IAEA-TECDOC-1529

# *Management of Reprocessed Uranium*

## Current Status and Future Prospects

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International Atomic Energy Agency

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# FOREWORD

The International Atomic Energy Agency is giving continuous attention to the collection, analysis and exchange of information on issues of back-end of the nuclear fuel cycle, an important part of the nuclear fuel cycle. Reprocessing of spent fuel arising from nuclear power production is one of the strategies for the back end of the fuel cycle. As a major fraction of spent fuel is made up of uranium, chemical reprocessing of spent fuel would leave behind large quantities of separated uranium which is designated as reprocessed uranium (RepU). Reprocessing of spent fuel could form a crucial part of future fuel cycle methodologies, which currently aim to separate and recover plutonium and minor actinides. The use of reprocessed uranium (RepU) and plutonium reduces the overall environmental impact of the entire fuel cycle. Environmental considerations will be important in determining the future growth of nuclear energy. It should be emphasized that the recycling of fissile materials not only reduces the toxicity and volumes of waste from the back end of the fuel cycle; it also reduces requirements for fresh milling and mill tailings. In comparison, the method of direct disposal of spent fuel premeditates creation of larger capacity repositories for permanent disposal. The issue of recycle and reuse of valuable material is important for the nuclear fuel cycle in the context of sustainable growth of the nuclear energy. Recognizing the importance of this subject, the International Atomic Energy Agency initiated the preparation of this report to review and summarize information available on the management of reprocessed uranium.

Reprocessed uranium has a potential value for recycling either directly or after appropriate treatment. This report analyses the existing options, approaches and developments in the management of reprocessed uranium. It encompasses the technical issues involved in managing reprocessed uranium such as RepU arisings, storage, chemical conversion, re-enrichment, fuel fabrication, transport, reactor irradiation, subsequent reprocessing and disposal options.

This TECDOC was result of the endeavours of the experts who attended the two working group meetings. In addition, there were additional contributions from other experts (listed at the end of the publication). The contributions of all who brought valuable help in drafting and reviewing the report (also listed at the end of this publication) are greatly appreciated. The IAEA wishes to express its gratitude to A. Max (Germany) for chairing the group of consultants and for his special efforts in drafting this publication. The IAEA officers responsible for this publication were H.P. Nawada and C. Ganguly of the Division of Nuclear Fuel Cycle and Waste Technology.

# EDITORIAL NOTE

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# 1. INTRODUCTION

The growth of nuclear energy is linked to the public perceptions on environmental impact associated with nuclear power plants and associated fuel cycles. Various fuel cycle options in different countries are being evaluated to further improve nuclear energy in resource utilization, safety, environmental friendliness and non-proliferation. The technical and economic evolution of the nuclear fuel cycle will depend on the future growth rate of nuclear energy, and on national and utility choices between options for each fuel cycle step. Many believe that the "once-through fuel cycle" by direct disposal of spent fuel in geological formations is an economically viable nuclear fuel cycle option in the immediate future. Some also feel that the option of the closed fuel cycle with a view to recycle the fissile materials and dispose of the fission products as high-level waste is the best option and that this closed fuel cycle (CFC) option should be developed as a long term option. This document focuses on the technical and practical issues of the management of reprocessed uranium (RepU) in the context of collection and analyses of information related to CFC.

On a sustainable development perspective, recycling is an attractive option for improving the efficiency of natural resource management and reducing radioactive waste accumulation. Furthermore, in a scenario of nuclear energy revival with a significant share for energy generation, including the development of nuclear systems for producing process heat, district heating, desalinated water and hydrogen, the demand for fissile materials may eventually exceed the quantities of uranium economically recoverable.

Environmental considerations will be important in determining the future of recycling. Reprocessing of spent fuel could form a vital part of this advanced fuel cycle methodologies to separate and recover valuable fissile materials. The recycling of RepU and Pu reduces the overall environmental impact by the entire fuel cycle. The recycling of fissile materials not only reduces the toxicity and volumes of waste; but it reduces also requirements for fresh milling and mill tailings. The direct disposal of spent fuel premeditates creation of larger capacity repositories for permanent disposal than compared to that required for the waste generated from the closed fuel cycle.

Reprocessing has long been considered as an important sensitive issue due to the proliferation potential, political and societal implications associated with it. There is considerable experience in the civil reprocessing of irradiated fuel on an industrial scale in several countries. As of the beginning of 2003, spent fuel totalling almost 255 000 tonnes of heavy metal (HM) have been discharged from power reactors. About 171 000 tonnes HM remain in storage as spent nuclear fuel, while the remainder has been reprocessed. In several countries (such as France, India, Japan, Russian Federation, etc.) spent fuel has been viewed as a national energy resource. Some countries hold reprocessed uranium as the result of their commercial reprocessing service contracts for reprocessing of spent fuel with others. In some of these countries, the use of recycled materials is already taking place.

The nuclear industry has in place facilities to recycle reprocessed uranium on an industrial scale [1]. A significant operating experience is continuously accumulated by the industry in each step of RepU recycling. Appropriate actions have been carried out to properly and safely operate the RepU management facilities and plants. Its purification and conditioning for storage, re-enrichment and/or direct utilization are now routine operations. Activities range from the small-scale reprocessing of fuel from research or experimental reactors to large-scale, industrial plants offering an international service for standard oxide fuel from LWRs, WWERs, PHWRs, AGRs and GCRs.

From the perceptions of the proliferation threat, the disposition of separated plutonium receives more attention than the disposition of reprocessed uranium from governments, the media, and the public. However, the consideration for the use of reprocessed uranium as reactor fuel is different and simpler than for using MOX fuel.

The technical issues involved in managing reprocessed uranium are RepU arisings, storage, chemical conversion, re-enrichment, fuel fabrication, transport, reactor irradiation, subsequent reprocessing and disposal options, as well as assessment of holistic environmental impacts.

The scope of this document is presented in the following scheme (Figure 1). The objective is to overview the information on the current status and future trends in the management of RepU and to identify major issues to be considered for future projects. Economic analyses, commercial interests, military facilities, minor research and development (R&D) programmes, and highly enriched uranium (HEU) (except when it is converted to low enriched uranium (LEU)) will not be discussed in this publication.

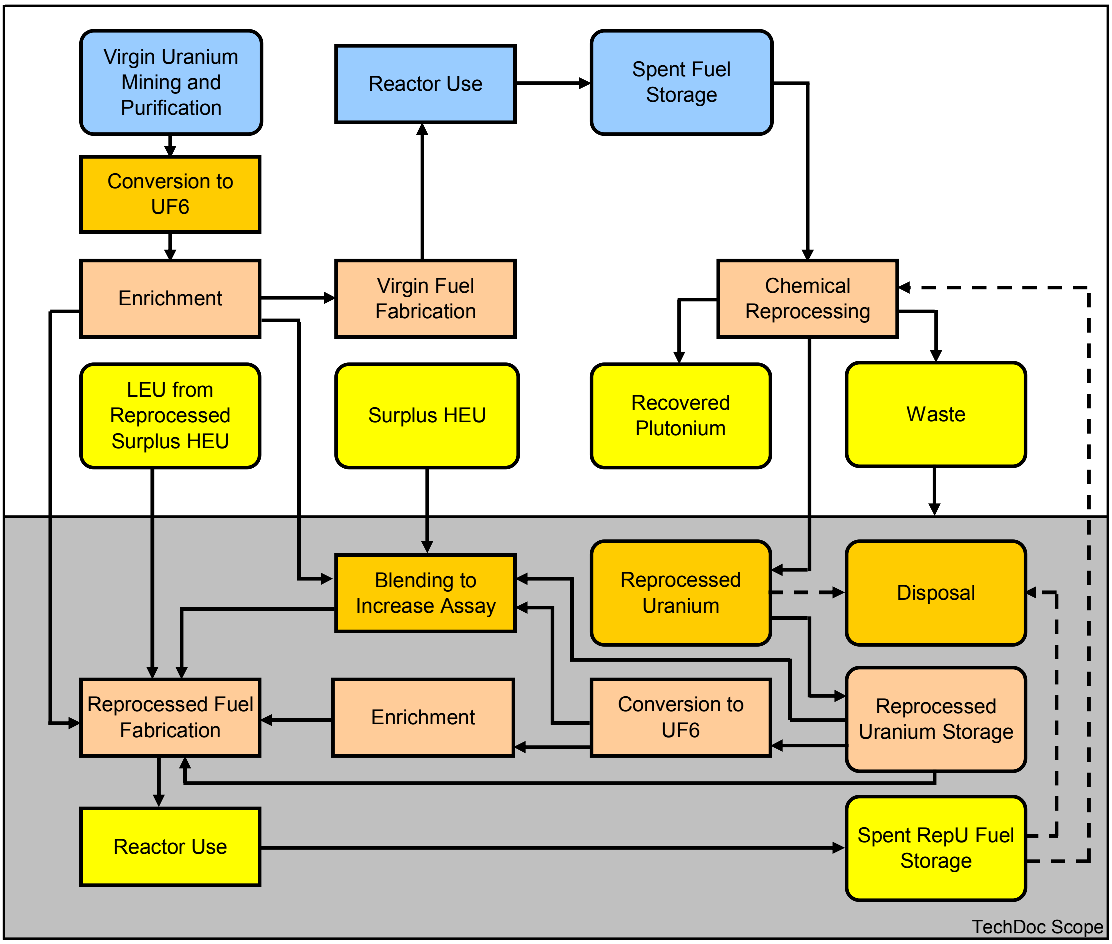


*Fig. 1. Management of reprocessed uranium: RepU flowchart and document scope.*

# 2. TECHNICAL CHARACTERISTICS OF REPROCESSED URANIUM

The characteristics of RepU are described under the two separate headings of 'Isotopic Composition of RepU' (see Section 2.1) and 'Chemical Impurities' (including physical properties where the form is appropriate) (see Section 2.2).

The isotopic composition of RepU is dependent on the initial $^{235}$U content in the fresh fuel (prior to irradiation), irradiation history in reactor (including burnup) as well as cooling and storage periods of spent fuel. Most of the isotopic composition data presented in this Chapter are based on computational results considering necessary parameters and factors.

The levels of chemical impurities in the uranium product are influenced by the fuel cladding and any deliberate fuel doping (e.g. Gadolinium), but are primarily dependent on the efficiency of separation and purification in the reprocessing facilities and product finishing facilities.

## 2.1. Isotopic composition of RepU

Although reprocessing by itself, does not alter the ratio or quantity of U isotopes from that present in the irradiated fuel, radioactive decay does have a time dependent effect.

Commercial-scale reprocessing using either continuous processes or large batch processes may however result in significant mixing between batches of uranium derived from dissolved fuels of quite different fuel irradiation histories. This potential mixing of the reprocessed uranium arising from adjacent reprocessing campaigns may adjust the ratio of uranium isotopes. In addition, some reprocessing practices incorporate the addition of a uranium nitrate solution in the four valent state ($U^{4+}$) up to 5% wt of the process batch to adjust the plutonium valency state to assist separation of Pu from U. The isotopic composition of this $U^{4+}$ (typically natural uranium) therefore affects the isotopic composition of the RepU product.

Naturally occurring uranium contains only the three isotopes $^{234}$U, $^{235}$U and $^{238}$U. The levels of these U isotopes in natural uranium are shown in the Table 1 and radioactive decay characteristics of these uranium isotopes are shown in the Table 2.

TABLE 1. RADIOACTIVITY OF URANIUM ISOTOPES PRESENT IN NATURAL URANIUM

| U Isotope | Natural U (wt %) | Natural U (atom %) | % by Activity | Specific Activity (Bq/g) | Activity in 1 g Natural U (Bq) |
|---|---|---|---|---|---|
| $^{232}$U | Not Present | | | | |
| $^{233}$U | Not Present | | | | |
| $^{234}$U | 0.0053 | 0.0054 | 48.9 | 231,300,000 | 12,356 |
| $^{235}$U | 0.711 | 0.72 | 2.2 | 80,011 | 568 |
| $^{236}$U | Not Present | | | | |
| $^{237}$U | Not Present | | | | |
| $^{238}$U | 99.284 | 99.275 | 48.9 | 12,445 | 12,356 |

[Standard Atomic Weight for Uranium = 238.02891]

TABLE 2. DECAY CHARACTERISTICS OF U ISOTOPES PRESENT IN NATURAL U

| Parent Nuclide | U Isotope | Decay Product | Half-Life | Radioactive Decay | |
|---|---|---|---|---|---|
| | | | | Type | Effective MeV |
| $^{238}$Pu | $^{234}$U | $^{230}$Th | 2.455 E+5 years | α | 4.859 |
| $^{239}$Pu | $^{235}$U | $^{231}$Th | 7.038 E+8 years | α | 4.679 |
| $^{242}$Pu | $^{238}$U | $^{234}$Th | 4.468 E+9 years | α | 4.270 |

Following an irradiation course in a nuclear reactor and a period of cooling to allow for the decay of highly active, but short-lived decay products, the reprocessed uranium presents a significantly different isotopic composition vis-à-vis with the natural uranium. Four new uranium isotopes are introduced as a result of irradiation of the fuel and the subsequent decay of irradiation products. These four uranium isotopes are shown in the Table 3.

TABLE 3. CHARACTERISTICS OF U ISOTOPES INTRODUCED BY IRRADIATION IN REACTOR

| Parent Nuclide | U Isotope | Decay Product | Half-Life | Radioactive Decay | |
|---|---|---|---|---|---|
| | | | | Type | Effective MeV |
| $^{236}$Pu | $^{232}$U | $^{228}$Th | 68.9 years | α | 5.414 |
| $^{236}$Pu | $^{233}$U | $^{229}$Th | 1.592 E+5 years | α | 4.909 |
| $^{240}$Pu | $^{236}$U | $^{232}$Th | 2.342 E+7 years | α | 4.572 |
| $^{241}$Pu | $^{237}$U | $^{237}$Np | 6.75 days | β | 0.519 |

Since the isotopes $^{232}$U, $^{233}$U, $^{236}$U and $^{237}$U are not present in natural uranium they are therefore considered as the signature for the RepU.

The amounts of these U isotopes present in RepU depend on a number of factors like the type of fuel used for different reactors, e.g. Advanced Gas-Cooled Reactor (AGR), Pressurized Water Reactor (PWR), Pressurized Heavy Water Reactor (PHWR) and Boiling Water Reactor (BWR), the degree of initial $^{235}$U enrichment, level of burnup at the time of the fuel discharge from the reactor, origin of the fuel (natural, enriched or RepU) and aging periods of spent fuel in cooling water ponds. Almost all irradiated fuels are typically aged (cooled) for five years in specially engineered ponds in order to ensure that the highly active fission products (with short half-lives) have decayed sufficiently so as to permit the fuel for reprocessing without any issues concerning radiological safety. However, for various reasons this cooling period is extended to 10–15 years or even longer.

In view of the influence of uranium even isotopes influence on the reactivity of the reactor core and their radiological impact, their assay is regularly measured along the reprocessed uranium treatment and recycling process. The measurement techniques and related uncertainty for routine measurement are different for the three even isotopes of uranium are presented in Table 4.

TABLE 4. MEASUREMENT TECHNIQUES AND RELATED UNCERTAINTIES FOR URANIUM ISOTOPIC COMPOSITION IN UNH

| Uranium Isotope | Measurement Type | Accuracy |
|---|---|---|
| $^{232}U$ | Absolute value can be measured directly by α-counting or calculated after γ-pick decay measure. | Range of uncertainty 5 % |
| $^{234}U$ | Analytical value is obtained by thermal ionization mass-spectrometry. | ± 0.5 % |
| $^{235}U$ | Analytical value is obtained by thermal ionization mass-spectrometry. | Accuracy better than 1 % |
| $^{236}U$ | Analytical value is obtained by thermal ionization mass-spectrometry with $^{235}U$ and $^{234}U$. | ± 0.5 % |

### *2.1.1. Description of uranium isotopes*

In addition to the three naturally occurring isotopes of uranium ($^{234}U$, $^{235}U$, and $^{238}U$), reprocessed uranium contains several synthetic uranium isotopes which were created in the reactor. These include $^{232}U$, $^{233}U$, $^{236}U$, and $^{237}U$. Of these natural and synthetic isotopes of uranium, $^{235}U$ and $^{238}U$ are the major isotopes, and the rest are considered as the minor isotopes. This section considers the properties, concentration and impact of the minor isotopes of uranium in the RepU. The formation, subsequent decay chain, and possible parent nuclides, are now considered for each isotope. Each of the uranium isotopes is a member of one of the four possible Radioactive Decay Series known as the 4n, 4n+1, 4n+2, 4n+3 Series involving successive α-decay and β-decay reactions ultimately leading to the formation of a stable isotope of lead or bismuth. These Radioactive Decay Series are included as Appendices I., II., III., and IV. The formation, subsequent decay chain, and possible parent nuclides, for each isotope are described below.

$^{234}U$ and $^{236}U$ are neutron absorbing isotopes. Hence, their presence in the reactor fuel sourced from the RepU imposes the requirement of over-enrichment of $^{235}U$ to compensate for its presence. This affects the economics for the use of RepU as the reactor fuel because of additional separative work requirement.

## Uranium-232

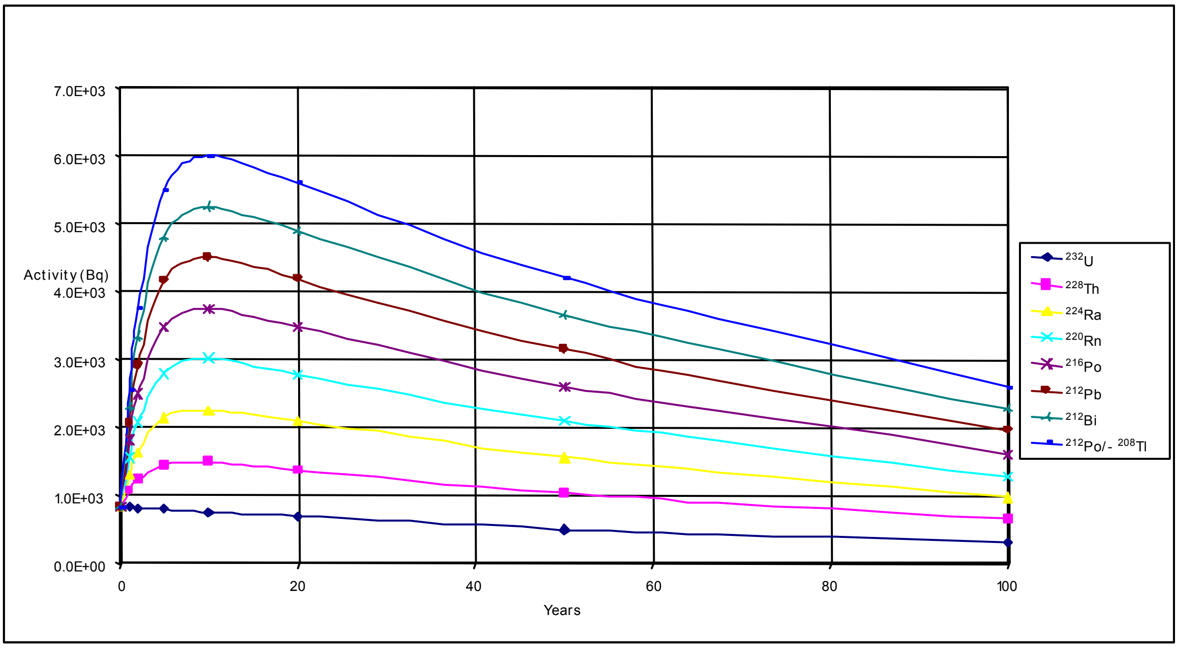


*Fig. 2. Total activity of $^{232}$U and its daughter products as function of time assuming the initial $^{232}$U content of 1 nano-gram $^{232}$U per gram of total U (stacked diagram).*

In a commercial power reactor $^{232}$U is created via several routes, each of which involves several neutron absorption and decay steps. Because the formation routes are rather long and complicated, $^{232}$U appears in very small quantities. $^{232}$U is formed from the α-decay of $^{236}$Pu (half-life 2.86 years) which is the predominant formation route. The typical concentration of $^{232}$U in the RepU is in the range 0.5 to 5 nano-grams $^{232}$U per gram uranium. $^{232}$U is also formed while the spent fuel is allowed to cool in the ponds. Possible parent nuclides for $^{232}$U include $^{232}$Np (electron capture) and $^{232}$Pa (β-decay). The initial α-decay product of $^{232}$U (half-life 68.9 years) is $^{228}$Th. The initial $^{232}$U concentration after reprocessing is dependent on the initial enrichment of the fuel and the irradiation level.

Even though $^{232}$U concentration in the RepU is usually observed between 0.5 to 5 parts per billion (ppb) of $^{238}$U, it is radiologically the most significant of all the U isotopes present. The nuclide $^{232}$U is not itself a major radiological hazard, but some of its daughter products are. $^{232}$U has a 68.9 year half-life, and decays through a series of much shorter-lived daughter products to the stable nuclide $^{208}$Pb. This decay chain goes through three nuclides that emit intense beta and gamma radiation: $^{212}$Pb, $^{212}$Bi, and $^{208}$Tl. $^{208}$Tl gives off a particularly strong gamma radiation, with an average energy of 3.4 MeV per disintegration. Because of this strong gamma radiation at the end of the decay chain, the hazard from $^{232}$U daughter products is dependant on the amount of time that has passed since reprocessing.

In an operating reactor $^{232}$U captures a neutron to form the fissile isotope $^{233}$U. That neutron is recovered when the $^{233}$U fissions. Because of this, and because of its very low concentration, $^{232}$U has no significant impact on reactor operations.

The decay chain 4n Series (Thorium Chain) which includes $^{232}$U is presented in Appendix I.

The build up of activity from $^{232}$U daughter products (to a maximum after ≈ 10 years) for typical irradiated fuel from a PWR with the time is illustrated graphically in Figure 2 as an example. This build up of activity has significant implications for the storage of $UO_3$ or $U_3O_8$ and its subsequent processing for the fuel manufacturing operations.

### Uranium-233

$^{233}$U is formed from the β-decay of $^{233}$Pa (half-life 26.967 days), which, in turn, is formed either from the β-decay of $^{233}$Th (half-life 22.3 minutes) or the α-decay of $^{237}$Np (half-life 2.144 E+06 years). Other possible parent nuclides include $^{233}$Np (electron capture) or $^{237}$Pu (α-decay). The initial α-decay product of $^{233}$U (half-life 1.592 E+5 years) is $^{229}$Th. As with $^{232}$U, the long and complex creation routes tend to keep the concentration of $^{233}$U relatively low. The typical concentration of $^{233}$U in the RepU is about 2 nanograms $^{233}$U per gram uranium (2 parts per billion).

The $^{233}$U in the RepU is not a significant radiological hazard. It is an alpha emitter with the similar decay characterstics to $^{234}$U, but as it is present in such small quantities the $^{233}$U has a negligible contribution to the radiation levels of reprocessed U. Nevertheless $^{233}$U is fissile, so it provides a small benefit when the RepU is reused in a reactor.

The decay chain 4n+1 Series (Neptunium Chain) which includes $^{233}$U is shown in the Appendix II.

### Uranium-234

$^{234}$U is a naturally occurring isotope. The concentration of $^{234}$U in natural uranium is typically 54 micrograms $^{234}$U per gram of uranium (54 parts per million), although a recent analyses of some ores mined using the in-situ leach technique have shown significantly higher $^{234}$U concentrations. $^{234}$U is preferentially enriched along with the $^{235}$U in both of the enrichment processes namely centrifuge as well as diffusion, consequently requiring enrichment operations to be carefully controlled to keep the $^{234}$U concentration below the internationally accepted product specification of 1% of the $^{235}$U concentration ($10*10^3$ µg $^{234}$U/g$^{235}$U). Therefore, fuel with 4% $^{235}$U will typically have just under 0.04% $^{234}$U. During reactor operation the $^{234}$U absorbs a neutron to create $^{235}$U, which can then fission and release neutrons. The concentration of $^{234}$U in the RepU will vary with the burnup, but is typically around 0.018 wt%.

An illustration of the enhanced levels of $^{234}$U in AGR $UO_2$ fuel with 2.207 % and 3.42% $^{235}$U initial enrichment levels is given in the Table 5.

TABLE 5. U ISOTOPE COMPOSITION FOR CERAMIC GRADE $UO_2$ — AGR FUEL

| **U Isotope** | **Natural U as Feed Material** | **For enriched U with 2.207 wt % $^{235}$U** | **For enriched U with 3.42 wt% $^{235}$U** |
|---|---|---|---|
| $^{234}$U | 0.0055 % wt w.r.t. U | 0.019 % wt w.r.t. U | 0.030 % wt w.r.t. U |
| $^{235}$U | 0.718 % wt w.r.t. U | 2.207 % wt w.r.t. U | 3.42 % wt w.r.t. U |
| $^{238}$U | 88.13 % wt | | |

Source: Westinghouse; w.r.t. = with reference to

Possible parent nuclides of $^{234}$U are $^{234}$Pa (by β-decay), $^{234}$Np (by electron capture) and $^{238}$Pu (by α-decay). The decay chain 4n+2 Series (Radium Chain) which includes isotope $^{234}$U is presented in the Appendix III. The initial α-decay product of $^{234}$U (half-life 245 000 years) is $^{230}$Th.

The presence of $^{234}$U in the RepU is predominately a personal protection issue due to its moderate half-life and strong alpha emission. However the levels of $^{234}$U in the reprocessed uranium (and therefore its hazard) are not significantly different from natural uranium.

### Uranium-235

$^{235}$U is present in natural U to the extent of 0.711 wt%. The $^{235}$U content in the RepU depends on the initial $^{235}$U enrichment of the fuel before irradiation and to the extent of burnup. The concentration of $^{235}$U is typically 0.3 to 0.4 % for the RepU which is derived from reprocessing of the spent PHWR fuels. However, in the case of RepU derived from processing of the spent LWR fuel (with the initial enrichment 2.2–5.0 %) the $^{235}$U content is usually in the range between 0.65 % and 1.1 %. But, with the ever increasing tendency to operate reactors to higher fuel burnups (45 GWd/t through to 60 GWd/t), the $^{235}$U content in RepU is expected to reduce to less than 0.6%.

Possible parent nuclides of $^{235}$U are $^{235}$Pa (by β-decay), $^{235}$Np (by electron capture), and $^{239}$Pu (by α-decay). The decay chain 4n+3 Series (Actinium Chain) which also traces $^{235}$U is shown in the Appendix IV. The initial α-decay product of $^{235}$U (half-life 703.8 million years) is $^{231}$Th.

### Uranium-236

$^{236}$U and $^{237}$U are produced by successive neutron captures from $^{235}$U. Possible parent nuclides of $^{236}$U are $^{236}$Pa (by β-decay), $^{236}$Np (by electron capture), and $^{240}$Pu (by α-decay).

The decay chain 4n Series (Thorium Chain) that includes $^{236}$U is shown in the Appendix I. The initial α-decay product of $^{236}$U (half-life 23.42 million years) is $^{232}$Th which has an even longer half life. It has, therefore, no significant radiological impact in the usage of RepU.

The amount of $^{236}$U present is directly proportional to the burnup levels, but is typically around 0.5 wt%. The effect of the $^{236}$U content in RepU on its subsequent use as a reactor fuel is further discussed in the Section 2.4.

### Uranium-237

$^{237}$U is included for the sake of completeness. Its impact on RepU is minimal, as it has a very short half-life (6.75 days). Immediately after reprocessing the RepU exhibits a high β activity due to the presence of $^{237}$U. The initial decay product of $^{237}$U (β-γ emitter) is the long-lived isotope $^{237}$Np and therefore the activity disappears rapidly.

Possible parent nuclides of $^{237}$U are $^{237}$Pa (β-decay) and $^{241}$Pu (α-decay). The initial β-decay product of $^{237}$U is $^{237}$Np (half-life 2.14E+06 years and the longest lived member of the 4n+1 Radioactive Decay Series) which, in turn, decays to $^{233}$Pa (α-decay) with half-life of 27 days and then to $^{233}$U (β-decay). The decay chain 4n+1 Series (Neptunium Chain) that traces $^{237}$U is presented in the Appendix II.

**Uranium-238**

The most abundant U isotope is $^{238}$U comprising 99.284 wt% of natural U.

$^{238}$Pa (β-decay), and $^{242}$Pu (α-decay) are possible parent nuclides of $^{238}$U. The decay chain 4n+2 Series (Radium Chain) for $^{238}$U is presented as Appendix III. The initial α-decay product of $^{238}$U (with a half-life 4.468E+9 years) is $^{234}$Th.

The daughter products of $^{238}$U that are of radiological concern are $^{230}$Th, $^{226}$Ra and $^{222}$Rn, all of which have long half-lives.

### *2.1.2. Typical RepU isotopic composition for irradiated fuel*

Under normal fuel management circumstances by the time the fuel is discharged from the reactor after irradiation, most of the $^{235}$U atoms are burnt and a proportion of the $^{238}$U atoms are also transmuted by neutron capture. This section concentrates on the levels of $^{234}$U, $^{235}$U, $^{236}$U and $^{238}$U isotopes in the spent fuel freshly discharged from a PWR reactor. While $^{232}$U and $^{233}$U are not of much relevance in this short time regimes, but these isotopes are of greater significance during the cooling period (see Section 2.1.3).

The dependence of $^{234}$U isotopic content in a typical irradiated PWR fuel on the extent of burnup for a given initial $^{235}$U content in the fresh fuel is depicted in the Figure 3. The figure indicates that the initial concentration of $^{234}$U is primarily related to the original $^{235}$U content in the fresh fuel and $^{234}$U gets burned gradually during the reactor irradiation.

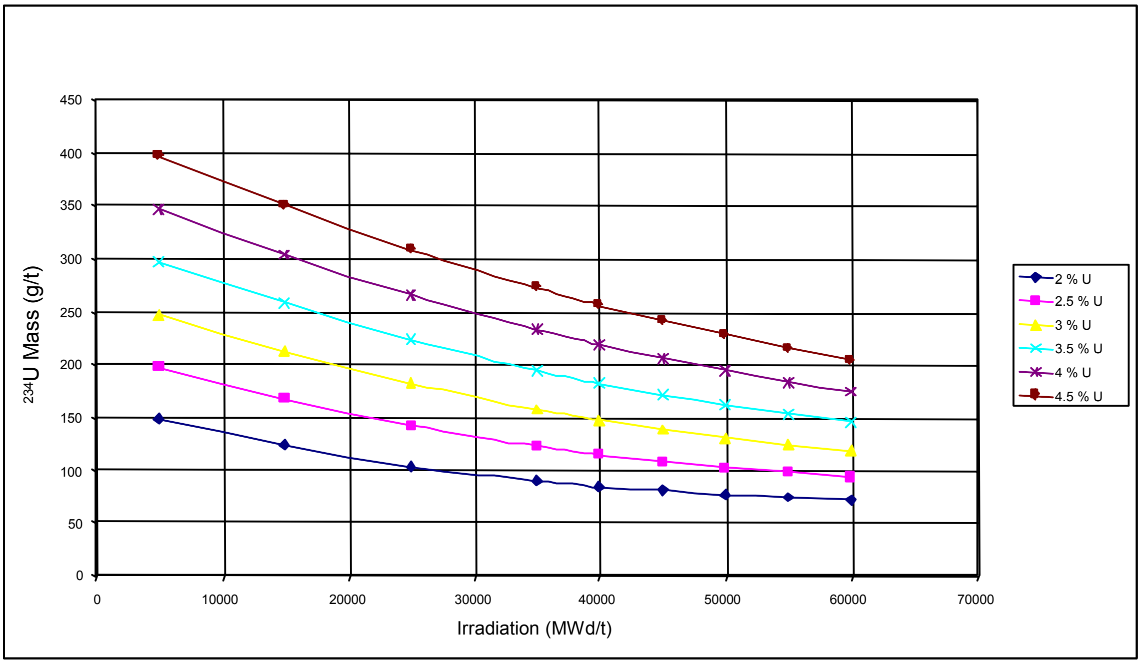


*Fig. 3. PWR fuel residual $^{234}$U content as function of the extent of burnup level for selected initial $^{235}$U contents as indicated; fuel rating 40 MW/t HM.*

The extent of depletion of $^{235}$U in the freshly discharged spent fuel from a PWR as a function of the burnup levels for six selected initial enrichments is shown in Figure 4. The dependence of build-up of $^{236}$U in irradiated PWR fuel on the extent of discharge burnup for given six $^{235}$U initial enrichments is illustrated in Figure 5. The figure reveals that the accumulation of $^{236}$U is interlinked to both the initial $^{235}$U enrichment in the fuel and to the level of irradiation.

