prices for natural uranium and enrichment are low due to vast secondary market supply for uranium (uranium on the market, that has been mined long time ago for military and purposes and is considered excess to military needs) and enrichment services. On a strict comparison of procurement cost for uranium- and MOX-fuel, there is no incentive currently to utilities to use MOX-fuel instead of uranium fuel and this situation is not expected to change any time soon.

The procurement cost however is only a part of the cost considerations that govern the use of MOX-fuel for a utility.

3. USE MOX BECAUSE PLUTONIUM IS AN ENERGY RESOURCE

About 20 to 30 years ago the basic structure of the fuel cycle in support of the then rapidly expanding nuclear power industry was established with the view, that uranium resources were limited and required the rapid introduction of new reactors, that used the limited resources of uranium many times more efficiently: breeder reactors, which also operate on MOX-fuel, burning the plutonium separated from spent fuel used in the early generations of LWR's.

Consistent with this view, and for the simple lack of alternatives, many utilities concluded contracts for the reprocessing of their spent fuel, in anticipation that there would always be a market for the plutonium and uranium that were to be separated from their spent fuel. Consistent with this scenario the pools to store spent fuel at the reactor site were designed to hold a relatively small number of spent fuel assemblies, enough to cool the spent fuel for a short period of time before it was to be shipped off to the reprocessing plant.

The assumptions underlying this scenario proved wrong. The growth of the nuclear power generating industry was overestimated as were the availability of cheap uranium and the difficulties of introducing a cost competitive fast breeder reactor system underestimated. As a result, few breeder reactors were operated, using the plutonium separated from spent LWR-fuel. In this situation stocks of separated plutonium grew to levels which are of concern to some countries. It was felt, that such stockpiles of separated plutonium were more difficult to protect against theft and misuse than the plutonium that is mixed with high active fission products in the spent fuel. This is a concern typically expressed by governments. The utilities had a different concern:

Plutonium is radiotoxic and - if fabricated into MOX-fuel - requires elaborate protection measures that guarantee, that no plutonium is inhaled or digested by workers in the plutonium fabrication facility. Plutonium from power reactors -contrary to plutonium from production reactors- also contains an important fraction of the isotope plutonium 241 that decays into americium, a strong γ-emitter. As a consequence, ageing plutonium originating from power reactors -in excess of 5 years in many cases- is the source of a strong γ-radiation, which requires remote handling during fabrication for worker protection or must undergo americium separation prior to processing. LWR-plutonium, that is not used within a few years after separation therefore becomes either unusable or very expensive to use.

Therefore most utilities, that have separated plutonium from earlier reprocessing of their spent fuel, have chosen to recycle the plutonium in their own reactor rather than storing it for future use beyond the point in time, when americium purification becomes necessary. After that time limit separated plutonium turn out to be a liability to the utility rather than a resource.

As a consequence, the economics for using MOX-fuel compared to uranium fuel for that utility that has separated plutonium is different: The procurement cost of a uranium assembly are added to the storage cost of unused separated plutonium and eventually the cost of separating the americium compared to the procurement of MOX-fuel. A utility that has separated plutonium therefore will almost always choose to recycle the separated plutonium as MOX-fuel even if this occurs at a procurement cost disadvantage for MOX-fuel.

Fortunately, there is additional value in going this route.

- Managing spent fuel storage capacity

Reprocessing spent fuel and recycling the contained uranium and plutonium reduces the uranium requirement by about 35 % compared with the so called once through cycle and reduces the required space for intermediate storage of spent fuel to about 25 to 30 % of the space required to store the equivalent amount of spent fuel without reprocessing. This is a pressing issue in particular in some Asian countries, where the development of nuclear power appears constrained by their ability to store the resulting spent fuel for the long term. For this reason these countries have an ongoing interest in the reprocessing of at least some of their spent fuel.

- Designing cores with MOX-fuel

On average over the entire residence time in the reactor, MOX-fuel is generally designed to produce the same amount of energy as uranium fuel, loaded into the reactor at the same time. Compared to uranium fuel, MOX-fuel is less reactive when first loaded in the reactor and more reactive later in life. This difference provides the core designer with added flexibility to design core loadings that are optimized for power distribution, economic fuel use and low neutron flux in the reactor vessel, which can limit the useful life of a nuclear power plant.

Whilst the last of these advantages is insufficient on its own today to justify the use of MOX-fuel in lieu of cheaper uranium fuel, these are advantages that can be exploited once the use of MOX-fuel is decided for other reasons.

Up to now we have explored the situation of a utility, that has separated plutonium for whatever reason. There are many utilities, that do not have separated plutonium. Is there a case for going to MOX-fuel for these utilities as well?

4. USE MOX FUEL TO REDUCE WORLD INVENTORIES OF WEAPONS PLUTONIUM

With the reduction of nuclear warheads stipulated by the Strategic Arms Reduction Treaties, START I and START II significant amounts of weapons material have become surplus to defense needs of the parties to the treaties, Russia and the USA. This concerns both high enriched uranium (HEU) and plutonium. HEU has already found its way to the civil market for nuclear fuel for power reactors foe example through an agreement by the US to purchase 500 t of Heu after dilution to low enriched uranium, suitable for power reactor use. Other stockpiles of uranium are reduced at the same time resulting in about half of current civil reactor uranium requirements coming from other than conventional sources e.g. mining uranium. This uranium is readily usable in the civil fuel cycle and has a well established commercial value benefiting the former owners of the weapons material.

The situation is more complex for weapons plutonium. Both the US and Russia have each declared 50 t of weapons plutonium surplus to their defence needs a few years ago. In the US, the National Academy of Science has concluded, that a combination of use as MOX-fuel in reactors for 38 of the 50 t and glassification together with fission products for the remaining 12 t would be the most effective way to convert the weapons plutonium into a form that presents no greater proliferation risk, than spent fuel from civil reactors (spent fuel standard). Russia has always considered its plutonium, which was produced at great cost to the society, an energy resource, which would therefore be put to use as MOX-fuel in Russian reactors.

The US have proposed a "spent fuel standard" concept whereby a part of the US surplus weapons plutonium is planned to be converted into spent MOX- fuel. "Weapons grade plutonium" (plutonium separated from production reactor fuel and produced for the purpose of making nuclear weapons) is almost pure plutonium 239. Once converted into spent MOX-fuel, the isotopic composition of the plutonium is similar to that of spent uranium fuel from power reactors, making it less attractive for weapons-purposes. In addition, plutonium contained in spent fuel is protected against misuse by a very intense radiation field from the fission products, that are also present in the spent fuel.

These are the basis of the **Spent Fuel Standard**: By using plutonium as MOX-fuel in a power reactor the proliferation risk of weapons grade plutonium becomes similar to that of spent power reactor fuel, for the two reasons mentioned.

Neither the US nor Russia however has the industrial facilities to produce MOX-fuels on a industrial scale and significant investment and many years will be needed before any surplus weapon plutonium can be converted into a product that meets the spent fuel standard. The situation is complicated by the fact, that the relative economics do not favour MOX-fuel (see chapter1) on a straight procurement cost comparison. An incentive would therefore have to be created to induce utilities to use MOX-fuel in lieu of the cheaper uranium fuel. This is particularly true as there are few other intrinsic benefits in burning MOX when it is a third parties' plutonium, such as conserving expensive spent fuel storage capacity.

In addition there are various unresolved policy issues between the two countries that slow down progress:

- a need for symmetry in rate of converting conversion weapons plutonium between the US and Russia, and
- the potential conflict with the US's-non reprocessing policy.

Whilst numerous hurdles have yet to be overcome before ex-weapons plutonium will be used as MOX-fuel in civil reactors, this is a very effective way to improve the proliferation resistance for weapons plutonium in the short term.

5. USE MOX-FUEL TO TRADE A SHORT-TERM PROLIFERATION RISK FOR LONG-TERM PROLIFERATION SECURITY

Most fission products have a short half lives (time needed for half of the fission products to decay) in comparison to the plutonium. The radiation field disappears in the course of 100 to 300 years as the fission products in the spent fuel decay into stable elements. This feature of spent fuel effectively limits the value of the storage of plutonium in the form of spent fuel for reasons of proliferation resistance to about 100 to 300 years after discharge.

There is yet another effect that limits the value of this disposal route. Some of the shorter lived plutonium isotopes (Plutonium 238 and 240, which are not fissionable) decay faster than the most important fissionable plutonium isotope 239. Over long time periods the quality of plutonium changes towards increased percentage of fissionable plutonium 239, making it more attractive for weapons purposes and diversion at a time, when the protection from the radiation field of fission products contained in the spent fuel has long disappeared.

Several technical concepts are being proposed to address this situation. One option is to separate the plutonium from the spent uranium fuel through reprocessing and burning it as conventional MOX- fuel. This is a simple, technically proven process to effectively reduce the fraction of fissionable plutonium isotopes in the plutonium in half, making it less attractive for weapons purposes.

Reprocessing facilities and MOX-fabrication facilities operate under stringent non-proliferation protection measures and security arrangements have proven effective in the past, as no diversion of any plutonium has knowingly occurred from the large industrial facilities. By taking this well defined and controlled short term proliferation risk associated with the operation of reprocessing and MOX-fabrication plants the long term proliferation risk of unreprocessed spent uranium fuel in more than 30 countries can certainly be meliorated.

If the proliferation risk associated with spent MOX-fuel with its low percentage of fissile isotopes in the plutonium is considered too great a risk to take in the long term, there are more advanced technologies under research and development, such as using the plutonium as MOX-fuel in a inert matrixes. This concept promises virtually complete combustion of all plutonium isotopes without generation of new plutonium from the uranium 238 matrix in conventional MOX-.fuel. Another possibility is the partitioning of plutonium and other Transuranium elements from the spent fuel and the transmutation into other elements in purpose built facilities. Several years will be needed to demonstrate the compatibility of such fuel concepts with the existing power reactors. A substantial effort will be required to convert the existing organisation of the fuel cycle to more proliferation resistant advanced fuel cycles. Industry will only commit to this effort, if supported by a broad international consensus on the desired reasonable degree of proliferation resistance.

On balance it seems inconsistent to ban plutonium and MOX-use from the civil nuclear fuel cycle for reasons of proliferation risk posed by the operation of reprocessing and MOX-fabrication plants and to leave plutonium in very large quantities in the spent power reactor fuel for an indefinite period of time in a large number of counties. The options to reduce the long term proliferation risk are limited:

- either separate the plutonium from the spent fuel and recycle it in some form of MOX fuel
- or store the spent fuel permanently in a few internationally controlled storage facilities.

If none of these option meets the necessary broad international support, different routes will be followed by individual countries to deal with the proliferation risks of the nuclear material under their control. What matters for the management of spent reactor fuel is the potential diversion of spent reactor fuel for weapons purposes by a group of people with moderate resources and not the capabilities of states with advanced nuclear weapons capabilities. The realistically assessed proliferation risks associated with various types of spent commercial fuel should therefore be the basis for a convention on the management of the short- and long-term management of proliferation risks, a Non Proliferation Convention, which should meet broad international consensus.

French plutonium management program

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Abstract. The French plutonium management program is summarized in this paper. The program considers nuclear generation as a major component of national electric power supply and includes the reprocessing of the spent fuel.

SUMMARY

French nuclear energy policy, set up and confirmed by various governments from the beginning of this industry, considers nuclear generation as a major component of national electric power supply and includes the reprocessing of the spent fuel.

The French Government and the Parliament are involved in the nuclear policy mainly through the action of Ministry of Industry and Ministry of Environment for the former and Parliamentary Office for Assessment of Scientific and Technological Options for the latter.

The main actors of the nuclear industry are the Commissariat à l'Energie Atomique (CEA), national agency in charge of basic research and tests, Electricité de France (EDF), national utility operating the power plants, COGEMA in charge of the nuclear fuel cycle, FRAMATOME designer and manufacturer of nuclear reactors and ANDRA, national agency for waste management.

The Reprocessing-Recycling strategy pursued in the last thirty years has allowed the development of a fully mature industry in the field of spent fuel management, with a world-wide international business development. At present in France 955 tons of enriched uranium fuel are loaded each year, among which 850 tons of spent fuel are reprocessed allowing the recovery of 8 tons of plutonium used in MOX fuel fabrication. This MOX fuel is loaded in 900 MWe PWRs. Already part of the reprocessed uranium is reenriched and recycled.

At present 18 reactors in France are loaded with MOX fuel and the whole set of 900 MW-class reactors (28) are expected to be loaded with MOX in the future.

In Europe, COGEMA controls 97 % of the MOX fuel manufacturing which is processed in three facilities: DESSEL in Belgium, CADARACHE and MELOX in France.

The Reprocessing-Recycling strategy is considered as presenting many advantages in various fields, such as recovery of valuable materials, conditioning of final non reusable waste, reduction in waste radiotoxicity. Moreover it leads to a significant reduction in the volume of final waste. This reduction has been given the benefits of strong improvements obtained through the completion of large research and development actions along the years. Several economical studies have not found any significant difference in costs between this option and the direct disposal option.

Beyond the present status of the recycling industry, Research and Development actions regarding the back-end of the fuel cycle are in progress in various directions: widening of the field of MOX use (BWR, EPR, etc.), higher burnups (70 GWd/t), plutonium multiple recycling, etc.

1. ORGANIZATION: GOVERNMENT AND INDUSTRY

French Nuclear Energy Policy and Reprocessing-Recycling

French energy policy from the very beginning has placed the generation of electric nuclear power as a major component of the supply of energy. This policy included the reprocessing of spent fuel, considered as not questionable due to the risk of uranium shortage and the development of breeders reactors. All governments that succeeded one another in France have regularly approved the options taken in the 1970s. The French Parliament has also regularly confirmed its support of the strategy for energy proposed by the Government. When it became clear that the breeder reactors policy had to be interrupted, reprocessing was questioned, but accounting for several major factors, feasibility of recycling of plutonium in PWR reactors, favourable outlooks of a back-end strategy for final waste, and economical aspects, the Reprocessing-Recycling strategy was confirmed and adopted by all main actors of the French Nuclear Industry.

Main actors

The French organisation in the field of Nuclear Energy is based on the following actors: the CEA, Commissariat à l'Energie Atomique, national research centre, ANDRA, national nuclear waste agency, Electricité de France, the national utility in charge of electric energy supply, which operates electric power generating plants (58 PWR reactors), FRAMATOME, the nuclear reactors manufacturer, and COGEMA in charge of the nuclear fuel cycle, starting from uranium mining, enrichment, fuel manufacturing, reprocessing and final waste conditioning.

Government and Parliament bodies

Concerning political and administrative entities and technical and safety aspects, the main body is the DSIN (Direction de la Sûreté des Installations Nucléaires, Directorate for Nuclear Installations Safety) which is directly responsible both to the Ministries of Environment and Industry. This Directorate is in charge of all safety matters in relation with nuclear energy, including transportation of nuclear material and waste management. DSIN uses as main technical advisor the IPSN (Institut de Protection et de Sûreté Nucléaire).

The DSIN role includes the assessment of the consistency of the choices made for the back-end fuel cycle options, particularly the **plutonium management.**

Parliamentary Office for Assessment of Scientific and Technological Options

The Government and the Parliament are acting through various bodies. On the parliamentary side we find the **Parliamentary Office for Assessment of Scientific and Technological Options**, composed of Members of Parliament, whose missions are:

- to inform the Parliament about the consequences of the scientific and technological choices in order to make all important decisions clear,
- to facilitate the legislative function by working upstream of the bill and law drafting process,
- to enforce the parliamentary control over the executive.

This Office issues information and guideline reports in order to inform and guide the Government as well as the Members of Parliament about the French energy policy. Two lately issued reports illustrate this important role of the Office:

- the Bataille-Galley report on development in research on wastes of March 1996,
- the Bataille-Galley report on fuel cycle back-end dated 11^{th} of March 1999,

whose main conclusions were that the fundamental options implemented in France regarding the back-end fuel cycle, namely spent fuel reprocessing and fissile material (plutonium and uranium) recycling should be maintained.