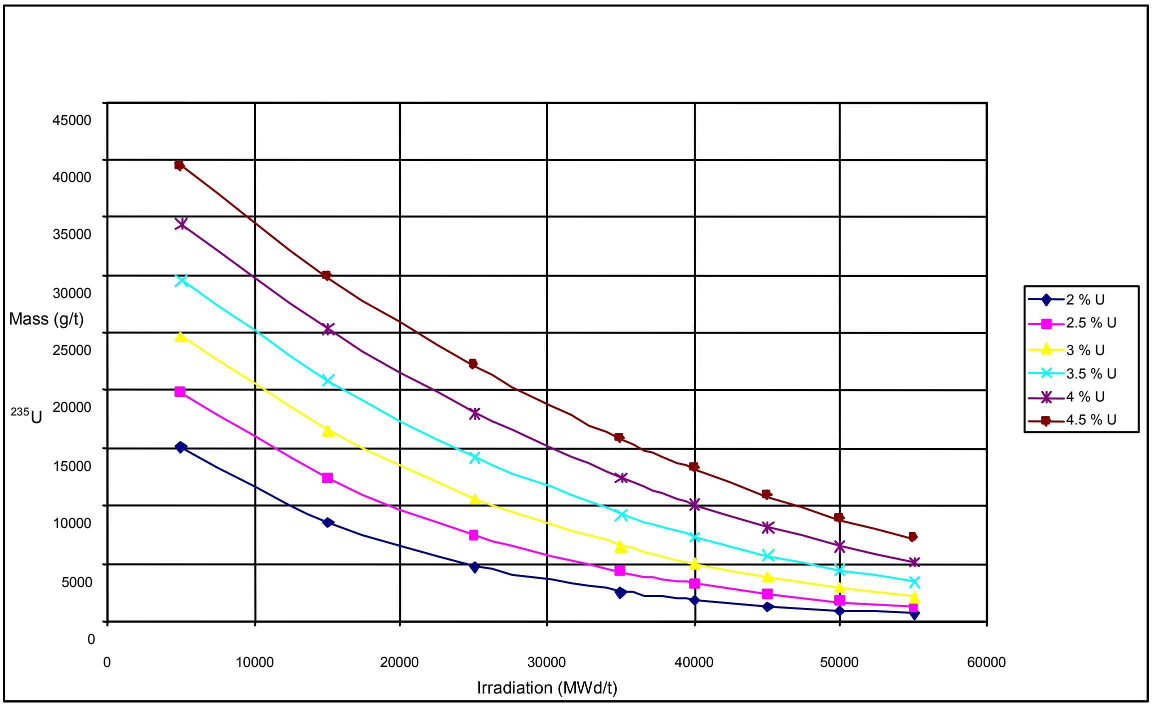


*Fig. 4. Residual $^{235}$U content in the PWR fuel as a function of burnup level for given initial $^{235}$U content; fuel rating 25 MW/t HM.*

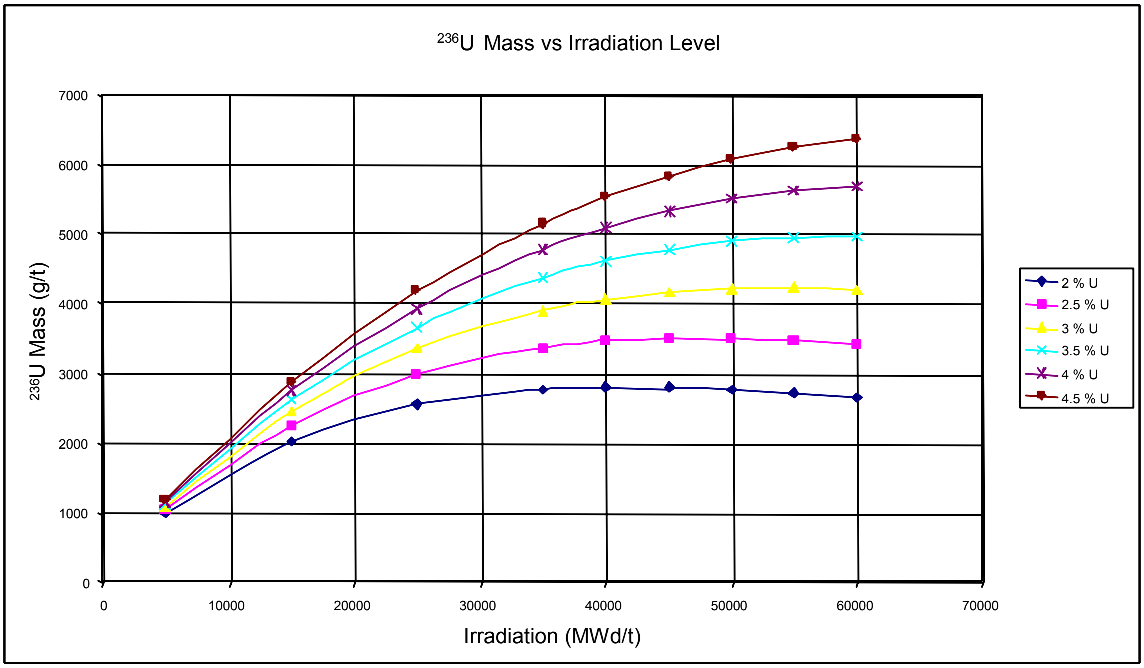


*Fig. 5. Relationship between the build up of $^{236}$U in an irradiated PWR fuel with the discharge burnup level for selected initial $^{235}$U enrichments for the fresh fuel as indicated in the figure; fuel rating 40 MW/t HM.*

When a $^{235}$U atom absorbs a neutron, fission occurs as the major reaction. But also a small fraction of neutrons are also absorbed by $^{235}$U resulting in the formation of $^{236}$U which is non-fissile. The amount of $^{236}$U formed during irradiation is proportional to the initial $^{235}$U content in the fuel. This build-up of $^{236}$U during irradiation is reflected in Figure 5. Further, the relationship between the ratio of $^{236}$U to $^{235}$U and the burnup level is depicted graphically in Figure 6. The figure indicates that low initial enrichment of $^{235}$U in LWR fuel results in relatively higher value for the ratio of $^{236}$U to $^{235}$U at higher burnup level.

$^{236}$U is a neutron absorber which reduces the neutronic activity of the fuel; it is a poison which can absorb another neutron to produce short-lived $^{237}$U which decays to non-fissile $^{237}$Np. This has considerable implications in reusing RepU as a reactor fuel warranting additional enrichment to compensate for this loss of activity. For instance, to make LWR fuel using the RepU (which is obtained from reprocessing of spent LWR fuel), the fuel should contain more than 4.5 % $^{235}$U to accomplish a target burnup of 45 GWd/t HM, while the same burnup could be achieved with only 4.1 % $^{235}$U enriched uranium fuel if it is obtained from “Enriched uranium derived from Natural Uranium” (ENU). This additional over-enrichment for “Enriched RepU” (ERU) to meet as LWR fuel compared to that of the normal LEU of ENU-origin to accomplish similar burnup levels is illustrated graphically in Figure 7.

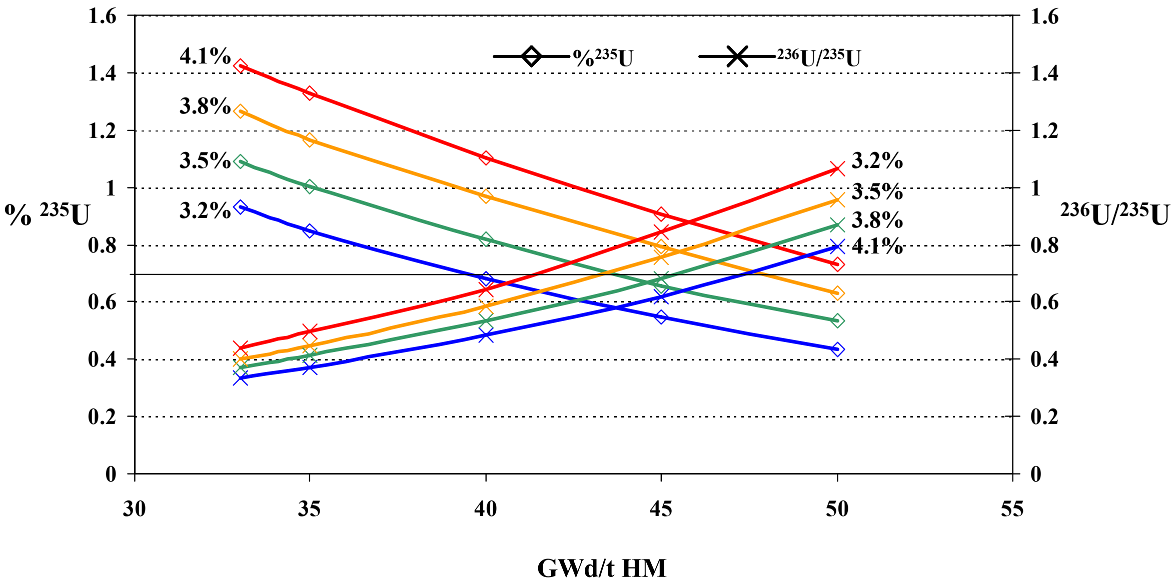


*Fig. 6. The dependence of the residual $^{235}$U content as well as the ratio of $^{236}$U/$^{235}$U in a typical spent LWR fuel with its burnup level for given initial enrichment of $^{235}$U content in the fresh fuel. Data from RWE NUKEM.*

The fertile $^{238}$U atoms that constitute the major fraction of any uranium based fuel are transmuted in the core of the reactor by the capture of neutron. The depletion of $^{238}$U atoms due to the transmutation in a PWR fuel as a function of burnup in the reactor is delineated in Figure 8.

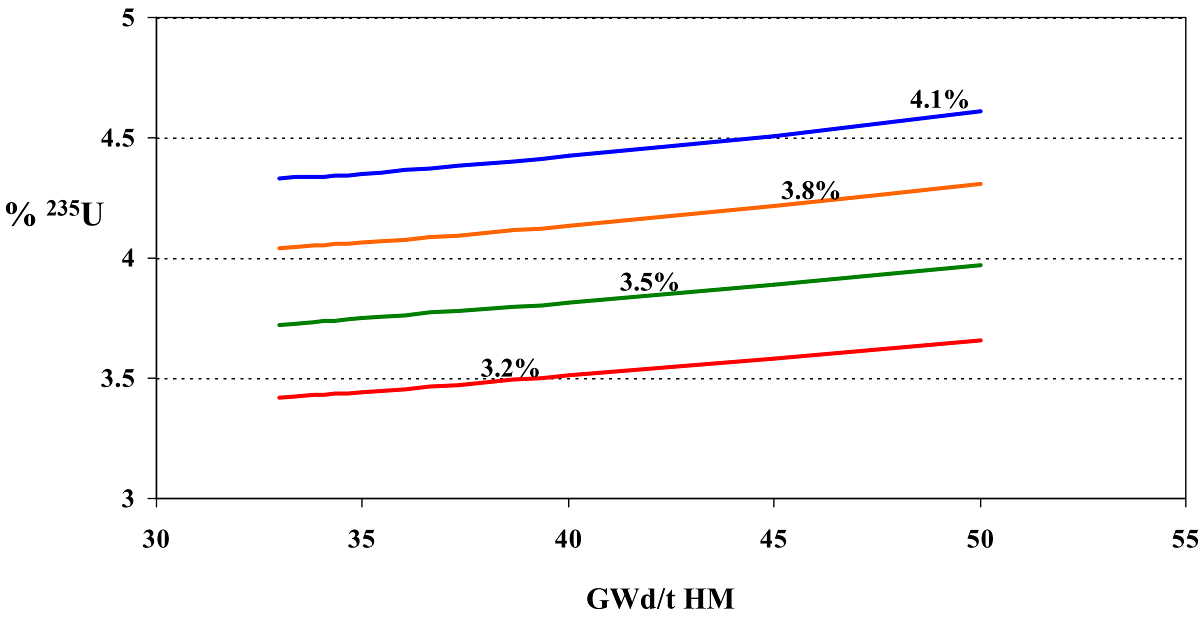


*Fig. 7. Additional enrichment required for RepU-based LWR fuel as a function of burnup level. Source: RWE NUKEM.*

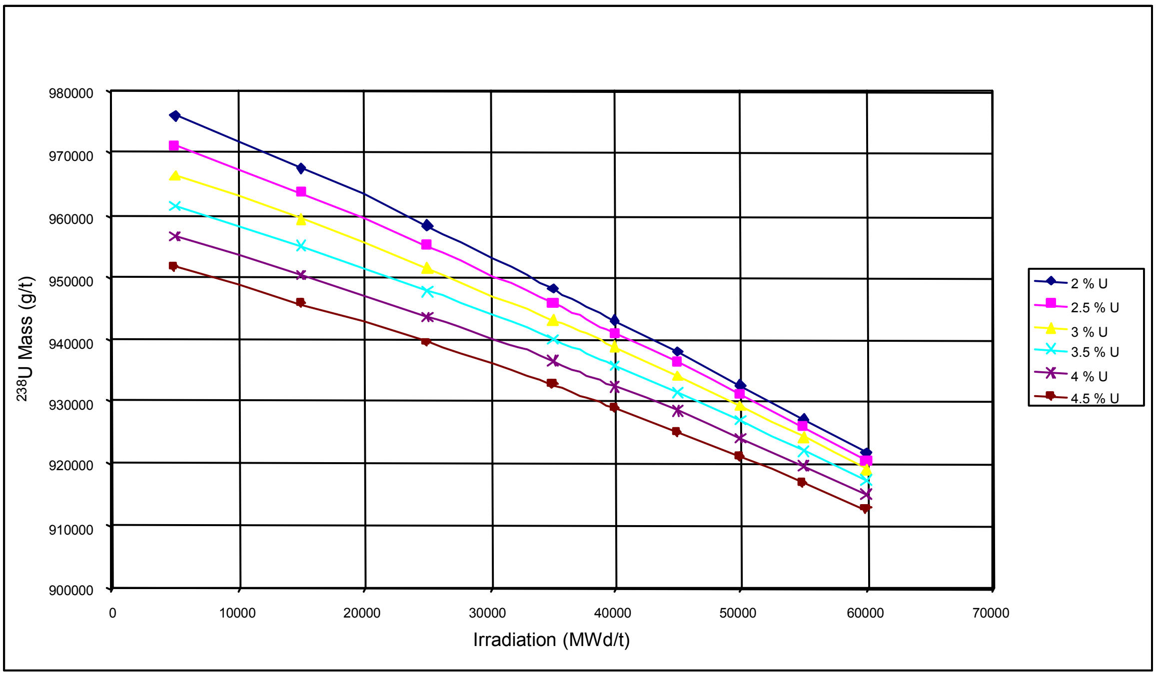


*Fig. 8. Depletion of* $^{238}$*U content in a PWR fuel as a function of irradiation levels for selected initial* $^{235}$*U content before loading into the reactor; fuel rating 40 MW/t HM.*

### *2.1.3. Impact of cooling and storage time of spent fuel on RepU isotopic composition*

There are no measurable changes for isotopes $^{235}$U, $^{236}$U and $^{238}$U contents by their radiological decay during the time of cooling of discharged spent fuel and storage prior to reprocessing operation. Hence, this section concentrates only on the variations in contents of $^{232}$U, $^{233}$U, $^{234}$U and $^{237}$U isotopes as the fuel is cooled in the storage ponds prior to reprocessing. Since the major fraction of short-lived actinides would decay during the cooling time, and there is no fresh creation of these actinides isotopes (unlike in the reactor irradiation), the growth rate of some U isotopes depends on its own radiological stability. For example, the decay of the remnant $^{236}$Pu (with a half-life 2.858 years) and $^{232}$Np (with a half-life 14.7 minutes) that are present in the spent fuel during the initial phase of cooling of the spent fuel would form increasing quantities of $^{232}$U. Subsequently the decay of $^{232}$U with a half-life of 68.9 years would result in its depletion. This effect of cooling time on the in-growth and decay of the $^{232}$U isotope in PWR fuel is shown graphically in Figure 9. On the other hand the $^{233}$U content in the spent fuel monotonically increases, due to its relatively slow decay. The in-growth of the $^{233}$U isotope in PWR spent fuel during the cooling is depicted in Figure 10. Similarly the levels of $^{234}$U change relatively slowly with time. The increased levels of $^{234}$U in the spent fuel compared to un-irradiated uranium are due to its in-growth from the decay of $^{238}$Pu. The in-growth of $^{234}$U isotope in a spent PWR fuel during cooling is illustrated in Figure 11. At low burnups the level of $^{234}$U is almost constant during cooling as can be seen is Figure 12. On the contrary, the content of the $^{237}$U isotope (which has a very short half-life) in the spent fuel steadily diminishes during cooling (Figure 13).

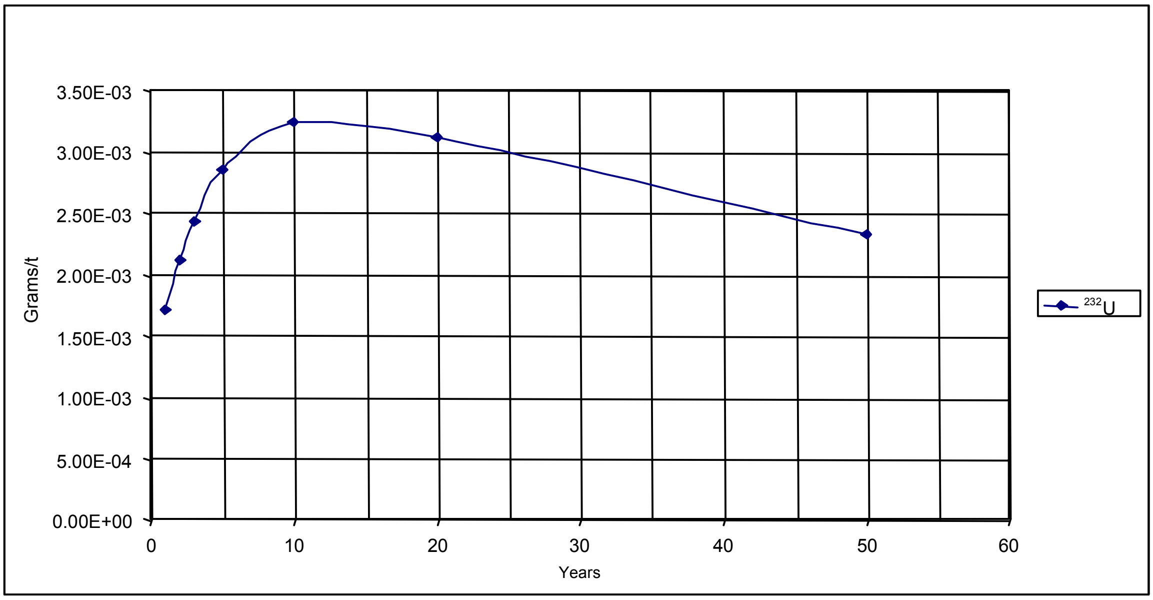


*Fig. 9. Description of in-growth and decay of $^{232}$U isotope in a typical PWR spent fuel as a function of the cooling time with the registered discharge burnup of 48 GWd/t HM and the fuel had an initial enrichment of 4.5% $^{235}$U prior to irradiation.*

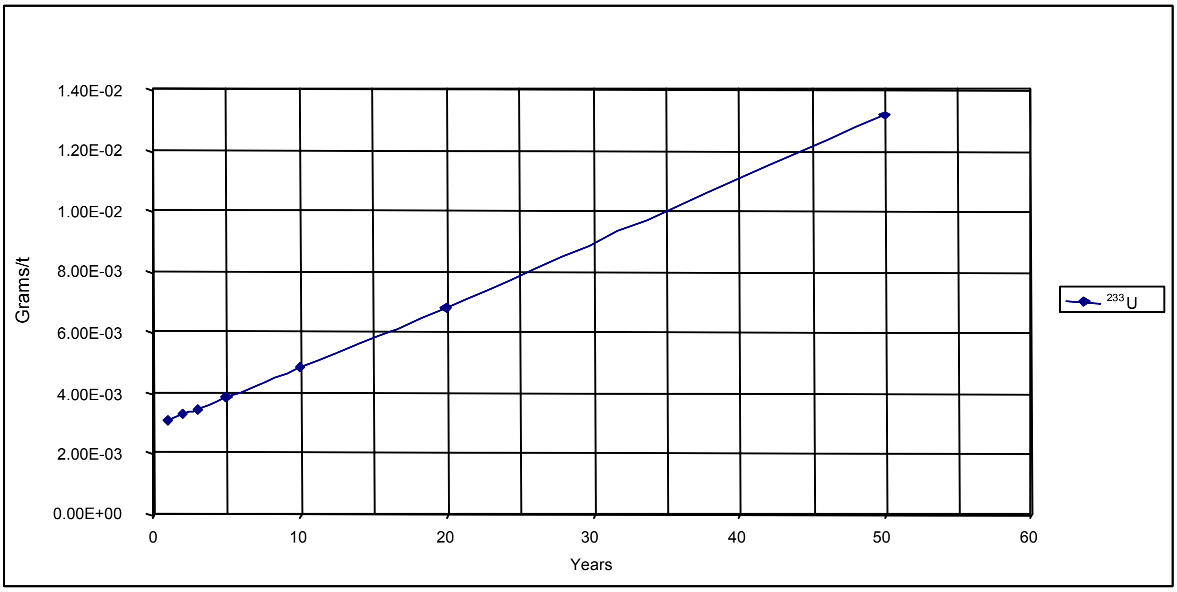


*Fig. 10. In-growth of* $^{233}U$ *isotope during the cooling time of a typical PWR spent fuel with the recorded discharge burnup of 48 GWd/t HM; the fuel had an initial enrichment of 4.5%* $^{235}U$ *prior to irradiation.*

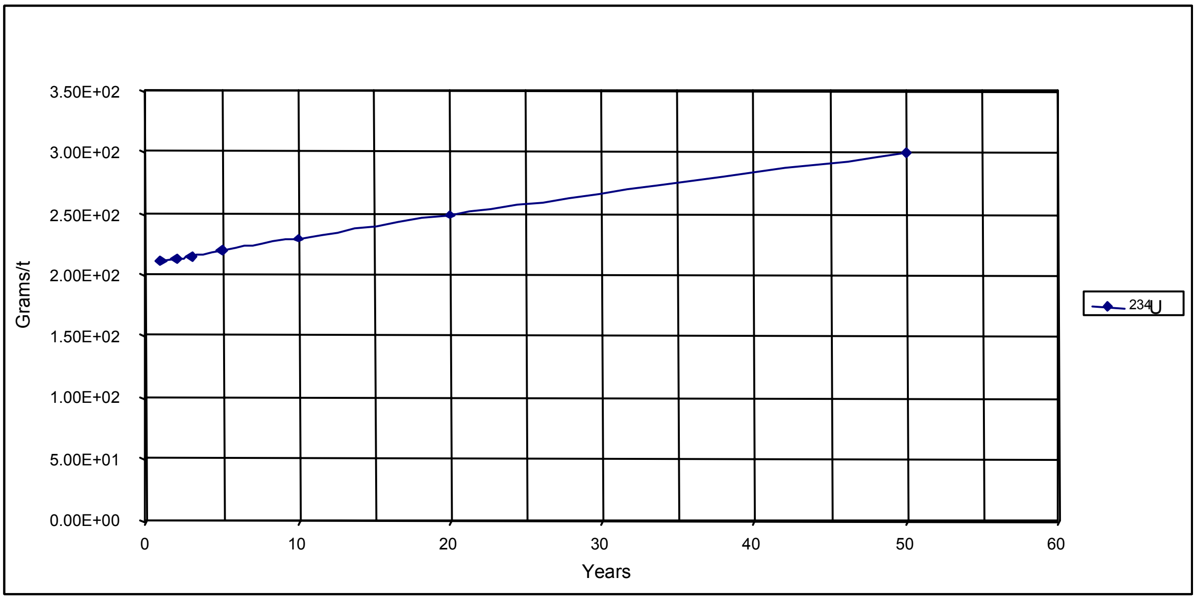


*Fig. 11. Accumulation of* $^{234}U$ *isotope during the cooling of a typical spent PWR fuel with the registered discharge burnup of 48 GWd/t HM; the fuel had an initial enrichment of 4.5% prior to loading into the reactor.*

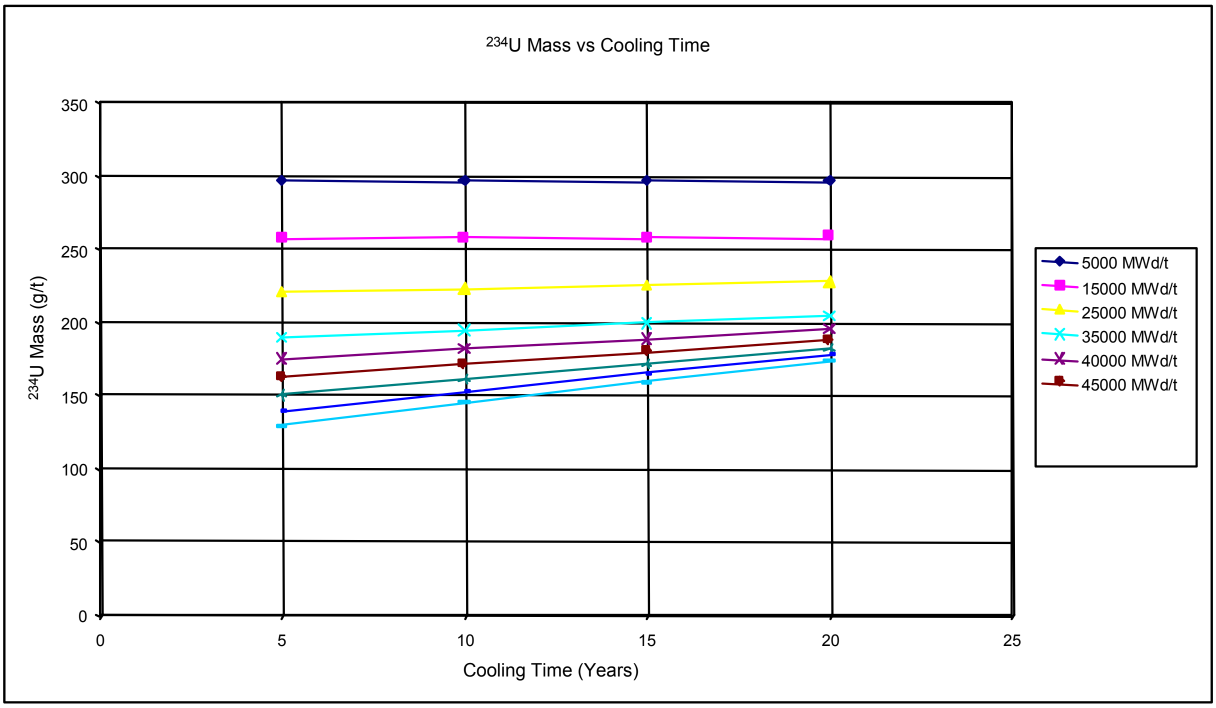


*Fig. 12. Variation of* $^{234}U$ *content as a function of cooling time for a typical spent PWR fuel for selected discharge burnups as indicated in the graph. The fuel was rated of 40 MW/t HM and fuel had an initial enrichment of 3.5 % prior to irradiation.*

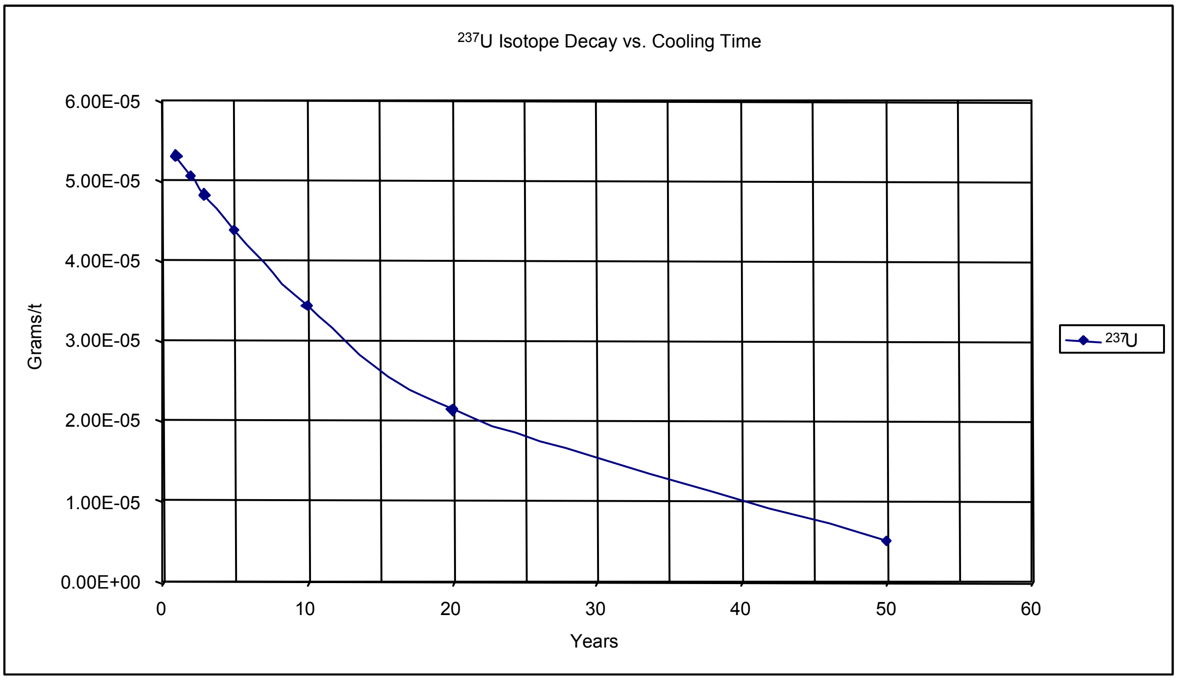


*Fig. 13. Depletion of* $^{237}U$ *isotope during the cooling time for a typical spent PWR fuel discharged from PWR. The burnup of the fuel was 48 GWd/t HM at the time of discharge and the fuel had an initial enrichment of 4.5%* $^{235}U$ *prior to loading into the reactor.*

### *2.1.4. U isotope composition of irradiated fuels (5 years cooled)*

Several factors such as the nature of the uranium used for the fuel (e.g.: natural uranium based or RepU-based), the initial $^{235}$U enrichment of the fresh fuel, the type of the reactor (AGR, LWR, PHWR, etc.) and the discharge burnup of the spent fuel, cooling time of the spent fuel would influence the evolution of the U isotopic composition of the spent fuel. To illustrate the influence of these factors on the U isotopic composition, some typical cases covering spent fuels from AGRs and LWRs as well as RepU based spent fuels which are cooled for 5 years are presented below in the tabular form.

TABLE 6. INFLUENCE OF THE DISCHARGE BURNUP ON THE ISOTOPIC COMPOSITION OF URANIUM FOR A TYPICAL SPENT AGR FUEL WHICH IS COOLED FOR 5 YEARS. THE INITIAL $^{235}$U ENRICHMENT OF THE FRESH FUEL IS 3.7 % $^{235}$U AND THE DISCHARGE BURNUP OF THIS FUEL ARE AS SPECIFIED IN TABLE

| **U Isotope** | **Burnup 35 GWd/t HM** | **Burnup 40 GWd/t HM** |
|---|---|---|
| | Grams | Grams |
| $^{232}$U | 1.03 E-03 | 1.38 E-03 |
| $^{233}$U | 1.59 E-03 | 1.67 E-03 |
| $^{234}$U | 2.54 E+02 | 2.34 E+02 |
| $^{235}$U | 7.94 E+03 | 5.60 E+03 |
| $^{236}$U | 6.07 E+03 | 6.36 E+03 |
| $^{237}$U | 1.73 E-05 | 1.91 E-05 |
| $^{238}$U | 9.43 E+05 | 9.40 E+05 |

Note: Basis of data is with respect to 1 tonne of initial heavy atoms

TABLE 7. ISOTOPIC COMPOSITION OF URANIUM IN SPENT LWR FUEL WHICH IS COOLED FOR 5 YEARS. THE INITIAL $^{235}$U ENRICHMENT OF THE FRESH FUEL AND THE DISCHARGE BURNUP OF THIS FUEL ARE AS SPECIFIED IN TABLE

| **U Isotope** | **Burnup 40 GWd/t HM - Initial Enrichment 4.0 % $^{235}$U** | **Burnup 48 GWd/t HM - Initial Enrichment 4.0 % $^{235}$U** | **Burnup 48 GWd/t HM - Initial Enrichment 4.5 % $^{235}$U** | **Burnup 60 GWd/t HM - Initial Enrichment 4.1 % $^{235}$U** |
|---|---|---|---|---|
| | Grams | Grams | Grams | Grams |
| $^{232}$U | 1.71 E-03 | 2.78 E-03 | 2.86 E-03 | 4.28 E-03 |
| $^{233}$U | 2.90 E-03 | 3.28 E-03 | 3.81 E-03 | 3.54 E-03 |
| $^{234}$U | 2.11 E+02 | 1.83 E+02 | 2.19 E+02 | 1.63 E+02 |
| $^{235}$U | 1.00 E+04 | 7.73 E+03 | 1.03 E+04 | 4.92 E+03 |
| $^{236}$U | 4.96 E+03 | 5.29 E+03 | 5.83 E+03 | 5.68 E+03 |
| $^{237}$U | 3.48 E-05 | 4.40 E-05 | 4.38 E-05 | 4.99 E-05 |
| $^{238}$U | 9.33 E+05 | 9.24 E+05 | 9.21 E+05 | 9.13 E+05 |

Note: Basis of data is with respect to 1 tonne of initial heavy atoms

However, the uranium isotope composition of a typical spent ERU fuel (ERU = Enriched Reprocessed Uranium) from LWR which is cooled for 5 years has shown marked increase in some of the uranium isotopes content (see Table 8). An approximate comparison of the U isotopic composition of ERU-based spent fuel with the ENU-(Enriched Natural Uranium) based fuel for a similar burnup and initial enrichment characteristics could be made using the Table 7 and 8. For instance, in a typical RepU-based spent fuel the contents of $^{232}U$, $^{234}U$ and $^{236}U$ would be higher by ≈ 450%, ≈ 360% and ≈ 330% respectively compared to that of ENU-based fuel.

TABLE 8. U ISOTOPIC COMPOSITION OF IN 5-YEAR COOLED SPENT FUEL ENRICHED REPU LWR FUEL WITH INITIAL ENRICHMENT OF $^{235}U$ AND BURNUP AS INDICATED IN TABLE

| **U Isotope** | **Burnup 50 GWd/t HM - Initial Enrichment 4.5 % $^{235}U$** |
|---|---|
| | Grams |
| $^{232}U$ | 1.16 E-02 |
| $^{233}U$ | 1.27 E-02 |
| $^{234}U$ | 7.90 E+02 |
| $^{235}U$ | 9.13 E+03 |
| $^{236}U$ | 1.92 E+05 |
| $^{237}U$ | 4.07 E-05 |
| $^{238}U$ | 9.06 E+05 |

Note: Basis of data is with respect to 1 tonne of initial heavy atoms

## 2.2. Chemical impurities

The **P**lutonium **U**ranium **R**ecovery by **EX**traction (PUREX) process achieves an excellent level of separation of plutonium from uranium and the removal of highly active fission products. Nevertheless, it is not possible to separate completely the other materials such as cladding and miscellaneous internals such as design springs before the initial dissolution of the spent fuel in concentrated nitric acid. Hence, some trace quantities of impurities that are not completely removed by the PUREX process are of particular significance in defining the constitution of the chemical impurities. The level of impurities present in RepU depends on the efficiency of the separation process e.g. PUREX process. These minor chemical impurities in RepU would also include very small quantities of the transuranic isotopes such as $^{238}Pu$, $^{239}Pu$, $^{240}Pu$, $^{241}Pu$, $^{242}Pu$; $^{237}Np$; $^{241}Am$; $^{242}Cm$, $^{244}Cm$ and the fission products such as $^{95}Zr/^{95}Nb$; $^{103}Ru$, $^{106}Ru$; $^{134}Cs$, $^{137}Cs$; and $^{99}Tc$. Some of these impurity elements i.e. boron, silicon, vanadium, tungsten, technetium, neptunium, ruthenium, chromium, and molybdenum form volatile fluorides during the process of conversion and re-enrichment of RepU. Thus certain proportion of these impurities may possibly follow the process route with the $UF_6$ stream during the conversion, enrichment and storage process. Refer to Appendix V. for typical analyses of $UO_3$ obtained from reprocessed uranium and stipulated ASTM standards for conventional natural uranium product.

The specifications for the finished RepU product for interim storage, whether Uranyl Nitrate Hexahydrate (UNH) subsequently converted to $U_3O_8$ via the intermediate Ammonium Di-Uranate (ADU), $UO_3$ or $U_3O_8$, is largely determined by two parameters:

(a) the relevant ASTM specifications for $UF_6$, $UO_2$ powder and $UO_2$ pellets that are referenced below, and

(b) the process to be used for conversion to $UF_6$ for re-enrichment, or the blending process to produce enriched material.

The acceptability levels of impurities in the product for intermediate storage depends on the process to be adopted for manufacturing nuclear fuel from the RepU. If, for example, the RepU material has to be converted to $UF_6$ prior to enrichment then the specification will take into account the known decontamination factors across the manufacturing process, in particular, known volatile and non-volatile fluorides.

In addition, for both $UO_3$ and $U_3O_8$ as the RepU finished product, there are limits in the specifications on the nitrate content (reflecting the degree of completeness of the precursor thermal decomposition); moisture content (storage properties) and sulphate content.

It is interesting to note that at each stage of gas extraction of $UF_6$ from its $UF_6$ storage/transport cylinder will result in a further purification of RepU from the impurities present, as some fraction of the impurities would remain in the source cylinder. This is a complimentary process for further purification of RepU before the enrichment and fuel fabrication steps.

The following American Standards for Testing Materials (ASTM) Standards [2] can be referred to for drawing the specifications for RepU:

ASTM C 787 – 06 Standard Specification for Uranium Hexafluoride for Enrichment.

ASTM C 788 – 03 Standard Specification for Nuclear Grade Uranyl Nitrate Solution or Crystals.

ASTM C 776 – 00 Standard Specification for Sintered Uranium Dioxide Pellets.

ASTM C 1348 – 01 Standard Specification for Blended Uranium Oxides with a $^{235}U$ Content of less than 5 % for Direct Hydrogen Reduction to Nuclear-Grade Uranium Dioxide.

ASTM C 1334 – 96 (Reapproved 2000) Standard Specification for Uranium Oxides with a $^{235}U$ Content of less than 5 % for Dissolution prior to Conversion by Direct Hydrogen Reduction to Nuclear-Grade Uranium Dioxide.

### 2.3. Chemical form of the RepU

As discussed in the Section 2.2, the chemical form of the RepU intermediate depends on the manufacturing process used to produce the intermediate for storage, when it is required for further processing and what process route is to be adopted to manufacture the RepU fuel.

Although UNH is the product from the reprocessing plant, it is rarely stored in this form due to its physical and chemical properties. If the RepU is to be stored it is converted into $UO_3$ or $U_3O_8$ or $UO_2$ or $UF_6$ or U metal. If it is for immediate conversion to RepU fuel it will be converted to $UO_2$, or $UF_6$

There is another important consideration, which is concerning the physical properties of the chemical intermediates. For example, for using RepU directly either in MOX fuel fabrication or in CANDU reactor fuel fabrication, the $UO_2$ product with very good physical characteristics (such as sinterable and free-flowing) is required. If the RepU is to be processed via a $UF_6$ enrichment route, then the physical properties of the $UO_3$, $U_3O_8$ or $UO_2$ precursors are of much less significance as it involves additional steps such as conversion and blending which could be adjusted to give desirable physical properties of the final product.

The conversion route for the production of $UO_2$ from RepU for MOX includes the precipitation of ammonium diuranate (ADU), its thermal decomposition to $U_3O_8$ and subsequent hydrogen reduction to obtain a sinterable and free-flowing ceramic grade $UO_2$.

## 2.4 Chemical additions to the RepU product

In the manufacture of $UF_6$ from $UO_3$ derived from thermal denitration of uranyl nitrate liquor in a fluidised bed reactor, a small amount of sulphate (as sulphuric acid) is added to the UNL prior to denitration. The addition of sulphate is to enhance the internal surface area of the $UO_3$ powder formed on the thermal denitration of uranyl nitrate and thus improve its reaction yield with HF during the hydro-fluorination reaction.

$$UO_3 \xrightarrow{H_2} UO_2 \xrightarrow{HF} UF_4 \xrightarrow{F_2} UF_6$$

## 3. REPROCESSED URANIUM PRODUCTION: PAST, CURRENT AND FUTURE

### 3.1. How is reprocessed uranium produced?

In general, the spent UOX fuel from thermal reactors contains approximately 81 to 83 wt % of uranium (whereas ≈ 13 wt % oxygen), the remaining 4 to 6 wt % being plutonium, (some minor actinides) as well as fission products depending on the level of burnup. Conventional nuclear fuel reprocessing of spent fuel consists of dissolving fuel materials in aqueous medium and separating uranium and plutonium initially from the fission products using solvent extraction process. The separated products namely the uranium and plutonium are further purified by the solvent extraction process for the removal of impurities. Finally the wastes are treated for concentrating and immobilizing (vitrification). The separated uranium and plutonium are returned to the nuclear fuel cycle as either uranium oxide or MOX fuels.

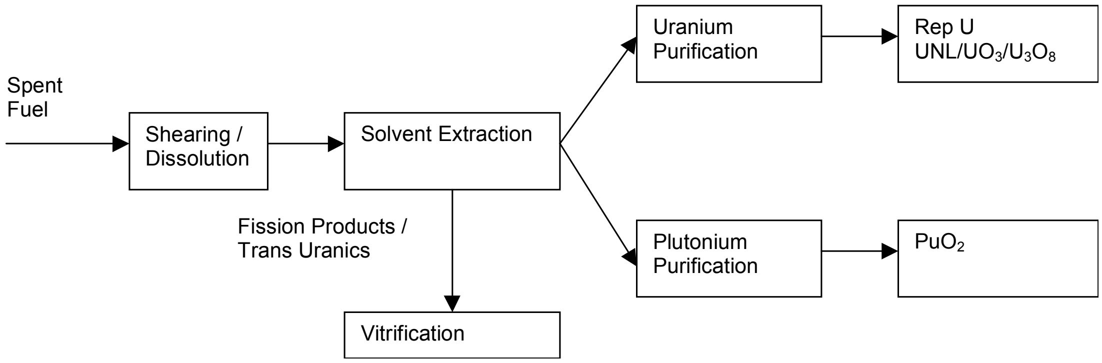


*Fig. 14. Simplified flow diagram for reprocessing of spent fuel.*

## 3.2. Reprocessing facilities: Past, present and future

### *3.2.1. Belgium*

EUROCHEMIC (European Company for the Chemical Processing of Irradiated Fuel) operated a pilot reprocessing facility in Dessel (Belgium), from 1966 to 1974. Over its operating life, the plant reprocessed 86 tonnes heavy metal (t HM) of spent fuel from the research reactors and 95.4 t HM of spent fuel from power reactors. The plant thus produced 30.6 t HM of uranium-aluminium alloy containing 1.36 t HM of HEU. Some fuel elements of the Material Test Reactor (MTR) were also reprocessed at the EUROCHEMIC plant.

### *3.2.2. China*

Since middle of the 1970's, R&D on reprocessing was carried out on large scale for civilian purposes. In the early 1980's, a multi-purpose reprocessing pilot plant project was incorporated in the national economy plan consisting of a receipt and storage facility, a main reprocessing facility with a maximum throughput of 400 kg LEU/d, a hot cell laboratory with a 0.90 kg HEU/d capacity, and a Machinery Testing Workshop (MTW), as well as some auxiliary facilities. With the exception of the MTW coming on stream in 1993 in advance, construction of all buildings is being carried out actively. It is planned that by supplementing some waste management facilities, the pilot plant could be later reconstructed to a small-scale production plant with a capacity of 80 to 100 t HM/year. After gaining extensive experience and accumulating sufficient amount of spent fuel, a large scale commercial plant, possibly with a 800 t HM/year capacity, would be commissioned around 2020 in order to match with the nuclear power capacity at that time [27].

### *3.2.3. France*

In France, the following three commercial reprocessing facilities have generated very significant reprocessing experience.

#### *UP1 at Marcoule*

From 1958 to 1976, the UP1 plant reprocessed fuels for CEA needs (G1, G2, G3 and Chinon reactors). Then commercial activities were initiated on the site in 1976, when UP1 began reprocessing spent fuel from the French natural uranium-fuelled, graphite moderated, gas-cooled reactors (GCRs) (Chinon-2 and -3, Saint Laurent des Eaux-1 and -2 and Bugey-1) and Hifrensa. Production in the UP1 plant was terminated at the end of 1997 after 40 years of operation. Since 1998, the plant has been undergoing final shutdown operations, retrieval and repackaging of accumulated waste and dismantling of the plant. Overall, UP1 handled 18 260 t HM of spent fuel and produced approximately 17 300 t HM of reprocessed uranium.

#### *UP2 and UP3 at La Hague*

The UP2 plant was commissioned in 1966 at La Hague. First dedicated to process spent fuel from gas-graphite reactors, it was adapted in 1976 to reprocess spent UOX fuel. UP2 reached its nominal capacity of 400 t HM/year in 1987. It was refurbished in 1994 to increase its capacity to 800 t HM/year, and it reached this new nominal capacity in 1996 (UP2-800).

At the La Hague site, UP3, a facility with a capacity of 800 t HM/year and initially dedicated to foreign customers, was commissioned in 1990. As of today the two operating plants located at La Hague, UP2-800 and UP3, can be considered as a single industrial platform that has a licensed capacity of 1 700 t HM/year. Characteristics of the reprocessed uranium produced at La Hague plant are given in Table 9.

TABLE 9. UNH PRODUCED AT LA HAGUE PLANT

| **Chemical Form** | $UO_2(NO_3)_2 – 6\ H_2O$ |
|---|---|
| **Physical Form** | Liquid |
| **U Concentration** | ≤ 400 gU/l |

About 5 000 t HM of spent gas-cooled reactor fuel was reprocessed at the La Hague plant between 1966 and 1987, producing about 4 600 t HM of separated reprocessed uranium. Around 19 400 t HM of spent LWR fuel was reprocessed at the La Hague plant as of 31 December 2003, producing about 18 400 t HM of separated reprocessed uranium (see Table 10).

TABLE 10. SPENT LWR FUEL REPROCESSED AT LA HAGUE PLANT BY COUNTRY, AS OF 31 DECEMBER 2003

| Country from where the Spent Fuel Originated | Spent LWR Fuel Reprocessed as of 31 Dec. 2003 (t HM) | Separated RepU as of 31 Dec. 2003 (t U) |
|---|---|---|
| France | 10 038 | ~9 540 |
| Germany | 4 816 | ~4 580 |
| Japan | ~2 900 | ~2 840 |
| Switzerland | 659 | ~630 |
| Belgium | 672 | ~640 |
| Netherlands | 293 | ~280 |
| Total | 19 422 | ~18 500 |

In accordance with the report published in 2003 by the World Nuclear Association's (WNA) Working Group on Reprocessed Uranium and MOX Fuel [3], the future arising of reprocessed uranium in France over the period 2003–2025, based on a 100% reprocessing capacity utilization, would be about 1 600 tU/year.

### *3.2.4. Germany*

The WAK (Wiederaufarbeitungsanlge Karlsruhe) pilot reprocessing facility in Eggenstein-Leopoldshafen was operated from 1970 to 1991. Over 20 years, about 208 t HM of spent fuel from power and research reactors were reprocessed.

### *3.2.5. India*

The spent fuel reprocessing activity in India is almost four decades old [4]. It began with the commissioning of the Trombay Plutonium Plant in 1964 for reprocessing of spent fuel from the research reactor CIRUS. Later, the plant underwent a complete decontamination cycle and was refurbished with enhanced processing capacity. This refurbished plant has been under operation since 1983 and reprocesses the aluminium clad metallic uranium spent fuel from the research reactors.

Two more reprocessing plants have been added to reprocess the zirconium clad, uranium-oxide spent fuel from Pressurized Heavy Water Reactors (PHWR). These are PREFRE-1 at Tarapur (in operation since 1978) and Kalpakkam Atomic Reprocessing Plant (KARP) at Kalpakkam (which has been commissioned recently).

The reprocessing capacities will be increased in the next few years to meet the recycling requirements of India's power program. KARP has provisions for augmentation of capacity and there are plans for nuclear construction of two more plants for the reprocessing of spent PHWR fuel. The spent fuel from the fast reactors will also be reprocessed for the recycling of plutonium.

### *3.2.6. Italy*

From 1970 to 1983, 600 MTR fuel elements from national research reactors and 1.5 t HM of CANDU (Canadian Deuterium-Uranium Reactor) fuel from the Pickering PHWR were reprocessed at ENEA's EUREX (Enriched Uranium Extraction) reprocessing facility.

### *3.2.7. Japan*

In Japan, the Tokai Reprocessing Plant at Tokai Mura (Ibaraki Prefecture) has been operated by Japan Nuclear Cycle Development Corporation (JNC) (now called as Japan Atomic Energy Agency (JAEA) after merging both JNC and Japan Atomic Energy Research Institute (JAERI). Test at the plant began in 1977, while full-scale operation started after 1980. The facility has been in operations ever since. As of 14 November 2003, 1 023 tonnes of LWR spent fuel has been reprocessed at the Tokai Reprocessing Plant (see Figure 15). The average burnup of the spent fuel processed at this plant is 28 000 MWd/t HM.

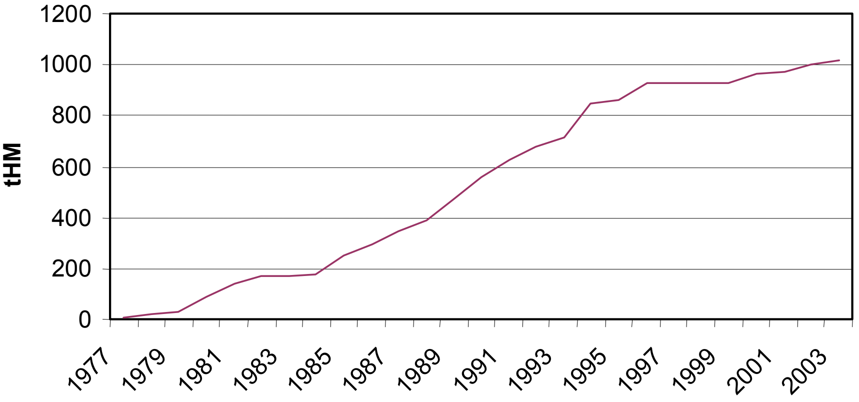


*Fig. 15. Tokai reprocessing plant's spent fuel reprocessing record.*

At the Tokai Reprocessing Plant, 20 to 40 t HM/year of LWR spent fuel have been reprocessed in recent years. The spent fuel from the Fugen HWR is scheduled to be processed by 2010.

Meanwhile, Japan Nuclear Fuel Ltd. (JNFL) is commencing operation of a commercial reprocessing plant with a nominal capacity of 800 t HM/year at Rokkasho Mura in the Aomori Prefecture. The main characteristics of Rokkasho and the Tokai Reprocessing Plants are shown in Table 11.

TABLE 11. MAIN FEATURES OF JAPANESE REPROCESSING PLANTS

| | **Tokai Reprocessing Plant** | **Rokkasho Reprocessing Plant** |
|---|---|---|
| Process | PUREX | PUREX |
| Capacity | 0.7 t HM/day | 800 tU/year |
| %$^{235}$U | < 4% | Pre-irradiation 5%;<br>Post-irradiation 3.5% |
| Burnup (MWd/t HM) | 28 000 | 55 000 (max.) 45 000 (per day) |
| Cooling Time (Before reception) | > 180 days | 1 year |
| Product Form (U) | $UO_3$ | $UO_3$ |
| Product Form (Pu) | (Pu -U) Oxide | (Pu -U) Oxide |

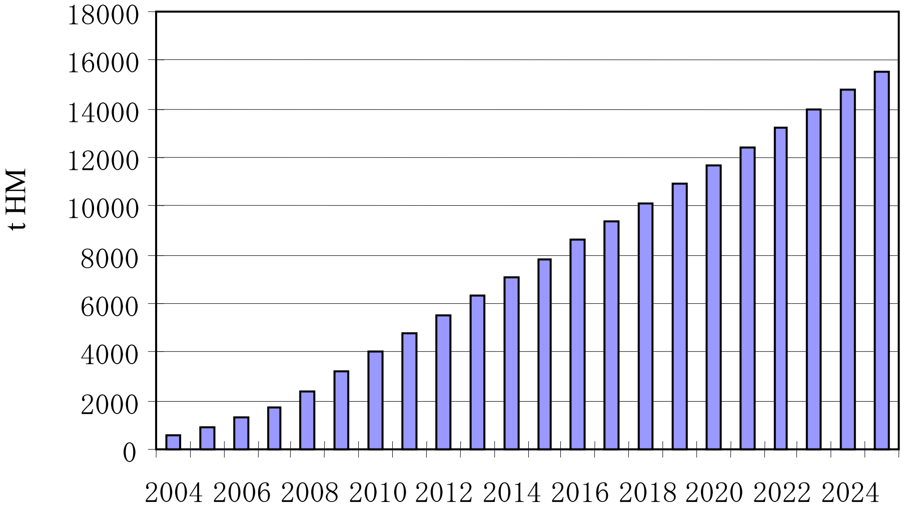
As stated above, the operation of the Rokkasho Reprocessing Plant started in 2006. Table 12 shows the previous spent fuel reprocessing plan of the plant.

TABLE 12. PREVIOUS REPROCESSING PLAN AT THE ROKKASHO REPROCESSING PLANT IN TONNES HM (AS OF SEPTEMBER 2003) [5]

| | **Acquisition Plan of SF** | **Scheduled Reprocessed Amount** | **Stockpile of SF** |
|---|---|---|---|
| **1998** | 8 | | 8 |
| **1999** | 24 | | 32 |
| **2000** | 96 | | 128 |
| **2001** | 340 | | 468 |
| **2002** | 312 | | 780 |
| **2003** | 116 | | 896 |
| **2004** | 500 | 30 | 1366 |
| **2005** | 500 | 270 | 1596 |
| **2006** | 800 | 350 | 2046 |
| **2007** | 850 | 480 | 2416 |
| **2008** | 850 | 640 | 2626 |
| **2009** | 800 | 800 | 2626 |
| **2010** | 800 | 800 | 2626 |

In accordance with this above reprocessing plan, about 750 tonnes HM at Tokai Mura and 3 370 tonnes HM at Rokkasho Mura, totalling about 4 120 tonnes of spent fuel were planned to be reprocessed by 2010. The total production of RepU was previously expected to reach 15 000 tU by 2025.

Since the burnup of the fuel to be reprocessed at the Rokkasho Reprocessing Plant (45 000 MWd/t HM) is higher than that of the fuel reprocessed at the Tokai Reprocessing Plant (28 000 MWd/t HM), the concentrations of $^{232}$U and $^{236}$U in the reprocessed uranium from the Rokkasho Reprocessing Plant are excepted to be higher than those of the Tokai Reprocessing Plant.



2004:Rokkasho Reprocessing Plant Operation

Amount of spent fuel forecast until 2010 contains contribution of Tokai Reprocessing

*Fig. 16. Previously anticipated reprocessed uranium (RepU) stockpile in Japan.*

The reprocessed uranium derived from a low burnup fuel gets consideration for recycling after re-enrichment process, while reprocessed uranium from the high burnup spent fuel could be recycled as MOX fuel. There are no immediate plans to use reprocessed uranium for the LWRs in Japan now. Therefore, Japanese utilities are storing RepU in preparation for the use in the future.

A second commercial reprocessing facility is planned, based on the operating experience of the Rokkasho Mura facility, results of future technology development, supply and demand trend of plutonium, etc. The plant capacity and the reprocessing technology will be chosen in 2010 [5].

### *3.2.8. Russian Federation*

The RT-1 complex of the “Mayak” Production Association carries out reprocessing of spent fuel from VVER-440 reactors in Russia and the Ukraine, as well as from the Russian BN-600 reactor and from spent fuel from research reactors and nuclear power reactors of sea vessels.

RT-1 deals with the range of spent fuel compositions using a kind of PUREX separation process producing both highly enriched reprocessed uranium and low enriched reprocessed uranium that are further used to manufacture nuclear fuel. At the plant, the RepU is re-enriched up to 2.6 % $^{235}$U for the fabrication of fuel for RBMK-1000. Re-enrichment is achieved by mixing uranyl nitrate solutions resulting from reprocessing of different types of spent fuel that contains uranium with different enrichment levels including those with ~20 % enriched uranium.

The Mayak Production Association delivers the RepU in the form of $U_3O_8$ to OAO MSZ (Open-Type Joint-Stock Company "Mashinostroitelny Zavod") complying to the following specifications given in Table 13.

TABLE 13. SPECIFICATIONS OF OAO MSZ REPROCESSED URANIUM

| **Uranium Isotope** | **Limit** |
|---|---|
| $^{232}U$ | 2.7 ppb U |
| $^{234}U$ | 0.16% |
| $^{235}U$ | 14-17% |
| $^{236}U$ | 1.5% |

The RT-1 plant has a design capacity of 400 t HM/year. Currently the amount of spent fuel received for reprocessing at RT-1 plant is considerably below the design capacity, and the actual reprocessing throughput is 120-150 t HM/year.

There are plans to revamp the RT-1 plant in order to improve the technical and economic performance of spent fuel recycling at the plant. An integrated investment project has therefore been prepared which includes:

- Creation of capacities to reprocess spent fuel from VVER-1 000 and foreign PWR reactors, to increase the load of the reprocessing complex;
- Improving the process flowchart to reduce the specific volume of the liquid radioactive waste; and
- Construction of installations for waste reprocessing and conditioning.

By 2008, when modernization is completed, the plant will reprocess up to 300 t HM/year of spent fuel with environmentally acceptable parameters of emissions and discharges.

In the long term perspective, Russia plans construction of the RT-2 reprocessing complex at the Mining and Chemical Combine (MCC) at Zheleznogorsk (Krasnoyarsk region) for reprocessing of spent fuel from VVER-1000 reactors.

### *3.2.9. United Kingdom*

Spent fuel from the Magnox reactors must be reprocessed for technical reasons. B204 was the United Kingdom's very first reprocessing plant, a military plant that was operative from 1951 to 1964. B204 had an annual capacity for processing 300 tonnes of high burnup fuel or maximum of 750 tonnes of low burnup fuel. In 1964, B204 was replaced by a bigger reprocessing plant viz., B205 (see below) which reprocessed fuel for civil use as well. With a greater production capacity than that of B204, this new plant could reprocess fuel from all of the Magnox reactors in the United Kingdom.

The B204 reprocessing plant was subsequently converted into a pre-handling plant for the purpose of pre-handling spent uranium oxide nuclear fuels from the new generation AGRs, prior to reprocessing in B205. Fuel from foreign BWRs and PWRs was also pre-handled at B204 prior to reprocessing in B205. The conversion work for this pre-handling plant was completed in 1969 and operated until 1972, with plans to be restarted later. However, after an accident in September 1973, B204 never resumed operation.

The Sellafield Magnox Reprocessing Plant (B205) was completed and began commercial operation in January 1964 (see Table 14). The plant is still operating and has a capacity of 1 500 tonnes HM/year. BNFL has currently two contracts for reprocessing of overseas Magnox fuel (Japan and Italy) at B205. Based on planned reprocessing throughputs of 1 000 t HM per year, future arisings of $UO_3$ are approximately 950 tonnes of U/year (up to and including 2012).

In May 2000, BNFL announced its intention to close down the great majority of the Magnox reactors by the end of 2010. As a consequence of this, BNFL plans to shut down the Magnox reprocessing plant around 2012 when all the Magnox fuel has been reprocessed. In order to carry out these plans, BNFL has estimated that the B205 plant will have to reprocess an average annual total of 1 000 tonnes of spent Magnox fuel until 2012.

As a consequence of the accident in the pre-handling plant (B204) in September 1973 (see above), it was no longer possible for BNFL to reprocess the ever-increasing more common uranium oxide fuel. This disrupted BNFL’s commercial plans to reprocess fuel on commission from foreign countries. Therefore, plans were made for an additional reprocessing plant that could accommodate this type of fuel.

The plans concerning the new Thermal Oxide Reprocessing Plant (Thorp) were released in March in 1977. It was in 1992 that Thorp construction was completed. Following two years of public consultations and the judicial review, Thorp commercial operations commenced in March 1994. Thorp has an annual reprocessing capacity of 800–900 tonnes HM/year depending on fuel characteristics, such as reactor type (AGR, PWR or BWR), fuel cladding design and burnup (see Table 14). As of January 2005, Thorp has reprocessed over 5 000 tonnes of LWR and AGR fuels (see Table 15). It is expected that Thorp base-load will generate slightly more than 7 000 t reprocessed uranium. Out of this 7 000 tU approximately 4 500 t U will be owned by overseas customers. Further quantities of reprocessed uranium will arise from Thorp post base-load reprocessing.