National Assessment Committee

Another important component of the French Organisation is the National Assessment Committee (Commission Nationale d'Evaluation), composed of 12 independent French and international experts named by the Government, the Parliament and the High Council for Nuclear Safety and Information. This Committee was stipulated in the 1991 law on high-level long-lived radioactive waste and formally created in 1993. The main task of the Committee is to organize regularly deep audits of all nuclear actors and issue a yearly synthesis report on high-level waste management. Its missions are in fact extended to the overall back-end fuel cycle which includes plutonium management.

Broad consensus and late political decisions

As explained here above, the policy towards nuclear energy has always been confirmed democratically and the last decisions made by Mr. Lionel Jospin's Government do not diverge from this line in confirming that « *the choice of nuclear energy as a major component of the national electric power supply will be maintained* », as stated at the highest level in 1998 on the occasion of the 9^{th} December meeting of an interministerial committee. One may also note the conclusions of the parliamentary debate on energy on the 21^{th} January 1999. With the exception of the « Green Party », which is no surprise, the French commitment to the nuclear options meets a very broad consensus amongst the French political staff.

2. PANORAMA OF THE BACK-END FUEL CYCLE

General

As stated in the general overview of political and administrative aspects of the nuclear energy framework in France, spent fuel has been continuously reprocessed in France since the beginning of the nuclear program. Despite some noticeable changes in the nuclear policy, as either the shift from Gas Cooled Reactors to Light Water Reactors as basic generators of electric power, in the late sixties, or the more recent phasing-out of Superphenix breeder reactor, all actors of the nuclear industry consider spent fuel reprocessing and valuable fissile material recycling as the only satisfactory solution for the back-end problem. At present the recycling of plutonium in MOX fuel to be loaded in the core of PWR reactors, whose decision was taken in 1984, is the current policy. The feasibility of plutonium recycling in PWRs had long since been demonstrated through various programs and experimentation, and in 1987 actual loading of reactors with MOX fuel began.

So France has acquired a very large experience in all technologies related to Reprocessing-Recycling, spanning over 30 years. This strategy has reached an unquestionable industrial reality, and has allowed the development of a mature industry as well as an international business development.

3. REPROCESSING

COGEMA operates at La Hague two large reprocessing facilities, UP2 and UP3. They have been designed with a nominal capacity of 800 tons of fuel each and their combined production has reached this target since five years. These plants basically deal with LWR fuel, but they are capable of other fuel, as Fast Breeder Reactor fuel, and MOX fuel and such operations have already been made.

The waste generated by the process has been drastically reduced in volume during the last years. Through improvement of the existing technologies and the adoption of new concepts, the specific volume of the ultimate residues has been reduced from 3 cubic meters per ton of uranium in the 80s to the present figure of less than half a cubic meter, with a content in plutonium less than 0.1 % of the original, whereas the forecasted figure for the direct disposal option is about 2 cubic meters.

4. RECYCLING

MOX fabrication plants

As concerning the MOX fuel designed for LWR reactors, the COGEMA Group supplies the production of three plants:

- two of them, CADARACHE and MELOX are operated by COGEMA and are located in France;
- the third, located at DESSEL in Belgium is operated by BELGONUCLEAIRE.

The MOX platform, with this set of these three plants, delivers 97 % of the MOX consumption in Europe, corresponding to 31 reactors (32 European reactors are currently loaded with MOX fuel). The COGEMA Group will be able in the future to supply all recycling needs of its back-end customers.

COGEMA has also started the production of MOX fuel for several Japanese utilities.

The process used in these plants is called the MIMAS process and has become a world-wide reference. The Department of Energy of the United States has selected MIMAS for its program of elimination of weapon-grade plutonium into MOX fuel (which demonstrates another benefit of the Reprocessing-Recycling strategy: getting rid of non civilian plutonium).

Reactors served

In France, with the 58 PWR reactors of EDF under operation, 955 tons of uranium fuel are loaded each year. Out of them, 850 tons are reprocessed in order to recover about 8 tons of plutonium that are needed for the manufacturing of MOX fuel elements. At present 18 PWR reactors receive and use MOX fuel elements and 28 are planned to be loaded with MOX fuel in the future.

Uranium recycling

It is important to note that the reprocessing technology allows also the recovery of slightly enriched uranium, which is a valuable material that may be further reenriched and reused in cores. The COMHUREX plant in southern France is one of the world few units to convert the uranium coming from the reprocessing plant under the form of uranyl nitrate into uranium hexafluoride (UF6) so as to be reenriched and thus allows the fabrication of URE (Enriched Reprocessed Uranium fuel). This is currently done at the FBFC-Romans plant which produces two fuel reloads per year. Two reactors in France are authorized to load reprocessed and reenriched uranium, CRUAS 4 was first loaded in 1987 and since 1995 CRUAS 3 and 4 burn URE exclusively.

5. ADVANTAGES OF REPROCESSING-CONDITIONING-RECYCLING (RCR)

The advantages of the Reprocessing-Conditioning-Recycling strategy have been placed in a prominent position through many studies, and large industrial experience feedback.

Overall economy

On the economical and strategic point of view, it appears that this option is a guarantee for a supply of nuclear fuel at stable conditions in the long term, not subject to fluctuation as are other energy resources such as fossil fuel.

Furthermore, in terms of costs, though the economy of the RCR option has often been questioned - and mainly by nuclear opponents...-, many studies have compared the once-through option (direct disposal of spent fuel) to the RCR and no significant difference appears between them. The OECD launched a thorough study of the overall costs and its result was that the difference is lower than 10 %. It must be stressed that the main investments have been made, whereas no remarquable installation for direct disposal exists nor has really proven its feasibility, and the cost of this option is still speculative. Moreover, the trend towards an increase of fuel discharge burnups will clearly improve the competitiveness of the MOX fuel and thus enforce the RCR option.

Though the RCR industry has proven to be mature under several major aspects, it is still a young moving forward industry and many improvements are expected from R & D in the near future, that will enhance both safety and productivity so as to make it more and more competitive.

On the utility point of view, it is also clear that all supplementary investment in spent fuel pool capacity via reracking or pool extension at the reactor are avoided, as in and out flows of spent fuel are permanently adjusted.

Final waste

As regarding the final waste, the reprocessing technology leads to small volumes of non-reusable materials, which are sorted and separated from the valuable ones and may be safely conditioned. Many improvements to this process have been performed during the late years.

The overall radiological balance is improved by plutonium recycling in MOX fuel. A very small quantity of plutonium is present in the final waste and the radiotoxicity of the irradiated MOX fuel is on the long-term lower than the radiotoxicity of the plutonium that would have not been used in MOX plus the uranium fuel which would have been used in place of MOX.

6. FUEL CYCLE BACK-END IN FRANCE - THE FUTURE

Research and development

As already stated above, the RCR industry in France devotes strong efforts in R&D studies so as to improve the back-end fuel cycle advantages in various fields. These efforts bear on the optimisation and adaptation of MOX fuel fabrication processes, the increase of discharge burnups, the increase of the in-core MOX fuel ratio, the multiple recycling of plutonium.

Concerning the MOX fuel fabrication, studies have been launched within the Russian-and-US program of elimination of weapon grade plutonium, by way of making it usable for burning in reactors.

Technological developments are in course with the aim of the improvement of MOX fabrication plants operation and efficiency, through upgrading and renewal in the mid-term.

A large and very promising R & D program is at present also under progress regarding the increase of burnups. This increase as said above shall make the MOX more competitive. A new type of fuel assembly is under study that will be capable of reaching 70 GWd/t by the year 2010.

At present at MOX using reactors, the reload includes only a third of a core in MOX fuel assemblies. Studies for the increase of the in-core MOX fuel ratio up to 100 % are undertaken. It is also intended to assess the increase of MOX fuel content in BWR and EPR.

A very important topic, which shall give an answer to some questions, lies in the multiple recycling of plutonium. At present it can clearly be stated that the feasibility of MOX fuel reprocessing has been proven, as La Hague plant has already reprocessed at an industrial scale spent MOX fuel on the occasions of the 1992 and 1998 reprocessing campaign. Moreover, several other components of the MOX strategy, such as spent MOX fuel transportation have already been given a confirmation.

The next step will be the second recycling of plutonium in a light water reactor core.

Scenario studies

Within the framework of R & D programs described above, an industrial working group in which participate ANDRA, CEA, COGEMA and FRAMATOME has been set up in order to overview the strategic options for the fuel cycle back-end.

This group works on the following scheme: a set of basic scenarios that can be envisaged for the future have been selected, based on the following points:

- are large quantities of plutonium acceptable or not in deep geological disposal,
- the best concept for deep geological repositories, and comparison with other ways of managing waste,
- authorized burnups,
- available techniques (RNR, purger laser, actinides partitioning).

In line with the ideas developed, CEA and COGEMA have undertaken a specific study aiming at the assessment of what should be in the mid and long term the total plutonium inventory according to six different scenarios, which account for different concepts. Among these concepts one finds CAPRA and MIX.

CAPRA (Consommation Accrue de Plutonium en Réacteur) is a research program, initially based upon the use of Fast Neutron Reactors with the aim of increasing the consumption of plutonium in reactors, at present the studies made on a multinational basis keep the same objective with other reactors. The goal lies in designing and operating high plutonium content reactors.

MIX is a new concept of nuclear fuel based upon the use of a mix composed of enriched uranium and plutonium. This fuel should be well adapted to multi-recycling use.

At present it is possible to give a partial information regarding assessment of future plutonium inventory via the outcome the study of six following scenarios:

- open cycle,
- a single recycling of plutonium in MOX fuel,
- multi-recycling of plutonium in MOX fuel,
- implementation of CAPRA,
- use partial MIX fuel elements,
- use of 100 % MIX fuel elements.

According to the result of this study, the plutonium inventory should:

- increase in the case of open cycle or a single recycling,
- remain stable in the case of CAPRA processing or multi-recycling of plutonium in MOX fuel,
- decrease in the case of implementation of a MIX strategy.

Another noticeable outcome of the study of these scenario concerns the radiotoxicity of the final waste. When comparing the five latter options to the open cycle in terms of radiotoxicity (through the ingestion exposure pathway) of the final waste, it appears clearly that, taking the open cycle as reference, the ratios are far in favour of all options based upon the reprocessing-recycling strategy, in the mid term and particularly in the very long term, with a factor of 10 after 10 000 years.

7. CONCLUSION

The Reprocessing-Conditioning-Recycling policy, at its current status has widely sustained the nuclear development in France. It relies on improved and well-mastered processes implemented on an industrial scale. It provides a steady solution to the back-end fuel cycle: low volume and lower toxicity of High Level Waste. Thus on account of the numerous advantages offered, the fundamental option for dealing with spent fuel in France is well justified.

As no irreversible option has yet been taken, this back-end solution keeps fully open the future. The disposal solution for the high level waste is to be taken by the Parliament in 2006, so the Fuel Cycle completion will be reached.

Thanks to a substantial investment on R & D, in order to improve and optimize the production capability, to develop and check new concepts, new products and new services, the RCR industry, though mature, has an enormous potential for further development.

Status of MOX fuel fabrication, utilization, design and performance in LWRs

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Abstract. The overview looks at the present status of Pu recycle as MOX fuel in thermal power reactors as presented mainly at the IAEA Symposium on MOX Fuel Cycle Technologies for Medium and Long Term Deployment held in Vienna in May 1999, and some other international meetings. Present status in MOX fuel fabrication, performance and utilization technologies are discussed. Recycling of plutonium as MOX fuel in LWRs has become a mature industry. The technology is well understood and the facilities, institutions and procedures are in place (capacity extensions being planned) to meet the anticipated arisings of separated plutonium from the production of nuclear power. MOX fuel performance and its modelling are discussed in comparison with those of UO_2 fuel.

1. INTRODUCTION

Today the plutonium recycling with LWRs has evolved to industrial level in some countries relying on the recycling policy. This happened mainly because of the worldwide delays in development of fast reactor's programmes. More than 30 years of reactor experience using MOX fuel as well as the fabrication of 2 000 MOX assemblies with the use of 85 t of Pu separated from spent fuel from power reactors and good performance records indicate now that the recycling of plutonium as MOX fuel in LWRs has become a mature industry. The technology is well understood and the facilities, institutions and procedures are in place to meet the anticipated arisings of separated plutonium from the production of nuclear power.

The number of countries engaged on plutonium recycling could be increasing in the near future, aiming the reduction of stockpiles of separated plutonium from earlier and existing reprocessing contracts. Economic and strategic considerations being the main factors on which to base such decision to use MOX. The reactor-based weapons-grade plutonium disposition approaches proposed by the US and Russia are build upon proven commercial MOX fuel technologies and may further contribute to optimization of Pu recycling technologies. An International Symposium on MOX Fuel Cycle Technologies for Medium and Long Term Deployment, organized by the IAEA in co-operaton with OECD/NEA and held in Vienna in May 1999, was a milestone to look in past, present and future of Pu recycling in nuclear power reactors [2].

2. MOX FUEL FABRICATION

Currently, six plants for MOX fuel fabrication are in operation in Belgium, France, Japan, UK and India (see Table I, [1]). Worldwide MOX fuel fabrication capacity at the end of 1998 amounts about 220 tHM per year. In the UK, a large scale MOX fabrication plant (SMP) has been constructed and is awaiting consent to start operation. In Russia, the first pilot plant (with a capacity of 10 t HM/y) for fabricating MOX fuel is under construction inside the RT-1 plant. A new MOX plant (Complex 300) is planned to commence operation in 2010 [3,4]. There are plans for the construction of a new MOX plant in Japan and of a demonstration facility in China. The available MOX fabrication capacities worldwide are projected to be over 600 tHM/y in the coming decade owing to deployment of new facilities and expansion of capacities of existing facilities, where it is anticipated that some 25 to 30 t of plutonium will be recycled per/year.

Fuel fabrication records as of 31 December 1998 for LWRs and FBRs [5] have shown that more than 550,000 LWR fuel rods for more than 3 300 fuel assemblies were produced in

facilities outside the United States [5]. FBR MOX production totals about 680 000 rods for more than 3 900 assemblies [5].

TABLE I. MOX fuel fabrication capacity (tHM/y), as of end 1998 [1]

Country	*Site*	*Plant*	*1998*	*2000*	*2005*	*2010*
Belgium	Dessel	P0	35	40	40	40
France	Cadarache	CFC	35	40	40	40
	Marcoule	MELOX	120	200[a]	200[a]	250[a]
India	Tarapur	AFFF	5	10	10	10
Japan	Tokai	PFPF	15[b]	15[b]	5[c]	5[c]
	Rokkasho-mura	MOX FFF	-	-	100	100
Russian Fed.	Chelyabinsk	inside RT-1	-	-	10	10
	Chelyabinsk	Mayak, Complex 300				40
UK	Sellafield	MDF	8	8	8	8
	Sellafield	SMP	-	120	120	120
Total			218	433	533	623

[a] date not fixed.
[b] for ATR Fugen and FBR Monju.
[c] for FBR Monju.

As MOX fabrication has progressed from its initial stages several process improvements have been made, especially in the area of powder production. The objectives of these improvements included producing a powder product with improved characteristics for pelletizing and improved fuel performance, especially related to fission gas release. [6-10]. Reduction of the size of Pu agglomerates is an important task in powder technology optimization[5, 11-13]. Enlargement of UO_2 grain size and diminishing Pu/U diffusion coefficients might be obtained through the use of small quantities (500-2000 ppm) of oxide (Ti, Cr and some others) additives [10, 12].

Process improvements have also been made with respect to reducing powder dispersion, thereby improving hold-up and personnel radiation protection [5, 11, 12]. Waste handling and scrap recycle have also been improved [5, 10, 11]. These are both important with respect to cost and safeguards.

Several aqueous [14, 15] and pyrochemical [16] processes of conversion of weapons-grade plutonium to MOX are under investigation in Russia. Ammonium co-precipitation of Pu and U with parallel involvement of surface active substances (SAS) is identified as the most promising (GRANAT, Combine “Majak”). A pyrochemical technology of the conversion of metallic Pu into MOX FBR fuel in molten salts has been developed in NIAR, Dimitrovgrad [16]. Vibro-pack method is used instead of pelletizing process. The letter is under development now to fabricate MOX fuel rods for WWER reactors.