Most of the fuel from the British Energy’s (BE) Advanced Gas-Cooled Reactors (AGRs) is contracted to be reprocessed in the Thorp plant, but recent contracts allow for long term storage of some fuel at Sellafield. BE has not yet taken any decision on the fate of the fuel from Sizewell-B PWR which has onsite storage capacities for 18 years of spent fuel arisings.

TABLE 14. REPROCESSING FACILITIES IN THE UNITED KINGDOM

| | **B205** | **Thorp** |
|---|---|---|
| Fuel Type | Magnox (Metal Encased in Alloy) | Oxide<br>(LWR and AGR) |
| Commissioning Date | 1964 | March 1994 |
| Forecast Decommissioning Date | 2012 | Undecided |
| Capacity / Year | 1 500 tU | 800-900 tU |

The majority of the Thorp throughput is AGR fuel from UK reactors. The constitution of overseas fuel is dominated by that delivered from Japan (see Table 15). With only minor exceptions the Rep U produced from the reprocessing of these fuels belongs to customers in the country of origin of the fuel.

TABLE 15. SPENT AGR AND LWR FUEL REPROCESSED AT SELLAFIELD, AS OF 31 MARCH 2004

| Country where the Fuel was Irradiated | Spent AGR and LWR Fuel Reprocessed as of 31 March 2004 (t HM) |
|---|---|
| UK (AGR) | 1 910 |
| UK (LWR) | 0 |
| Germany, Switzerland, Japan, and Others | 3 160 |
| **Total** | **5 070** |

### *3.2.10. United States of America*

About 625 t HM of spent fuel, including 245 t HM of LEU spent fuel, were reprocessed at the NFS West Valley facility from 1966 to 1972 [6]. Meanwhile, a reprocessing plant with a 300 t HM/year capacity was built at Morris, Illinois, but it was never commissioned. Construction of another reprocessing plant at Barnwell, South Carolina (planned capacity 1 500 t HM/year) was stopped in 1977.

## 3.3. Inventories by countries and potential disposition path

Information on the RepU inventories by country as well as on the potential future disposition paths is given in Section 4.4. An overview over the RepU derived from spent commercial light water reactor fuel (of Belgian, French, German, Japanese, Dutch, Spanish, Swedish and Swiss origin) and recycling until 2010 is given in Table 24. Furthermore, an overview over RepU derived from spent commercial non-light water reactor fuel (of French, India, Japanese, Spanish and British origin) and recycling until 2010 is given in Table 25.

Detailed information on the property and the location of the RepU which has not yet been recycled is largely withheld by the owners of the material. The same applies to the quality of the material (isotopic composition, etc.).

## 3.4. Future spent fuel arising

As summarized in the recent review [7], the total amount of spent fuel cumulatively generated worldwide by the beginning of 2003 was close to 255 000 t HM. Of these about 171 000 t HM were stored in storage facilities across the world at the beginning of 2003.

TABLE 16. SPENT FUEL STORED WORLDWIDE AS OF 1 JANUARY 2003

| Region | Stored (t HM) | Region | Stored (t HM) |
|---|---|---|---|
| Western Europe | 36 100 | America | 83 300 |
| Eastern Europe | 27 700 | Asia & Africa | 23 900 |

Meanwhile, the spent fuel generation rate, which is now at about 10 500 t HM/year, is expected to reach about 11 500 t HM/year by 2010. Projections indicate that the cumulative amount generated by the year 2010 may be close to 340 000 t HM. By 2020, the time when many of the presently operated nuclear power reactors will be close to the end of their licensed operation life time, the total quantity of spent fuel generated will be approximately 445 000 t HM.

# 4. MANAGEMENT OF REPROCESSED URANIUM

## 4.1. Management options and associated technical issues: Past, current status, and future trends

### *4.1.1. Storage*

In France, the storage duration at each step of the recycling process of RepU is kept as short as possible in order to keep the radiological impact due to the decay products of $^{232}U$ as low as reasonably achievable (ALARA principle). Meanwhile the RepU, which is not intended for immediate recycling, is converted into $U_3O_8$ and stored in drums at the Pierrelatte site.

In the United Kingdom reprocessed uranium is converted directly into the form of $UO_3$ for storage purpose. Thorp product is fed into stainless steel drums with a capacity of approximately 150 kg U which are placed into a purpose built store at Sellafield. The RepU product of Magnox reprocessing was originally fed into mild steel drums and after interim storage at Sellafield was transferred to Capenhurst for long term storage. Magnox product has more recently been fed into stainless steel drums and older drums are being over-drummed to meet current storage standards and placed into long term storage at Capenhurst.

### *4.1.2. Recycling*

#### 4.1.2.1. Accepted product specifications

Two internationally recognized standardization bodies, the International Standards Organization (ISO) [8] and ASTM International (ASTM) [2], create and maintain standards related to the nuclear industry along with several other industrial standards that contribute to making the development, manufacturing and supply of any industrial products and services more efficient, safer and cleaner. ISO, through the efforts of Technical Committee 85, provides many standard practices and procedures, particularly relating to analytical methods [8]. ASTM Committee C26 is also involved in the creation of standard practices and procedures, in some cases issued jointly with the ISO. In addition, ASTM C26 maintains standard specifications for low enriched uranium (LEU) in a variety of chemical forms. These ASTM specifications are the most widely-accepted product specifications for LEU.

The key ASTM product specifications for reprocessed uranium are:

- C787 “Standard Specification for Uranium Hexafluoride for Enrichment”; and
- C996 “Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% $^{235}U$”.

These specifications define the acceptable nuclide levels in reprocessed uranium and enriched reprocessed uranium, respectively. These nuclide levels are shown in the below mentioned Table. Although these standards specifically apply to uranium hexafluoride, the nuclide limits given in these specifications are accepted as the reference in the standard specifications for reprocessed uranium in other chemical forms.

Acceptable uranium nuclide levels in reprocessed uranium and enriched reprocessed uranium

| **Nuclide** | **ASTM C787 (Reprocessed Uranium)** | **ASTM C996 (Enriched Reprocessed Uranium)** |
|---|---|---|
| $^{232}$U | 0.005 µg/gU | 0.050 µg/gU |
| $^{234}$U | 480 µg/gU | 2000 µg/gU |
| $^{236}$U | 8400 µg/gU | |
| $^{99}$Tc | 0.500 µg/gU | 5 µg/gU |

The nuclide limits given above apply only to reprocessed material. The referenced specifications also provide nuclide limits for unirradiated uranium, and provide limits on elemental contaminants. The elemental contaminant limits apply to both unirradiated and reprocessed uranium. Different elemental limits apply depending on the chemical form of the uranium. Current specifications could be considered as technical requirements as of today. These specifications could be modified in future with possibly increasing quantities of reprocessed uranium to be processed and with new facilities and/or technologies, as well as depending on the evolution of spent fuel characteristics.

The following ASTM standards may also be of interest:

- C788 “Standard Specification for Nuclear-Grade Uranyl Nitrate Solution”;
- C1334 “Standard Specification for Uranium Oxides with a $^{235}$U Content Less of Than 5% for Dissolution Prior to Conversion to Nuclear-Grade Uranium Dioxide”;
- C1348 “Standard Specification for Blended Uranium Oxides with a $^{235}$U Content of Less Than 5% for Direct Hydrogen Reduction to Nuclear-Grade Uranium Dioxide”; and
- C753 “Standard Specification for Nuclear-Grade, Sinterable Uranium Dioxide Powder”.

In addition to ISO and ASTM specifications, national regulations and internal corporate specifications must be considered in any transaction.

#### 4.1.2.2. Direct recycling

Direct recycling of the RepU is its utilization in reactors without re-enrichment of RepU. This offers the benefit of handling uranium with lower concentrations of $^{232}$U and $^{234}$U, the two isotopes that call for major attention due to their radiological implications. Direct recycling is of greater importance to countries that have power programmes based on natural uranium as fuel. In this mode of resource utilization, RepU is used as a substitute for natural uranium (NU) or Slightly Enriched Uranium (SEU) (generally SEU referred to as a subclass of LEU that contains low $^{235}$U content namely in the range of 0.9 to 2 wt% $^{235}$U). The RepU fuel in the form of depleted uranium is used for flux flattening and in the form of SEU for increasing fuel burnup. The details of various programmes associated with direct recycling of RepU are described in the following paragraphs.

**Flux flattening in Heavy Water Reactors (HWR)**

The initial core of an HWR starting with a core composed entirely of fresh natural $UO_2$ fuel will have an unacceptably high power in the central core region, unless some means of flux flattening in the absence of differential burnup is provided. One of the means to achieve the desired flux flattening is by placing depleted uranium fuel bundles in the central region of the core. The RepU obtained by reprocessing of the spent fuel from the natural uranium-based HWRs has a depleted $^{235}U$ content (less than 0.7%). This depleted uranium has been used on a large scale for providing sufficient flux flattening in the initial fresh cores of Indian PHWRs.

**Use of RepU to Increase Fuel Burnups in HWR**

The RepU obtained from the spent LWR fuel has frequently a slightly higher fissile concentration than natural uranium. The actual $^{235}U$ content and isotopic composition of the RepU depends on several factors such as the initial enrichment of the fresh fuel and the discharge burnup of the spent fuel which are already described in the Chapter 2. In general, the enrichment level in the RepU obtained from reprocessing of LWR spent fuel is around 0.9%, which can be categorized as Slightly Enriched Uranium (SEU). Using this level of enrichment instead of natural uranium in heavy water reactors, the average fuel burnup can be increased from about 7 MWd/kg HM to about 14 MWd/kg HM.

The use of RepU derived from reprocessed spent PWR fuel in HWRs can be considered as an illustration of HWR/LWR synergy on a global basis. The major reprocessing of LWR spent fuel has been in Europe and Japan. The cumulative amount of this fuel reprocessed till the year 2000 is about 25 000 tonnes HM. This material would provide sufficient fuel for 500 CANDU 6 reactor years of operation. A detailed assessment of the use of RepU in CANDU reactors is currently underway as part of the collaborative program between Atomic Energy Canada Limited (AECL), British Nuclear Fuels plc. (BNFL), and Korea Atomic Energy Research Institute (KAERI). The various issues related to the use of RepU from spent PWR fuel in HWR like economics, reactor management, fuel design and performance experience are described in detail in elsewhere [9].

**Use of RepU in fast reactors**

RepU as depleted uranium finds application in India's fast reactor programme [10]. The RepU from PHWRs will be used as blanket material in the FBR in India. The 500 MWe FBR has both radial as well as axial blankets. The axial blanket is integrated in the fuel assemblies and measures 300 mm each at the top as well as at the bottom. The radial blanket is formed by 186 blanket assemblies, which encircles the fuel core in three rings.

#### 4.1.2.3. Physical re-enrichment (LWR, RBMK, AGR)

Since the RepU has an assay close to that of natural uranium, its use in most of the reactors (PWR, AGR, RBMK) would require that the RepU to be re-enriched to have higher $^{235}U$ content. Physical re-enrichment consists of using the same technology as the one used for enriching natural uranium. In this regard, it necessitates the conversion of reprocessed uranium into $UF_6$ prior to the enrichment process.

**Conversion**

The conversion of RepU depends on the utility's recycling strategy. If delayed, recycling is the preferred option, reprocessed uranium is stored in a stable oxide form. In the case RepU is delivered by the reprocessing plant in the form of the uranyl nitrate, a preliminary conversion into $U_3O_8$ is required for the purpose of storage of RepU. If the utility wants to recycle its reprocessed uranium immediately, the material (UNH or $UO_3$) delivered by the reprocessing plant is directly converted into $UF_6$ by employing the same conversion technology applied for natural uranium conversion.

Irrespective of the option retained by the utility, the conversion into $UF_6$ results in complementary purification of some of its chemical impurities (see Table 17). And it is of great importance before subsequent recycling activities. Decontamination of reprocessed uranium occurs in the flame reactor during the conversion of $UF_4$ into $UF_6$ and during the cooling of $UF_6$:

- Elements such as carbon, phosphorus, silicon, boron or sulfide give very volatile fluorides which are neutralized in washing columns;
- Transuranic elements and thorium give non-volatile fluorides that can be collected at the bottom of the flame reactor or filtered as dust in $UF_6$; and
- Fission products such as ruthenium, zirconium, niobium, cesium or $^{232}U$ decay products behave differently, some being easy to trap, some more difficult.

The purification efficiency of the RepU can reach 90%. Further decontamination and purification can be reached by trapping some isotopes during an additional transfer of $UF_6$ from one container to another.

TABLE 17. CONCENTRATION OF IMPURITIES IN URANYL NITRATE AND IN FRESH UF6

| | **Uranyl Nitrate** | **Fresh $UF_6$** |
|---|---|---|
| Total Transuranic Isotopes (TU) | 100 Bq/gU | < 3 Bq/gU |
| $^{232}U$ and Decay Products | 1,500 Bq/gU | ~ 15 Bq/gU |
| Fission Products | 200 Bq/gU | < 10 Bq/gU |

Conversion of UNH into $U_3O_8$ does not lead to any purification of the RepU from transuranic elements and fission products.

**Enrichment**

Presently, there are two enrichment technologies in use: Gaseous diffusion and centrifugation. Both rely on cascade systems. To enrich RepU, the centrifuge enrichment process has some advantages compared to the diffusion enrichment process, e.g.: i) in general the centrifugal method consumes 50-60 times less energy per SWU than the diffusion method; ii) a diffusion plant needs higher U inventories; and iii) it is difficult to isolate and dedicate a part of the plant for RepU enrichment in a diffusion plant. In the cascade, all uranium isotopes, except $^{238}U$, are enriched together with $^{235}U$:

- The lighter isotopes $^{232}U$ and $^{234}U$ all reach the top of the cascade and are more enriched than $^{235}U$; and

- Roughly two thirds of the $^{236}$U reach the top of the cascade and $^{236}$U is therefore slightly less enriched than $^{235}$U.

The following chart shows the isotopic composition of reprocessed uranium vs. the $^{235}$U assay in the enrichment cascade.

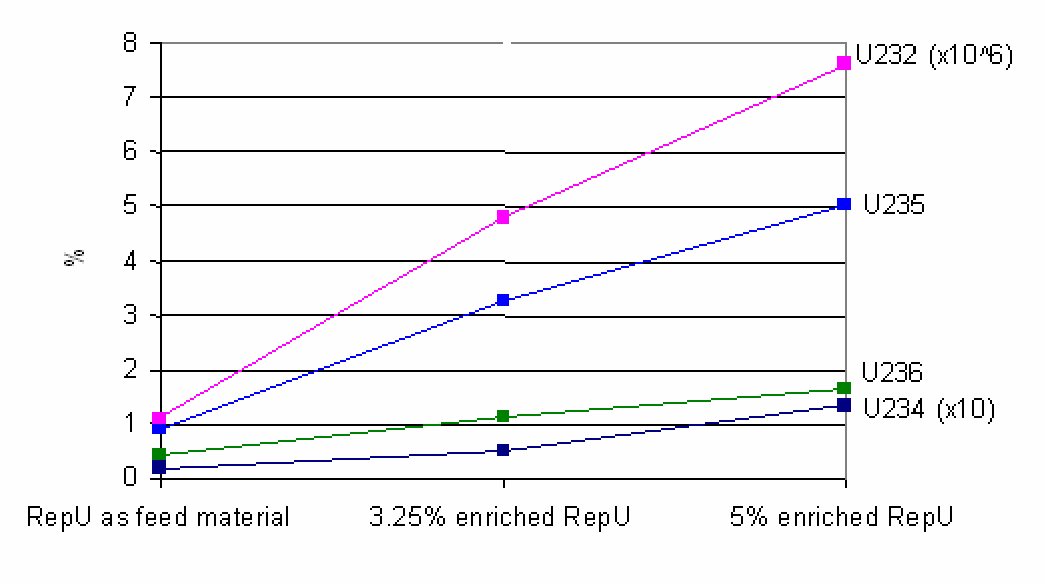


*Fig. 17. Isotopic composition of enriched RepU (ERU) vs. the required $^{235}$U assay in the enrichment cascade (the feed RepU comes from the reprocessing of spent fuel from 900 MWe PWRs with a discharge burnup of 33 000 MWd/t HM).*

The radioactivity of enriched RepU is higher than that of enriched natural U. That is why the use of re-enriched RepU usually requires dedicated processing facilities, including additional shielding to protect operators against radiation. This may also imply rigid management controls to prevent cross contamination with fuel manufacture from natural U. However, although the $^{232}$U assay is largely increased after enrichment, the operators do not encounter any difficulties with enriched reprocessed uranium (ERU), as it is quickly filled in $UF_6$ cylinders. Moreover, when the cylinders are full, the radiations emitted by $^{232}$U decay products are partially absorbed by the uranium. Nevertheless, several measures can be taken to reduce the radiation at this very step:

- Reducing time between $UF_6$ conversion and reprocessed uranium re-enrichment;
- purifying the material just before shipment, by transferring $UF_6$ from one container to another in case the $UF_6$ has been stored for a long time prior to enrichment (special filters can also be used to improve the purification factor); and
- diluting feeding heels by filling the empty feeding containers with enrichment tails that do not contain $^{232}$U nor its decay products.

The presence of the neutron absorbers $^{234}$U and $^{236}$U in the reprocessed uranium calls for a substantial additional enrichment of the fuel to compensate for this loss of neutronic reactivity during reactor irradiation. Therefore, at equal burnups, the $^{235}$U assay must be higher in ERU fuel than in Enriched Natural Uranium (ENU) fuel. This problem of over-enrichment needed for ERU fuels to reach the burnup equivalence with ENU fuel has been already briefly mentioned in the Section 2.1.2 (including the illustration Figure 7). One example quoted is for RepU enriched to 4.02 % of $^{235}$U being equivalent to 3.7 % enrichment from the natural U (see Table 18). Tables 18 and 19 provide comparisons of ERU fuel isotopic composition at different enrichments.

TABLE 18. U ISOTOPE COMPOSITION FOR RE-ENRICHED REPU AT $^{235}$U CONTENT 3.25% WT EQUIVALENT AND 3.7% WT EQUIVALENT

| RepU U Isotopes | RepU as Feed Material Origin PWR 900 MWe Burnup 33 GWd/t | ERU Enrichment Level of 3.25 % wt equivalent | ERU Enrichment Level of 3.7 % wt equivalent |
|---|---|---|---|
| $^{232}$U | 1.1 ppb | | |
| $^{234}$U | 0.021 % wt | 0.088 % wt | 0.10 % wt |
| $^{235}$U | 0.92 % wt | 3.52 % wt | 4.02 % wt |
| $^{236}$U | 0.42 % wt | 1.14 % wt | 1.31 % wt |
| $^{238}$U | 98.64 % wt | 95.25 % wt | 94.56 % wt |

(SOURCE: COGEMA/AREVA NC REPU HANDBOOK)

TABLE 19. U ISOTOPE COMPOSITION FOR RE-ENRICHED REPU AT $^{235}$U CONTENT 3.25% WT AND 5.0% WT

| RepU U Isotopes | RepU as Feed Material Origin PWR 900 MWe Burnup 33 GWd/t | ERU Enrichment Level of 3.25 % wt actual | ERU Enrichment Level of 5.0 % wt actual |
|---|---|---|---|
| $^{232}$U | 1.1 ppb | 4.8 ppb | 7.6 ppb |
| $^{234}$U | 0.021 % wt | 0.085 % wt | 0.133 % wt |
| $^{235}$U | 0.92 % wt | 3.25 % wt | 5.0 % wt |
| $^{236}$U | 0.42 % wt | 1.09 % wt | 1.55 % wt |
| $^{238}$U | 98.64 % wt | 95.25 % wt | 93.23 % wt |

(SOURCE: COGEMA/AREVA NC REPU HANDBOOK)

#### 4.1.2.4. Blending with HEU

The presence of uranium even isotopes and fission products makes the processing of RepU rather special. The existing limits for $^{232}$U, $^{234}$U and $^{236}$U as well as license restrictions for the RepU processing facilities give some reasoning to increase the enrichment of RepU by blending it with uranium of higher $^{235}$U enrichment levels, than by the centrifugal enrichment method.

Examples of the uranium isotopic composition of fuel for boiling water reactors (BWR) and pressurized water reactors (PWR) using the blended RepU / LEU materials are given below.

TABLE 20. EXAMPLES OF THE ISOTOPIC COMPOSITION OF THE FUEL FOR BWR AND PWR WITH BLENDED REPU/LEU (IN %)

| Uranium Isotope | BWR | PWR |
|---|---|---|
| $^{232}$U | $1.1 \times 10^{-7}$ | $0.75 \times 10^{-7}$ |
| $^{234}$U | 0.052 | 0.05 |
| $^{235}$U | 4.75 | 4.7 |
| $^{236}$U | 0.54 | 0.38 |

There are two blending technologies to convert RepU into ceramic grade powder with required enrichment:

In the **first** one, uranium oxides are converted by the so called “wet” process. The process consists of the following operations:

- Preliminary blending in dry environment of uranium oxide powders of RepU with uranium oxide powders with medium or high $^{235}U$ enrichment;
- dissolving in nitric acid with further filtering;
- uranyl nitrate extraction and re-extraction in a cascade of centrifugal extractors;
- precipitation of polyuranite with further filtering; drying and calcinations;
- reduction of uranium dioxide in hydrogen atmosphere with further sifting and magnet separation;
- homogenization of powders in orbital-screw blender; and
- quality control and completion of powder lots.

The **second** technology utilizes the gas-flame method. The process consists of the following operations:

- Preliminary blending in dry environment of uranium oxide powders of RepU with uranium oxide powders with low or high 235U enrichment;
- conversion into uranium hexafluoride (UF6);
- evaporation or sublimation of UF6;
- conversion into oxyfluorides in hydrogen steam atmosphere with further separation of oxyfluorides from metal-ceramic filters;
- defluorination and reduction of oxyfluorides up to dioxide in hydrogen atmosphere and water steam;
- sifting and magnet separation of dioxide;
- homogenization of powders in orbital-screw blender; and
- quality control and completion of powder lots.

RepU processed by the blending method meets the best of the customer's requirements concerning the content of even uranium isotopes from the viewpoint of reactor efficiency and physics.

### 4.1.2.5. RepU as matrix for MOX fuel

**LWR fuel**

Mixed uranium plutonium oxide (MOX) fuel is a homogeneous solid solution of $PuO_2$ and $UO_2$. Like natural uranium or depleted uranium, reprocessed uranium can be used as such matrix material to form the MOX. France's MOX fuel fabrication plant MELOX has thus been designed to use this kind of matrix. However, since reprocessed uranium is slightly more enriched than depleted uranium, MOX fuel manufactured with a reprocessed uranium matrix would use less plutonium than depleted uranium-based MOX fuel, which is not the target of the utilities today. Moreover, the use of MOX fuel in the reactor introduces a penalty since the safety margin has to take into account the uncertainty on the isotopic composition and the fissile material content. An additional penalty is introduced by the reprocessed uranium whose characterization is less detailed than that of natural uranium as well.

On the other hand, increasing burnup for UOX and MOX fuels and parity between the two fuels will be limited by the maximum plutonium content in MOX fuel (12.5%). In that case, the use of reprocessed uranium as a matrix would lead to a higher $^{235}$U content in the fuel, i.e. a higher fissile material content. Since a 0.6% $^{235}$U assay in MOX fuel is equivalent to a 0.3% assay in UOX fuel, the use of reprocessed uranium could increase the burnup by 3 000 – 4 000 MWd/t HM.

Meanwhile, if the utilities decided to use reprocessed uranium matrixes, the quantities of reprocessed uranium recycled would be relatively low.

### PHWR fuel

The RepU obtained from the processing of spent PHWR fuel has depleted fissile uranium concentration which can accommodate higher enrichments of plutonium, when used as matrix for MOX fuel. In India, studies have been carried out for the use of the (RepU – Pu) MOX fuel in 220 MWe PHWRs [11]. One of the fuel designs which has been found to be suitable contains MOX fuel with 0.8% $PuO_2$ and the reprocessed uranium having 0.25% $^{235}$U in all the 19 pins of the fuel bundle. The overall fuel requirement also reduces to about 65% of the all-natural uranium core and the natural uranium requirement reduces to about 33% of the present level.

### PFBR fuel

The 500 MWe Proto-type Fast Breeder Reactor (PFBR) fuel sub-assemblies will have RepU from spent PHWR fuel as matrix for the MOX fuel [12]. It has RepU with depleted fissile uranium mixed with about 25% plutonium content. The core of PFBR has 181 fuel subassemblies with the initial core containing about 9 tonnes of MOX fuel and requiring an annual refuelling of about 3 tonnes of MOX fuel.

#### 4.1.2.6. Belgian experience with the blending of RepU with Low Enriched Uranium (LEU)

In Belgium, first experience in the recycling of RepU in the power reactors was achieved in 1989 (see also Section 4.4.1.2). A quantity of 16.3 tonnes RepU in the form of $UF_6$ with a residual enrichment of 1.9% $^{235}$U was mixed with 27.4 tonnes U in the form of enriched natural $UF_6$ (LEU) with a product assay of 4.9% $^{235}$U. The 16.3 tonnes RepU came from spent fuel of the French 310 MWe Chooz-A PWR which was shut down in 1991. The blending of the RepU with the ENU resulted in 88 fuel assemblies or 43.7 tonnes of low enriched uranium (LEU) with an average assay of 3.85% $^{235}$U. The average isotopic composition of the RepU and of the refabricated fuel is given in Table 21.

TABLE 21. ISOTOPIC COMPOSITION OF REPU AND OF FUEL WITH BLENDED REPU/LEU IN BELGIUM

| Uranium Isotope | RepU | New Fuel Made of RepU and ENU |
|---|---|---|
| $^{232}U$ | 1.2 ppb | 0.4 ppb |
| $^{234}U$ | 220 ppm | 360 ppm |
| $^{235}U$ | 1.9% | 3.85% |
| $^{236}U$ | 4 115 ppm | 1 600 ppm |

The blended product was divided into six lots corresponding to the containers in which the material was blended. As two lots had a high and low $^{235}U$ content, respectively, it was decided to mix the $UO_2$ powder produced by conversion of these two lots.

During the conversion, the bulk of the decay products of the $^{232}U$ was trapped on the bottom of the $UF_6$ containers, thereby reducing the radiation level for the subsequent fuel fabrication steps. The fabrication process was reportedly not modified and the fabrication costs increased only slightly, compared to the process using ENU only.

At the end of 1989, the fuel fabrication campaign at FBFC's Romans facility was finalized. The transport of the fuel assemblies was allowed within the limits of the ENU fuel assembly's container licence. The campaign provided a complete annual reload for the (then) 1 001 MWe Doel-4 PWR and half an annual reload for the 900 MWe Tihange-2 PWR.

#### 4.1.2.7. Japanese experience with the blending of RepU

Japan has the following experience with the direct blending of RepU with natural uranium and with the blending of RepU with plutonium:

**Direct blending of RepU with natural uranium**

About 3 tonnes of RepU regained by the reprocessing of spent fuel from the Japan Power Demonstration Reactor (JPDR) at the Tokai Reprocessing Facility in the period 1977-1981 was mixed with natural uranium and used as fuel for the Advanced Thermal Reactor FUGEN.

**Blending of RepU with plutonium**

A portion of of the RepU regained at the Tokai Reprocessing Plant in the form of uranyl nitrate was converted by the microwave de-nitration method developed by JNC (now renamed as JAEA), and it was used from 1982-1988 as a fuel for FUGEN. (Note: Pu-U mixture conversion is employed from the viewpoint of increased proliferation-resistance.)

## 4.2. Reactor management issues

### *4.2.1. Experience with enriched RepU Fuel in LWRs*

For both Western PWRs and BWRs, high performance fuel assemblies have been accomplished, based on the recycling of reprocessed uranium (ERU) fuel assemblies.

An outline of selected utilities' experience with enriched RepU (ERU) in LWRs gained until mid-2004 is given in Table 22 and 23.

Figure 18 shows the thermal power released per initial tonne HM of the spent fuels viz MOX; RepU- and ENU-based spent fuels during cooling of the spent fuel.

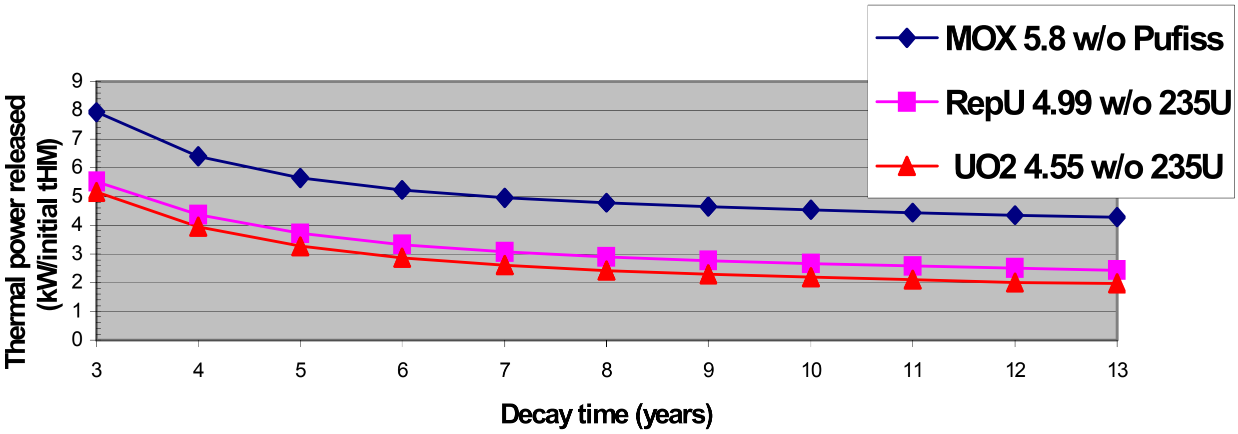


*Fig. 18. Heat load for different spent oxide fuels as a function of the cooling time. (Source: NOK, May 2004).*

TABLE 22. EXPERIENCE WITH ENRICHED REPU FUEL IN LWRS IN SELECTED COUNTRIES

| Item | Utility | | | |
|---|---|---|---|---|
| | EdF | EPZ | Kansai | NOK |
| | France | Netherlands | Japan | Switzerland |
| • The same burn-up was achieved with re-enriched RepU fuel as with enriched natural uranium fuel. | Yes | No1) | Yes | Yes |
| • A special authorisation/license was required before loading re-enriched RepU fuel in reactor cores, due to the increase of the U-235 content in the fuel. | Yes1) | No2) | Yes1) | No |
| • The same fuel transport casks for re-enriched RepU fuel and enriched natural uranium fuel were used. | Yes | Yes | Yes | Yes |
| • The same reactor core management configuration for re-enriched RepU fuel and enriched natural uranium fuel with standard equipment were used. No operation restrictions were in use. | Yes2) | Yes-No | Yes | Yes |
| • The "neutronic" computer codes were qualified and gave the same accuracy for re-enriched RepU fuel cores and enriched natural uranium fuel cores. | Yes | Yes3) | Yes | 1) |
| • Reactivity control was adjusted using boron content in reactor primary circuit. | No | Yes4) | Yes | No |
| • Additional control rods were used (similar to reactors using MOX fuel). | Yes | No | No | No |
| • No difference of reactor effluents production (gaseous, liquid or solid) was found. | No | Yes | No2) | Yes |
| • No special arrangements were performed for the handling and storage of irradiated re-enriched RepU fuel. | Yes | Yes | No3) | Yes |
| • Interim storage and transport of spent RepU fuel did not pose any specific problems, because it is not much different from spent enriched natural uranium fuel. | Yes | Yes | Yes | No |
| • Spent RepU fuel had no significantly different characteristics than other advanced fuel, like spent MOX fuel or high burn-up fuel. | Yes | Yes5) | Yes | Yes |

**Electricité de France (EdF), France:**

(1) This is true in France for any new type of fuel or any new core management configuration

(2) Additional control rods required (see below)

**EPZ, Netherlands:**

(1) Based on reactor physics calculations, a lower burnup is to be expected. Loading was in 2003 and unloading is not expected before 2006.