IAEA/EURATOM safeguards monitoring in MOX fabrication facilities is well organized [5, 17-19]. An unattended NDA measurement system recently deployed at Belgonucleaire was described and the savings in cost and personnel time resulting from use of the system were presented. The IAEA’s Tank Monitoring System (TAMS) and the benefits of its use with respect to timely (every 15 seconds) and accurate data collection was noticed. The safeguards will be strengthened with the enforcement of the IAEA's 93+2 Programme, resulting in additional constraints.

3. MOX FUEL UTILIZATION

Today, more than 30 thermal reactors have used MOX-fuel complying with a partial core loading pattern [20]. The main part of the NPP using MOX are 33 PWRs (19 in France [12], 9 in Germany, 2 in Belgium and 3 in Switzerland), but 2 BWRs use also MOX (in Germany). Table 2 lists the current status of MOX fuel utilization in thermal reactors worldwide. The commercial application of MOX fuel in LWRs has been started in the mid 1980s when some modification or withdrawal of fast reactor programmes was enforced. The developed technologies of recycling and fuel fabrication were applied for Pu-recycling in LWR fuel, in the mean time focusing on stabilisation of the separated plutonium inventory.

Currently, the use of MOX fuel has been established on an industrial scale in a number of countries. In Belgium, France, Germany, Japan and Switzerland, a considerable number of thermal power reactors (PWRs and BWRs) are either licensed (i.e. 40, of which 34 have MOX fuel loaded) or have applied for a license (about 13) to use MOX fuel at levels of up to 30% of the reactor core.

Up to now, more than 2000 FAs have been loaded in LWRs in Europe [20, 21]. In France, since 1987 about 1200 MOX assemblies have been loaded in 19 PWRs (900 MWe - 17x17), and about 600 FAs achieved 3 cycles, and a few FAs were authorized for 4 and 5 cycles for experimental purposes [12]. In Germany, since 1985, about 800 MOX FAs have been loaded in 9 PWRs (16x16 and 18x18) and 2 BWRs : Gundremmingen B and C (1344 MWe-9x9). In Switzerland, about 160 MOX FAs have been loaded in PWRs : since 1978 in Beznau 1 and 2 (PWR 350MWe - 14x14) and recently in Gosgen (PWR 1000MWe - 15x15). In Belgium, since 1995 about 70 MOX FAs have been loaded in Doel 3 and Tihange 2 (PWR 1000MWe - 17x17). Loading of MOX FAs are postponed in Japan after falsification of MOX pellets by BNFL. Experience with MOX reloads in BWRs and PWRs from 1981 to 1988 is given in Table 3 [20, 21].

TABLE II. Present Status of large scale MOX fuel utilisation in thermal reactors[a]

	Number of Thermal Reactors			
	Operating reactors	Licensed to use MOX FAs	Loaded with MOX FAs	Applied for MOX license
Belgium	7	2	2	
France	57	20	19	8[b]
Germany	19	12	11	4
Japan	52	3	1	1
Switzerland	5	3	3	
Total	130	40	36	13

[a] There are a number of reactors, notably in Europe and India, not included in this Table, which are licensed to use MOX fuel on an experimental basis;

[b] Technically capable reactors planned to be licensed.

Highest MOX fuel burnup (42-60 MWd/kg HM) was reached in Switzerland, Beznau 1 and 2 [20]. In Germany, mean MOX FA discharge exposure is about 40 GWd/t, maximum MOX FA exposure is about 48 GWd/t in commercial operation. The trend in this field is increasing and now many of the MOX FAs under irradiation will reach mean discharge exposure around 48 GWd/t. In France, the average burnup rate per MOX reload is maintained at around

37 MWd/kg HM and the maximum assembly burn-up rate at 41 MWd/kg HM, compared with the 50 MWd/kg HM reached by UO_2 assemblies obtained in annual cycle (3 cycles for MOX and 4 cycles for UO_2). By the beginning of the next century, the implementation of an optimised core management will increase MOX burn-up rate close to 50 MWd/kg HM.

TABLE III. EXPERIENCE WITH MOX RELOADS IN BWR AND PWR FROM 1981 TO 1998 [20]

COUNTRY/ REACTOR/ FA-TYPE	**No. OF REACTORS**	**No. OF MOX FA RELOADED**	**MAX. AV. PU_{FISS} IN W/O / CARRIER MATERIAL**	**MAX. FA – EXPOSURE AT EOC IN MWD/THM**
Belgium				
PWR (17x17-24)	2	56	4.9 / U_{tails}	43900
France				
PWR (17x17-24)	17	992	4.5 / U_{tails}	44250
Germany				
PWR (18x18-24)	2	4	4.6 / U_{tails}	8000
PWR (16x16-20)	5	364	4.2 / U_{tails}	44900
PWR (15x15-20)	1	32	3.0 / U_{nat}	42000
PWR (14x14-16)	1	41	3.8 / U_{nat}	37000
BWR (9x9-1)	2	116	3.0 / U_{tails}	32000
Switzerland				
PWR (15x15-20)	1	28	4.8 / U_{tails}	23000
PWR (14x14-17)	2	152	4.1 / U_{tails}	51000

All above-mentioned reactors were designed to use UO_2 fuel. The larger amount of plutonium in the core shifts the neutron spectrum towards higher energy levels, thereby reducing the efficiency of the reactivity control systems. The reduced neutron absorber worth needed in some cases to improve the control rod pattern, to increase boron concentration of the boron make-up storage tank and to increase boron concentration of the refuelling water storage tank. In some specific cases, use of enriched boron is needed. Taking into account these neutronic considerations, the number of MOX assemblies is limited on each reactor (MOX recycling rate), the limit depending upon the reactor initial design and its capability of evolution.

4. MOX FUEL DESIGN

MOX fuel is designed to perform to the same operational and safety criteria as uranium fuels under equivalent conditions. This is also confirmed by the parallel development of design codes to accommodate the special characteristics of MOX fuel. The MOX skeleton is generally identical to the skeleton of a typical uranium assembly. As an example - AFA-2G MOX FA (EDF-Framatome-CEA), designed for FA burnup of 43-45 MWd/kg HM, integrated the improvements in the current UO_2 FAs with a modified fuel rod design (mainly lower initial He pressure and a slightly increased free volume). These modifications were directed on compensation of higher fission gas release from MOX fuel compared to UO_2 fuel. French “MOX parity” programme is targeted on 4 cycle operation of UO_2-MOX core (EDF 900 Mwe PWRs) with FA (AFA-3G type) discharge burnup of 52 MWd/kg HM Further burnup increase is also planned by CEA-Framatome-EDF [12].

MOX assembly contains only mixed oxide rods. The plutonium oxide is mixed with natural or depleted uranium oxide to maximise the quantity of plutonium per assembly.

To achieve better balanced rod power distribution in the assemblies, several plutonium contents (three for PWRs) are used in the MOX assembly (zoning). The low plutonium content zone is located at the periphery of the MOX assembly in order to compensate for local power peak in the interface with the uranium assemblies induced by the large increase of the fission and absoption cross-sections of the MOX compared to the UO_2. The BWR MOX fuel assemblies are even more complex with up to 6 MOX fuel rod types to compensate the spectrum changes between the fuel channels. As for all BWR fuel assemblies also the MOX FAs usually have some Gd poisoned rods.

In order to meet the energy equivalence with enriched uranium fuel, the average plutonium content is adjusted using an equivalence formula that takes account of the isotopic composition of the plutonium. As the main part of the reprocessed fuel originates from the PWR reactors, the total plutonium contains between 60% and 70% of fissile isotopes of plutonium (239 and 241). Recently, the maximum Pu content in MOX FAs for EDF reactors was increased from 5.3 to 7.08 % to allow use of Pu coming from reprocessing of highly irradiated UO_2 FAs [12].

Regarding core management, the goal of each utility is to obtain the equivalence between MOX and UO_2 in terms of fuel performance and reactor operation.

5. MOX FUEL PERFORMANCE: EXPERIMENTS AND MODELLING

National and international programmes to evaluate performance MOX fuel in comparison to that of UO_2 have been carried over a period of 35 years [22]. These programmes, which are still ongoing, are providing the data necessary to compare MOX fuel behaviour with that of UO_2, and to develop specific MOX fuel performance models and verify core design codes. A wide range of variables has been investigated in these test irradiations e.g. fabrication processes, cladding materials, rod geometries, operating conditions [22]. Also, PIE of commercial MOX FAs added a lot of data on MOX fuel performance and reliability. It has been concluded from both data sources that :

- MOX fuel reliability remains as good as that of UO_2 FAs;
- overall performance of the MOX fuel has proved to be as good as that of equivalent UO_2 fuel but with the added benefit of improved resistance to PCI.

Because of extremely big UO_2 fuel performance database and respective well validated codes, it is important to define differences in structure and performance of MOX and UO_2 fuels. Very recent IAEA Technical Committee Meeting on Nuclear Fuel Behaviour Modelling at High Burnup and its Experimental Support (Windermere, 19-23 June 2000) reaffirmed, that fuel performance models developed for UO_2 and based on UO_2 experience are generally applicable to MOX. Only several models need to be adapted to cope with quantifiable differences between MOX and UO_2 fuels, and only some of these characteristics are affected by the microstructural pattern of the MOX fuel. It has been indicated that MOX fuel modelling has reached a high degree of development, based on a good understanding of the differencies between MOX and UO_2 fuel. Serious progress was reported on codes capable to evaluate and predict MOX fuel performance including COPERNIC (Framatome-EDF-CEA, [23]), ENIGMA-B (BNFL, [24]), FPAC (NFI, [25]), COSMOS (KAERI, [26]). Detailed description of MOX fuel international and national R & D programmes is presented in [22].

Radial power and burnup profiles. The differences arise from the presence of significant quantities of Pu in MOX fuel at the beginning of life, and their subsequent evolution under irradiation [24]. Also, Pu has a greater neutron absorption cross-section than uranium. This is why radial power and burnup profiles are "flatter" in MOX pellets than in UO_2 pellets and Linear Heat Generation Rate (LHGR) is rather higher during 3rd and 4th irradiation cycles. As an example, typical LHGRs for 4-year cycle is given at Fig. 1 [6]. It is seen that for the 4 th cycle LHGR is about 170 W/cm, for UO_2 FAs, located near by, it is about 120 W/cm.

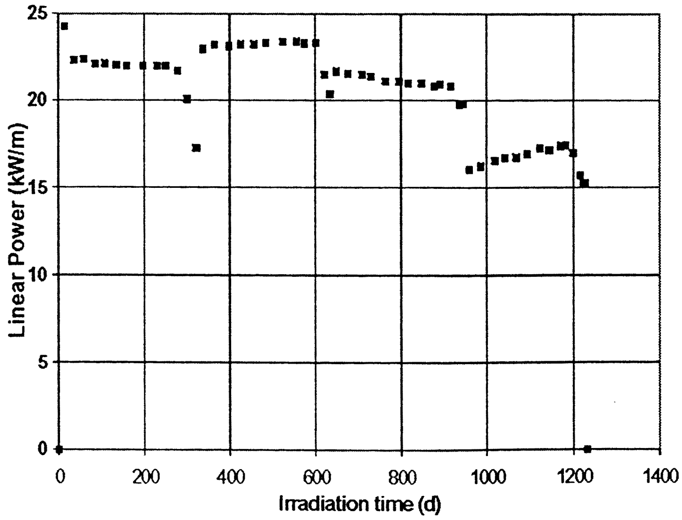

FIG. 1. Example of evolution of the rod averaged LHGR [6].

Fission product and helium generation. The differences arise because of the factors already mentioned above. The primary source of He generation in oxide fuels during irradiation is alpha decay trans-uranium elements, primarily Cm-242. The measured and calculated He production in MOX fuel is about 4 times that of UO_2 fuel [27] because of shorter chain to transpose Pu to CM-242 than for uranium [24, 27]. Volume fraction of helium in the total gas released into free volume of irradiated MOX fuel might exceed 30% [see references in 27].

Thermal conductivity. There is a slight degradation of the thermal conductivity of the MOX as compared to UO_2. However there is no agreement for the quantitative trend of the degradation as has been noticed at the above-mentioned IAEA TCM on Nuclear Fuel Behaviour Modelling. For example, KAERI [26] describes this degradation as a two-phase material with different conductivities in the matrix and the Pu-rich agglomerates(reduction of thermal conductivity ranges from 10 to 7%). BNFL [28] applies a uniform degradation of 8%. Franatome-EDF-CEA applies a linear degradation with the Pu content [23].

Fuel creep. Creep rate is deeply related to PCI behaviour of fuel rods. Out-of-pile creep rate of MOX pellets is pretty larger than that of UO_2 pellets which seems to lead to much mitigation of PCI comparing to UO_2 [29]. On the other hand, it was found by a comprehensive review [30] of creep rates of irradiated fuel pellets that radiation-enhanced creep rates of both oxide pellets were not so significantly different from each other at the temperature of operation.

Fission gas release. Because of complexity of MOX fuel structure, both before and during irradiation, and higher center-line temperature of MOX fuel in comparison to UO_2 fuels, it is difficult to have simple judgement on fission gas release from these fuels for the whole range of heat rates, burnups and fabrication technologies. The onset of intensive FGR is observed at lower burnups for MOX rods than for UO_2 rods for the same heat rates because of higher center-line temperature (Fig. 2). Figure 2 presents CEA-Framatome-EDF data on FGR for

UO_2 and MOX rods with similar power history irradiated during 1-4 cycles in EDF 900 MWe reactors. Conclusion has been drawn that no FGR enhancement occurs in MOX fuel due solely to the burnup effect. It can be also noticed from this figure that difference in FGR from MOX fuel and UO_2 fuel decreases when homogeneity of Pu distribution is improved (comparison of MIMAS ADU and AUC MOX fuels, see part). It has been also confirmed by low FGR from irradiated (up to 35 MWd/kg HM) SBR MOX fuel with good homogeneity in initial Pu distribution [31]. Above-mentioned "MOX parity" program foresees improvement of MIMAS technology to reduce the level of FGR by enlarging grain size, obtaining better homogeneity in Pu distribution and lowering of self-diffusion rate.

Fuel rod growth, water side corrosion: These characteristics of MOX fuel rods are fairly the same as those of UO_2 fuel rods [12, 22].

Transient behaviour. Transient tests of high burnup MOX fuel rods (up to 50 GWd/t HM) indicated that integrity of the fuel rods was maintained without defects up to power levels of 416 - 474 W/cm with a rate of 100 W/cmxmin. This result would imply better transient performance of the MOX fuel rods than the UO_2 rods [32]. The fission gas release from MOX pellets in transient conditions is clearly compatible with those from UO_2 pellets [33, 34].

RIA behaviour: Three tests (Na-6, 7 and 8) were performed in France on Cabri reactor on different burnup MOX fuel rods (Table 4, [35]). REP-Na7 resulted in a clad rupture. Though being at a much more higher enthalpy level than expected in reactor from theoretical neutronic calculations, this rupture gives a suspicion of a fission gas effect enhanced in MOX with regard to UO_2, leading to a cladding pressure loading. Detailed PIE of failed rod is under way. Further RIA tests for MOX rods are planned.

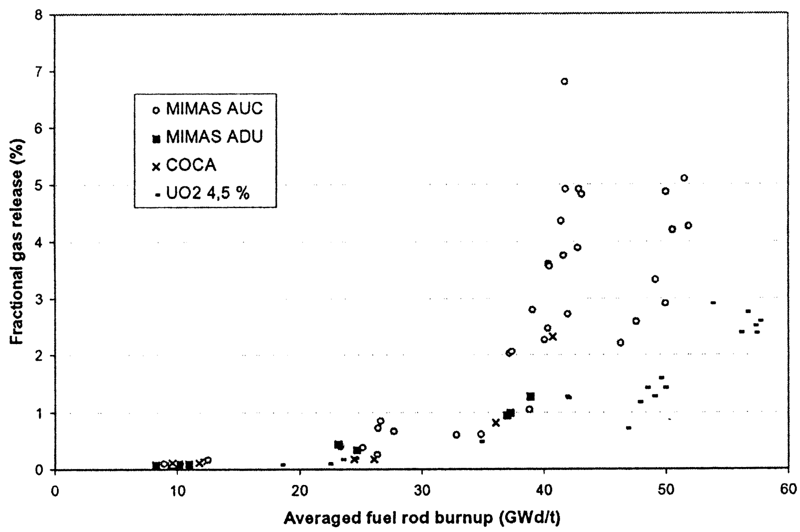

FIG. 2. Fission gas release in irradiated MOX fuel rods (CEA-Framatome-EDF [6]).

Summing up the above-mentioned available information on MOX fuel performance and having in mind presently accepted national and international MOX fuel improvement programmes, it is possible to conclude that the task of "MOX parity" for high burnup operation will be successfully solved in near future.

TABLE IV. The MOX-fuel tests of the Cabri REP-Na test matrix [35]

Test (date)	*Tested rod*	*Pulse (ms)*	*Energy at pulse end (cal/g)*	*Clad Corrosion (μ ZrO_2)*	*Results and remarks*
Na-6 (3/96)	**EDF MOX, 3c span 5 47 GWd/t**	**40**	**165 (at 1.2 s) (690 J/g)**	**35**	No rupture **Hmax = 148 cal/g** $\Delta\phi/\phi$: 2.65 % (mean max) FGR = 21,6 %
Na-7 (2/97)	**EDF MOX, 4c span 5 55 GWd/t**	**40**	**175 (at 1.2 s) (732 J/g)**	**50**	**Rupture at 120 cal/g** **Hmax = 140 cal/g** **Pressure peaks** **Fuel dispersal** Examination currently carried out
Na-9 (4/97)	**EDF MOX 2c span 5 28 GWj/t**	**34**	**241 (at 1.2 s) (953 J/g)**	**<20**	No rupture **Hmax = 210 cal/g** **$\Delta\phi/\phi$ = 7.3 % mean max** FGR ≈ 35% (to be confirmed) Examination currently carried out

***Note**: the burn-up indicated in the table is the maximum burn-up of the test rodlet : the father rod has a burn-up 10 to 12% lower.*

ACKNOWLEDGEMENTS

The author expresses his thanks to all speakers and Chairmen of the Sessions of the IAEA Symposium on MOX Fuel Cycle Technologies for Medium and Long Term Deployment which materials he used preparing this overview.

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ADVANCED FUEL CYCLES

Optimization of closed nuclear fuel cycle including transmutation of minor actinides

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Abstract. Up to now no definite internationally recognized quantitative criterion of minor actinides transmutation efficiency was worked out though it would be highly desirable. The parameters which should be taken into account in working out such criteria are discussed on the base of calculations of isotope kinetics of MA transmutation in thermal and fast neutron spectra with americium recognized as primary target. Even isotopes of plutonium are considered as 'ballast actinides" because their accumulation results in deterioration of the fuel properties of recycled actinides composition. The mass of uranium as irreparable natural raw material spent to transmute a unit mass of MA and average energy released per neutron absorbed by the fuel mixture are also taken into account.

1. INTRODUCTION

Basic parameters describing the process of MA transmutation are:

ΔM – total reduction (absolute or relative) of the mass of the transmuted actinide (TA).

Δm – reduction of the same mass by turning TA into fission products (burning).

The isotopic composition (concentrations c_i) of actinides produced in the SNF from TA.

The same for the actinides produced from other heavy metals of the fuel (isotopic evolution of the fuel matrix).

n - total number of neutrons absorbed by TA and other actinides.

E - total energy release.

T_L - fuel life-time in the core.

T_C – cooling time before reprocessing.

2. ANALYSIS

The actinides involved in the process may be split into a few groups:

(1) Fissioning $^{233,235}U$, $^{239,241}Pu$ (the last one decaying into ^{241}Am).
(2) Fertile, low radioactive ^{232}Th, ^{238}U.
(3) Minor actinides – Np, Am, Cm.
(4) Even isotopes of U and Pu, both present in significant quantities like $^{238,240,242}Pu$, $^{234,236}U$ and rare ^{236}Pu and ^{232}U with highly hazardous daughter nuclides. This group may be considered as "ballast actinides" (BA) together with MA.
(5) The quantity of irreproducible natural resource – uranium – spent to destroy unit mass of the transmuted nuclide is an important figure. Since $^{234,236}U$ are unseparable from the main isotopes they should be considered together. It's more obvious for low radioactive ^{236}U ($T_{1/2}$=23 million years) decaying into ^{232}Th than for ^{234}U.

Following numerical are most frequently used to estimate the transmutation efficiency:

$\Delta M/T_{к}$ – transmutation rate (kg/year or per cent/year);

$\Delta m/T_{к}$ – burning rate (kg/year or per cent/year);

$\Delta M/E$ – transmutation intensity (kg/GW year);

$\Delta m/E$ – burning intensity (kg/GW year);

TABLE I. THE RESULTS OF TRANSMUTATION OF AM-CM AND PILING UP OF BALLAST ACTINIDES (BA) IN 20% AMOX-FUEL, G/T

Element	20% AMOX, irrad., days		Cooling, years	
	0	874	3	10
Np	0	5.893E+02	5.787E+02	5.787E+02
Pu	0	1.788E+04	2.151E+04	2.087E+04
Am	1.746E+05	1.370E+05	1.364E+05	1.350E+05
Cm	5.439E+03	9.835E+03	5.322E+03	4.173E+03
Total BA mass, M	1.800E+05	1.654E+05	1.638E+05	1.607E+05
BA ΔM	0	1.460E+04	1.620E+04	1.930E+04
U-235	6.000E+05	4.432E+05	4.432E+05	4.432E+05
U-238	2.000E+05	1.909E+05	1.909E+05	1.909E+05
U-236	0	2.458E+04	2.458E+04	2.458E+04
U	8.000E+05	6.587E+05	6.587E+05	6.587E+05
U mass reduct. ΔU	0	1.413E+05	1.413E+05	1.413E+05

TABLE II. THE RESULTS OF TRANSMUTATION OF AM-CM AND PILING UP OF BALLAST ACTINIDES (BA) IN 50% AMOX-FUEL, G/T

Element	50% AMOX, irrad., days		Cooling, years	
	0	872	3	10
Np	0	4.673E+02	4.608E+02	4.608E+02
Pu	0	3.973E+04	4.974E+04	4.795E+04
Am	4.364E+05	3.298E+05	3.283E+05	3.251E+05
Cm	1.359E+04	2.594E+04	1.358E+04	1.065E+04
Total BA mass, M	4.500E+05	3.968E+05	3.920E+05	3.841E+05
BA ΔM	0	5.320E+04	5.800E+04	6.590E+04
U-235	3.750E+05	2.640E+05	2.640E+05	2.640E+05
U-238	1.250E+05	1.184E+05	1.184E+05	1.184E+05
U-236	0	1.728E+04	1.728E+04	1.728E+04
U	5.000E+05	3.997E+05	3.997E+05	3.997E+05
U mass reduct. ΔU	0	1.003E+05	1.003E+05	1.003E+05

TABLE III. BURNING OF AMERICIUM IN A THERMAL SYSTEM (VVER-1000, F = 4 × 10^{14} N/S*CM2) AND IN A FAST SYSTEM (BN-800, F = 10^{16} N/S*CM2).

Thermal						
Time, days	0	400	800	1200	1600	2000
Actinides, g	1.0E+03	9.53E+02	8.94E+02	8.40E+02	7.89E+02	7.38E+02
Fissions	0.0E+00	1.17E+23	2.62E+23	3.96E+23	5.24E+23	6.50E+23
Captures	0.0E+00	1.04E+24	1.77E+24	2.31E+24	2.75E+24	3.12E+24
BU, MWt-d/t	0	5.98E+04	1.25E+05	1.84E+05	2.38E+05	2.91E+05
Burn-Up, %	0	4.67	10.463	15.83	20.968	26.027
Energy, MeV	0.0E+00	3.23E+25	6.78E+25	9.94E+25	1.29E+26	1.57E+26
Mev/neutron		2.80E+01	3.34E+01	3.66E+01	3.93E+01	4.17E+01
Thermal						
Time, days		2000	4000	6000	8000	10000
Actinides, g		7.37E+02	4.98E+02	3.14E+02	1.95E+02	1.22E+02
Fissions		6.50E+23	1.25E+24	1.71E+24	2.01E+24	2.19E+24
Captures		3.11E+24	4.42E+24	5.24E+24	5.76E+24	6.07E+24
BU, MWt-d/t		2.91E+05	5.32E+05	7.15E+05	8.33E+05	9.06E+05
Burn-Up, %		2.60E+01	5.00E+01	6.84E+01	8.04E+01	8.77E+01
Energy, MeV		1.57E+26	2.87E+26	3.85E+26	4.49E+26	4.88E+26
Mev/neutron		4.17E+01	5.06E+01	5.54E+01	5.78E+01	5.90E+01
Fast						
Time, days	0	400	800	1200	1600	2000
Actinides, g	1.0E+03	9.14E+02	8.03E+02	6.90E+02	5.84E+02	4.91E+02
Fissions	0.0E+00	2.15E+23	4.91E+23	7.75E+23	1.04E+24	1.27E+24
Captures	0.0E+00	7.23E+23	1.26E+24	1.66E+24	1.97E+24	2.20E+24
BU, MWt-d/t	0	9.16E+04	2.03E+05	3.15E+05	4.19E+05	5.09E+05
Burn-Up, %	0	8.604	19.641	31.01	41.592	50.932
Energy, MeV	0.00+00	4.94E+25	1.10E+26	1.70E+26	2.26E+26	2.75E+26
Mev/neutron		5.26E+01	6.26E+01	6.98E+01	7.51E+01	7.90E+01
Fast						
Time, days	0	2000	4000	6000	8000	10000
Actinides, g		4.90E+02	2.04E+02	9.46E+01	4.87E+01	2.64E+01
Fissions		1.27E+24	1.99E+24	2.26E+24	2.38E+24	2.43E+24
Captures		2.20E+24	2.80E+24	3.01E+24	3.11E+24	3.16E+24
BU, MWt-d/t		5.09E+05	7.85E+05	8.91E+05	9.35E+05	9.56E+05
Burn-Up, %		5.09E+01	7.96E+01	9.06E+01	9.52E+01	9.74E+01
Energy, MeV		2.74E+26	4.23E+26	4.80E+26	5.04E+26	5.16E+26
Mev/neutron		7.90E+01	8.83E+01	9.09E+01	9.17E+01	9.20E+01

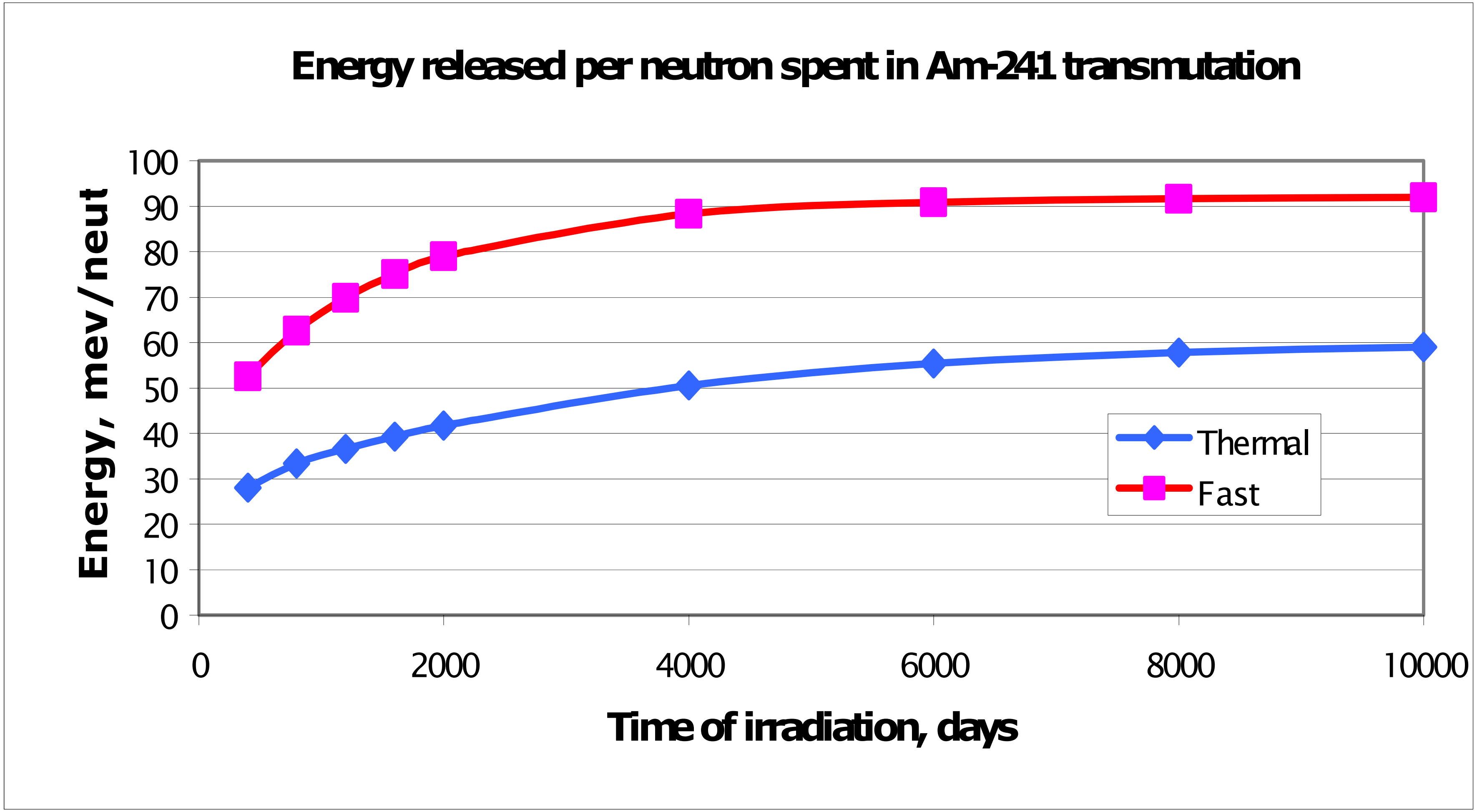

FIG. 1. Energy released per neutron spent in Am-241 transmutation.

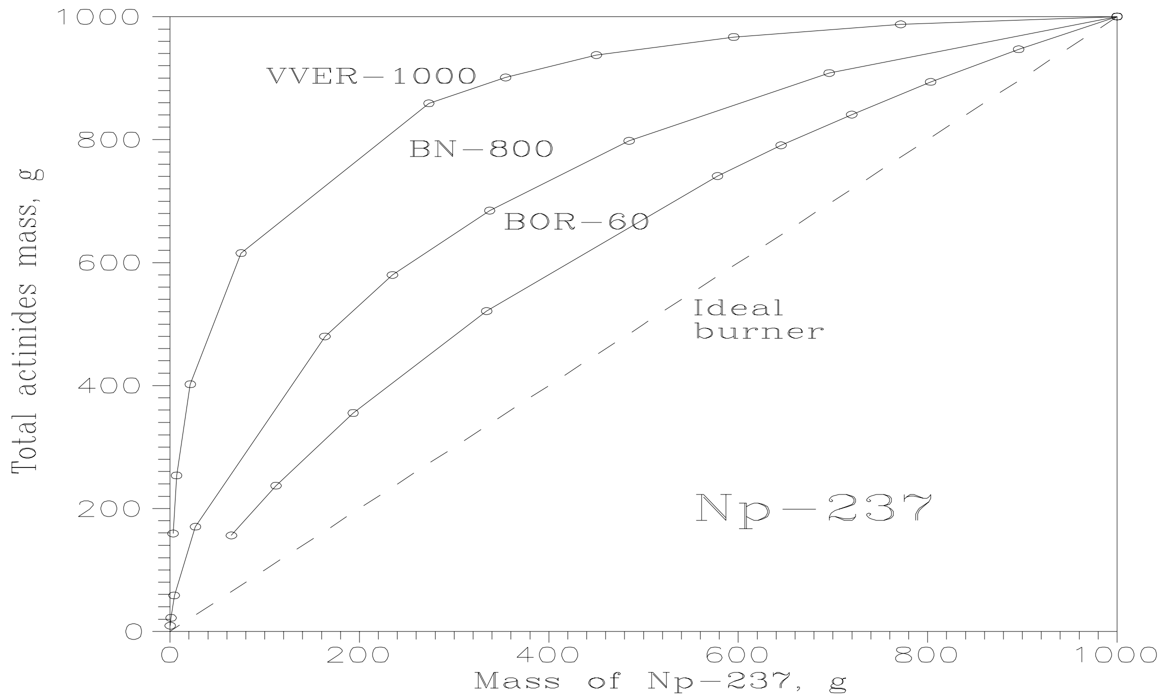

FIG. 2. Total actinide mass M as a function of residual mass of Np-237 in the process of burning of 1 kg of Np for 10 000 days in neutron spectra corresponding to different types of Russian reactors.