(2) U-236 was compensated by larger reload size, since plant license does not permit excess U-235 enrichment.

(3) Framatome ANP did the qualified calculations. They could demonstrate the quality (Reference: Previous projects)

(4) As usual

(5) Expected, not yet confirmed

**Kansai Electric Power Co., Japan:**

(1) At some reactor cores, Kansai increased the enrichment levels and obtained an additional license

(2) No reactor effluents production was found

(3) No special arrangement were performed

**Nordostschweizersiche Kraftwerke AG (NOK) Switzerland:**

(1) The correct isotopic composition was used for neutronic calculations. The codes accurately predict neutronic performance, but were not subject to a specific qualification for that performance.

TABLE 23. EXPERIENCE WITH ENRICHED REPU FUEL IN LWRS IN SELECTED COUNTRIES

| **Item** | **Utility** | | | |
|---|---|---|---|---|
| | OKG | RWE | Synatom | TEPCo |
| | Sweden | Germany | Belgium | Japan |
| • The same burn-up was achieved with re-enriched RepU fuel as with enriched natural uranium fuel. | Yes | No[1)] | Yes | Yes[1)] |
| • A special authorisation/license was required before loading re-enriched RepU fuel in reactor cores, due to the increase of the U-235 content in the fuel. | No | Yes | No[1)] | Yes[2)] |
| • The same fuel transport casks for re-enriched RepU fuel and enriched natural uranium fuel were used. | Yes | Yes | - | Yes[3)] |
| • The same reactor core management configuration for re-enriched RepU fuel and enriched natural uranium fuel with standard equipment were used. No operation restrictions were in use. | Yes | Yes | - | Yes |
| • The "neutronic" computer codes were qualified and gave the same accuracy for re-enriched RepU fuel cores and enriched natural uranium fuel cores. | Yes | Yes | - | Yes |
| • Reactivity control was adjusted using boron content in reactor primary circuit. | N.A. | No[2)] | - | No[4)] |
| • Additional control rods were used (similar to reactors using MOX fuel). | N.A. | No | No | No[5)] |
| • No difference of reactor effluents production (gaseous, liquid or solid) was found. | Yes | Yes | - | No |
| • Re-enriched RepU fuel showed similar heat release and gamma emissions, compared to enriched natural uranium fuel. | Yes | - [3)] | - | Yes[6)] |
| • No special arrangements were performed for the handling and storage of irradiated re-enriched RepU fuel. | [1)] | Yes | - | No |
| • Interim storage and transport of spent RepU fuel did not pose any specific problems, because it is not much different from spent enriched natural uranium fuel. | [2)] | - [3)] | - | Yes |
| • Spent RepU fuel had no significantly different characteristics than other advanced fuel, like spent MOX fuel or high burn-up fuel. | Yes | - [3)] | No[2)] | Yes |

**OKG Aktiebolag, Sweden:**

(1) Not proven yet, the fuel still in core

(2) Not proven yet, the fuel still in core. But so far no signs of difference.

N.A. = not available

**RWE Power AG, Germany:**

(1) Up to now only two cycles of experience, but expected

(2) Usage in BWR

(3) Up to now only two cycles of experience, but expected

**Electrabel/Synatom, Belgium:**

(1) This is considered a minor modification by the authorities in Belgium

(2) Beware of isotopic content though…It cannot be recycled through reenrichment, save maybe with lasers, because of too high U-236 content.

**Tokyo Electric Power (TEPCo), Japan:**

(1) The same burn up was achieved with 8x8 BJ RepU fuel assemblies as with 8x8 BJ uranium fuel assemblies

(2) A special license was required in case of the increase of the U-235 average enrichment for reactivity compensation

(3) But the classifications of re-enriched RepU fuel transport cask and enriched natural uranium fuel transport cask were different

(4) Reactivity control was adjusted using the insertion depth of control rods (Usage in BWR)

(5) No additional control rods were used by comparison with natural uranium core reactor

(6) RepU is not much different from natural uranium fuel in terms of heat release and gamma emissions

### 4.2.2. *Selected reactor management issues for heavy water reactors*

The recycling of RepU in a heavy water reactor (HWR) calls for negligible / very low concentrations of $^{232}U$ and $^{234}U$ in the RepU. The radiological consideration associated with these two uranium isotopes, therefore, does not have much bearing on the direct recycling of RepU. The various fuel cycle activities like fabrication, transport, storage, and other related operations do not require any modification from those being practiced for natural uranium [13].

In HWRs, on-power re-fuelling provides flexibility in fuel management. However, in-depth reactor physics studies related to fuelling schemes operating with a transition core and with a mixed core are required to be carried out. The reactor management approach for the various cases of recycling of RepU in HWR is described below [13].

#### 4.2.2.1. PHWR flux flattening

The typical fuel loading scheme for using the RepU bundles with the depleted fissile uranium content for the 220 MWe PHWR is shown in Figure 19. The guiding principles in deciding the core configuration and selection of the re-fuelling strategy are:

- attainment of early full power and equilibrium core conditions;
- maximum discharge burnup; and
- operating within the fuel bundle and channel power limits.

The 220 MWe PHWRs in India attains the equilibrium core configuration in about 600 equivalent full power days (EFPDs), with refuelling operation beginning after about 120 EFPDs. The selection of channels and the number of bundles to be refuelled are decided on the basis of maximum fuel utilization with bundle power limited to its operating limit of 462 kW.

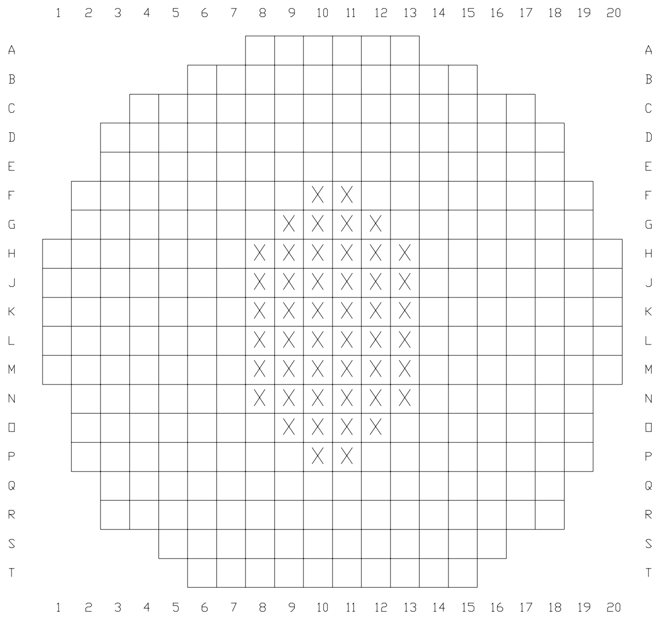


*Fig. 19. Initial core configuration of Narora Power Plant (NAPP); where x denotes –RepU bundles; and unmarked positions denotes natural uranium.*

#### 4.2.2.2. RepU from spent Light Water Reactor fuel (SEU) in CANDU reactors

The RepU obtained from spent LWR fuel has a $^{235}U$ between 0.5% and 1.0%. Fuel management studies have been carried out for the utilization of this fuel in CANDU reactors. These studies have been carried out for reactor cores equipped exclusively with SEU bundles and reactor cores with a mix of natural uranium and SEU bundles. For all the SEU fuel core configuration, the fuel burnup is nearly doubled. A two or four bundle shift, bi-directional fuelling scheme results in a flatter/superior axial bundle power distribution. The studies on all SEU fuel cores until today have mainly focussed on the equilibrium core. For the mixed natural uranium/SEU core, studies have been carried out for both equilibrium and transition core configurations. The mixed core offers the advantage of using SEU for flattening the power profiles in the core.

#### 4.2.2.3. MOX fuel using RepU in PHWRs

Reactor physics optimisation studies have been carried out for the loading of MOX fuel using RepU in Indian 220 MWe PHWRs. The loading scheme was evolved based on limiting parameters like maximum bundle power, maximum channel power, and maximum coolant outlet temperatures. The most suitable core configuration found to consists of 116 central channels loaded with natural uranium bundles and the remaining 190 channels with the MOX fuel. The reactivity-worths[1] of the shut down devices and regulating devices were found to be comparable to that of the all-natural uranium core.

### *4.2.3. Reactor management issues for RBMKs*

Operational experience of RepU fuel has shown that there are no additional significant technical or safety problems with this fuel in comparison with the natural uranium-based fuel [14].

### *4.2.4. RepU from British Magnox reactors recycled into British AGRs*

Around 17 000 tonnes U of $UO_3$ derived from the reprocessing of irradiated Magnox fuel has been converted to $UF_6$, re-enriched and recycled as AGR reactor fuel in the UK. The low burnup of Magnox fuel gives rise to very low levels of $^{236}U$ and $^{232}U$ in RepU. Consequently, even in re-enriched RepU the levels of $^{236}U$ and $^{232}U$ remain relatively low. These low levels of $^{236}U$ and $^{232}U$ resulted in no distinction being made between the process for fabrication of AGR fuel derived from natural uranium and that fabricated using re-enriched ex-Magnox RepU. The only significant process route modifications were related to precautionary measures to deal with potentially higher levels of ruthenium in waste streams and the potential build up of technetium in plant items.

The lack of distinction between RepU- and natural uranium-derived material made during fabrication resulted in no distinction being made between the two in terms of fuel handling and in anticipated performance in the AGR reactor. Today, of course, worker dose uptake limits are much lower and the very small incremental dose arising from the increased levels of $^{232}U$ compared to natural U would have to be taken account of in the fuel route management strategy.

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[1] Reactivity worth: The effect that a parameter or component has on the reactivity of a reactor.

#### *4.2.5. Conclusion*

There is little to differentiate between the in-reactor performance of the re-enriched RepU (ERU) fuel in comparison to the enriched natural uranium (ENU) fuel. The important reactor core parameters, such as power peaking, reactivity feedback coefficients, control rod worth and transient response are not altered to any significant extent, and in many cases no modification to the reactor control and safety systems are necessary.

However, individual experts are referring to the following drawbacks:

- Due to a higher $^{235}U$ enrichment requirement for ERU than in the equivalent natural uranium fuel (ENU), some safety margins may be reduced and difficulties may arise, requiring some modifications [15].

- In re-enriched RepU (ERU) fuel, the equivalent of 5% enriched uranium cannot be reached, due to the $^{236}U$ content. Therefore, very high burnups cannot be attained in full ERU equilibrium reactor cores [16].

- The increased decay heat of irradiated ERU fuel requires consideration in the planning of interim dry storage and final disposal. These considerations could lead to extra costs [16].

Nevertheless, the accumulated experience demonstrates that there are no insurmountable technical drawbacks in using enriched reprocessed uranium in existing reactors.

### 4.3. RepU management facilities by country

#### *4.3.1. Germany*

The German company Reaktor-Brennelement Union (RBU) has manufactured fuel elements with pure reprocessed uranium. The fuel fabrication plant was operated by Siemens from 1965 to 1995.

Advanced Nuclear Fuels (ANF) manufactures ERU fuel at its Lingen fuel fabrication plant. A specific license to produce 50 tonnes U/year was awarded by the German safety authorities in 2003. This licence is limited to fuel rod manufacturing using ERU pellets fabricated at OAO MSZ, Russia from downblended fissile material. The manufacturing process is the same as the one implemented for ENU fuel. No specific treatment is used for ERU scraps.

#### *4.3.2. France*

##### 4.3.2.1. TU2 and TU5 Facilities, Pierrelatte

Two de-nitrification facilities, TU2 and TU5, are located at the AREVA NC plant in Pierrelatte. At TU2, uranyl nitrate hexahydrate (UNH) can be converted either into $U_3O_8$ for further storage or into $UO_2$ for further use as matrix for MOX fuel.

**Process**
At TU2, UNH is first diluted, then precipitated as ammonium diuranate (ADU) by ammonia. After filtration, ADU paste is dried and atomized. $U_3O_8$ is finally produced by calcination in a rotary furnace. In case the required end-product is $UO_2$, $U_3O_8$ is reduced in a rotary furnace.

At TU5, UNH is first precipitated by hydrogen peroxide. After washing and filtration, $UO_3$ is calcinated in a rotary furnace to produce $U_3O_8$.

Since reprocessed uranium is not currently used as a matrix for MOX fuel, the RepU product produced at TU2 and TU5 facilities is eventually stored in form of $U_3O_8$ in steel drums.

**Management of scraps and other process residues**

At TU5, filtration residues are concentrated and hydrogen peroxide is eliminated. Solutions are then treated as follows: Nitric acid is obtained by distillation, while uranium is denitrified and recycled in the technological process. At TU2, ammonia is obtained from the residual $NH_4NO_3$.

**Radiation protection**

Reprocessed uranium batches are surrounded with radiation protection shields made of depleted $U_3O_8$ that absorb most of the $\gamma$-radiations emitted by $^{232}U$ decay products.

**End products**

| **Chemical Form** | $U_3O_8$ | $UO_2$ |
|---|---|---|
| **Physical Form** | Powder | Powder |
| **Characteristics** | Suitable for direct fluorination | Flowable, sinterable |

**Licensed capacity**

The nominal processing capacity for $U_3O_8$ production of the TU2 and TU5 facilities are 1 200 tU/year and 1 700 tU/year respectively.

**Technical specifications**

TU2 and TU5 can process feed products with $^{235}U$ assays up to 1.2% and 1%, respectively. The isotopic composition and the impurities content of $U_3O_8$ product are similar to the ASTM specification for UNH. Besides, some physical characteristics are required for safe storage:

- U content $> 80\%$ wt;
- Moisture $< 0.5\%$; and
- Particles diameter $< 1$mm.

**Production record**

The TU2 facility was commissioned in 1986. The TU5 facility was commissioned in 1996. Overall more than 14 000 tU of UNH have been converted into $U_3O_8$ until mid-2004.

#### 4.3.2.2. Structures 2000 and 300, Pierrelatte

A special facility dedicated to the conversion of reprocessed uranium into $UF_6$ is located at the COMURHEX conversion plant. The so-called “INB 105” facility is actually composed of two workshops, Structure 2000 and Structure 300.

**Process**

At the Structure 2000 workshop, UNH is first precipitated as ammonium diuranate (ADU) by ammonia. And it is recovered by filtration and then dried and calcinated. Oxide is then reduced into $UO_2$ by a countercurrent of $H_2$ in a rotary furnace. $UO_2$ is converted into $UF_4$ by a countercurrent of HF in two rotary furnaces. $UF_4$ is loaded into tanks for intermediate storage before conversion into $UF_6$ at Structure 300.

At the Structure 300 workshop, $UF_4$ is crushed and then fluorinated into $UF_6$ by a cocurrent of $F_2$ in a flame reactor. $UF_6$ is crystallized in primary cold traps. Un-burnt residues are recovered by gravity and filtration and then recycled. The excess of $F_2$ and remaining traces of $UF_6$ are scrubbed by potassium hydroxide (KOH).

After heating, $UF_6$ is transferred into an intermediate storage tank. $UF_6$ is homogenized and then filled in 48Y cylinders. $UF_6$ can be stored in cylinders pending its shipment to the enrichment plant.

**Management of scraps and other process residues**
All the scraps resulting from the operation of the COMURHEX Pierrelatte facility are recycled in a technological process.

All un-burnt impurities collected during the conversion of $UF_4$ into $UF_6$ are stored on site under special conditions until the $\gamma$-activity diminishes to acceptable levels, before starting recycling on site.

Solid waste is shipped to ANDRA's Centre de l'Aube repository, while process wastes are treated at the AREVA NC Pierrelatte facility.

**Radiation protection**
In the process building, static barriers are combined with ventilation to ensure proper confinement of the products. In order to reduce the individual dose of workers, process equipment is equipped with radiation protection shields and workers wear lead aprons.

**End product**

| | |
|---|---|
| **Chemical Form** | $UF_6$ |
| **Physical Form** | Solid |
| **Isotopic Composition** | See ASTM C787 |

**Licensed capacity**
The plant's licensed capacity is 330 tonnes/year.

**Technical specifications**
Under the plant's safety requirements, the maximum $^{235}U$ assay is 2.5% in the Structure 2000 workshop and 1% in Structure 300. However, processing of materials with $^{235}U$ assay over 2.25% requires a specific authorization from the French safety authorities. Besides, the plant's discharges are regulated by specific authorizations. Reprocessed $UF_6$ produced by COMURHEX meets the ASTM C787 specification.

**Production record**
INB-105 was commissioned in 1972. It has converted about 3 600 tU of reprocessed uranium until mid-2004.

### 4.3.2.3. TE Workshop, Pierrelatte

TE is a $UF_6$ sampling and transfer workshop at the AREVA NC plant in Pierrelatte which can process both natural grade $UF_6$ and RepU grade $UF_6$. Different operations can be carried out including:

- $UF_6$ transfer in liquid phase;
- $UF_6$ transfer in gas phase;
- $UF_6$ partition into several containers;
- isotopic adjustment to $^{235}U$ assay specification;
- gas or liquid sampling; and
- pumping of $UF_6$ container's heel.

For this purpose, several process lines are operated. After each campaign the lines and collector pipes are carefully cleaned.

In the reprocessed uranium recycling process, the TE workshop is mainly used to reduce the radiological impact of $^{232}U$ decay products at the fuel fabrication stage. Transfer of $UF_6$ in the gas phase is performed just before the shipment of re-enriched $UF_6$ from Pierrelatte to the fuel manufacturing plant.

**Process**
A contaminated 30B cylinder is placed in a closed hot-cell equipped with an electric heater and heated up to 100°C. Gaseous $UF_6$ is then transferred through a heated pipe into a clean 30B container.

**Radiation protection**
On full cylinders γ-doses are around 100 to 200 µGy/h (micro Gray/hour), while on empty cylinders values over 3 000 µGy/h can be found for low points (viz., bottom portion of the cylinders). Such γ-doses are well above the authorized limit of 2 mSv/h (milli Sievert/hour) for transport containers. Such containers are stored on specific areas with a calibrated distance between cylinders until the γ-activity has reached acceptable levels. They are then sent to the AMC workshop where they are washed and cleaned (see the Section 4.3.2.5).

### 4.3.2.4. FBFC, Romans

**Process**
At the FBFC plant in Romans, the $UO_2$ powder is produced by dry route conversion. The ERU fuel manufacturing process is the same as the one implemented for ENU fuel. However, the ERU fuel is fabricated on lines with the special radioprotection features, as explained below, which are carefully cleaned between two fabrication campaigns.

**Management of scraps and other process residues**
No specific treatment is used for ERU scraps and effluents; the same purification treatment used for the scraps and effluents arising out of ENU fuel processing is applied for ERU scraps and effluents. All process scraps are recycled in the production process. Burnable residues are incinerated. The resulting ashes can be recycled together with the scraps. The liquid effluents due to ERU are stored for one to two months before release in order to obtain a decrease of radioactivity due to the residual impurities such as $^{232}U$ decay products. The resulting

products are either incinerated or cemented and then shipped to ANDRA's Centre de l'Aube repository.

**Radiation protection**

Usually, feed material is purified at the AREVA NC's TE workshop in Pierrelatte prior to shipment to the FBFC plant. Second purification occurs when $UF_6$ is vaporized and unloaded from the cylinders at the FBFC plant. Meanwhile the storage of $UF_6$ is kept as short as possible in order to limit the activity of $^{232}U$ decay products. Within two months, the enriched $UF_6$ is converted into $UO_2$. Moreover, the fabrication schedule is kept as short as possible in order to limit the external dose rates.

In the process building, static barriers are combined with ventilation to ensure proper confinement of the products. Besides, fabrication lines are equipped with radiation protection shields, some process operations were automated and automatic control devices were implemented to reduce as much as possible the individual dose of workers operating the lines. Process rooms are also equipped with Air Sampling Devices and Continuous Atmospheric Contamination devices.

As far as storage is concerned, ERU fuel rods are stored as high as possible in order to increase the distance with the workers. ERU fuel assemblies are stored directly in shipment containers.

**End product**

The FBFC plant can manufacture different fuel designs (14x14, 17x17…), with different $^{235}U$ assays up to 5%.

**Licensed capacity**

The plant's licensed capacity is 150 tonnes U/year for the powder workshop and the fuel assembly fabrication workshop.

**Technical specifications**

It is required that the isotopic composition of ERU not vary too much from one batch to the other. Different batches of $UO_2$ can therefore be mixed to homogenize the isotopic compositions. Besides, under a new decree dated July 2000, the FBFC plant must control the uranium, fission products and transuranic elements content in both gaseous and liquid discharges.

**Production record**

As of end of 2002, a total of about 350 t HM of ERU fuel has been fabricated at FBFC since the plant was commissioned in 1993. As of today, FBFC provides two of the French Cruas NPP reactors with ERU reloads.

#### 4.3.2.5. Plans for new facilities

The AREVA Group signed an agreement with the Urenco shareholders on 24 November 2003, under which it will acquire a 50% equity interest in the Enrichment Technology Company (ETC). In October 2004, the European Commission approved AREVA's acquiring a stake in URENCO's subsidiary ETC. AREVA will thus have the necessary means and access to the technology to launch the project to construct its future uranium enrichment plant. This plant, to be named Georges Besse II, will be built at the Tricastin site. Production capacity will be increased gradually starting in 2007 and reach its nominal level around 2016.

AREVA may therefore decide to dedicate part of the production capacity to the enrichment of reprocessed uranium.

#### *4.3.3. Japan*

JNC (now JNC is renamed after merging with JAERI as JAEA) operated a conversion facility and an enrichment facility in Ningyo-toge which were shut down in 1999 and 2001, respectively. About 350 tonnes of reprocessed uranium from the Tokai Reprocessing Plant was used at Ningyo-toge for the re-enrichment examination.

**Fuel fabrication facilities**

ERU fuel can be fabricated at four fabrication plants: The Mitsubishi Nuclear Fuel Company's (MNF) facility in Tokai Mura (Ibaraki Prefecture), the Global Nuclear Fuel Japan (GNF-J) facility in Yokosuka (Kanagawa Prefecture), and the Nuclear Fuel Industries, Ltd.'s (NFI) facilities in Kumatori (Osaka Prefecture) and Tokai Mura (Okayama Prefecture). However, conversion of $UF_6$ into $UO_2$ is performed only at the MNF facility.

**Process**

At the MNF facility, the $UO_2$ powder is produced by wet conversion. The ERU fuel manufacturing process at all facilities is same as the one implemented for ENU fuel.

**Radiation protection**

Reprocessed uranium was handled without special radiation protection measures in the fabrication process.

**End product**

ERU fuel assemblies are the final products.

**Licensed capacity**

| Fuel Vendor | Plant | Fuel Design | Capacity |
|---|---|---|---|
| MNF | Tokai Mura | PWR | ~22 tU/year |
| NFI | Kumatori | PWR | ~30 tU/year |
| NFI | Tokai Mura | BWR | ~22 tU/year |
| GNF-J | Yokosuka | BWR | ~25 tU/year |

**Technical specifications**

The typical specifications for handling reprocessed uranium at the fuel fabrication step are as follows:

| Isotope | Specification/Concentration |
|---|---|
| $^{232}U$ | < 10 ppb U |
| U(α) | < 3.30E+05 Bq/gU |
| $^{237}Np$ | < 1.00E-01 Bq/gU |
| Pu(α) | < 1.00E-01 Bq/gU |
| Pu(β) | < 3 Bq/gU |
| $^{99}Tc$ | < 10 Bq/gU |
| $^{106}Ru$ | < 10 Bq/gU |
| $^{125}Sb$ | < 2 Bq/gU |

**Production record**
As of December 31, 2004, about 62 tU of PWR ERU fuel and 5 tU of BWR ERU fuel have been manufactured in Japan, using about 340 tU of reprocessed uranium.

### *4.3.4 Kazakhstan*

The Ulba Metallurgical Plant (UMP) produces fuel pellets for RBMK reactors. UMP receives reprocessed uranium in the form of $U_3O_8$ from Russia's MAYAK combine and send the end product to the OAO Mashinostroitelny Zavod (OAO MSZ) plant, also in Russia (see Section 4.3.6.3).

**Process**
At UMP, $UO_2$ powder is produced by the wet conversion. The ERU fuel manufacturing process is the same as the one implemented for the ENU fuel.

**Radiation protection**
No special measures are implemented for the ERU fuel campaigns.

**End product**
$UO_2$ pellets for RBMK reactors are the final products.

**Licensed capacity**
The plant's licensed capacity is 800 tonnes U/y.

**Technical specifications**
$UO_2$ pellets delivered to OAO MSZ are manufactured to the following specification:

| Uranium Isotope | Limit |
|---|---|
| $^{232}U$ | 1.2 ppb U |
| $^{234}U$ | - |
| $^{235}U$ | 2.65-2.87% |
| $^{236}U$ | 0.42% |

**Production record**
UMP manufactures annually 150 to 200 tU of ERU pellets for RBMK reactors.

### *4.3.5. Netherlands*

Urenco has utilised part of the existing capacity at its Almelo plant to provide RepU re-enrichment services on a contractual basis to utilities.

**Process**
The ultracentrifugation process is followed at Almelo Plant for the enrichment of RepU.

**Management of scraps and other process residues**
Reprocessed uranium enrichment and associated residues handling is permitted in the Almelo Site Licence.

**Radiation protection**
At Almelo, shielding of RepU cylinders and remote handling of cylinders have been deployed to reduce operator doses. RepU cylinder storage areas also have additional shield walls.

**End product**

| **Chemical Form** | $UF_6$ |
|---|---|
| **Physical Form** | Solid |
| **Isotopic Composition** | As per ASTM C996<br>$^{235}U < 5\%$ |

**Licensed capacity**
The effective capacity of the RepU unit is slightly over 300 000 SWU/year. However, this size was chosen in light of the expected RepU feed arisings. Urenco can expand this capacity to any reasonable given demand if sufficient lead-time is provided.

**Technical specifications**
Urenco processes will accept feed material for enrichment which is within RepU feed limits defined in ASTM specification C787 so that the enriched RepU complies with the requirements of ASTM specification C996.

**Production record**
Since 1992, Urenco has handled 1 000 tU of ex-oxide reprocessed material at Almelo.

### *4.3.6. Russian Federation*

Russian RepU management facilities are located at two sites. The Siberian Chemical Combine (SCC) in Seversk has industrial facilities for the radiological purification and the conversion of uranyl nitrate into $UF_6$ and re-enrichment of $UF_6$. The OAO Mashinostroitelny Zavod (OAO MSZ) in Elektrostal (see Section 4.3.6.3) can manufacture ERU fuel for both reactor types viz., RBMK and LWR.

#### 4.3.6.1. Conversion Facility, Siberian Chemical Combine (SCC), Seversk

**Process**
The fluorination process is employed for the production of $UF_6$.

**Radiation protection**
No special measures are taken at the production steps.

**End product**
$UF_6$

**Licensed capacity**
The plant's licensed capacity is 500 tU/year.

**Technical specifications**
The conversion facility can handle reprocessed uranium either in the form of solution in nitric acid, or oxides or uranium tetrafluoride.

**Production record**
n.a.

### 4.3.6.2. Enrichment Facility, Siberian Chemical Combine (SCC), Seversk

**Process**
The ultracentrifugation in a dedicated cascade is the process followed for enrichment of RepU.

**Radiation protection**
No special measures are implemented for the radiological protection.

**End product**
The final product is in the form of enriched $UF_6$.

**Licensed capacity**
The plant's licensed capacity is 3 000 000 SWU/year.

**Technical specifications**
Re-enriched $UF_6$ delivered by SCC to OAO MSZ meets the following specifications:

| Uranium Isotope | Limit |
|---|---|
| $^{232}U$ | 1.9 ppb |
| $^{234}U$ | 0.04% |
| $^{235}U$ | 2.65% |
| $^{236}U$ | 0.4% |

**Production record**
On request by AREVA NC, the Siberian Chemical Combine successfully implemented in the 1990s direct enrichment of reprocessed uranium by the traditional route. The resulting enriched $UF_6$ had the following composition:

| Uranium Isotope | Limit |
|---|---|
| $^{232}U$ | $< 10^{-6}\%$ |
| $^{235}U$ | < 4.95% |
| $^{236}U$ | < 1.4% |

### 4.3.6.3. OAO Mashinostroitelny Zavod (OAO MSZ), Elektrostal

**Process**
At MSZ, the $UO_2$ powder is produced by wet conversion. The ERU fuel manufacturing process is the same as the one implemented for ENU fuel. However, fabrication of the ERU fuel assemblies for LWRs is carried out on dedicated lines to take care of the specifications of the fuel design.

**Radiation protection**
No special measures are implemented for the ERU fuel campaigns.

**End product**
OAO MSZ manufactures ERU fuel assemblies for RBMK and VVER reactors. In addition, it manufactures ERU fuel assemblies for Western LWRs under an agreement with Siemens / Framatome ANP (AREVA NP).

**Licensed capacity**
The plant's licensed capacity is 1 600 tU/year for the fuel assembly production.

**Technical specifications**

The technical specifications of ERU fuel manufactured by OAO MSZ for different reactors are given below:

| Reactor Type | Isotopic Composition | | | |
|---|---|---|---|---|
| | $^{232}$U | $^{234}$U | $^{235}$U | $^{236}$U |
| RBMK | 1.9 ppb | - | 3% | 1% |
| VVER | 1.7 ppb | - | 5% | 1% |
| LWR | 5 ppb | 0.1% | 5% | 0.84% |

The RepU quantities delivered to Russia under the agreement between Siemens / Framatome ANP and OAO MSZ in the period 1998 – 2005 are given in Figure 20.

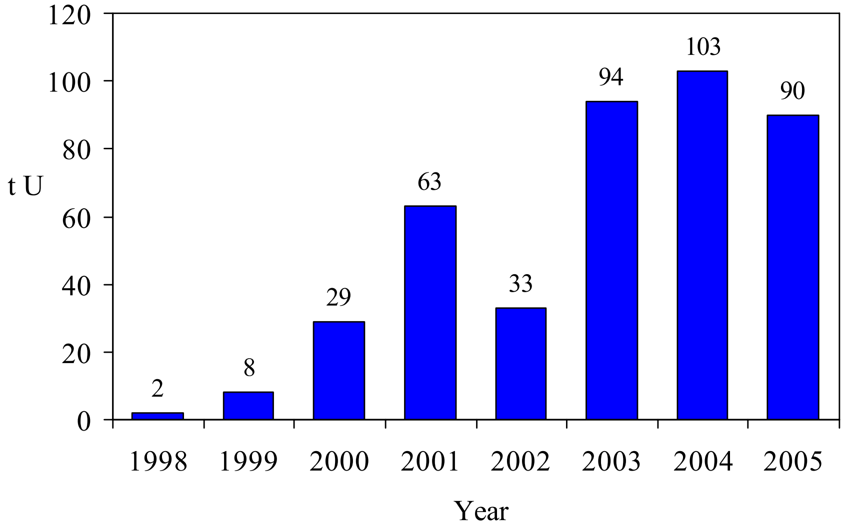

*Fig. 20. The amount of RepU delivered to Russian under the Agreement between Siemens / Framatome ANP (AREVA NP) and OAO MSZ, 1998 – 2005.*

MSZ manufactures annually about 200 tU of ERU fuel for Russian RBMK reactors. The plant's production record for LWR fuel is shown in Figure 21.