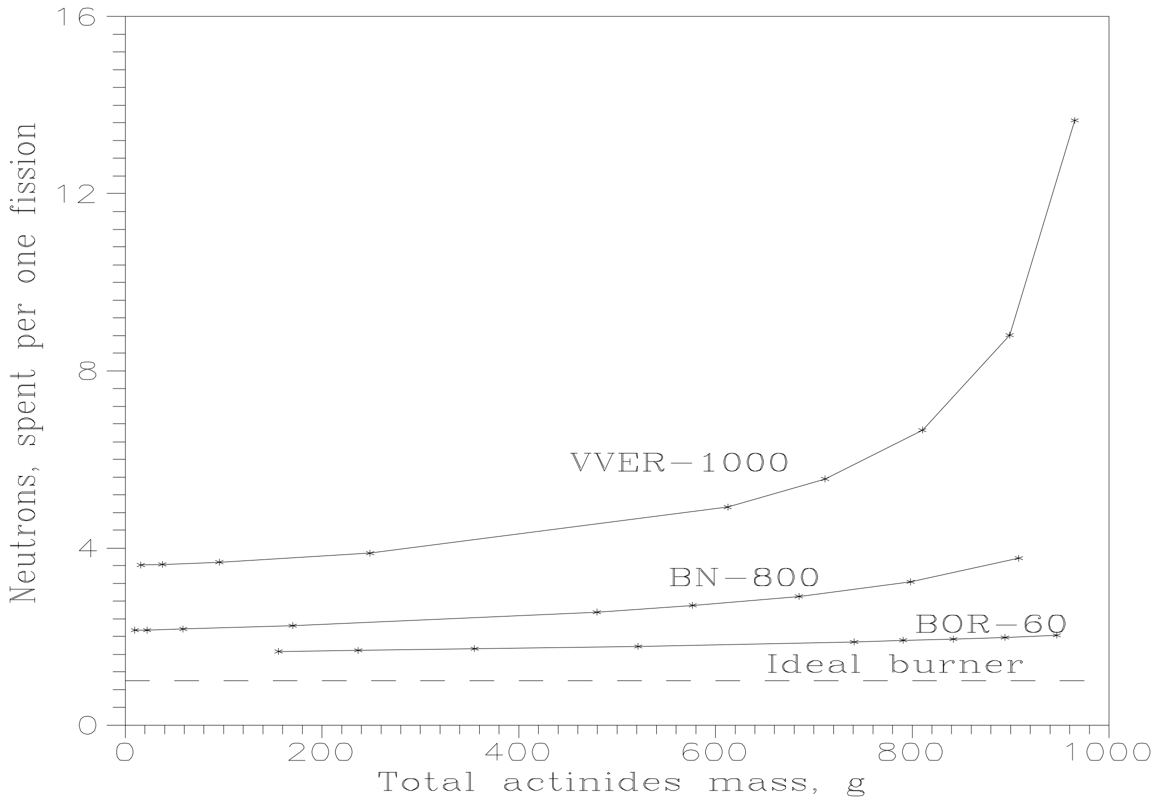

FIG.3. The number of neutrons spent per one actinide fission in the same process.

According to said above ΔM is completely unsatisfactory mass parameter because it ignores accumulation of BA both from transmuted nuclide and from the fuel matrix. Besides, except the case of inert matrix, this value is not directly observable, it may only be calculated. Value Δm also seems inadequate, due to ignoring the second of the two factors. Both values do not take into account the accumulation of even U and Pu isotopes mentioned above. As an example calculations were made of isotope inventory of AMOX fuel during the campaign and cooling for planned “RECYCLE” experiment on BOR-60 reactor. The results are presented in Tables I-III and Figs 1-3.

Fuel composition: U enriched to 75%; 20 or 50% mass of Am-Cm-RE fraction of SNF of VVER-1000 reactor (burnup 40 GW days/t, cooled for 10 years).

Am transmutation rate is about 22-24% per campaign, but the mass of accumulated Cm is about 12% of the mass of transmuted Am, the mass of accumulated Pu – 37%-47% (almost entirely non-fissioning isotopes, i.e. BA).

3. RESULT

No single satisfactory numerical criterion of the efficiency of MA transmutation was proposed up to now, so various parameters influencing the efficiency are discussed for the case analysed.

Three figures are most important:

M – initial mass of BA;

ΔM – its total reduction;

ΔU – the mass of burnt uranium.

Two relative parameters of transmutation efficiency $\Delta M/M$ (transmutation rate) and $\Delta U/\Delta M$ (uranium spent for unit transmuted mass) may be used. Their values are:

For 20% AMOX-fuel: $\Delta M/M = 0.107$ and $\Delta U/\Delta M = 7.35$;

For 50% AMOX-fuel: $\Delta M/M = 0.146$ and $\Delta U/\Delta M = 1.52$;

Both parameters grow with MA concentration in AMOX-fuel, second one much faster. Another significant value is the energy produced per neutron spent.

4. SUMMARY

Two agents are involved in actinides transmutation and burning: nuclides and neutrons. To optimize the process both agents should be classified – nuclides by their physical properties, neutrons by their energy.

Some techno-economical values should be assigned to every group in this classification, either qualitative or, better, quantitative.

The simplest classification of actinides seems to be into two groups with two subgroups in each: useful (fissioning and fertile) and ballast (minor actinides and threshold isotopes of U and Pu).

The simplest classification of neutrons is into two groups – thermal and fast.

Results of the transmutation may be evaluated first by the changes in the mass balance of useful and ballast actinides and second by the number of neutrons spent in the process.

The mass of U as natural resource spent for transmutation should also be taken into account.

It would be useful to work out a concept for pricing neutrons the way fuel is priced.

Fast neutron systems seem to be better actinide transmuters than thermal ones for any set of efficiency criteria.

Nuclear recycle: One of the key factors ensuring the sustainable development of China's nuclear power

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Abstract. Since early 1970's China has established a complete nuclear fuel cycle system for defense purpose. CNNC and its formers deal mainly with business in nuclear power and nuclear fuel cycle fields. Now COSTIND (CAEA) implements the governmental functions. Based on the features of energy sources, China is steady developing nuclear power as the appropriate supplement to meet demand of electricity in the southeast coastal regions. In the middle of 1980's, considering a number of elements, covering spent fuel arising from PWR dominantly, uranium source availability, development of FBR, radioactive waste management safely and existing good nuclear industry foundation etc., the closed fuel cycle strategy was formulated and announced internationally. In view of complexity of reprocessing technology and the limited spent fuel arising in the early period of nuclear power growing up, a multi-purpose pilot plant will be first set up. The plant involved in spent fuel AFR storage, and HEU/LEU fuel reprocessing separately has been constructed since 1991. At present more than a half of the project job has been completed while the preparation for long-distance transport of spent fuel is actively pushed on. It is expected that a centralized wet storage facility will be finished at the end of next year but the whole plant will be put into active commissioning in 2003. With the extension of the cumulative arising of spent fuel, a large scale reprocessing plant would be established around 2020. Reprocessed uranium will be surely re-enriched in the past manner while reactor-grade plutonium would be fabricated into MOX fuel to supply to FBR program.

1. INTRODUCTION

In the middle of 1950's China launched industrially nuclear activities. The Ministry of the Second Machine Building was founded in 1958, in charge of administration and construction of the national nuclear industry for defense purpose. As a result of nearly 20 years setting up, a complete nuclear fuel cycle system, involving reprocessing and reprocessed uranium recycle was established. Since early 1980's China has placed emphasis of its nuclear industry strategy on serving national economy development. The above mentioned Ministry was replaced by the Ministry of Nuclear Industry in 1982 and later by China National Nuclear Corp. (CNNC) in 1988 again.

Last year's spring CNNC submitted its whole governmental functions to the Committee of Science, Technology and Industry for National Defense (COSTIND), composed newly and called the China Atomic Energy Authority (CAEA) in case of involvement of the foreign affairs in nuclear field, under the State Council. Now COSTIND, on behalf of the government, is responsible for constitution of strategy, regulations and plan, as well as professional administration etc. in the defense industry fields.

Since last June CNNC has split two individually enterprises: one is China Engineering and Construction Corporation (group) as project construction contractor; the other China nuclear industry corporation (group) still keeps its English name (in short CNNC), engaged in the rest nuclear business, covering mainly nuclear power and nuclear fuel cycle. The new CNNC remains under the State's control but runs on a "market basis".

2. DEMAND FOR NUCLEAR POWER AND PLANNING

2.1. Features of Energy Sources in China

China's economy is nowadays forging rapidly ahead with the increase of energy source demand. Though rich in energy sources in China, their composition is not so rational, and the geographical distribution is quite uneven. Coal and hydropower resources as the main energy resources distributed in regions in the north and the northwest, and the southwest respectively while oil and natural gas resources are comparatively less. The primary energy sources are insufficient in the economically developed coastal regions.

In order to mitigate on the increase of pressure incurred by the long distance transport of coal, to improve environment protection and to optimize energy resource composition, nuclear power should be considered as an appropriate supplement of the energy sources in these regions.

2.2. The Present Status of Nuclear Power

The existing two NPPs (Qingshan, Zhejiang province, phase one 30OMWe PWR and Daya Bay, Guangdong province, 2X900 MWe PWRS) have been put into commercial operation since 1994. Nuclear share accounted for about 1.3% of the total electricity generation of 1 135 TWh in 1997.

In this five-year (1996-2000) plan for national economy development, there are four NPP's with eight units (660 MWe) under construction. Qingshan phase two with 2X600 MWe PWR units) is the second self-designed and -constructed NPP. Qingshan phase three with 2x700 MWe CANDU units is imported from Canada. Ling-ao NPP closed to Daya Bay plant is almost the latter's duplicate. Tianwan NPP recalled just, i.e. the original Liangyungang one, Jiangsu province, with 2x1000 MWe VVER units is introduced from Russia. Recently their construction all are underway while the latest action is the first pouring of concrete for Tianwan NPP on 20th last month. It is expected that they will be successively come into operation since 2002. The total nuclear installed capacity will amount to 8.7 GWe by 2005.

2.3. Future Development Plan for Nuclear Power

Authorities concerned have recently formulated the next five-year's plan and development strategies up to the year 2020 in nuclear field. Based on the current determined principles, China's nuclear power could be developed properly for the near and medium terms only. According to estimation from some experts, there would be electricity shortage that should be made up by nuclear or other energy for a considerable long period.

At the moment the latest situation of nuclear power is being faced with fierce challenges touching upon temporary mitigation of electricity demand, the huge investment requirement, and localization of design and equipment etc., while its scale would not be as large as estimation before. However, nuclear power as a supplement to the conventional one still has a great potential in the future as it has the distinct advantage in relieving the earth from environmental and ecological issues. As matters stand, a few of additional projects are being vigorously prepared.

3. ELEMENTS ESTABLISHING STRATEGY OF CLOSED NUCLEAR FUEL CYCLE

In the middle of 1980's China determined that nuclear power reactor would be dominated by PWR meanwhile based on China's actual situation, the closed nuclear fuel cycle strategy through reprocessing and recycle, was also formulated and announced internationally.

3.1. Fully and Effectively Utilization of Uranium Resource

The annual 60 tHM of spent fuel has been currently discharged from the operating NPP's and the figure will be constant by 2002. As of the middle of his year the accumulative quantity has reached about 330 tHM. By 2005 the annual PWR spent fuel arising will sharply increase up to 168 tHM at the same time the accumulative one 930 tHM. Beside these, 176 tHNVa would be meantime discharged from CANDU reactors and 440 tHM accumulated respectively. If the total capacity of China's nuclear power comes to 20 GWe by 2010 and 40 GWe by 2020 separately and with two CANDU reactors being built only, the annual spent fuel arising will approximately amount to 600 tHM in 2010 and 1000 tHM in 2020. In this case the accumulative arising amounts to about 3 800 tHM and 12 300 tHM respectively. These appear the tremendous figures.

On the other hand, according to the foreign publish on uranium resources of China, there are extensively uranium deposits distributed in various geological formations in seven regions throughout the country, and reasonably assured plus estimated additional resources exceed 57000 tU.

As well known, abundance of uranium-235 in spent fuel from PWR is even slightly higher than in the natural uranium. Therefore, uranium from reprocessing spent fuel described above with more than 95% recover rate is not a small proportion in comparison with the present uranium resources. To save the resources they must be utilized fully and effectively.

3.2. Development of Fast Breeder Reactor

Development of Fast Breeder Reactor (FBR) can generally increase utilization rate of uranium resource from about 1%, in case of reprocessing and plutonium recycling to thermal reactors. to 60 - 70%. France, Russia and Japan etc. have gained essential experience in FBR's PJD, design, construction and operation. Total time of the world's FBR operation has amounted to about 300 reactor-years.

Since it is quite difficult to meet the demand for domestic uranium resource for the far future, for instance 50 years and beyond, and other transuranium nuclides with long life, separated from reprocessing process, are preferred to be burned out, China has been paying attention to R&D on FBR. It would possibly become the second generation of nuclear energy source. Therefore, reactor-grade plutonium recovered from reprocessing will be supplied to FBR as MOX fuel.

3.3. Reduction of Uranium Mining, Milling and Enrichment Costs

Because most of 200 or more uranium deposits of China are small scale, and of low grade and economically being exploited with a certain difficulty, costs of uranium mining and milling are relatively expensive. Moreover, abundance of uranium-235 in spent fuel from PWR rests with the range between 0.8% and 0.9%, slightly more than in natural uranium. Consequently, utilization of reprocessed uranium can reduce costs of the above processes.

3.4. Disposal of Radioactive Wastes Safely

Foreign reprocessing experience has shown that after essentially separation of uranium and plutonium, all radioactive wastes from reprocessing can been safely managed and their volume has been decreased significantly. According to French operation experience, La Hague reprocessing plants' waste volume will reduce by a factor of six next year, compared with the volume arising a decade ago, while the volume of waste requiring geological disposal is already four times less than the corresponding volume of the spent fuel itself.

Therefore, by separating uranium and plutonium, the closed cycle reduces the amount of radioactive nuclides in the waste to be disposed of and, on time scales relevant to the potential effect on the biosphere, decreases the radio-toxicity of this waste by about an order of magnitude. In case adopting partitioning and transmutation technology for TRU nuclides, being currently researched by some countries including China, meets with success, reprocessing waste management will become relatively a simple and safe problem.

3.5. Existing Foundation in Reprocessing

Development of civil reprocessing in China is in an advantageous position. Our country has a vast land and there are a few ideal sites for building reprocessing plant, where the population density is very low, and meteorological and geographical conditions all are suitable to this process. China has also had a certain technical base, i.e. a specialized team for scientific research and design, as well as near 20 year's successful operation experience of the plants at the almost same level in reprocessing technology as the world's advanced countries in 1970's. Nowadays, under IAEA's safeguard institution, some matured techniques introduced from abroad appropriately would be of benefit to us.

4. THE PRESENT STATUS AND PROSPECTS OF BACK END

4. 1. Transport and Storage of Spent Fuel

With the 10 year's typically at-reactor storage period of spent fuel, NPPs are mostly situated in the south and the east coastal areas while the present reprocessing establishment is located in the northwest, 3 000 to 4 000 km away. The transport issue of spent fuel has to be dealt with. A feasibility study on the subject from Daya Bay plant to Lanzhou Nuclear Fuel Complex (LNFC) has been completed. The study results have recommended that a combined transport option by both sea and rail would be preferable, using big payload casks and making two round trips annually. Alternatively, a gate-to-gate transport option by road is a realistic solution because of the enormous investment of the previous option and the very limited business in the near future. However, it is necessary that a complete transport system, including casks and its maintenance facility, a purpose-built marine terminal, ships and wagons etc., should be set up for the long term. The first transport of spent fuel from Daya Bay plant will take place in 2003.

Placed at LNFC, construction of the first stage of a Centralized Wet Storage Facility with a 550 tHM (500 t for PWR fuel and 50 t for other fuel) capacity started in spring of 1994. Now the facility project has been nearly completed and will be put into active commissioning with the first movement of spent fuel from research reactor into it late next year. Entering early next century, its storage capacity might be extended modularly and inter-linked with the future commercial reprocessing plant through a designated channel.

4.2. Spent Fuel Reprocessing

Since the middle of 1970's R&D on civil reprocessing have been conducted at laboratories. In early 1980's a multi-purpose pilot plant was incorporated in the national economy plan. The project consists of the receipt and storage facility mentioned above, a main reprocessing facility with a maximum throughput of 400 kgLEU/d, a hot cell lab with a 900 gHEU/d capacity, and a Machinery Testing Workshop (MTW), as well as some auxiliary facilities.

With the exception of MTW coming on stream in 1993 in advance, construction of its all buildings are being carried out actively. So far, more than half of the project job has been finished. Commissioning of the whole plant is anticipated by the year 2002. It is planning that given supplementing some waste management facilities, the pilot plant could be reconstructed to a small-scale production plant with a capacity of 80 to 100 tHM/a by 2006. After obtaining extensive experience and a sufficient amount of spent fuel accumulated, a large scale commercial plant, possibly with a 800 tHM /a capacity, would be commissioned around 2020 in order to match with the nuclear power capacity at that time.

4.3. Recycle of Reprocessed Uranium and Reactor-grade Plutonium

In the past years all uranium products, as either ammonium uranyl carbonate or uranium trioxide forms, from military reprocessing was converted and re-enriched as a part of feed material. In the future uranium reprocessed as U03 will be still recycled in the same manner.

At present an experimental fast breeder reactor project with a 65 MWt capacity has been entered in a list of the State High Technology Program and is under construction in China Institute of Atomic Energy, Beijing. It will be completed early next century. Hence, reactor-grade plutonium recovered as MOX fuel would supply to the project and the further FBRS. Currently it is being also considered to build a demonstration facility for MOX fuel fabrication in LNFC at the appropriate time.

5. CONCLUSION

China's nuclear power dominated by PWR is being lively developed.

Taking into account the following facts: saving uranium resource, economically utilization of uranium, developing FBR and safely disposing radioactive waste, as well as placing in advantageous position, China has established the closed fuel cycle strategy through reprocessing and recycle.

According to the strategy, some facilities relating to back end of nuclear fuel cycle are being built and pursued ahead continuously. Recycle of both uranium and plutonium should be one of the key factors ensuring the sustainable development of China's nuclear power.

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Fuel cycle strategies for growth of nuclear power in India

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Abstract. Nuclear power has been identified as an essential component to meet the growing energy demand of India. The three stage fuel cycle strategy to achieve this with the available resources envisages the use of natural uranium in PHWRs in the first stage, the plutonium-uranium/plutonium-thorium cycles in Fast reactors/Advanced HWRs in the second stage, followed by exploitation of essentially U233 in the third stage. The technologies necessary for this programme, mainly through the back-end of the fuel cycle including reprocessing, waste management and recycle of Pu have been developed accordingly, as a direct result of the closed fuel cycle policy followed by us from the very beginning. This paper addresses the considerations involved in several activities taken up in our programme, their current status and plans for the future.

1. INTRODUCTION

It is expected that bulk of the world nuclear growth in the coming decades will be in Asia, of which India is no exception, due to the projected population growth and consequent energy demand. At present our total installed electricity generation capacity is 90 000 MWe, of which the nuclear contribution is 1 840 MWe. By the year 2020, these capacities are planned to be 450 GWe and 20 GWe respectively[1].

This nuclear component is expected to be met by a mix of Pressurised Heavy Water Reactors (PHWRs), Light Water Reactors (LWRs), VVERS, Fast Breeder Reactors (FBRS) and Advanced Heavy Water Reactors (AHWRs). This structure of our plan has evolved due to our limited natural uranium (nat.u) reserves and need to exploit the abundant thorium resource.

The resource base of U is about 60 000 tons of assured resources with about 30 000 tons of estimated additional reserves[2]. The thorium resources are estimated at about 590 000 tons. The closed fuel cycle philosophy has hence been our natural choice from the very inception of our programme. A comprehensive back-end fuel cycle has been an essential component of our work plans. The related strategies worked out and studies being undertaken to realise that objectives are outlined hereunder:

2. THE THREE-PHASE POWER PROGRAMME

The nuclear power programme in India envisages the setting up of nat.U fuelled PHWRs in the first phase. Utilisation of Pu obtained by the reprocessing of the spent fuel from the PHWRs is the second phase in which FBRs using mixed oxide fuel will be set up, in parallel with the construction of AHWRS, which would use Th and a small feed of Pu. The third phase involves exploiting the Th-U^{233} Cycle in breeders, which could sustain a generation of 200 000 GWe using our large Th reserves.

At present, we have, a couple of BWRs and 9 PHWRs in operation, generating a total of 1 740 MWe . Construction is in progress on 5 PHWRs for a capacity augmentation of 1 660 MWe, with 3 of them nearing completion shortly. The projected growth for plant construction is shown in Table-1. These perspectives are in line with the projected international vision that FBRs will become available, driven by increased uranium prices, and reprocessing will be accepted, although it should economically compete with interim storage of spent fuel, and Pu would become a scarce commodity [3].

TABLE I. Projected Growth of Nuclear Generation Capacity up to 2020s

	Installed Capacity /No. of Reactors in operation											
Year	BWR		PHWR		FBR		VVER		LWR		Total	
2000	320	2	2.400	12	-	-	-	-	-	-	2.720	14
2005	320	2	3.400	14	-	-	-	-	-	-	3.720	16
2010	320	2	4.280	18	500	1	2.000	2	-	-	7.100	23
2015	320	2	7.780	25	500	1	5.600	6	220	1	14.420	35
2020			9.280	28	2.500	5	8.300	9	440	2	20.840	46

3. BACK-END CONSIDERATIONS

The need for effective utilization of limited resources, while ensuring meeting the environmental obligations, has been the driving force for adopting the Reprocessing and Recycle option. Exercising this option for a closed fuel cycle enables the resource base to be exploited for large scale nuclear power generation. Efficient utilisation of Pu is the key for the success of the closed cycle philosophy. Also, the Reprocessing and Recycle option will, while ensuring the sustainability of nuclear power, result in the reduction of actinide/minor actinide inventories and lead to safe management of rad. waste. The Th/U fuel, while necessitating a new reactor type, has benefits in this area-, and FBRs, in the burner mode, also facilitate this objective [4]. The long-term radiological risk is reduced very significantly as a result of the extraction of U and Pu from the spent fuel.

3.1 Reprocessing and Plutonium Fuel Fabrication

Reprocessing with plutonium recycle option has endowed the programme with several mid-course options in both U & Th fuel cycles with Pu forming the vital link between the two [5]. At present there are three reprocessing plants in operation one for research reactor spent fuel at Trombay (nat.U fuel) and for power reactor zircaloy-clad oxide spent fuel one each of 100T capacity at Tarapur (in operation since 1978) and Kalpakkam (recently commissioned). PUREX process is adopted in these plants. Further, provision has been made for augmentation of capacity at the recently commissioned plant at Kalpakkam. Also planned is a separate reprocessing facility for the mixed carbide spent fuel from the fast research reactor or FBTR and 2 additional plants for the PHWR spent fuel. Keeping in mind the need to reduce the toxicity potential of the waste to be disposed off, and to have better partitioning efficiencies, R&D activities are being pursued with these objectives. Alternative extractants, reduction of the actinide content of the wastes including minor actinides, etc. are under investigation. Simultaneously, work is on hand for the development of Co-conversion routes for mixed oxides, e.g. the sol-gel process, which would supply free flowing microsphere granules as feed for MOX fuel fabrication. This dust-free process goes a long way in enabling automated MOX fuel fabrication by omitting the hazardous powder handling steps of the process flow -sheet and will be amenable for thermal reactor or fast reactor fuel fabrication programmes. The sol-gel processing step can be integrated into the reprocessing plant, which will have operational advantages.

3.2 Fast Reactor Fuels and MOX Programme

For our fast test reactor FBTR at Kalpakkam, mixed carbide of 70% PuC-UC composition was chosen as the fuel for the first core, after detailed feasibility studies were conducted on this new fuel type. This fuel has performed satisfactorily, and the core comprising of 25 fuel assemblies of 61 pin design has accumulated a burn up of more than 50 000 MWd/Te, Recently, it has been relicensed for burnup extension to 65 000 MWd/Te. The post-irradiation examination of the maximum rated fuel to evaluate the expected life-limitation due to pellet-clad contact by fuel swelling has indicated potential burnup extension well beyond the licensed limits[6]. It is now planned to carry out experimental irradiation in the FBTR of compositions Of UO_2-PUO_2 typical of commercial fast reactors for the proposed Prototype Fast Breeder Reactor (PFBR) at Kalpakkam. Fabrication of the test pins is under way. Studies on mixed nitrides are also under consideration as doubling time could be an important consideration in the coming decades. Mixed nitride fabrication would be comparatively easier than mixed carbide, as also reprocessing.

As for thermal recycle, feasibility studies were carried out for enabling MOX use in our BWRs and PHWRS. Based on these, MOX fuel has been introduced in a small scale in the BWRs at Tarapur. This MOX fuel assembly design incorporates three Pu enrichments in the 6 X 6 fuel bundles. This fuel design replaces the standard LEU fuel rod design with similar performance and is planned for use up to 30% of core assembly. The lead assemblies, fabricated in a MOX fuel fabrication plant set up at Tarapur[7] have seen a burnup of more than 16 000 MWd/Te, and work is on hand to progressively enhance the number of MOX fuel assemblies in these reactors. Studies were also conducted for PHWR fuel bundles with MOX in the inner 7 rods of the 19-rod bundle design[8]. While a reference bundle design has been made, irradiation studies are planned to be taken up shortly. In this scheme, it is expected that there will be considerable savings in nat.uranium consumption, primarily due to the enhancement of the design average core burn-up from 6 800 MWd/Te to about 10 500 MWd/Te. The annual Pu requirement for this scheme is quite small for the 220 MWe reactor. There will be considerable savings in the back-end costs and the load on reprocessing and waste management will be significantly reduced, as also work load on the fueling machines for these on-load-refueling reactors.

3.3 Radioactive Waste Management

The programme on safe management of radioactive wastes envisages two distinct modes of final disposal-, near-surface engineered, extended storage for low and intermediate level active wastes and deep geological repository for high level and alpha bearing wastes. A waste Immobilization Plant for the treatment of high level waste has been in operation at Tarapur, wherein a semi-continuous pot glass process is applied for calcination and melting 'in process vessels as the means for vitrification. Two additional plants are under construction at Trombay and Kalpakkam. A comprehensive view on vitrification takes into account melt development, acceptable product characteristics and processing techniques. The melt developments are constantly reviewed and eformulated to accommodate changes in the inactive constituents associated with the high level liquid waste and the equipment and techniques to accomplish it. To enable high melting glass forming, joule heated ceramic melters and cold crucible practices are being introduced, changing from the current induction melting practice.

The solid storage surveillance facility (SSSF) has recently been commissioned at Tarapur for the interim storage of high level wastes. For the purpose of final disposal, the choice is focused on igneous rock formation and some selected sedimentary deposits. Investigations are in progress for the evaluation of candidate sites for a repository.

3.4 The future options in Waste Management

The partitioning and transmutation options are also under study in India. For the present, these studies are limited to the partitioning of the long-lived actinides from the high level wastes. At an appropriate time, a long-term policy on the final utilization/transmutation of the recovered actinides would be evolved, based on the technologies that would be available in future.

CMPO based solvent extraction and extraction chromatographic studies with high level wastes are in progress. It is intended to develop flow sheets for partitioning of relevant actinides from the wastes to reduce their alpha burden. KSM-17(equivalent to PC-88 A) based extraction chromatography has given viable results for trivalent actinides separation from trivalent rare-earths, which appear together in the CMPO process[9]. Electro oxidation of Ce followed by its removal by TBP or KSM-17 extraction is possible for the separation and reduction of Ce content in Am.

4. Th AND U^{233} CYCLE

Thorium bundles are being irradiated in the PHWRs for the purpose of initial flux flattening at reactor startup. It is planned to recover U^{233} by reprocessing using the thorex process which has been under use for research reactor irradiated Th/ThO_2 rods and studies are planned for the fabrication of U^{233} based fuel. The Advanced Heavy Water Reactor currently under design envisages obtaining almost 80% of the energy generated using Th and the balance through Pu[10] and the fuel assembly design is being appropriately made. It is expected that we would be taking up fabrication of this type of fuel in the near future. Thoria is also the chosen blanket element in the FBTR, wherefrom quantities of U^{233} could be separated. Development work on fabrication of U^{233} based fuels is hence an essential task. Remote refabrication is envisaged for this purpose and a pilot plant scale facility is planned to be set up in the coming years. For the present, a low energy neutron radiography facility, KAMINI, has been in operation at Kalpakkam using U^{232}-Al plate type fuel elements fabricated at Trombay[l 1]. Steps are also planned for clean up of U^{233} to keep U^{222} levels low enabling the fuel cycle to manageable limits. This would be a major solution to our near near-term and long-term plans for thorium utilization.

5. KEY FACTORS FOR THE BACK-END STRATEGY

The key factors for the long-term back-end strategy in order to implement the 3-phase nuclear power programme efficiently flow from the directions of work programme outlined in the previous sections. It is essential that the reprocessing and waste management operations which make available the Pu and U^{233} for the second and third phases, are carried out efficiently with minimum burden to the environment . Further, for ensuring economic viability, the reprocessing capacity would be planned and added to make Pu available for use on Just-in-Time (JIT) basis. Such a scheme would also alleviate the problems posed by presence of AM^{241} in stored Pu. Some capacity of Away-From-Reactor (AFR) storage has to be built up for the interim storage of spent fuels. Recycled uranium (depleted) would be ploughed back into the fast breeder fuel cycle. Schemes for the burning of minor actinides in fast reactors or by transmutation techniques are to be evolved and implemented. In order to execute the second phase of the programme effectively, MOX and fast reactor fueling schemes would be organized on a larger scale. For efficient implementation, higher burnup fuel designs have to be evolved and tested after establishing that the fabrication procedures are simplified to enable automated fabrication. The PFBR and the AHWR are vital links for the 2nd and 3rd phase, and their success needs to be ensured. As such, for the two reactor types, considerable design effort is under way and several component fabrication/performance studies are also being proceeded with. Aspects of the advanced fuel cycle, e.g. the Th-U^{233} cycle for the long-run are being addressed and there is optimism that the issues are quite resolvable, although there is no underestimation of the magnitude of the task.