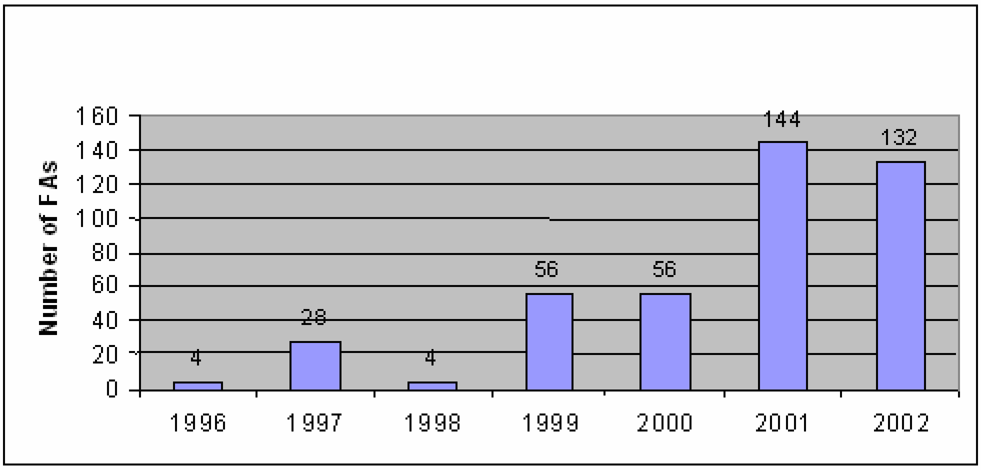

*Fig. 21. RepU Fuel Manufactured under the Agreement between Siemens / Framatome ANP and OAO MSZ as of 31 December 2003 [17].*

#### 4.3.6.4. Plans for new facilities

In order to reduce the $^{232}$U content in low enriched uranium hexafluoride, the Siberian Chemical Combine carried out research works on the possibility to purify RepU from the isotope $^{232}$U, utilizing the gas centrifuge isotope separation technology. Evaluations showed the principle possibility of such purification. This facility would consist of a main cascade and a purification cascade. Sampling from the main cascade would go to the purification cascade, and tails of the purification cascade would be the final product i.e., enriched RepU with low contents of $^{232}$U.

### *4.3.7. United Kingdom*

The Springfields process (conversion, fabrication) that was used to recycle Magnox reprocessed uranium between 1970 and 1982 has been decommissioned. BNFL undertook construction of a Line 3 Hex Plant (originally known as ECHO) in the late 1990s, but the project was abandoned in early 2003. Meanwhile 3 000 tU of ex-Magnox reprocessed material have been re-enriched by Urenco at its Capenhurst plant. Reprocessed uranium could be handled there in the future if current safety cases are appropriately addressed, submitted to and approved by the regulator (UK Nuclear Installations Inspectorate).

### *4.3.8. United States of America*

Since the 1960s, the US Department of Energy (USDOE) had included the acceptance of reprocessed uranium in all of its contracts, and substantial quantities of reprocessed uranium had been returned to USDOE as feed material. Some of this has actually been fed to the Paducah diffusion plant, the remainder being held in inventory.

## 4.4. Recycling programs by country

There are different approaches for nuclear fuel cycle back-end policies, such as:

- Reprocessing of spent nuclear fuel and the recycling of the separated materials (plutonium and reprocessed uranium) into uranium/plutonium mixed oxide fuel and ERU fuel;
- storage of spent fuel over periods of 30–50 years in wet and dry intermediate storage facilities and subsequent disposal as high level waste (HLW) in geologic repositories (‘once-through fuel cycle’); and
- deferral of a final decision on whether to choose reprocessing or final disposal of the spent fuel.

In a number of countries, different approaches are overlapping or existing in parallel, depending on the individual IAEA Member States’ back-end fuel cycle management strategies and the evolvement over time of these strategies. As very long time periods for interim storage of spent fuel is being contemplated by several Member States, the distinction between the deferral and once through concepts are becoming less distinctive.

Historically (until year-end 2003), about 22 250 tonnes RepU were derived from spent fuel of commercial light water reactors (LWR) in Belgium, France, Germany, Japan, Netherlands, Spain, Sweden, and Switzerland (see Table 24). According to this Table, the bulk of this quantity (about 43%) is belonging to Electricité de France (EdF). Japanese (27%) and German utilities (23%) are placed second and third in terms of RepU received from the reprocessing of spent LWR fuel. Of the total of about 22 250 tonnes RepU, until year-end 2003 roughly 6 850 tonnes (or 31%) had been recycled into NPPs, while another about 8 825 tonnes RepU (or 40%) were still available for recycling.

Between year-end 2003 and year-end 2010 another 7 096 tonnes RepU are expected to arise from the reprocessing of spent LWR fuel. The bulk of this quantity (about 85%) will come from French spent PWR fuel.

TABLE 24. REPROCESSED URANIUM (REPU) DERIVED FROM SPENT COMMERCIAL LIGHT WATER REACTOR FUEL AND RECYCLING (UNTIL 2010) (IN TONNES U)

| **Country** | **Utility/Company** | **Total RepU gained from Reprocessing (until 31 Dec. 2003)** | **RepU Recycled into NPPs (until 31 Dec. 2003) (incl. Swapped and Purchased Material)** | **RepU Still Available from Recycling (incl. Swapped and Purchased Material)** | **RepU Expected to Arise under Existing of Firmly Planned Reprocessing Contracts (until 31 Dec. 2010)** | **RepU Expected to be Recycled during the Period (1 Jan. 2004-31. Dec. 2010)** |
|---|---|---|---|---|---|---|
| Belgium | Electrabel | 643 | 525 | - | - | - |
| France | EdF | 9 600 | 2 900 | 6 700 | 6 000 | 400 – 2 800 |
| Germany | E.ON, EnBW, HEW and RWE | 5 078 | 2 696 | 1 911 | 646 | 2 467 |
| Japan | Various | 6 060 | 335 | n.a. | n.a. | undecided |
| Netherlands | EPZ | 270 | 270 | n.a. | 60 | 60 |
| Spain | Various | 0 | 0 | 0 | ~97 | n.a. |
| Sweden | OKG | 34 | 20 | 14 | 0 | 0 |
| Switzerland | Various | 565 | 365 | 200 | 293 | 328 |
| **Total** | | **22 250** | **7 111** | **8 825** | **7 096** | **3 255 – 5 655** |

n.a. = not available

Furthermore, historically (until year-end 2003), about 53 000 – 55 000 tonnes RepU were derived from spent commercial non-light water reactor fuel processing (see below Table 25). The bulk of this quantity is belonging to British companies (previously primarily to British Nuclear Fuels plc. (BNFL), now to the Nuclear Decommissioning Authority (NDA) accounting for roughly two thirds of the total.

Between year-end 2003 and year-end 2010, another 15 000 tonnes RepU are expected to arise from the reprocessing of spent non-LWR fuel in Japan and the United Kingdom. Most of this total will come from the reprocessing of BNFL's Magnox reactors.

Hereinafter the recycling programs are described for only those countries which have been, or which are still engaged in the reprocessing of spent fuel from power reactors. There is experience in the handling and treatment of RepU also in other countries. However, in most cases such experience was gained from laboratory-scale reprocessing of spent fuel and treatment of RepU and was not yet directed at, and was not part of industrial-scale recycling of RepU. The detailed discussion on recycling programmes of Belgium [18], France [19], Germany [20], [21], India [13], Japan [4, 22], The Netherlands [23], Russia [14], Spain [24], Sweden [25] and Switzerland [26] is given below.

TABLE 25. REPROCESSED URANIUM (REPU) DERIVED FROM SPENT COMMERCIAL NON-LIGHT WATER REACTOR FUEL AND RECYCLING (UNTIL 2010) (IN TONNES U)

| Country | Total RepU gained from Reprocessing (until 31 Dec. 2003) | RepU Recycled into NPPs (until 31 Dec. 2003) (incl. Swapped and Purchased Material) | RepU Still Available from Recycling (incl. Swapped and Purchased Material) | RepU Expected to Arise under Existing of Firmly Planned Reprocessing Contracts (until 31 Dec. 2010) | RepU Expected to be Recycled during the Period (1 Jan. 2004-31. Dec. 2010) |
|---|---|---|---|---|---|
| France | 18 260[1] | n.a. | n.a. | n.a. | n.a. |
| India | n.a. | n.a. | n.a. | n.a. | n.a. |
| Japan | 1 529 | - | - | 146 | 146 |
| Spain | 1 840 | 0 | 0 | 0 | 0 |
| UK | ≃35 000[2] | >16 000 | ≃20 000[2] | ≃15 000[3] | n.a. |
| **Total** | **53 000-55 000** | | | | |

[1] Historically reprocessed (from domestic and foreign reactors), [2] Ex-AGR material (from domestic and foreign reactors),
[3] Expected to arise until 2012

### *4.4.1. Belgium*

#### 4.4.1.1. Evolvement of reprocessing and recycling policies

Previously, Belgium's spent fuel management was based on the reprocessing of spent fuel and on the earliest possible recycling of the regained plutonium and reprocessed uranium.

The fuel cycle company Synatom, being responsible for nuclear fuel procurement and for taking care of spent fuel management (including transport, storage, and reprocessing of spent fuel), signed in 1976 and 1978 its first three reprocessing contracts with COGEMA (now AREVA NC), covering discharged fuel from the Doel-1 and -2 PWRs as well as from the Tihange-1 PWR. In total, 140 tonnes of spent fuel were reprocessed in the French UP2 facility at La Hague, and the plutonium was used in the European fast breeder reactor programs. The regained RepU was recycled to the US Department of Energy (USDOE) (for Belgian NPPs) which at that time still accepted RepU as feed material under contracts with its enrichment customers (see below) [18].

In 1978, and in full agreement with both the Belgian government and the country's parliament, a fourth reprocessing contract was signed with COGEMA for the reprocessing of 530 tonnes of spent fuel in the new French UP3 facility. This quantity represented most of the spent fuel discharged from Doel-1, -2, and Tihange-1 in the period 1979–1990. Later on, another optional contract was signed for reprocessing of spent fuel from Tihange-1, covering a quantity of 225 tonnes heavy metal (HM).

Belgium's nuclear back-end policy was discussed in parliament throughout 1992 and 1993. Finally, in December 1993, the government decided to let Synatom honour the existing baseload contract between Synatom and COGEMA's UP3 facility. The decision in favour of this contract was expected as the contract was binding and included no escape clauses. But the government imposed a five-year moratorium on the execution of the optional contract signed for reprocessing of spent fuel from Tihange-1. Later, in 1998, the reprocessing of Belgian irradiated fuel was abandoned at the request of the government. Eventually, the optional contract signed for reprocessing of spent fuel from Tihange-1 has never been executed.

#### 4.4.1.2. Belgian experience with the recycling of RepU

In Belgium, first experience in the recycling of RepU in power reactors was achieved in 1989. A quantity of 16.3 tonnes RepU in the form of $UF_6$ with a residual enrichment of 1.9% $^{235}U$ was mixed with 27.4 tonnes U in the form of enriched natural $UF_6$ with a product assay of 4.9% $^{235}U$. The 16.3 tonnes RepU came from spent fuel of the French 310 MWe Chooz-A PWR which was shut down in 1991. The blending of the RepU with the ENU resulted in 88 fuel assemblies or 43.7 tonnes of low enriched uranium (LEU) with an average assay of 3.85% $^{235}U$.

At the end of 1989, the fuel fabrication campaign at FBFC's Romans facility was finalized. The transport of the fuel assemblies was allowed within the limits of the ENU fuel assemblies container licence. The campaign provided a complete annual reload for the (then) 1 001 MWe Doel-4 PWR and half an annual reload for the 900 MWe Tihange-2 PWR.

Recycling of RepU (average $^{235}U$ assay of 0.8-0.9%) by re-enrichment in dedicated cascades of Urenco's enrichment facilities started in the fall of 1994 for the Doel-1 PWR. The trial batch of FRAGEMA AFA design (4.25% $^{235}U$) was successfully loaded, after limited licensing efforts. Thereafter the Doel-1 PWR has been fuelled continuously with re-enriched RepU fuel assemblies until 2003 when the last re-enriched RepU fuel reload was loaded.

#### 4.4.1.3. Statistics

Historically, re-enriched RepU fuel assemblies containing 525 tonnes RepU (out of 643 tonnes RepU received under reprocessing contracts) were loaded into the Belgian NPPs. Synatom have no longer RepU available for recycling (see Table 25).

### *4.4.2. China*

China has primarily established a nuclear fuel cycle system covering – among others – reprocessing of spent fuel. However, currently spent fuel assemblies from Chinese power reactors are temporarily stored at on-site pools. They will be transported to spent fuel interim storage sites for “concentrated” storage and be reprocessed in due course.

A pilot reprocessing plant for spent power reactor fuel is currently under construction at Lanzhou. It will have a final annual capacity of about 100 tonnes HM. China is expected to gain experience in the recycling of RepU on a commercial scale only after the completion of this facility. This will be later in of this decade at the earliest.

### *4.4.3. France*

#### 4.4.3.1. Evolvement of reprocessing and recycling policies

The Ministry for Economy, Finance, and Industry (Ministere de l'Economie, des Finances et de l'Industrie, MINEFI) supervises energy policy in France. Production of energy from nuclear power in France reached 420.6 TWh in 2003. This nuclear electricity production means that about 1 150 tonnes of spent fuel is unloaded each year.

Like a number of other countries, France has decided in favour of the reprocessing and recycling option for its spent fuel. Following the strategic decisions taken in the 1970s, a consistent, high-technology industrial tool for each stage in the nuclear fuel cycle has to be instituted. This arrangement comprises a fuel reprocessing complex at La Hague with a total annual capacity of 1 700 tonnes HM (this figure varies according to the burnup fraction of the fuels involved). In addition, the French nuclear power plant fleet comprises a total of 58 standardised reactors. This number includes 20 NPP units operating with MOX fuel produced by recycling, and another eight whose design enables this fuel to be used after minor operational modifications. Furthermore, two reactor units are fuelled with RepU (see below). This choice on loading MOX and RepU fuel was made primarily for energy and environmental reasons.

To avoid accumulating quantities of separated plutonium for which there would be no use, the fuel is reprocessed as and when utilization for the extracted plutonium arises ("equal flow" principle). This leads today to reprocess annually about 850 tonnes of fuel, out of the 1 150 tonnes unloaded from the reactors in France, and enables the plutonium being recycled in the form of about 100 tonnes of MOX fuel per year. Hence, the reprocessed fuel has to be stored in cooling pools.

The "equal flow" policy involved long duration interim storage of "excess" spent fuel, i.e. about 200-250 tonnes HM per year. In particular, Electricité de France (EdF) gets reprocessed preferentially the "conventional" uranium oxide spent fuel.

### 4.4.3.2. French experience with the recycling of RepU

Historically, about 18 260 tonnes of non-LWR fuel — mainly from gas-graphite reactors (GGR) - were reprocessed at Marcoule (UP1 facility) and La Hague (UP2 facility) (see Table 25). 8 700 tU of non-LWR RepU was stored in France at year-end 2003.

The technical and economical aspects of recycling RepU have been studied by a working group set up in 1984 by EdF and the main French fuel cycle companies, namely COGEMA, COMURHEX, and FRAGEMA/FBFC. The mandate of the working group was to draw general conclusions on an industrial scale recycling of RepU by taking into account the following considerations:

The RepU had to be chosen among irradiated fuel assemblies whose compositions were representative for the next ten years and whose expected mean values of their characteristics match with respect to an initial enrichment assay of 3.10% $U^{235}$, minimum burnup of 33 000 MWd/t HM and three years cooling period.

Moreover, the RepU had to present the worst uranium isotopic composition in terms of potential neutronic and radiation impacts to make a conservative estimate. Initially, the RepU lot chosen for the program contained 0.832% $^{235}U$, 0.375% $^{236}U$, 0.016% $^{234}U$, and 1.38 ppb $^{232}U$.

In order to validate the results of previous studies which stated that there were no major technical difficulties in using RepU at various steps of the nuclear fuel cycle, the working group established in April 1985 to study a small scale experiment whose purpose was to confirm the technical feasibility of RepU recycling in PWRs.

Two types of fuel assemblies (Type 1 and Type 2 (see below)) were manufactured in FBFC's plant in Romans in southern France, both being of the 17 x 17 type for EdF's 900 MWe PWRs. The RepU for all these assemblies was reconverted in 1985 by COMURHEX in Pierrelatte and re-enriched by Urenco in 1986. The re-enriched product contained 6.3 ppb

$^{232}U$, 0.081% $^{234}U$, 3.567% $^{235}U$, and 1.191% $^{236}U$. This fuel was equivalent to ENU fuel with 3.25% $^{235}U$.

Four fuel assemblies represented Type 1 and comprised 100% RepU. These fuel elements had a combined weight of about 1.85 tonnes U, enriched to about 3.567% $^{235}U$ each, and were representative of RepU from centrifugation.

The four fuel elements of Type 2 fuel were representative of RepU from gaseous diffusion plants. About 300 kg of RepU, enriched to 3.567% $^{235}U$, were blended with LEU (natural uranium enriched to 3.25% $^{235}U$). The 300 kg RepU covered approximately 16.4 % of the total weight of the four assemblies.

The eight demonstration fuel assemblies were loaded in April 1987 in the 880 MWe Cruas-4 PWR. They were unloaded after 3 cycles of operation at the beginning of 1991, with final burnup levels between 30 000 and 40 000 MWd/tHM. After successful completion of this test, EdF decided in 1991 to carry out a new experiment on a larger scale and thus placed the necessary orders to recycle several hundred tonnes of RepU in the years thereafter.

In 1994, EdF started to load re-enriched RepU fuel on an industrial scale in the 915 MWe Cruas-3 PWR. In total 24 RepU assemblies were loaded. In 1995, Cruas-3 and the 915 MWe Cruas-4 PWR were loaded with re-enriched RepU fuel. In 1996, three Cruas NPP units (Cruas-2, -3, and -4) received reloads containing re-enriched RepU fuel.

Since 1997, Cruas-3 and -4 are regularly loaded with re-enriched RepU fuel. Both reactors are licensed to have a 100% RepU core, with a 1/4 core reload per cycle of 12 months. Accordingly, since 1997 between 150 and 400 tonnes RepU are recycled each year.

Most of the RepU recycled by EdF into its Cruas PWRs has been physically re-enriched under contracts with Urenco. However, in the years 1997-1998, under a contract between COGEMA and Techsnabexport (TENEX), RepU was re-enriched also at the Siberian Chemical Combine at Seversk (Tomsk-7), Russia.

#### 4.4.3.3. Statistics

Until December 2003, about 9 600 tonnes of RepU arised from the reprocessing of spent French PWR reactor fuel (see Table 24). Thereof about 2 900 tonnes were already recycled into French PWRs, mainly Cruas-3 and -4. Around 6 700 t RepU were converted into stable compounds and are awaiting potential later use.

Until year-end 2010, another 6 000 tonnes of RepU are currently expected to arise from the reprocessing of French spent PWR fuel. Thereof between 400 and 2 800 t RepU depending on operator's choice are expected to be recycled into French NPPs.

### *4.4.4. Germany*

#### 4.4.4.1. Evolvement of reprocessing and recycling policies

Germany's spent fuel management policy has undergone a number of changes. Until 1994, the Atomic Energy Act (Atomgesetz, AtG) included the requirement of reusing the fissile material in the spent fuel assemblies. This requirement changed with the amendment of the AtG in 1994, according to which the NPP operators then had the option of either reuse by means of reprocessing, or else opt for interim storage pending direct spent fuel disposal after completion of an appropriate final disposal facility.

Until 30 June 2005, delivery of spent fuel for the purposes of reprocessing will be prohibited in accordance with the new "Act on the Structured Phase-out of the Utilization of Nuclear Energy for the Commercial Generation of Electricity" (which was enforced in spring 2002), and only direct disposal of the spent fuel assemblies then existing in Germany will be possible. Until that date, the quantities of spent fuel assemblies contractually agreed with the reprocessing facilities abroad (UP3 at La Hague (France) and Thorp at Sellafield (UK)) must have been taken to the said facilities by the NPP operators. There are no legal stipulations as to when the spent fuel assemblies shipped from German NPPs to reprocessing facilities abroad will be reprocessed and/or until when reprocessing activities have to be finished.

For those spent fuel assemblies disposed of by way of reprocessing, the proof of reuse of the separated plutonium during reprocessing in the form of MOX fuel must be kept. This is designed to ensure that all plutonium in separated form can be precluded from misuse throughout the remaining residual terms. However, no such proof is requested for the RepU arising from the reprocessing of German spent fuel.

The German federal government proceeds on the assumption that all spent nuclear fuel delivered to France (La Hague) and the United Kingdom (Sellafield) until 30 June 2005 will be reprocessed. As there is as yet no final repository available for high radioactive waste in Germany, those spent fuel assemblies not being disposed of by way of reprocessing will be stored intermediately at the site where they were irradiated until the decision on repository construction is taken, in order to minimize the transportation of spent fuel.

The spent fuel is stored dry in casks (for exceptions at Obrigheim NPP). The spent fuel assemblies from decommissioned Russian-design reactors (VVER) in the German new federal states (former German Democratic Republic, GDR) are likewise to be stored dry in casks at a central storage facility in Greifswald (German federal state of Mecklenburg-West Pomerania). Additionally, there are two central storage facilities already available at Ahaus and Gorleben, respectively, which are operational and on stand-by. Usually, the spent fuel assemblies from research reactors will be returned to their country of origin for disposal. If that is not possible, these elements too will be intermediately stored until their final transportation to the repository. The German federal government is aiming to establish a repository in deep geological formations for the disposal of all kinds of nuclear waste by the year 2030.

#### 4.4.4.2. Evolvement of reprocessing and recycling practices

Since the mid-1970s, the German nuclear utilities have committed reprocessing services with COGEMA/AREVA NC and BNFL, totalling about 8 650 t HM. A total of 5 532 t HM was committed under so-called "Old Reprocessing Contracts", while 3 010 tonnes were covered under "New Reprocessing Contracts".

The "Old Reprocessing Contracts" include the following:

- Fixed quantity, fixed price reprocessing services with COGEMA's former UP2 facility totalling 1 541 tonnes HM. These services include quantities which were transferred from UP3 to UP2;

- Fixed quantity, “cost-plus-fee” reprocessing services with COGEMA’s UP3 facility totalling 3 111 tonnes HM (“UP3 Basedload Contracts”); and

- Fixed quantity reprocessing services with BNFL’s Thorp facility, totalling 884 tonnes HM (“Thorp Baseload Contracts”).

However, the “Old Reprocessing Contracts” do not include about 100 tonnes HM which were reprocessed in COGEMA’s UP2 facility under early contracts between German utilities and the France’s Commissariat a l’Energie Atomique (CEA) or United Reprocessors GmbH (URG). These early contracts were later taken over by COGEMA.

After the cancellation of the German Wackersdorf reprocessing plant project (WAW) in 1989, the German utilities concluded additional reprocessing contracts with COGEMA and BNFL These “New Reprocessing Contracts” (also referred to as “Post-Baseload Contracts”) include cancellation clauses and initially covered spent fuel discharges of a total of 3 010 tonnes HM through 2005. There were further options for spent fuel discharges until 2015. However, by cancellations and contract adaptations the total quantity currently contracted was reduced to 1 429  t HM or less than 50% of the previous quantity.

Out of the German utilities’ Baseload and Post-Baseload Contracts with COGEMA and BNFL covering 5 424 t HM a total of 3 153 t HM (or 59%) had already been reprocessed until year-end 2002 (see below Table 26). However, according to industry experts, actual spent fuel deliveries to La Hague and Sellafield may remain slightly behind contractual volumes, due to technical reasons (non-availability of appropriate transport containers) and other reasons.

#### 4.4.4.3. German experience with the recycling of RepU

While already in 1971 and 1972 a total of 34 MOX fuel elements with fuel rods containing RepU as matrix material were loaded into the 15 MWe BWR Kahl (now decommissioned), a demonstration project for the direct recycling of RepU was launched only about ten years later, in 1982. Through the then German reprocessing company Deutsche Gesellschaft für Wideraufarbeitung von Kernbrennstoffen mbH (DWK), the German utilities participated in the project. The fuel element fabricator Reaktor-Brennelemente Union (RBU) (later renamed Siemens Brennelementewerk Hanau, SBH) and the power plant vendor Kraftwerk Union (KWU) (later renamed Siemens/KWU) participated, too. The program comprised the testing of RepU fuel elements in two reactors. The program demonstrated the technical feasibility of recycling of RepU. In both cases, the reprocessed UNH was converted into $UF_6$ by COMURHEX and then re-enriched by Urenco. Fuel element fabrication was done by RBU at Hanau.

Under the program, the first German RepU fuel element, a so-called “Siemens/KWU lead test assembly”, was manufactured in 1982/83. The burnup of the corresponding spent fuel discharged from the 340 MWe Obrigheim PWR (KWO) was below 20 000 MWd/t HM, and the re-fabricated RepU fuel element fabricated thereof was loaded into KWO in October 1983 for four reactor cycles. The RepU recycled in the 810 MWe Neckarwestheim-1 PWR (GKN-1) was based on spent fuel with a burnup of about 33 000 MWd/t HM. In 1986, four lead test assemblies were loaded into the reactor and additional four followed in 1987 (see Table 26). Each of the two reloads remained in the reactor core for four reactor cycles. The isotopic composition of the RepU lead test assemblies is given below.

TABLE 26. ISOTOPIC COMPOSITIONS OF GERMAN REPU LEAD TEST ASSEMBLIES (WEIGHT PERCENT)

| Reactor | Year of Loading | $^{232}U$ | $^{234}U$ | $^{235}U$ | $^{236}U$ | $^{238}U$ |
|---|---|---|---|---|---|---|
| KWO | 1983 | 0.3E-6 | 0.07 | 3.50 | 0.85 | 95.58 |
| GKN-1 | 1986 | 0.5E-6 | 0.07 | 3.76 | 1.03 | 95.14 |
| GKN-1 | 1987 | 0.55E-6 | 0.06 | 3.72 | 0.81 | 95.41 |

After the final shutdown of the Siemens fuel fabrication plant at Hanau, Siemens started cooperation in 1994 with the Russian company Open-Type Joint-Stock Company Mashinostroitelny Zavod (OAO MSZ), now a subsidiary of the Joint Stock Company (JSC) TVEL, for the fabrication of RepU fuel assemblies. The first test rods from this cooperation were inserted in the Obrigheim PWR in 1995, followed by four test assemblies in 1997.

Currently, OAO MSZ (in cooperation with AREVA NP, previously Siemens, later Framatome ANP GmbH) is the only fuel fabricator in the world offering a commercial scale blending operation for RepU with enriched uranium and the fabrication of power reactor fuel elements out of this fuel mix. The blending and the subsequent fuel fabrication are done at OAO MSZ at Elektrostal, Moscow.

The milestones in the history and the results of cooperation between OAO MSZ and Framatome ANP GmbH/Siemens concerning the processing of RepU are as follows:

- 1995: Qualification of OAO MSZ to be the manufacturer of PWR fuel pellets and fuel rods under Siemens specifications;
- 1995: Fabrication of pilot PWR fuel rods;
- 1996: Qualification of OAO MSZ to be the manufacturer of PWR fuel assemblies under Siemens specifications;
- 1996: Fabrication of pilot PWR fuel assemblies;
- 2000: Qualification of OAO MSZ to be the manufacturer of fuel pellets with Gadolinium (Gd), BWR fuel rods, and BWR fuel assemblies under Siemens specifications; and
- 2000: Fabrication of pilot BWR fuel assemblies.

Concerning JSC TVEL's and AREVA NP's short- and medium-term cooperation in the field of re-enriching RepU, the RepU quantities from European utilities (mainly from German, Swiss, Swedish and Dutch utilities) available for processing at Elektrostal are currently expected to provide for the extension of the cooperation at its current level until about 2008.

All RepU-LEU produced by OAO MSZ from Western European spent fuel and currently being retained by OAO MSZ has to be recycled into RBMKs. This situation will prevail for the next 5-6 years, particularly, as prior to 2010 the Russian VVER-1000 reactors are not expected to be loaded with RepU-LEU fuel for licensing reasons.

Today, the German nuclear utilities have somewhat different RepU recycling strategies:

- RWE Power AG’s strategy is to recycle RepU as soon as ERU fuel fabrication and loading capacities are available (‘contemporary recycling’);

- Large-scale recycling of RepU fuel manufactured by OAO MSZ is carried out in the Gundremmengen-B BWR, jointly owned by RWE Power AG (75%) and E.ON Kernkraft GmbH (25%);

- RepU arising under E.ON Kernkraft GmbH’s reprocessing contracts will be recycled in the form of ERU fuel fabricated by OAO MSZ and loaded in E.ON’s Unterweser PWR and Brokdorf PWR; and

- Gemeinschaftskernkraftwerk Neckar GmbH (GKN) and Kernkraftwerk Obrigheim GmbH (KWO) both owned by the utility Energie Baden-Wuerttemberg AG (EnBW) decided to switch over from the re-enrichment of RepU at Urenco (in the 1980s) to the blending of RepU with enriched uranium at OAO MSZ. Fuel elements fabricated there were regularly loaded into the Neckarwestheim-2 PWR (GKN-2) and the Obrigheim PWR (KWO). After the final shutdown of KWO loading of such fuel elements in GKN-2 is still going on.

#### 4.4.4.4. Statistics

Until December 2003, 5 078 tonnes of RepU has produced from the reprocessing of German spent commercial light water reactor fuel (see Table 25). Thereof about 2 696 tonnes were already recycled into German LWRs. The remaining RepU was converted into stable compounds and is awaiting potential later use.

Between year-end 2003 and year-end 2010, another about 646 tonnes RepU are currently expected to arise from the reprocessing of German spent LWR fuel. About 2 467 tonnes RepU are expected to be recycled into the domestic NPPs.

### *4.4.5. India*

#### 4.4.5.1. Evolvement of reprocessing and recycling policies

India has a somewhat unique situation in terms of nuclear fuel resources. It has modest uranium reserves, but abundant thorium sources. This makes the adoption of the closed nuclear fuel cycle policy a necessity as well as a superior option, given the fact that the closed fuel cycle with reprocessing and the recycling of uranium and plutonium allows better utilization of the country’s limited uranium resources.

A three-stage power programme has been formulated to utilise effectively the limited uranium and the large thorium resources:

- The first stage involves utilization of natural uranium in pressurized heavy water reactors (PHWRs);
- The second stage involves the fast reactors, which will utilise plutonium obtained from the reprocessing of spent fuel from PHWRs; and
- The third stage involves the reactor system based on the $^{233}$Th/U fuel cycle.