6. CONCLUSION

The present status of the Indian nuclear power programme is addressing the essential elements of the three phase power programme, based on the resource constraints on uranium and abundant availability of thorium. The near term strategy is crucial and would demonstrate the adaptability of the plans for the long-term growth anticipated to meet the bulk energy demand foreseen. The back-end area has, as such, many possibilities and challenges, and will have to be addressed taking into account the progressive demonstration of the key technologies.

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Factors determining the UK's back-end nuclear fuel cycle strategy and future nuclear systems

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Abstract. Nuclear generating capacity in the UK is static with no units currently under construction. The AGRs and the UK's only PWR, Sizewell B, are operated by British Energy Generation Ltd (BEGL) and British Energy Generation (UK) Ltd (BEG(UK)L), who are subsidiaries of British Energy plc (BE) which was privatised in July 1996. Ownership of the Magnox stations, which were excluded from this privatisation, has now been transferred to BNFL.Government policy on spent fuel management in the UK is that it is for the owners of the spent fuel to decide on the appropriate spent fuel management options, based on their own commercial judgement, subject to meeting the necessary regulatory requirements. The main factors which have predominantly determined UK utility decisions on spent fuel management, to date, have been based on the technical considerations of the spent fuel characteristics, economic attractiveness of the options and at reactor site spent fuel storage capacities. To date, reprocessing has been the dominant form of spent fuel treatment in the UK. Spent fuel storage facilities consist of a mixture of at-reactor stores and large, centralised ponds associated with the reprocessing activities which take place at the Sellafield site. BEGL and BEG(UK)L have contracts for the lifetime arisings of AGR fuel which allow for all AGR spent fuel to be sent to Sellafield for reprocessing or long-term storage. The prompt reprocessing of all Magnox fuel will continue, and spent PWR fuel will continue to be stored at the reactor site in the short to medium term. It is likely that a combination of factors, which are discussed later in this paper, will continue to affect back-end nuclear fuel cycle strategy and future nuclear systems.

1. BACKGROUND AND GENERAL ISSUES

Nuclear power in the UK represents some 18% of installed capacity but currently produces some 31% of electricity supplied. The generating capacity comprises some 8 320 MW AGR and one 1 200MW PWR operated by British Energy's (BE) subsidiaries British Energy Generation Ltd (BEGL) and British Energy Generation (UK) Ltd (BEG(UK)L), and 3 350MW Magnox operated by British Nuclear Fuels plc (BNFL). The details of the nuclear power stations currently in operation in the UK are given in Table 1.

No new nuclear capacity is currently under construction, however, a joint Royal Society and the Royal Academy of Engineering report on the future of nuclear power, published in June 1999, concluded that 'it is vital to keep the nuclear option open.' The report contained some important conclusions and recommendations on how nuclear power should play a continuing long-term role in the UK's energy mix.

2. CURRENT UK SPENT FUEL MANAGEMENT POLICY

Current UK Government policy is that it is for the owners of the spent fuel to decide on the appropriate spent fuel management option based on their own commercial judgement, subject to meeting the necessary regulatory requirements.

The main factors which have determined UK utility decisions on spent fuel management, to date, have been based predominantly on the technical considerations of the spent fuel characteristics, economic attractiveness of the options and at reactor site spent fuel storage capacities.

TABLE I. OPERATIONAL NUCLEAR POWER STATIONS IN THE UK

Name	Type	No. of Reactors	Net Capacity (MW)	Start of Operation	Current Accountancy Lifetime (Years)
Calder Hall	Magnox	4	200	1956	50
Chapelcross	Magnox	4	200	1959	50
Bradwell	Magnox	2	240	1962	↑
Dungeness A	Magnox	2	440	1965	Average of
Hinkley Point A	Magnox	2	460	1965	37 years
Oldbury	Magnox	2	440	1967	Operating
Sizewell A	Magnox	2	420	1966	Lifetime
Wylfa	Magnox	2	950	1971	↓
Dungeness B	AGR	2	1 140	1983	25
Hartlepool	AGR	2	1 180	1983	25
Heysham 1	AGR	2	1 100	1983	25
Heysham 2	AGR	2	1 240	1988	35
Hinkley Point B	AGR	2	1 170	1976	35
Hunterston B	AGR	2	1 240	1976	35
Torness	AGR	2	1 250	1988	35
Sizewell B	PWR	1	1 200	1994	40

Magnox - Spent Fuel Management Strategy

The reprocessing of UK spent fuel, at the Windscale piles, was initially prompted by plutonium requirements for military purposes. A large scale reprocessing facility was constructed at Sellafield in the early 1950s and a second reprocessing plant was built and commissioned at Sellafield in 1964. This plant serves the requirements of the UK's Magnox nuclear power programme and has also reprocessed spent Magnox fuel from Italy and Japan.

The UK does not have uranium reserves which are economically extractable, therefore reprocessing was seen as the key to the UK's energy independence, and during the formative years of its development and implementation the concept and practice of reprocessing became an established stage of the UK's spent fuel management strategy.

Magnox fuel elements consist of bars of natural uranium metal, approximately 1 metre long, clad in Magnesium alloy. The Magnox reactor system was designed with a wet discharge route and interim pond storage in anticipation of early reprocessing. Both the uranium metal and the magnesium alloy cladding are susceptible to corrosion during pond storage and as a consequence Magnox fuel is generally reprocessed within a year of discharge. Although in principle Magnox fuel could be dry stored, the retrofitting of expensive drying facilities or modifications to station fuel discharge routes, such that fuel could be discharged dry, would prove uneconomic given the limited lifetime of the Magnox stations. This fact was acknowledged by the UK Government Select Committee, in 1986, which concluded that, "continuation of fuel reprocessing to the end of the Magnox programme remains prudent policy", this situation remains to date. The only dry store for Magnox fuel was constructed at the Wlyfa site to provide a contingency buffer prior to fuel being sent to Sellafield for reprocessing.

Recycle of the uranium recovered during Magnox reprocessing in AGR initial cores continued to support the reprocessing of Magnox fuel and as a result, 1 650tU of AGR fuel was produced from over 15 000tHM of Magnox Depleted Uranium (MDU). Current uranium market conditions are such that further MDU recycle is not economic at present.

All Magnox fuel will continue to be reprocessed at BNFL's facility at Sellafield following a short period of interim storage.

AGR - Spent Fuel Management Strategy

An AGR fuel element contains 36 pins, consisting of UO_2 pellets clad in a stainless steel tube, arranged in a circular lattice and sheathed in a graphite sleeve. As early reprocessing was envisaged during the design of the reactors the AGR stations have small at-reactor pond stores, hence all spent AGR fuel is sent to Sellafield.

Following a public inquiry in 1977 approval was given for the construction at Sellafield of a reprocessing plant, THORP. The plant, which began operation in 1994, has the capability to reprocess oxide fuel from AGR and LWR stations.

Historically reprocessing contracts were signed, predominantly, on a cost-plus basis i.e. cost of providing the product/service plus a percentage profit for the reprocessor. The contracts for the majority of Thorp baseload business were signed on a cost-plus basis, however, post-baseload contracts are likely to be fixed price.

The cost plus contracts that were signed between BNFL, CEGB and SSEB for reprocessing were renegotiated in the early 1990s, resulting in a fixed price agreement between BNFL and the UK utilities for baseload and post-baseload reprocessing.

AGR fuel could theoretically be stored in dry stores and indeed, Scottish Nuclear (now part of British Energy plc) undertook a study of the feasibility of this option and applied for planning permission to build a dry store at their Torness site. However, following a review of options they concluded that a mixture of reprocessing and long term storage at Sellafield provided the most cost-effective spent fuel management solution.

The current situation in the UK is that all AGR spent fuel will be sent to Sellafield where it will be reprocessed or stored underwater. On signing the 1997 British Energy Spent Fuel Management Contract, British Energy Chief Executive Bob Hawley announced that "The contracts have been awarded to BNFL following a thorough review by Nuclear Electric Ltd. of the economics of alternative spent fuel management strategies including dry storage". British Energy's current contracts with BNFL are all fixed price contracts.

PWR - Spent Fuel Management Strategy

Currently there is only one PWR in the UK, Sizewell B. The spent fuel storage pond can accommodate 30 years spent fuel arisings. British Energy plc will consider in due course arrangements for further management of spent PWR fuel in the light of the prevailing commercial and regulatory environment.

Fast Reactor - Spent Fuel Management Strategy

Fuel from the Dounreay Prototype Fast Reactor (PFR) has been reprocessed at Dounreay since 1979, with the plutonium arisings transferred to Sellafield for storage. The UKAEA are currently evaluating options for the future management of the remaining PFR fuel.

3. KEY FACTORS IN DETERMINING A BACK-END STRATEGY

It is difficult to say specifically which factors have played a major part in the decision of a particular utility in determining a spent fuel management route. The combination and weighting of factors will be different for each country, and each utility within a country. Some utilities are bound by national government strategies whereas others, such as those in the UK, have the freedom to choose the most appropriate strategy based on their own analysis and conclusions.

The following factors are just some of those which might be taken into account when determining future back-end fuel cycle strategy.

Economics

As electricity utilities strive to reduce the cost of electricity generation the economics of the nuclear fuel cycle play an ever increasing role in utility decisions. However, back-end costs currently represent no more than about 20% of the total fuel cycle costs, and fuel cycle costs represent no more than about 20% of overall generation costs. Therefore, overall back-end costs are only a few percent of generating cost, and are likely to remain insignificant in the context of overall costs.

The economics of direct disposal, reprocessing and recycle are specific to each utility i.e. for some direct disposal is the lowest cost option, for others reprocessing and recycle is the lowest cost option.

Facility Availability

If fuel is to be stored, a country/utility will have to consider the benefits and disadvantages of either construction of on-site stores at each reactor compared to a central store. Implications in terms of public reaction, economics, transport between sites, and final disposal location will be just some of the issues that will have to be taken into account.

An international storage facility or repository would provide a country, or utility, with an additional spent fuel management option. As with in-country stores an international facility or repository will generate a number of issues and factors which will require consideration by the utility such as public perception, political stability of the repository host country, transport and costs.

Risk

If utilities are constrained to use only one spent fuel management route, they are at greater risk of having to close their reactor because of spent fuel management constraints, potentially resulting in enormous costs. The impact on a utility of any small differences in fuel cycle costs are insignificant compared to the potential cost of station closure.

No utility has yet committed 100% of fuel arisings to reprocessing and as such these utilities have two options, whereas some countries/utilities have committed 100% to direct disposal, and thus have only a single spent fuel management option available.

Anti-nuclear groups target the back-end of the fuel cycle because they see it as the easiest point of attack that may ultimately lead to the closure of nuclear power stations.

Environmental Issues and Waste Management

The radiological impacts of direct disposal and reprocessing are considered to be both small and broadly comparable.

Changes in regulatory requirements will impact on current spent fuel management routes, and new systems will have to be designed to be compliant with these e.g. OSPAR.

BNFL believes that it is essential to continue to generate further substantial reductions in the cost and environmental impact of reprocessing and has set up the Radical Purex programme to achieve this. Partitioning and Transmutation, and non-aqueous reprocessing methods are both potential future systems which may reduce the waste implications of reprocessing.

Political Issues

Each country has its own policy on the management of spent fuel which may constrain utilities to one spent fuel management option or provide freedom for utilities to choose the most appropriate option to them. Within the UK, Government policy is that it is for the owners of the spent fuel to decide on the appropriate spent fuel management option based on their own commercial judgement, subject to meeting the necessary regulatory requirements.

The House of Lords Select Committee on Science and Technology carried out a review of the management of nuclear waste, following the refusal of planning permission for the construction of a Rock Characterisation Facility at the potential Nirex repository site in Cumbria. The recommendations were published in March 1999, and the UK Government published its response in October 1999 which proposed the publishing of a detailed and wide-ranging consultation paper in early 2000. A subsequent policy statement (White Paper) will be issued in light of this.

Safeguards and non-proliferation concerns

Any spent fuel management route must meet the requirements of international safeguards regimes. Plutonium must be safeguarded whether it has been separated during reprocessing or if it is unseparated as in the case of directly disposed fuel. The security required for the safeguarding of plutonium is relatively unchanged irrespective of quantities stored.

Proliferation concerns are a matter of perception regardless of international regimes. No safeguarded plutonium has ever been diverted from commercial reprocessing operated under international safeguards.

Resource Utilization/Recycle to fast reactor

The strategic importance of energy independence is another factor that may be taken into account when determining a back-end strategy, and will be influenced by a country's natural energy resources and desire for self sufficiency. Direct disposal of spent fuel in encapsulated form rules out re-use of the plutonium and uranium contained within it, whereas reprocessing/recycle can provide a country/utility with a future energy source. The energy content of 1t of spent fuel varies from 10 000 to 40 000t of coal equivalent, depending on the reactor type from which the spent fuel arises and whether the plutonium and uranium is recycled in AGRs or PWRs. Recycling in fast reactors would increase these values by a factor of about 40.

Reprocessing provides a platform for the development of a future fast reactor programme which requires some form of thermal reprocessing to provide plutonium for the initial fast reactor fuel.

The industrialisation of recycle facilities world-wide provides capacity to recycle plutonium in the form of Mixed Oxide Fuel (MOX), providing utilities with a further consideration in determining their back-end fuel cycle strategy. In terms of uranium recycle the current low prices and excess capacity for uranium and enrichment do not favour the recycle of uranium, however, the expectation is that future use of reprocessed uranium will increase as prices begin to rise or as availability is reduced.

4. CONCLUSION

Within the UK, Magnox fuel will continue to be reprocessed, AGR fuel will be sent to Sellafield for reprocessing or long term storage and in the short to medium term PWR fuel will continue to be stored at the Sizewell B reactor site.

Future UK back-end fuel cycle strategy will continue to be defined by a combination of factors, some of which may have been mentioned above, others which may arise through future developments both within the nuclear industry and changes within the external environment.

Development of a strategic plan for an international R&D project on innovative nuclear fuel cycles and power plants

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Abstract. The long-term outlook for nuclear energy should be considered in a broader perspective of future energy needs, operational safety, proliferation and environmental impacts. An Advisory Group Meeting (AGM) on Development of a Strategic Plan for an International R&D Project on Innovative Nuclear Fuel Cycles and Power Plants was convened in Vienna in October 1999 to assess the criteria, the needs for international cooperation, and to formulate a strategic plan for project integration.

1. INTRODUCTION

Currently many countries are interested in research and development (R&D) efforts on advanced and innovative technologies for future nuclear systems (reactor and its associated nuclear fuel cycle). To coordinate these efforts for the purposes of conserving resources and sharing knowledge and experience, the IAEA organized an Advisory Group Meeting (AGM) on Development of a Strategic Plan for an International R&D Project on Innovative Nuclear Fuel Cycles and Power Plants in Vienna in October 1999. The meeting was conducted to provide:

- Background and framework for strategic plan,
- Innovative project proposals, and
- Strategic plan elements and integration.

2. BACKGROUND AND FRAMEWORK FOR STRATEGIC PLAN

Table I lists the existing and planned activities within the Agency which are relevant to innovative nuclear fuel cycles and power plants. These activities encompass a wide spectrum of nuclear energy development, including reactors concepts, nuclear fuel cycle, radioactive waste management, safety, safeguards, nuclear power planning and economics. They form the basis for coordination and the need for a strategic plan.

3. INNOVATIVE PROJECT PROPOSALS

Based on the presentations made by the participating Members States on innovative concepts, the following nine candidate projects were proposed.