In the context of India’s nuclear programme, fast breeder reactors (FBR) are the ideal platform for the recycling of reprocessed uranium and plutonium. The recycling of RepU has also been implemented in the PHWRs for initial core flux flattening. This has been aided by two factors: firstly, it does not call for any significant changes in the thermal reactor fuel cycle facilities, and, secondly, it will not affect the recycling in fast reactors in future.

In India, various agencies of the Department of Atomic Energy (DAE) are responsible for all the activities associated with nuclear power. Nuclear Power Corporation of India Limited (NPCIL) is responsible for the construction and operation of nuclear power plants. In 2004, 12 PHWRs and 2 BWRs with a total capacity of 2 720 MWe were under operation and constituted about 2.7% of the country's total electricity generation.

The share of nuclear power in India will be increased to more than 10% in the coming few years by the planned addition of both thermal and fast reactors. At present, the reprocessing of the spent fuel from PHWRs is being carried out in Prefre-1 at Tarapur and KARP at Kalpakkam. The reprocessing capacities will be increased in the next few years to meet the recycling requirements. Natural uranium and low enriched uranium (LEU) fuel assemblies required for the thermal reactors (PHWRs and BWRs) are fabricated at the Nuclear Fuel Complex (NFC). NFC also fabricates reprocessed uranium fuel bundles used in PHWRs and the reprocessed uranium blanket assemblies for the fast reactors. The MOX fuel for the fast reactors, which will use reprocessed uranium as matrix will be fabricated at the Advanced Fuel Fabrication Facility (AFFF), at Tarapur. The fabrication of MOX fuel for the initial core of the prototype fast breeder reactor (PFBR) is presently being carried out in this facility.

#### 4.4.5.2. Indian experience with the recycling of RepU

Reprocessed uranium has been used on an industrial scale for providing sufficient flux flattening in the initial fresh cores of PHWRs. Table 27 below gives the details of the various reprocessed uranium fuel loading campaigns carried out in PHWRs.

TABLE 27. REPROCESSED URANIUM LOADED IN VARIOUS PHWRS

| Year of Criticality | Reactor | Number of Fuel Bundles | Quantity of Uranium Oxide (tonnes U) |
|---|---|---|---|
| 1980 | RAPS-2 | 656 | 9.8 |
| 1983 | MAPS-1 | 656 | 9.8 |
| 1985 | MAPS-2 | 384 | 5.7 |
| 1989 | NAPS-1 | 384 | 5.7 |
| 1991 | NAPS-2 | 384 | 5.7 |

Recently, in 2003, about 1 600 reprocessed uranium fuel bundles (about 20 tonnes U) were used for the initial core of MAPS-2 after "enmasse retubing", meaning that during a planned long shutdown all the coolant channels are replaced by new ones. This involves the discharge of all the fuel in the core. After the replacement of all the coolant channels, the core is loaded with new fuel bundles.

At least several of the PHWR units of 220 MWe each and the PHWR units of 540 MWe each currently under construction will also be utilising the reprocessed uranium fuel for their initial fresh core for the purpose of flux flattening.

### 4.4.5.3. Statistics

The official policy of the Indian Department of Atomic Energy (DAE) is not to publish the operational statistics of reprocessing plants. Thus, there are no official data concerning RepU quantities arising from the reprocessing of Indian spent fuel (see Table 25). However, given the semi-industrial size of the Indian reprocessing facilities it is reasonable to assume that the RepU quantities produced so far would be in the order of several hundred tonnes.

## *4.4.6. Japan*

### 4.4.6.1. Evolvement of reprocessing and recycling policies

The Japanese energy policy has been driven by considerations of energy supply security and the need to minimize dependence on fuel imports. This is why Japan's nuclear policy since 1956 has been to maximise the utilization of imported uranium. Until now, the reprocessing of spent fuel has been largely undertaken in Europe by BNFL and AREVA NC, with vitrified high-level wastes being returned to Japan for disposal. This reprocessing will finish in a few years, and full-scale operation of Japan Nuclear Fuel Limited's (JNFL) reprocessing plant at Rokkasho-mura is scheduled to start soon thereafter. Since 1998, spent fuel has been stored at Rokkasho-mura in anticipation of operation of the new reprocessing facility. Spent fuel shipments to Europe finished in 2001.

Plutonium recovered by reprocessing in the United Kingdom and France will be used mainly in LWRs as mixed-oxide (MOX) fuel. The related "pluthermal" program was initiated in 1994. There is no such dedicated program concerning the recycling of RepU.

The total contracted quantity of reprocessing services for spent fuel from the Japanese NPPs is in the range of 8 100 tonnes HM. Of this quantity, about 1 500 tonnes HM were under contract with BNFL for the 166 MWe GGR Tokai-1. A total of about 5 600 tonnes HM has been contracted primarily under baseload reprocessing commitments with BNFL (about 2 700 tonnes HM) and AREVA NC (about 2 900 tonnes HM). The 5 600 tonnes HM include a small portion of Japanese spent fuel which was already reprocessed in the UP2 plant in France. The balance of 1 000 tonnes HM was contracted with Power Reactor & Nuclear Fuel Development Corporation (PNC) (subsequently renamed as JNC and now it is part of JAEA) for reprocessing in the Tokai Reprocessing Plant.

### 4.4.6.2. Japanese experience with the recycling of RepU

Japan gained experience in different RepU handling and treatment technologies, such as the blending of the RepU with LEU of enriched natural uranium, the mixing of RepU with plutonium to produce MOX fuel elements, and the physical re-enrichment of RepU in dedicated centrifuge cascades. The latter technology seems to be the one which has so far been given greatest attention.

In 1989, PNC's prototype centrifuge enrichment plant started operation at Ningyo-toge. This plant had a capacity of about 200 000 SWU/year. Two utilities, i.e. Kansai Electric Power Co. (KANSAI) and Tokyo Electric Power Co. (TEPCO), together with PNC, carried out a demonstration test of re-enriching RepU in 1989. During this demonstration test, a total of 40 tonnes of RepU was re-enriched. Then the re-enriched RepU was fabricated into fuel elements and assemblies. And it was loaded into one of KANSAI's Ohi PWRs and into one of TEPCO's Fukushima-daini 1 BWRs.

The remaining RepU produced at Tokai Reprocessing Plant was converted into $UO_3$. Most of this material is still stored there. However, a portion of this material was converted into $UF_6$ and was re-enriched. It is stored at Ningyo-toge.

KANSAI negotiated a re-enrichment contract with Urenco. A total of 150 tonnes RepU ($^{235}U$ assay of 0.8–1%) was transported from the French company UREP as feed material to Urenco's Almelo enrichment plant in the Netherlands. The re-enrichment campaign resulted in about 25 tonnes of re-enriched RepU. The fuel fabricated thereof in Japan by Mitsubishi Nuclear Fuel Company/Nuclear Fuel Industries, Ltd. (MNF/NFI) was loaded into KANSAI's 826 MWe Mihama-3 PWR in 1995.

Other than the delivery of RepU from France via Urenco to Kansai Electric Power Co., no news has been reported about the delivery of RepU back from European reprocessing facilities to Japan.

While the Japanese government's policy is still to encourage the full use of Pu and RepU, the government respects discretion of each utility in scheduling for its recycling of RepU. As the Japanese government does not instruct the logistics of the recycling of RepU, each utility will move according to its own policy.

In view of the problems over MOX fuel usage in Japanese reactors, the usage of re-enriched RepU is not a topical issue at all.

#### 4.4.6.3. Statistics

Until end of March 2004, about 1 529 tonnes of RepU arose from the reprocessing of Japanese non-light water reactor fuel (see Tables 25 and 28). Until year-end 2010, another small quantity of RepU is currently expected to arise from the reprocessing of Japanese non-light water reactor fuel.

Furthermore, until March 2004, about 6 060 tonnes of RepU produced from the reprocessing of fuels from LWRs in Japan (see Tables 24 and 29).

The RepU that exists within Japan now is material derived from spent fuel at the Tokai Reprocessing Plant. About 350 tonnes RepU gained at Tokai Reprocessing Plant have been used for the technology development tests of RepU utilization and about 605 tonnes RepU (in form of $UO_3$) of the remainder are still stored at the facility (data as of November 2003). The above data exclude the Pu-U mixture conversion products kept in the facility.

TABLE 28. REPROCESSED URANIUM (REPU) DERIVED FROM SPENT FUEL FROM COMMERCIAL NON-LIGHT WATER REACTORS AND RECYCLING (UNTIL 2010) (IN TONNES U)

| Utility/ Company | Total RepU gained from Reprocessing | RepU Recycled into NPPs until December 2003 (incl. Swapped and Purchased Material) | RepU Still Available for Recycling (incl. Swapped and Purchased Material) | RepU Expected to Arise under Existing or Firmly Planned Reprocessing Contracts (until 31 Dec. 2010) | RepU Expected to be Recycled during the period (1 Jan. 2004 – 31 Dec. 2010) |
|---|---|---|---|---|---|
| JNC (previously PNC) | 69 (1) | – | – | 146 (3) | 146 (3) |
| Tokai Reprocessing Plant | | – | – | | |
| Japan Atomic Power Co. | 1 460 (2) | – | – | – | – |
| BNFL | | – | – | – | – |

(1) As of 1 June 2004; (2) As of end of March 2004; (3) Spent fuel from the FUGEN reactor

TABLE 29. REPROCESSED URANIUM (REPU) DERIVED FROM SPENT FUEL FROM COMMERCIAL LIGHT WATER REACTORS AND RECYCLING (UNTIL 2010) (IN TONNES U)

| Utility/ Company | Total RepU gained from Reprocessing until end of March 2004 | RepU Recycled into NPPs until 31 Dec. 2003 (incl. Swapped and Purchased Material) | RepU Still Available for Recycling (incl. Swapped and Purchased Material) | RepU Expected to Arise under Existing or Firmly Planned Reprocessing Contracts (until 31 Dec. 2010) | RepU Expected to be Recycled during the period (1 Jan. 2004 – 31 Dec. 2010) |
|---|---|---|---|---|---|
| AREVA NC | 2 840 | Approx. 335<br>(before enrichment1) | – | – | Undecided |
| BNFL | 2 330 | | | | |
| JNC | 890 | | | | |

### *4.4.7. Netherlands*

4.4.7.1. Evolvement of reprocessing and recycling policies

Both Dutch utilities operating NPPs have had reprocessing programmes for their spent fuel, to be carried out in Belgium, France and the UK. However, in contrast to all other countries which have been, or are still engaged in spent fuel reprocessing, the Dutch utilities did not plan the recycling of any plutonium into their reactors outside the European fast breeder reactor breeder programs for which plutonium was produced from Dutch spent fuel assemblies.

At first, the Netherlands actively participated in the Eurochemic reprocessing programme, for the operation of a plant at Dessel/Mol in Belgium, which started operation in 1957 and ended in 1974 when the plant was mothballed. The electric utility GKN, which operated the

55 MWe Dodewaard PWR until its shutdown, had 8.5 tonnes of spent fuel reprocessed in the plant. GKN also signed a 53-tonnes reprocessing contract with BNFL for other Dodewaard spent fuel.

Starting in the 1970s, the electricity utility EPZ (formerly called PZEM), which operates the 452 MWe Borssele PWR, signed two reprocessing contracts with the French company COGEMA (AREVA):

- The first contract related to spent fuel which was reprocessed at the UP2 reprocessing plant before 1990. This contract corresponded to 85 tonnes of spent fuel. The contract originally concerned only 79 tonnes HM, but an additional 6 tonnes of "defective" fuel were reprocessed because it was difficult to manage; and

- The second contracts concerned fuel to be reprocessed at the UP3 plant. The so-called Base-Load Contract (or Service Agreement) was signed in 1978 which corresponded to 140 tonnes of spent fuel which were reprocessed at the UP3 plant at La Hague from 1990 to 2000. This contract had the option of extension for the reprocessing of extra quantities. EPZ has used this option several times (the last time in 2004) to cover its requirements for the foreseeable future.

#### 4.4.7.2. Dutch experience with the recycling of RepU

There is no information available on what has been done with the RepU arising from the reprocessing of spent fuel from the Dodewaard PWR at the Belgian Eurochemic reprocessing facility and at BNFL's Thorp plant.

Until 1998, EPZ has sold the uranium arising from the reprocessing of its spent fuel. However, in October 2003, EPZ loaded 36 fuel assemblies of blended RepU (RepU blended with Russian HEU) in the Borssele PWR for the first time. These assemblies were produced by Russian company OAO MSZ at Elektrostal on a back-to-back contract with Framatome ANP, EPZ's fuel fabricator, using Framatome ANP structural components and Elektrostal-produced pellets. At the time of this report, this RepU fuel is performing well, but it is still in the first operating cycle and feedback on experience is not yet available.

#### 4.4.7.3. Statistics

Until December 2003, about 270 tonnes of RepU was produced from the reprocessing of Dutch PWR reactor fuel (see Table 24). Most of the RepU was sold to third parties.

Until year-end 2010, another 60 tonnes RepU are currently expected to arise from the reprocessing of Dutch spent PWR fuel. All of that material is expected to be recycled into the Netherland's only still operating Borssele PWR.

### *4.4.8. Russian Federation*

#### 4.4.8.1. Evolvement of reprocessing and recycling policies

In Russian Federation, the nuclear fuel cycle for VVER-440 reactors is almost fully closed. This is not true for spent RBMK and VVER-1 000 reactor fuels where considerable investment is required in Russia to facilitate the reprocessing and recycling of such fuels. Important elements of Russia's spent nuclear fuel management scheme are as follows:

- Reprocessing of spent fuel from is performed in the RT-1 facility at PO MAYAK (MAYAK Chemical Combine, Chelyabinsk-65 which — until 1990 — was referred to as Chelyabinsk-40);
  - VVER-440 reactors (average discharge enrichment 0.95-1.0% $^{235}$U; 0.3% $^{236}$U);
  - research reactors (average discharge enrichment believed to be about 70% $^{235}$U);
  - Russian ice-breakers and nuclear submarines (average discharge enrichment 17% $^{235}$U); and
  - fuel from the BN-600 fast breeder reactor (average discharge enrichment 20% $^{235}$U).
- The Mining and Chemical Combine (MCC) at Krasnoyarsk (Zheleznogorsk, Krasnoyarsk-26) carries out centralized intermediate storage of spent nuclear fuel from VVER-1 000 reactors in Russia and the Ukraine;
- Some spent VVER-1 000 reactor fuel is stored in at-reactor (AR) storage ponds. So is also all of the spent RBMK reactor fuel; and
- The existing capacity of at-reactor (AR) and away-from-reactor (AFR) storages for spent VVER-1 000 reactor fuel will be sufficient for several more years. The capacity for the AR storage of spent RBMK reactor fuel will be exhausted within few years, depending on the individual RBMK reactor sites.

#### 4.4.8.2. Reprocessing of spent VVER-440 reactor fuel and recycling of RepU

The RT-1 was commissioned in 1977. The plant's design output is 400 tonnes heavy metal (HM) per year. Due to intervention of local authorities the plant's throughput is currently limited to 250 tonnes HM/year. And in recent years the actual reprocessing volume was fluctuating between just 120 and 150 tonnes HM/year, according to MINATOM (now called Federal Atomic Energy Agency (FAEA) or ROSATOM).

As for as spent power reactor fuel is concerned, RT-1 can handle only fuel elements unloaded from VVER-440 reactors, as VVER-1 000 reactor fuel bundles are much longer and 3–4 times heavier than VVER-440 reactor fuel elements. According to MINATOM's information, RT-1 had processed 3 500 tonnes HM of spent fuel by the year 2001, including 3 100 tonnes HM from VVER-440 reactors.

At RT-1, recovered VVER-440 RepU solutions are mixed with $^{235}$U-rich solutions of reprocessed uranium from nuclear submarines and ice-breaker reactors, research reactors, and the fast reactor BN-600. Mixing the different solutions to directly obtain the desired enrichments for power reactor fuel avoids the need for re-enrichment elsewhere. However, the facilities of the RT-1 plant cannot produce LEU from reprocessed uranium with assays above 2.6% $^{235}$U. Thus, as the average product assays of fresh VVER-440 and VVER-1 000 reactor fuel are higher (on average 3.8% and 4.3% $^{235}$U, respectively), the RT-1 plant can produce RepU-based fuel only for RBMK reactors (average product assay 2.4-2.6% $^{235}$U). Therefore, there is no RepU-based fuel exported from RT-1 (via Russian fuel fabrication facilities) to the former Soviet-designed power reactors in the East European countries.

It is technically feasible for entire RBMK cores to be made up of fuel produced from reprocessed and recycled VVER-440 fuel. However, as the source of spent VVER-440 fuels from the Finnish, Slovakian, Hungarian, Bulgarian, and Armenian VVER reactors has all but stopped, RT-1's output of LEU from reprocessed uranium is now very limited.

Currently, RT-1 receives the spent fuel deliveries only from the following reactors:

- the Russian reactors Kola-1, -2, -3, and 4; and Novovoronezh-3 and -4; and
- the Ukrainian VVER-440 reactors Rovno-1 and -2.

The spent fuel arisings from these reactors in 2002 were about 58.2 tonnes HM for Russia and 15.7 tonnes HM for the Ukraine. Thus, the total arisings of 73.9 t HM remained almost a factor of six below the design capacity of RT-1. However, Russian sources indicate that substantially more Russian and Ukrainian VVER-440 fuel was reprocessed at RT-1 during the period 1996-2001, indicating that a substantial backlog of fuel was accumulated prior to 1996.

The inventory of spent VVER-440 reactor fuel at RT-1 currently stands at roughly 400 tonnes HM. On an average, the spent fuel assemblies are stored for around 15 years prior to being reprocessed at RT-1.

In the Russian Federation, RepU-based nuclear fuel is in use in RBMK reactors since 1981 (see Table 33). In 2002, pilot operations started concerning the recycling of RepU-based fuel in VVER-440 and VVER-1 000 reactors.

RT-1 produces RepU-based fuel with average uranium product assay of 2.4–2.6 % of $^{235}$ U for the 11 Russian RBMK reactor units currently in operation: Kursk-1, -2, -3, and -4; Sosnovy Bor-1, -2, -3, and -4; and Smolensk-1, -2, and -3. As per the literature, the average composition of the reprocessed material and the current average annual throughput of 150 tonnes HM from the RT-1 facility are made up as follows:

- 93% spent fuel from VVER-440 reactors;
- < 1% spent research reactor fuel;
- 3% spent fuel from nuclear submarines and ice-breaker reactors; and
- 3% spent fuel from BN-600.

The resulting liquor has an assay of 2.4–2.6% $^{235}$U, which is around the average assay of a fresh RBMK reactor fuel.

According to the licenses granted by the previous Russian Federal Nuclear and Radiation Safety Inspectorate (Gosatomnadzor, GAN); since August 2004 named Federal Ecological, Technological and Nuclear Regulatory Authority (FETNRA) up to 50% of the RBMK cores can be loaded with fuel assemblies containing reprocessed uranium. The proportion of in core RBMK fuel containing reprocessed uranium is currently about 40-50%.

### 4.4.8.3. Prospects for the future production of RepU from spent power reactor fuel

Early in 1999, the former MINATOM decided to redesign and modify the existing reprocessing plant RT-1 instead of building a new reprocessing plant for the spent fuel from

VVER-1 000 at Krasnoyarsk (RT-2 plant). The decision was based on the perception that long-standing plans to build a larger plant at Krasnoyarsk cannot be financed easily in the foreseeable future. The RT-2 project, which would have been the more appropriate solution to the problem of VVER-1 000 spent fuel reprocessing activities, has now been deferred indefinitely, as it seems unlikely to be realised in the next 25 to 30 years.

In 1999, former MINATOM officials pointed out that RT-1 would establish an on-site interim storage facility and an appropriate transport system, as well as an improved and extended system to prepare spent fuel assemblies for reprocessing as the first step of redesign. The next stage would involve creation of a "technologically complete" system for reprocessing spent fuel from, and subsequent fuel fabrication for VVER-440, VVER-1000, and RBMK reactors. Manufacturing of MOX fuel as well as the reprocessing of spent fuel from foreign PWRs and BWRs would be the long term goal. No substantial progress towards this aim has ever been reported. But according to Russian experts the spent nuclear fuel from VVER-440, VVER-1000 , BWR, and PWR reactors will be reprocessed at RT-1 soon. However, due to paucity of funds the reprocessing of spent VVER-1000 reactor fuel at RT-1 is expected to commence, in incremental steps, from 2010 onwards.

A feasibility study is underway for the modernization of RT-1 for: (i) enhanced technology and waste management; (ii) increase of the actual annual throughput; and iii) adjustment of the facility to the processing of spent fuel from VVER-1000 reactors [27, 28]. It is also believed that sometime after 2010 the RT-1 reprocessing plant may run short of spent fuel from research reactors and ice-breakers and nuclear submarines, so that UNH with appropriate $^{235}$U assays required for the production of fuel for RBMK reactor can no longer be produced at RT-1. This may happen, for the following reasons:

- By the late 1990s, 184 Russian submarines with nuclear reactors have been decommissioned. The retired submarines are currently undergoing dismantling; and

- Each submarine core carries on an average 1 tonne of enriched uranium, with an average discharge assay of 17% $^{235}$U. Thus, only 100-110 tonnes of RepU with an assay of 17% $^{235}$U will arise from the dismantling of the already retired submarines. If each year 10 tonnes of that material are used for boosting RepU from VVER-440 reactors, all RepU regained from retired submarine material would have been used up by 2012 / 2013.

This implies that beyond 2012/2013 the natural uranium equivalent of the RepU regained annually from reprocessing activities at RT-1 will stay at just 350 tonnes U. Likewise, the SWU content of the regained RepU will stay at just 100 000 SWU.

### *4.4.9. Spain*

#### 4.4.9.1. Evolvement of reprocessing and recycling policies

For technical reasons, spent fuel from Spain's only (now abandoned) gas-graphite reactor Vandellos-1 was reprocessed at the French UP1 facility at Marcoule and at the UP2 reprocessing plant at La Hague. Under the related reprocessing contracts, about 1 840 tonnes RepU were produced. This quantity was sold to third parties.

In the early days of the Spanish light water reactor program, the Spanish utilities intended to secure reprocessing services abroad. But these plans were shelved in 1983, partly due to the then abundantly available and reasonably priced natural uranium. Accordingly, only very limited services for the reprocessing of spent LWR fuels were secured. Under these contracts just 55 tonnes RepU arised before 31 December 2003 (see Table 24), another 47 tonnes in August 2004, and another about 50 tonnes are expected to arise in coming years. No more RepU is expected to arise thereafter.

Thus, the Spanish utilities' present policy for spent nuclear fuel and high-level radioactive waste management is continued interim storage, followed by direct final disposal into deep geologic formations. The 5th Spanish Radioactive Waste Management Plan, approved by the Spanish government in 1999, outlined that no decision on the final disposal of high-level radioactive waste (including spent LWR fuel) be made up to 2010 [24, 29].

#### 4.4.9.2. Spanish experience with the recycling of RepU

The Spanish utilities have no experience yet in the recycling of RepU from spent gas-graphite and light water reactor fuel. No final decision has yet been made on what to do with the 47 tonnes RepU delivered in 2004, and the about 50 tonnes RepU expected to arise in coming years.

### *4.4.10. Sweden*

#### 4.4.10.1. Evolvement of reprocessing and recycling policies

Swedish nuclear fuel cycle back-end policy was originally based on the assumption that reprocessing of spent power reactor fuel and plutonium recycling would form attractive and desirable elements of the nuclear fuel cycle. Hence, the construction of a reprocessing plant in Sweden was taken up.

As commercial NPPs were built, arrangements were made therefore to send the spent nuclear fuel abroad for reprocessing. However, during the late 1970s the policies changed, and reprocessing was for various reasons no longer considered an acceptable method for the management of spent nuclear fuel.

The current spent fuel management policy was established in the late 1970s which aims at direct disposal of spent fuel without reprocessing.

In 1969, the Swedish utility Oskarshamnverkets Kraftgrupp AB (OKG) signed a contract with the United Kingdom Atomic Energy Agency (UKAEA), which was later taken over by BNFL for reprocessing of spent nuclear fuel from OKG's NPP units in Windscale (later renamed Sellafield). In all, 140 tonnes of spent fuel was shipped to Sellafield between 1972 and 1982. The fuel was reprocessed in 1997 that resulted in 136 tonnes of RepU. OKG plans to manufacture and use about 100 MOX fuel elements.

Between 1978 and 1982, an agreement was made between the Swedish Nuclear Fuel Supply Company (SKBF, later renamed SKB) and COGEMA (now named AREVA NC) regarding the reprocessing of 672 tonnes of spent nuclear fuel from the NPP units at Barsebaeck, Ringhals and Forsmark. A total of 55 tonnes was shipped to COGEMA's reprocessing facilities at La Hague before the contracts were cancelled. The fuel was then exchanged for 24 tonnes of used MOX fuel from Germany. The exchange meant that Sweden did not have to build a repository for vitrified waste and Germany did not have to build a repository for used MOX fuel. The used MOX fuel from Germany is now stored in the Central Interim Storage Facility for Spent Nuclear Fuel (CLAB) facility at Oskarshamn.

#### 4.4.10.2. Swedish experience with the recycling of RepU

OKG has concluded a “Service Contract” with the Framatome ANP (now named AREVA NP). Under this contract, the 136 tonnes of RepU were transported to Russia. Part of the total quantity was blended with enriched uranium and the resulting fuel mix was fabricated into power reactor fuel elements at TVEL’s subsidiary Machine Building Company (OAO MSZ) at Elektrostal.

In May 2000, OKG received the first ERU fuel elements containing roughly 34 tonnes RepU-LEU with a product assay of about 4.2% $^{235}$U, under the “Service Contract” from OAO MSZ. The elements were loaded into the Oskarshamn-2 reactor in the same year, and the last ERU fuel elements are planned to be loaded in 2005. After irradiation, the material will undergo intermediate storage at the CLAB facility.

The irradiation experience gained so far on of ERU fuel shows no different behaviour vis-à-vis the standard oxide fuel.

#### 4.4.10.3. Statistics

Until December 2003, the Swedish utility OKG received from OAO MSZ about 34 tonnes of RepU in the form of fabricated ERU fuel (see Table 24). Out of this, about 20 tonnes of RepU were already recycled.

### *4.4.11. Switzerland*

#### 4.4.11.1. Evolvement of reprocessing and recycling policies

Switzerland’s previous nuclear legislation stemmed in part from the early years of civil use of nuclear energy and, therefore, contained no detailed provisions on spent fuel and radioactive waste management. Under this legislation the nuclear back-end strategy chosen by the Swiss NPP operators included both reprocessing and long term storage of spent fuel, the latter with a view to later reprocessing or direct disposal. A totally new Nuclear Energy Act addressing more specifically spent fuel and radioactive waste management was passed by the Swiss parliament [30]. This new Nuclear Energy Act (Kernenergiegesetz, KEG) became effective on February 2005. It addresses more specifically spent fuel and radioactive waste management. Even if this new Act does not officially speak out in favour of the phase-out of the reprocessing of spent fuel, it imposes a 10-year moratorium on the shipment of spent fuel to reprocessing plants, beginning on 1 July 2006. This 10-year period can be extended to 20 years by simple federal order. The strategy chosen by the NPP operators includes both reprocessing and storage of spent fuel, the latter with a view to later reprocessing or direct disposal. The reprocessing of spent fuel launched under this legislation takes place abroad, in France and the United Kingdom. But the residues, such as plutonium and RepU as well as the radioactive waste arising from the reprocessing of Swiss spent fuel, will be returned to Switzerland.

In Switzerland, five power reactors (3 PWRs, 2 BWRs) at four sites are in operation, totalling around 3 000 MWe. For the interim storage of spent fuel and of radioactive waste returned from reprocessing abroad, a dry storage building at Beznau NPP (ZWIBEZ) and a Central Storage Facility (ZWILAG) have been built. A total amount of about 3 000 tonnes of spent fuel is expected assuming 40 years nuclear power plant operation (NPP). Longer operating lifetimes are technically feasible. The contracts between the Swiss NPP operators and the foreign reprocessing companies (AREVA NC in France and BNFL in the UK) cover roughly

1 100 tonnes of spent fuel (see Table 30). By the end of 2002, about 1 060 tonnes of spent fuel have already been shipped from the Swiss NPPs to the reprocessing installations in France and the United Kingdom.

TABLE 30. SWISS REPROCESSING CONTRACTS WITH AREVA NC AND BNFL

| **Reprocessor** | **Reprocessing Plant** | **Quantity (t HM)** |
|---|---|---|
| COGEMA/AREVA NC | La Hague | 659 t HM reprocessed as of 31 December 2003 |
| BNFL | Thorp | 406 |

ZWILAG started storage operations in the year 2001. By the end of 2003, twelve transport and storage casks containing spent fuel assemblies and vitrified high level waste (VHLW) have been emplaced at ZWILAG. This facility has a capacity of 200 transport and storage casks containing either spent fuel assemblies or vitrified high level waste from reprocessing.

#### 4.4.11.2. Swiss experience with the recycling of RepU

The individual strategies of the Swiss utilities concerning the recycling of RepU are as follows:

- Bernische Kraftwerke AG (BKW) (later renamed BKW FMB Energie AG) operates the 355 MWe Mühleberg BWR. BKW has not yet started its own recycling program. No concrete planning about the recycling of RepU in its NPP Mühleberg is reported. BKW has at the moment a preference to swap or loan its reprocessed material with other partners in order to minimize its RepU stockpiles;

- Fuel supplies for Nordostschweizerische Kraftwerke AG (NOK) include the use of RepU in the two 350 MWe Beznau-1 and -2 PWRs. Recycling of RepU started in 1996. The RepU quantities to be recycled each year are decided on a case by case basis, depending on the annual core management and the relevant economical conditions. RepU reload quantities are inserted in both NPPs and are combined with MOX reloads. Reportedly, the first RepU fuel element batches were supplied by COGEMA, but currently the RepU fuel is supplied by Framatome ANP/AREVA NP and their subcontractor OAO MSZ, by blending RepU with enriched uranium;

- NOK also operates the 1 165 MWe BWR Leibstadt. The recycling of RepU in this reactor is not yet decided; and

- Kernkraftwerk Gösgen-Däniken AG (KKG) started in the mid-1990s the qualification program for the use of RepU in the 1 020 MWe Gösgen PWR. Until December 2003, KKG has received about 329 tonnes of RepU arising from the reprocessing of its spent fuel assemblies. The RepU arising from AREVA NC reprocessing contracts is in the form of $U_3O_8$, while the RepU arising from the BNFL reprocessing contract is in the form of $UO_3$.

Until year-end 2010, another 55 tonnes RepU under the AREVA NC reprocessing contracts are currently expected to be recovered. As part of the qualification process of the manufacturing facility OAO MSZ at Elektrostal (Russian Federation) for the KKG fuel assemblies, 6 test fuel rods with natural uranium enriched to 3.6% $^{235}$U were loaded in the reactor in 1995. The test fuel rods were supplied by Siemens, which has been Gösgen's fuel assembly supplier since the start of reactor operation in 1979. By the end of 2003, 164 tonnes RepU have been shipped to OAO MSZ for enrichment (by blending with enriched uranium) and fabrication into fuel assemblies. The remaining RepU would be shipped in another two years. The 55 tonnes RepU yet expected to be received under the AREVA NC reprocessing contracts (see above) will also be shipped to Elektrostal for fabrication into fuel assemblies. Starting in spring 2000, deliveries of complete reloads of RepU assemblies followed. The quantities delivered annually are in the range of 20–28 assemblies and will cover the needs of the Gösgen PWR until the year 2008, totalling around 220 RepU assemblies under AREVA NP fabrication contracts. Currently the Gösgen PWR is operating in a full recycling mode (MOX plus RepU assemblies) with a typical annual reload consisting of 40 to 44 assemblies (see Tables 31 and 32).