- Proliferation Resistance
- High Level Waste Reduction/Incineration/Transmutation
- Non-Aqueous Reprocessing
- Natural Circulation Phenomena
- Passive Systems and Reliability Studies
- Thorium Utilisation
- Accelerator Driven Systems
- Assessment of Russian Federation Fast Reactor Project
- Smart Monitoring and Control

TABLE I. INNOVATIVE FUEL CYCLES AND POWER PLANTS INITIATIVE - RELATED INTERNAL SCOOPE

Project	**Subprogramme Title**	***Project Title [tasks]***	**Section Head**	**Contact**
A.1.01	Nuclear Power Planning, Implementation & Performance	Nuclear Power Programme Planning and Economic Analysis (NENP) [3,4,5]	B. Gueorguiev	M. Rao
A.1.06	"	Management of NPP Operations in a Competitive Environment (NENP) [1-7]	"	T. Mazour
A.2.01	Nuclear Power Reactor Technology Development	Small and Medium Sized Reactor (SMR) Development (NENP) [1,2,3,6]	J. Kupitz	J. Kendall
A.2.02	"	Light Water Reactors (NENP) [1,8]	"	J. Cleveland
A.2.03	"	Liquid Metal Cooled Reactors (NENP) [1-8]	"	A. Stanculescu
A.2.04	"	Gas Cooled Reactors (NENP) [1-7]	"	J. Kendall
A.2.05	"	Heavy Water Moderated Reactors (NENP) [1,2]	"	R. Lyon
A.2.06	"	Cogeneration and Heat Applications (NENP) [1-5]	"	T. Konishi
A.2.07	"	Emerging Nuclear Energy Systems for Energy Generation and Transmutation (NENP) [1-9]	"	A. Stanculescu
B.1.03	Nuclear Fuel Cycle & Materials	Reactor Fuel Materials and Advanced Fuel Technology (NEFW) [4,5]	K. Fukuda	J.S.Choi
B.1.08	"	Nuclear Fuel Cycle Issues (NEFW) [1,2]	"	J.S.Choi
B.1.09	"	Plutonium Inventory and Emerging Problems [5]	"	J.S.Choi
B.2.01	Sources of Radioactive Waste	Waste Arisings [2]	M.Bell	R.Burcl
C.3.04	Nuclear Energy in Sustainable Energy Strategies	Analysis of Nuclear and Other Energy Options and Elaboration of Sustainable Energy Strategies (NENP) [3]	H. Rogner	V. Kagramanian S.Kononov
G.1.01	Nuclear and Atomic Data for Applications	Data Centre Activities [4,5,7,9]	D. Muir	V. Proniaev
G1.02	"	International Data Centre Network Co-ordination and Co-operation Projects [1,4,5]	"	"
G.1.03	"	Nuclear Data Assessment and Improvement for Applications [7,8]	"	P. Oblozinsky
G.1.04	"	Establishment of International Atomic and Molecular Interaction Database [9]	"	R. Clark
H.2.01	Design and Engineering Safety	Nuclear Power Plant Safety Design (NSNI) [14,15]	A Guerpinar	P. Labbe
H.5.01	Regulatory Activities Related to Nuclear Safety	Assistance to National Regulatory Authorities (NSNI) [7.9]	F. Niehaus	Z. Kriz
J.1.01	Safety of Disposable Waste	Safe Predisposal of Solid Waste [5,9,10]	G. Linsley	E. Warnecke
J.2.01	Safety of Dischargeable Waste	Control of Discharges of Radioactive Substances to the Environment [[7]	"	C. Torres
J.3.03	Safety of Residual Waste	Safe Decommissioning of Installations with Radioactive Substances [1,3]	"	P. Stegnar
I.4.01	Radiation Emergencies	Emergency Planning and Preparedness (NSRW) [1]	A. Wrixon	M. Crick
K.1.01	Safety Policies and Standards	Safety Policies (NSSCS, NSNI) [1]	J. Versteeg	A. Karbassioun
K.1.02	"	The Agency Safety Standards (NSSCS, NSRW, NSNI) [1]	J. Versteeg	A. Karbassioun
L.2.04	(Safeguards) Development & Support	Systems Studies and Approaches (SGCP) [2]	V. Pouchkarev	S. Morsy

TABLE II. CANDIDATE PROJECTS AND LEVEL OF INTEREST

Candidate Projects	Potential Participants[1]	Remarks
Proliferation Resistance	France, USA, UK, Turkey, Russia, Korea, Egypt, Japan, Czech[2], China, Canada, India	
High Level Waste Reduction /Incineration/ Transmutation	France, UK, Turkey, Russia, Korea, Japan, Czech, China, Canada, India	
Non-Aqueous Reprocessing	UK, Turkey[2], Russia, Japan, Canada, India	
Natural Circulation Phenomena	France, USA, UK, Turkey, Russia, Korea, Japan , Egypt, Czech[2], China, Canada, Argentina, India	
Passive Systems and Reliability Studies	France, USA, UK, Turkey, India, Russia, Korea, Japan, Egypt, Czech[2], China, Canada, Argentina	
Thorium Utilization	India, France, Canada, Korea, Russia, Turkey, UK, USA	
Accelerator Driven Systems	France, Czech, Korea, Turkey, UK, India	
Assessment of Russian Federation Fast Reactor Project	Russia, China, Korea, India	• France Bilateral, yes • UK, elements of interest • USA, defer to DOE • Czech defer • Japan defer
Smart Monitoring & Control	USA, France, Egypt, Canada, Argentina[2], UK[2], Korea	

[1] Indicate an expression of interest in participation, not a definite commitment to participate, projects would be open to additional participants.
[2] Participation deferred pending receipt of detailed proposal.

TABLE III. SUMMARY DESCRIPTION OF CANDIDATE PROJECTS

1. DEVELOPMENT OF TECHNOLOGIES TO FACILITATE IAEA SAFEGUARDS AGAINST NUCLEAR PROLIFERATION

Objectives

To identify technical options for innovative fuels, fuel cycles and reactor systems that could enhance the implementation of IAEA safeguards against nuclear proliferation.

International Added Value

Enhancement of the already existing non-proliferation measures

Scope

To provide an international forum for review and discussion of technical measures against non-proliferation.

End Result

A compilation and synthesis of technical options leading to the identification of proliferation-resistant attributes and safeguards friendly design options for innovative fuel cycles and power plants

2. HIGH LEVEL WASTE REDUCTION/INCINERATION/TRANSMUTATION

Objectives

To avoid and reduce the production of radioactive wastes and to reduce their radiotoxicity in the long term

International Added Value

Development of clean nuclear technologies, thus leading to the enhancement of the public acceptance of nuclear energy.

Scope

Where applicable, investigation of plutonium recycling and burning or transmutation in fast and thermal reactors. Identification of scenarios providing an equilibrium condition for the production and consumption of plutonium

Transmutation and incineration of long lived minor actinides and fission products ; investigation of the associated safety issues.

Volume reduction of radwaste

End Result

A compilation and synthesis of technical options regarding long term radwaste management

3. NON-AQUEOUS PROCESSING

Objective

Review the current methodologies for non-aqueous processing of spent fuel, and facilitate further improvement and optimisation of these to minimise quantity of waste streams as well as potential for proliferation.

International Added Value

Carrying out project under the auspices of IAEA facilitates a convergence of views on the desirable goals for these processes, and an early advancement of technologies required to achieve these, through efficient use of resources and knowledge from a wide range of international expertise

Scope

The scope includes all types of fuels proposed to be used in advanced and innovative reactors, including uranium, U-Pu, and thorium based fuels in metallic, oxide, or other forms. The work under the project includes:

Definition of desirable goals to be achieved by the processes.

Assessment of the current technologies and experimental advances, in the light of the goals, which will include potential for upscaling the processes in experimental stage.

Generation of additional data with various processes to facilitate such assessment.

Co-operative research in areas of common interest for improvements and data generation to facilitate an early achievement of the goals.

End Result

The end result of these activities will be a compilation of options for non-aqueous processing.

4. NATURAL CIRCULATION SYSTEMS

Objective

Assess the applicability of current methodologies for computing natural convection phenomena in innovative reactor designs. Provide recommendations for needed improvements in models and for relevant experimental data.

International Added Value

Carrying out project under auspices of IAEA provides:

Each contributor the benefit of the knowledge of the solutions of others for their own project

Synergistic viewpoint from looking at various coolants and system designs.

Efficient use of resources and knowledge from a wide range of international expertise.

Scope

Develop understanding of natural convection phenomena in innovative reactor designs. Review the state-of-the-art of modeling natural convection phenomena, and identify requirements for new models, new experimental data for correlations and for validation. Identify experiments for possible benchmarking of computer codes for natural convection cooling phenomena.

End Result

Overview of the important phenomena in natural circulation systems, existing relevant data and identification of needed model improvements and additional experimental data.

5. PASSIVE SAFETY DEVICES AND THEIR RELIABILITY STUDIES

Objectives

To identify short and medium term perspectives, needs and potentials of the passive safety systems. To assess their reliability and benefit in terms of safety and/or economics.

International Added Value

Identification of options and recommendations for future innovative reactor design and development with enhanced safety features.

Scope

The adoption of passive devices will be conditioned by the demonstration of their reliability and economic incentive as compared to competing active systems. The project will thus focus on promising passive safety systems (e.g., shutdown, decay heat removal, fission product containment, passive containment cooling system, steam injectors, depressurisation valves, back-up condensers, thermal-lock energy storage devices) and identify the experimental and theoretical R&D required for their validation and the methodology to assess their performances.

End Result

A compilation and synthesis of technical options regarding passive safety devices providing orientations to concentrate future efforts on a limited number of alternative passive safety systems.

6. THORIUM UTILISATION

Objective

To facilitate an early attainment of maturity of the technologies for thorium fuel utilisation in commercial next generation power reactors.

International Added Value

Carrying out project under the auspices of IAEA facilitates:

- Integration of the existing experience from diverse reactor concepts using thorium, including, HWRs, LWRs, FRs (Fast Reactors), and HTRs.
- Maintaining the impetus of R&D in this area, which is strongly linked to sustainability of nuclear power, but is not of immediate commercial interest in many countries.
- An early advancement of associated technologies through efficient use of resources and knowledge from a wide range of international expertise.

Scope

The scope includes:

- Nuclear data for thorium and relevant isotopic species, including U-233.
- International benchmarks for validating methods for thorium based lattices
- Technological approaches for large scale fuel manufacture
- Extended burn-up fuel performance
- Spent fuel strategies for thorium-based fuels, taking advantage of highly reduced waste streams.

End Result

The end result of these activities will be an assessment report on thorium utilisation.

7. ASSESSMENT OF RUSSIAN FEDERATION FAST REACTOR PROJECT

Objective

To assess the Russian Fast Reactor designs and associated fuel cycle with respect to economics, safety, proliferation resistance and waste management.

International Added Value

Draw attention and support of the concept from international experts.

Scope

Review and assess the technical design of the BREST-300 reactor and associated fuel cycle with respect to safety, proliferation resistance, and waste management; also evaluate the technical and economic feasibility of the BREST-1200 concept for commercial deployment.

TABLE III. (cont.)

End Result

The final product will be an assessment report and recommendations on the Russian Fast Reactor designs and their economic competitiveness.

8. FEASIBILITY OF ACCELERATOR DRIVEN SYSTEMS

Objectives

To investigate the technical feasibility of subcritical systems driven by an accelerator source, (ADS) with a view to incinerate minor actinides

International added value

Development of an inherently safe system particularly adapted for long lived high level waste incineration.

Scope

Hybrid systems have the highest theoretical capacity for actinide transmutation (46kg/TWh). This property enables ADS to be loaded with minor actinides in quantities which will be unacceptable in fission reactor systems. The scope of the project is therefore to identify the required technical options in the fields of accelerators, nuclear physics, neutronics of subcritical systems, materials and fuels. Other investigations will in particular be concerned with the development of technologies for the spallation target and its cooling systems

End Result

Technical options enabling the eventual development of an international project on ADS demonstrator.

9. SMART MONITORING AND CONTROL

Objective

Compile literature on design of computerized reactor control systems which rely on advanced sensors and on advanced software and robotics to provide reactor control and maintenance which may lead to improvements in reliability and availability. Assess the state-of-the-art and provide recommendations for application to innovative reactor design.

International Added Value

Carrying out project under auspices of IAEA provides:

Synergistic viewpoint from looking at various coolants and system designs.

Efficient use of resources and knowledge from a wide range of international expertise.

Scope

Review literature on advanced control system software and on advanced sensors. Assess ability of software to improve reliability and availability. Provide recommendations for further developments. Provide a review and assessment of advanced sensors and reactor control systems that incorporate advance software to improve reliability and availability.

End Result

The results will support the activities of innovative reactor designer to provide advanced control systems which improve system reliability and plant availability.

These projects were shown in Table II with indication of interest from Member States in participation in each project. To further define these activities, the objective, international value added, scope and end results expected were developed in summary form for each of these projects. The results of this process are provided in Table III.

4. STRATEGIC PLAN ELEMENTS AND INTEGFRATION

To coordinate efforts on innovative concepts of nuclear fuel cycles and power plants for the purposes of conserving resources and sharing knowledge and experience and to explore the desirability for an international project, a strategic plan consisting of the following elements is needed:

- Threshold criteria for acceptance of proposed projects,
- Identification of project needs for international co-operation,
- Application of existing Agency products, activities and plans; existing Agency structure and mechanisms for new activities; and identification of new Agency structure or mechanisms needed.

Threshold Criteria for an International Project on Innovative Nuclear Fuel Cycles and/or Power Plants under IAEA Auspices

An international project on innovative nuclear power plants and fuel cycles could range from support for a specific broadly supported design to basic research supportive of most or all technologies. Specifically, an international project could address complete reactor systems; and/or associated fuel cycles or portions thereof; or relevant basic science applications or applied technology. Proposed projects should:

- Demonstrate a strong commitment by its participating Member States
- Encourage broad international participation with a minimum number of participating Member States to be determined on a case-by-case basis
- Include provisions for additional participation
- Demonstrate international added value and significance
- Deal with innovative reactors and/or associated fuel cycles, or portions thereof, which could be commercially available by 2020
- Include a clear definition of the product of the proposed work, its relevance to innovative reactors and/or fuel cycles, and the benefits of the end results
- Be non site-specific
- Satisfy relevant international safety objectives, standards and criteria
- Be consistent with the applicable licensing requirements in the country promoting it.
- Have a finite duration, suggested to be between 24 and 60 months
- Be consistent with the Agency's defined role and activities

Identification of Project Needs for International Co-operation (in areas that can be addressed by the Agency)

In general, forums to exchange ideas, to identify common interests, and to share knowledge and experience for effective resource management on innovative nuclear fuel cycles and power plants should be established. Also, a visible international focus on innovative concepts should be maintained. If an international project on innovative fuel cycles and power plants should be established, the IAEA could provide assistance to its Member States (MS) on the following:

Economics

- The Agency could provide assistance in the identification of cost goals for potential applications of innovative nuclear fuel cycles and power plants on a national or regional basis. The Agency could also extend its methodology for cost evaluation to include innovative concepts.
- This effort could be carried out in collaboration with other international organizations (IEA, EC, NEA, etc.)

Safety/Licensing

- The Agency could provide an international consensus on safety and licensing guidelines for innovative nuclear fuel cycles and power plants projects, as well as to encourage its MS to streamline the existing licensing process.

Safeguards

- The Agency could assist its MS on the safeguardability of innovative nuclear fuel cycles and power plants projects.

Fuel cycle/Waste management

- The Agency could address the issues associated with the back-end of the nuclear fuel cycle and waste management in the context of the innovative nuclear fuel cycles and power plants.

Conduct of Agency Activities in Support of Innovative Nuclear Fuel Cycles and/or Power Plants

There have been many completed and on-going activities in the Agency's Department of Nuclear Energy as_well as other Agency departments that have relevance to this subject area. Any new activity of innovative nuclear fuel cycles and power plants should make use of this experience and also of on-going activities within the frame of existing International Working Groups. This would make any project in this area more focused and useful to a broader community.

To carry this out effectively, there is a need to filter available information and to facilitate identification of gaps in knowledge more precisely. There is also a need to co-ordinate the activities of all departments of the IAEA with the activities or projects. This will ensure that these efforts remain focused and lead to a common objective of achieving a sustainable, innovative nuclear technology.

In light of the above, an implementation function for innovative fuel cycles and reactor designs, which may include a dedicated individual and/or an inter-departmental team, should be established to facilitate co-ordination of its activities and to maintain continuity of various relevant efforts within IAEA

The internal and external co-ordination and information exchange activities will require a substantial and sustained effort. Adequate funding should be provided for the planning and co-ordination of these activities.

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