TABLE 31. ACTUAL DELIVERIES OF REPU ASSEMBLIES (AS OF MAY 2004)

| Delivery Year | Number of FAs | Enrichment ERU % U-235 | Equiv. Enrichment % U-235 in ENU |
|---|---|---|---|
| 2000 | 4 | 4.40 | 4.30 |
| 2000 | 28 | 4.40 | 4.30 |
| 2000 | 24 | 4.40 | 4.30 |
| 2001 | 28 | 4.50 | 4.40 |
| 2002 | 20 | 4.50 | 4.40 |
| 2003 | 28 | 4.75 | 4.60 |

Note: "Equiv. Enrichment" is the enrichment of natural uranium assemblies with equivalent reactivity.

TABLE 32. ACTUAL LOADING OF REPU ASSEMBLIES (AS OF MAY 2004)

| Loading Year | Number of FAs | Enrichment % U-235 |
|---|---|---|
| 2000 | 4 | 4.40 |
| 2001 | - | - |
| 2002 | 28 | 4.40 |
| 2003 | 20 | 4.40 |
| 2004 (prov.) | 40 | 4.40 – 4.50 |

#### 4.4.11.3. Statistics

Until December 2003, about 565 tonnes of RepU arised from the reprocessing of Swiss spent reactor fuel (see Table 24). Out of about 365 tonnes were already recycled. Part of this RepU arising from the reprocessing of Swiss spent fuel at La Hague was converted into stable compounds and is awaiting potential later use.

Between year-end 2003 and year-end 2010, another 293 tonnes RepU are expected to arise from the reprocessing of Swiss spent PWR fuel. During the same period around 328 tonnes RepU is expected to be recycled into the Swiss NPPs.

### *4.4.12. United Kingdom*

#### 4.4.12.1. British experience with the recycling of RepU

Experience in the UK is limited to the recycling of RepU from Magnox reactors, the “Magnox Depleted Uranium” (MDU). This MDU usually has an average $^{235}$U content of between 0.4 and 0.5% and is low on $^{232}$U and $^{236}$U isotopes, because of its low average burnup of just 5 500 MWd/tonne HM.

For technical reasons, all Magnox reactor fuel has been reprocessed at Sellafield since the start of the Magnox program in the 1960s (see above). Until mid-2004, more than 35 000 t RepU have been recovered as $UO_3$ of which slightly more than 16 000 t had been reconverted to $UF_6$ (see Table 25) and subsequently re-enriched at BNFL’s diffusion plant at Capenhurst (to assays of 0.7% $^{235}$U), followed by Urenco’s centrifuge plants for use in AGRs (to assays of 2.6-3.4% $^{235}$U). No RepU derived from spent Magnox reactor fuel has been used for the fabrication of fresh Magnox fuel, because of the complexity and higher cost of making the metal fuel from reprocessed material.

Until the mid-1990s, about 60% of all AGR fuel used has been made from MDU. This enriched fuel, in use since 1976, amounted to some 1 650 t in the form of LEU.

Re-enrichment of MDU from its low residual level to AGR enrichment levels has required no significant extra precautions to be taken during the fuel fabrication process. There has been no segregation between fuel derived from natural and reprocessed feed in the stages of fuel fabrication process as well as for fuel loading into AGR reactors. However, this segregation is considered during conversion and enrichment. This is because the extra effect of using AGR fuel derived from MDU has been insignificant and any effects such as extra enrichment required in combating the effects of $^{236}$U rests within the uncertainties surrounding reactor physics calculations.

The recycling of MDU was discontinued in 1996 on economic grounds (prevailing low prices for natural uranium and SWU).

#### 4.4.12.2. Statistics

At Sellafield, there is currently about 20 000 tonnes RepU (ex-Magnox) in stock, and until around 2012 (when all the Magnox fuel has been reprocessed) another 15 000 tonnes RepU (ex-Magnox) will arise. Of the 35 000 tonnes RepU, some 2 400 tonnes RepU (ex-Magnox) will be from BNFL's overseas customers.

Until 2006 about 7 000 tonnes RepU derived from reprocessed from power reactor fuel will have accumulated. This quantity was expected to increase to 10 000–10 400 tonnes RepU by 2010/2011. Of this total, 4 500 tonnes RepU will be from reprocessed fuel from BNFL’s overseas customers, while the balance of 5 500–5 900 tonnes RepU will be from British spent AGR fuel. However, as BNFL’s RepU storage capacities at Sellafield are limited to 9 000 tonnes, BNFL feels forced to develop and implement a strategy on what to do with the “excess” 1 000–1 400 tonnes RepU.

Thus far, only 200 tonnes RepU from Thorp were transferred to the Russian Federation for the recycling into fuel.

### *4.4.13. Overview*

Table 33 summarizes the knowledge gained with the recycling of RepU in several IAEA Member States.

TABLE 33. EXPERIENCE WITH THE RECYCLING OF REPU (IN TONNES U)

| Country | Utility (current Majority Owner) | Reactor | Year of RepU Loading | Special License Needed for Regular RepU Fuel Loading (Yes/No) | Regular RepU Fuel Loading in 2003/2004 (Yes/No) | RepU Fuel Loading Firmly planned until 2010 (Yes/No) | Comment |
|---|---|---|---|---|---|---|---|
| Belgium | Electrabel | Doel-1 | 1997 | | | | Regular annual reloading started in 1997 |
| | | Doel-4 | 1990 | | | | Campaign comprised one complete annual reload |
| | | Tihange-2 | 1990 | | | | Campaign comprised half an annual reload |
| France | Edf | Cruas-2 | 1996 | | | | Loading of RepU fuel |
| | | Cruas-3 | 1994 | | | | Loading of 24 fuel assemblies |
| | | | 1995 | | | | Two consecutive loadings of RepU fuel in 1995-1996 |
| | | | 1997 ff | Yes | Yes | No | Regular annual reloading of ¼ core per cycle of 12 months |
| | | Cruas-4 | 1987 | | | | Loading of 8 demonstration fuel assemblies |
| | | | 1996 | | | | Loading of RepU fuel |
| | | | 1997 ff | Yes | Yes | No | Regular annual reloading of ¼ core per cycle of 12 months |
| Germany | EnBW AG | Obrigheim (KWO) | 1983 | | | | Loading of one “Siemens/KWU lead test assembly” |
| | | | 1995 | | | | First test ***rods*** fabricated by OAO MSZ were inserted |
| | | | 1996 | | | | First test ***assembly*** fabricated by OAO MSZ was inserted |
| | | | 1997 | | | | Four further test assemblies fabricated by OAO MSZ were inserted |
| | | | | | Yes | | Regular loading of RepU fuel assemblies fabricated by OAO MSZ |
| | | Neckarwestheim-1 (GKN-1) | 1986 | | | | Loading of four lead test assemblies |
| | | | 1987 | | | | Loading of another four lead test assemblies |
| | E.ON Kernkraft GmbH | Unterweser (KKU) | 2000 | | | | Loading of four lead test assemblies |
| | | | 2003 | | Yes | | First reload in 2003. Regular loading of ERU fuel assemblies fabricated by OAO MSZ |
| | | Brokdorf (KBR) | 2000 | | | | Loading of four lead test assemblies |
| | | | 2003 | | Yes | | First reload in 2003. Regular loading of ERU fuel assemblies fabricated by OAO MSZ |
| | RWE Power AG/E.ON Kernkraft GmbH | Gundremmingen-B (KRB-B) | | | Yes | | Regular loading of RepU fuel assemblies fabricated by OAO MSZ |
| | EnBW AG | Neckarwestheim-2 (GKN-2) | | | Yes | | Regular loading of RepU fuel assemblies fabricated by OAO MSZ |
| India | Nuclear Power Corp. of India | Rajasthan-2 (RAPS-1) | 1980 | n.a. | No | n.a. | Loading of RepU bundles is carried out in PHWRs only during initial criticality and after enmasse coolant channel |

TABLE 33. (Continued)

| Country | Utility (current Majority Owner) | Reactor | Year of RepU Loading | Special License Needed for Regular RepU Fuel Loading (Yes/No) | Regular RepU Fuel Loading in 2003/2004 (Yes/No) | RepU Fuel Loading Firmly planned until 2010 (Yes/No) | Comment |
|---|---|---|---|---|---|---|---|
|  | (NPCIL) |  |  |  |  |  | replacement |
|  |  | Madras-1 (MAPS-1) | 1983 | n.a. | No | n.a. | Loading of RepU bundles is carried out in PHWRs only during initial criticality and after enmasse coolant channel replacement |
|  |  | Madras-2 (MAPS-2) | 1985 | n.a. | No | n.a. | Loading of RepU bundles is carried out in PHWRs only during initial criticality and after enmasse coolant channel replacement |
|  |  |  | 2003 | n.a. | No | n.a. | MAPS-2 loading in 2003 was after coolant channel replacement |
|  |  | Narora-1 (NAPS-1) | 1989 | n.a. | No | n.a. | Loading of RepU bundles is carried out in PHWRs only during initial criticality and after enmasse coolant channel replacement |
|  |  | Narora-2 (NAPS-2) | 1991 | n.a. | No | n.a. | Loading of RepU bundles is carried out in PHWRs only during initial criticality and after enmasse coolant channel replacement |
| Japan | Tokyo Electric Power Co. | Fukushima-I-3 | 1987 | No | No | Undecided | RepU re-enriched by JNC |
|  |  | Fukushima-II-1 | 1993 | No | No | Undecided | RepU re-enriched by JNC |
|  | Kansai Electric Power Co. | Ohi-2 | 1991 | No | No | Undecided | RepU re-enriched by JNC |
|  |  | Mihama-3 | 1995 | No | No | Undecided | RepU re-enriched by Urenco |
|  |  | Takahama-1 | 2003 | No | Yes | Undecided | RepU re-enriched by JNC and Urenco |
|  | Shikoku Electric Power Co. | Ikata-3 | 2003 | No | Yes | Undecided | RepU re-enriched by JNC |
|  | Japan Atomic Power Co. | Tsuruga-2 | 2002 | No | No | Undecided | RepU re-enriched by JNC |
|  | Kyushu Electric Power Co. | Sendai-2 | - | No | Yes | Undecided | RepU re-enriched by JNC |

TABLE 33. (Continued)

| Country | Utility (current Majority Owner) | Reactor | Year of RepU Loading | Special License Needed for Regular RepU Fuel Loading (Yes/No) | Regular RepU Fuel Loading in 2003/2004 (Yes/No) | RepU Fuel Loading Firmly planned until 2010 (Yes/No) | Comment |
|---|---|---|---|---|---|---|---|
| Netherlands | EPZ | Borssele | | | Yes | | One reload comprising 36 fuel elements was loaded in October 2003 |
| Russia | MINATOM / Rosenergoatom | Kursk-1 | 1990 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Kursk-2 | 1990 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Kursk-3 | 1990 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Kursk-4 | 1990 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Sosnovy-Bor-1 | 1980 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Sosnovy-Bor-2 | 1981 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Sosnovy-Bor-3 | 1983 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Sosnovy-Bor-4 | 1983 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Smolensk-1 | 1990 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Smolensk-2 | 1990 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| | | Smolensk-3 | 1990 | | Yes | | Up to 50% of the core can be loaded with RepU fuel |
| Sweden | OKG | Oskarshamn-2 (OKG-2) | 2000 | | No | | 34 tonnes RepU fuel were loaded in 2000: Last 14 tonnes RepU fuel are planned to be loaded in 2005 |
| Switzerland | NOK | Beznau-1 | 1996 | | Yes | Yes | |
| | | Beznau-2 | 1996 | | Yes | Yes | |
| | Kernkraftwerk Goesgen-Daeniken AG | Goesgen-Daeniken (KKG) | 2000 | | | | First test assemblies fabricated by OAO MSZ were inserted |
| | | | 2002 | | Yes | | Since mia-2002, complete reloads comprising RepU fuel assemblies are loaded regularly |
| United Kingdom | British Energy (BE) | Different AGRs | 1976 | | | | Regular recycling of re-enriched “Magnox Depleted Uranium” (MDU) into AGRs until 1996 |

## 4.5. Management of spent RepU fuel (multi-recycle)

### *4.5.1. Physically re-enriched PWR fuel*

Spent ERU fuel contains downgraded plutonium with a high $^{238}$Pu content and uranium with isotopes $^{235}$U, $^{234}$U and $^{236}$U higher than in spent ENU fuel. It can, however, be reprocessed with the usual technological process. Feasibility studies carried out by the AREVA NC have shown that the spent ERU fuel could be reprocessed in La Hague plant's existing facilities provided some additional constraints are taken into account.

Reprocessed uranium resulting from the reprocessing of spent ERU fuel can be further recycled into ERU fuel. It is necessary, however, to dilute the given the RepU to cope with the increasing content of both $^{232}$U and $^{236}$U.

To cope with the increasing content of $^{236}$U, dilution with HEU or selective enrichment could be envisaged to stay within the 5% $^{235}$U assay limit. Meanwhile reprocessed uranium can also be used as an inert matrix for Pu-based fuels. As for the separated plutonium, it could be recycled in fast neutron reactors.

### *4.5.2. Gas-graphite reactor fuel, case of the United Kingdom*

Reprocessing of spent AGR fuel which is manufactured using the RepU (the RepU obtained from Magnox spent fuel processing) has shown feasible in the United Kingdom (see Section 4.4.12). A significant proportion of the 1 650 tU of AGR fuel fabricated using RepU recovered from the reprocessing of Magnox fuel at Sellafield has been reprocessed in Thorp. However, RepU produced from Thorp has not been recycled in the United Kingdom AGR or PWR reactors so far.

## 4.6. Novel enrichment technologies

Much development efforts have been spent on LASER enrichment technology, but it remains very much as a technology at an experimental stage. LASER enrichment appears to offer two key advantages:

- Lower electricity costs than gaseous diffusion; and
- Lower capital costs than centrifuges.

The USEC's AVLIS (Atomic Vapour LASER Isotope Separation) process [31] which is based on the principle on the LASER photo-ionization to produce $^{235}$U isotopes with a positive charge (which is also called as ‘Atomic processes’), and the CEA’s SILVA process [32] have both got their development programmes terminated the former in 1999 and the latter in 2003. Even with the alternative Molecular Processes which is based on the principle of photo-dissociation by LASER of $UF_6$ to $UF_5$ (solid) viz. in the South African MLIS (Molecular LASER Isotope Separation) process and the Australian (Separation of Isotopes by LASER Excitation) SILEX process, only the development programme for SILEX remains on-going [33]. The Ministry of Economy, Trade and Industry in Japan decided to shelve its research and development on LASER uranium enrichment technology on 2 October 2001. The decision reflects questions about the technology’s applicability and economic feasibility. Since the late 1980s, the Japanese government has spent more than Yen 50 billion and the power industry over Yen 15 billion on developing LASER uranium enrichment technology.

In order to reduce the $^{232}$U content in low enriched uranium hexafluoride, in Russia the Siberian Chemical Combine (SCC) at Tomsk carried out researches on the possibility to purify RepU from the isotope $^{232}$U, utilizing the gas centrifuge isotope separation technology. Evaluations showed the principle possibility of such purification. This facility would consist of a main cascade and a purification cascade. Sampling from the main cascade would go to the purification cascade, and tails of the purification cascade would be the final product: enriched RepU with low contents of $^{232}$U in the final U product. Mathematical evaluation of re-enrichment and purification of 100 tonnes of RepU with characteristic U isotopic composition in feed, product and tails are illustrated in Table 34.

TABLE 34. CALCULATED EVALUATION OF ENRICHMENT AND PURIFICATION OF 100 TONNES REPU WITH TYPICAL ISOTOPIC COMPOSITION

| **Material** | **RepU (tonnes)** | **Isotope Content (in %)** | | | |
|---|---|---|---|---|---|
| | | **$^{235}$U** | **$^{232}$U** | **$^{234}$U** | **$^{236}$U** |
| Feed | 100 | 0.822 | $1.2 \times 10^{-7}$ | 0.016 | 0.343 |
| Sampling in the Main Cascade | 13.953 | 4.047 | $9.0 \times 10^{-7}$ | 0.094 | 1.207 |
| Tails (Product) of the Purification Cascade | 13.943 | 4.0 | $1.0 \times 10^{-7}$ | 0.070 | 1.208 |
| Sampling in Purification Cascade | 0.010 | 66.574 | $1.0 \times 10^{-3}$ | 33.00 | 0.414 |

### 4.7. Immobilization and disposal

Certain categories of reprocessed uranium (perhaps because of very low $^{235}$U or very high $^{236}$U content, incomplete separation from plutonium in reprocessing, or contamination with neutron absorbers) may not meet either the technical requirements or the economic criteria to justify recycle. Therefore such materials may neither be recycled nor further stored, so considered for immobilization and disposal. This disposal process would be irreversible and therefore needs to be assessed not in terms of asset management, but rather in terms of liability management.

In the United Kingdom, $UO_3$ is not considered to be a suitable form for long term storage and therefore conversion to an alternative more stable form is required pending decisions on its final disposition. In France, $U_3O_8$ is considered to be a suitable form for long term storage and therefore this conversion decision is not required.

A possible option to be considered for final disposition might be to return the reprocessed uranium to the mines from which the uranium oxide concentrate (UOC) was removed. This would require international agreements and would be subject to political and public relations issues. It is judged that further consideration of this option is outside of the scope of this document. An alternative is disposal in a national or international repository. It is possible, depending on the development of national policies, that such a repository would be combined with a national radioactive waste repository.

Conversion to a stable form and immobilisation would be necessary first steps prior to the disposal, but it would be essential that the technologies to be used for these processes are integrated with the disposal criteria of the selected repository.

A similar problem exists for the large stocks of depleted $UF_6$ arising from enrichment processes generating tails with very low levels of $^{235}U$. It is possible that there would be benefits arising from choosing a common stable form for both RepU and depleted $UF_6$. The most likely stable form for disposal is as $UO_2$. Immobilisation of the $UO_2$ in concrete in steel containers is likely to be the most suitable physical form for ultimate disposal.

# 5. MARKET/ECONOMIC ISSUES AND DRIVERS TO USE REPU

## 5.1. Share of RepU in uranium requirements balance

RepU currently plays a very minor role in satisfying world uranium requirements for power reactors. It currently displaces 2 000 to 2 500 tonnes of natural uranium per annum, in a total market of 65 000 tU per annum. It is unlikely that this will change fundamentally by 2025, unless there is a sustained period of high natural uranium prices.

## 5.2. Optimization of the uranium resources

World uranium reserves are currently more than adequate to fuel likely nuclear power programs over the coming decades. Even a substantial revival of nuclear power is unlikely to alter this as uranium is so abundant in the earth's crust. However, many uranium deposits are inaccessible and can only be exploited at substantial cost and the market may not offer sufficient price incentives to do so. Without higher uranium prices, new mines may not come on stream when needed, while exploration activity will be curbed.

In order to make full use of the world's uranium reserves, however, it is necessary to reprocess the world's spent fuel and make RepU available for use in reactors. This will maximise the use of the original uranium resource, but may not necessarily be the optimum position. This will depend on the economic criteria.

## 5.3. Drivers to use RepU

The motivations to recycle RepU in reactors may be strategic or economic or political or a combination of these. At a national level, energy policies are likely to be a major influence on the decision making of the owners of RepU. In particular, almost all countries have a national the energy policy or a specific nuclear energy policy which determines long term goals for issues such as energy mix, the role of nuclear energy and security of energy supply. Security of supply is an important attribute of nuclear power in many countries, concerned about dependence on oil and gas imports from outside. Nuclear fuel has historically been available in good volumes and from politically stable countries. In a number of countries it is recognised that maximising the use of the original uranium resource by reprocessing spent fuel and recycling the derived products can contribute to the security of energy supply.

It is appropriate to make a distinction between political choices related to operating nuclear power plants and reprocessing of spent fuel on the one hand and on the other more strategic decisions such as the recycle of RepU. In most countries, RepU inventory is owned by the utilities rather than treated as a national inventory and these utilities are faced with practical decisions about whether to treat RepU as an asset and whether it is part of their fuel inventory. For such utilities, decisions on the recycle of RepU are primarily strategic or economic.

The available evidence is that in many countries economic factors dominate decision-making by the utilities. If a decision is not taken by its owner to recycle RepU on an early timescale then it must either be stored or disposed of. Conversion of the RepU into a form suitable for medium term storage and the provision of such medium term storage in purpose designed storage facilities are services which are available from a limited number of suppliers. A disposal route for RepU has not been proven on any commercial scale and the specific requirements for conditioning of RepU prior to disposal have not been established. Clearly, both long term storage and disposal involve significant political and strategic decisions and may incur substantial economic costs.

Assuming there is already a stockpile of RepU available, the economic assessment to be undertaken by the utility involves a comparison between the costs of manufacturing fuel from natural uranium against the costs of manufacturing fuel from recycled RepU offset by the 'saving' achieved by not committing to an alternative strategy of long term storage or disposal. In circumstances where the decision has to include the costs of reprocessing and the costs of disposal options for the spent fuel (whether reprocessed or not), the calculation becomes more complex. Unfortunately, the costs of final disposal of spent nuclear fuel are not well-known, so any calculations made about “optimum” mixes of natural and RepU are somewhat speculative.

There is currently no international market for RepU with stockpiles being administered on a broadly national basis. The higher prices for natural uranium recently experienced will undoubtedly spur increased interest in RepU, but the constraints on its use make the development of a market unlikely. There is little evidence to indicate any movement from the current situation in which RepU continues to be held as part of the strategic inventory of certain power utilities and is not generally traded. Nevertheless, some utilities who have already committed to the use of Rep U fuel in their own reactors may be able and willing to consume RepU belonging to other utilities provided that an economically attractive arrangement can be achieved for both parties.

# 6. REGULATORY AND PUBLIC ACCEPTANCE ISSUES

## 6.1. Regulatory issues

### *6.1.1. Existing IAEA regulations*

At this point in time, there are no IAEA regulations specific to the use of RepU.

Conformity with the regulations for the safe transport of reprocessed uranium was examined to determine whether packages used for the transport of un-irradiated (enriched) uranium may also be used for reprocessed uranium transport [34]. As a result of this scrutiny, it has been concluded that the provisions in the current regulations that govern the transport of natural uranium can generally be applied to reprocessed uranium compounds and that reprocessed uranium can consequently be transported as a low specific activity (LSA) material. As per this 1994 document [34] the existing definition of unirradiated uranium is inadequate, because it would allow reprocessed uranium to be classified as unirradiated. Consequently, this document describes the proposed changes in the definition of unirradiated uranium so as to encompass uranium containing not more than $2 \times 10^3$ Bq of plutonium per gram of $^{235}$U, not more than 9 MBq of fission products per gram of $^{235}$U and not more than $5 \times 10^{-3}$ grams $^{236}$U per gram of $^{235}$U.

Recommendations on satisfying the requirements concerning particular responsibilities and functions of the regulatory body in the regulation of nuclear facilities in general are described in four interrelated IAEA Safety Guides [35–38]. These guides cover in general nuclear facilities such as enrichment and fuel manufacturing plants, nuclear power plants, spent fuel reprocessing plants and radioactive waste management facilities.

The first safety guide [35] in this series provides recommendations on the organizations and staffing of a regulatory body for nuclear facilities; its structure and organization; its interaction with other organizations; the appropriate qualifications required of the staff of the regulatory body; and the training to be provided for those staff.

Recommendations for regulatory bodies on reviewing and assessing the various safety related submissions made by the operator of a nuclear facility at different stages (siting, design, construction, commissioning, operation and decommissioning and closure) in the facility's lifetime to determine whether the facility complies with the applicable safety objectives and requirements are discussed in the second safety guide [36].

The third safety guide [37] in this series describes recommendations for regulatory bodies on the inspection of nuclear facilities, regulatory enforcement and related matters. The objective is to provide the regulatory body with a high level of confidence that operators have the processes in place to ensure compliance and that they do comply with legal requirements, including meeting the safety objectives and requirements of the regulatory body. However, in the event of non-compliance, the regulatory body should take appropriate enforcement actions.

The purpose of fourth safety guide [38] in this series is to provide recommendations for regulatory bodies and operators on the documentation to be prepared for regulatory processes for nuclear facilities, and on how to ensure that such documentation is of sufficient quality and provides correct information in an appropriate way to serve its intended purpose.

About specifying requirements related to the legal and governmental infrastructure for the safety of nuclear facilities and including all other issues of radiation (such as safe use of sources of ionizing radiation, radiation protection, the safe management of radioactive waste and the safe transport of radioactive material) are described in the IAEA Safety Guide No. GS-R-1 [39]. This publication establishes requirements for legal and governmental responsibilities in respect of the safety of nuclear facilities, the safe use of sources of ionizing radiation, radiation protection, and the safe management of radioactive waste and the safe transport of radioactive material. Thus, it covers development of the legal framework for establishing a regulatory body and other actions to achieve effective regulatory control of facilities and activities. The principles of fundamentals of safety are illustrated in the publications of the IAEA Safety Series [40–42].

In general, the regulations aim to provide a uniform and adequate level of safety that is commensurate with the inherent hazard presented by the radioactive material being transported [43]. To the extent feasible, safety features are required to be built into the design of the package. By placing primary reliance on the package design and preparation, the need for any special actions required to be taken during transportation, i.e. by the carrier, is minimized. Nevertheless, some operational controls are required for safety purposes. This document also covers details regarding safety regulations of transport of RepU.

The recommendations and best practices, which in the light of experience and the present state of technology should be satisfied to ensure the safety for all stages in the lifetime of

Uranium Fuel Fabrication Facilities (UFFFs) are discussed in a recent IAEA publication [44]. These recommendations are the actions, conditions or procedures for meeting safety requirements. This document provides guidance to designers, operators, and regulators to ensure the safety of UFFFs. This document deals with the handling, processing and storing of low enriched uranium (LEU: $^{235}U \leq 6$ %), derived from natural, highly enriched or reprocessed uranium.

The safety requirements which, in the light of experience and the present state of technology, shall be satisfied to ensure the safety, for all stages in the life cycle of fuel cycle facilities (namely mining and milling, conversion, enrichment, fabrication of fuel including mixed oxide fuel, reprocessing, waste treatment and storage facilities are described in the recent IAEA document [44]. The requirements embodied in this publication are applicable to new fuel cycle facilities and are also to be applied to existing ones to the extent that is reasonably practicable.

The IAEA is preparing a Safety Guide [46] for the safety of conversion and enrichment facilities which will apply to facilities for conversion of uranium concentrate (natural or reprocessed uranium) to $UF_6$, enrichment by diffusion or centrifuge process to LWR enrichment level (~6% $^{235}U$) or high enrichment (HEU) of $UF_6$ originating from natural or reprocessed uranium, storage of depleted, natural reprocessed and enriched $UF_6$.

### *6.1.2. Existing regulations concerning handling and processing of RepU*

Regulations concerning handling and processing of reprocessed uranium vary from one facility to the other. In France, the discharge limits are fixed by decrees. The limits are set by the Safety Authority, based on the limits applied for by the operator of a given facility and on the results of an Environmental Impact Statement. There is no periodical modification of these limits; however the Safety Authority can call for new discharge limits. The following tables (Table 35 and 36) show the discharge limits for FBFC fuel fabrication plant in Romans, France.

TABLE 35. LIQUID DISCHARGE LIMITS

| Regulated item | Limit | Regulated item | Limit |
|---|---|---|---|
| Total activity for all uranium isotopes | 7 GBq/year<br>200 Bq/L | Al | 20 mg/L |
| Total activity for Transuranic elements | 0,1 GBq/year | P | 50 mg/L |
| Total activity for Fission products | 3 GBq/year | Zr | 0,01 mg/L |
| pH | 6,0-8.5 | Cu | 0,5 mg/L |
| Temperature | 30°C | Cr VI | 0,1 mg/L |
| Suspended solids | 100 mg/L | Total hydrocarbons | 10 mg/L |
| DCO | 300 mg/L | Cyanides | 0,1 mg/L |
| DBO5 | 100 mg/L | F | 15 mg/L |
| Total nitrogen | 30 mg/L | Other metals | 15 mg/L |

TABLE 36. OTHERS LIMITS

| Regulated item | Limit |
|---|---|
| Derived Air Concentration (DAC) * | 0.53 Bq $\alpha/m^3$ |
| Concentration of uranium | $1.3x10^{-6}g/m^3$ |

*) Annual Limit on Intake (ALI) divided by the volume of air inhaled by reference man in a working year (i.e. $2.4x10^3$ $m^3$).

### *6.1.3. Implementation of transport regulations*

#### 6.1.3.1. France

Liquid UNH is shipped to Pierrelatte in LR65 tank containers specially designed for transport and storage of low specific activity liquids. LR65 containers have an LSA II international agreement under ISO standards. Short distance transport from the La Hague reprocessing plants to the Valognes railway terminal is performed by road, using trucks. From the railway terminal up to the Pierrelatte conversion plants, transport is performed by rail. A standard wagon can receive one LR65 container, but an extended platform wagon can also be used with two LR65.

Transport of reprocessed uranium in the form of $UF_6$ is subject to the same regulations as transport of natural $UF_6$. Reprocessed $UF_6$ with a $^{235}U$ assay below 1% is shipped in 48Y cylinders, while reprocessed $UF_6$ with a $^{235}U$ assay above 1% is shipped in 30B cylinders. The cylinders are shipped by truck, by rail or by ship, depending on the carrier and on the enrichment plant the UF6 is shipped to.

Transport of re-enriched reprocessed uranium in the form of $UF_6$ is subject to the same regulations as the transport of ENU and is carried out in the same type of cylinders (30B cylinders). The corresponding containers are B(U)F type.

Transport of ERU fuel assemblies is subject to the same regulations as ENU fuel assemblies and is carried out in the same type of containers (RCC type or FS 57 containers containing two fuel assemblies). The corresponding containers are IF type. Even safer containers, the so-called FCC type containers, are currently being introduced. In both cases a specific authorization is required for the shipment of ERU.

Reprocessed uranium in the form of $U_3O_8$ is shipped from Comurhex to the Machine Building Plant (MSZ) at Elektrostal (near Moscow) in industrial 213-liters drums loaded in 20-feet ISO containers. Containers are shipped first by rail and then by ship up to St. Petersburg. They are then shipped by train to Elektrostal.

Transport of spent ERU fuel is subject to the same regulations as transport of spent ENU fuel and is carried out in the same type of containers (TN12 or TN13 depending on the type of reactor).

#### 6.1.3.2. Japan

At the time when JNC was carrying out conversion of reprocessed uranium at its Ningyo-toge Workshops, reprocessed uranium was transported from the Tokai Reprocessing Plant by truck or trailer. The distance from this plant to the Ningyo-toge Workshops is approximately 900 km and transport takes about 18 hours.

Approximately 220 metric tonnes of reprocessed uranium were transported in 30 separate trips from June 1994 to October 1997. Safe transport was undertaken without any accident or disturbance.

The transport container for reprocessed uranium, which had been approved as a Type A(F) container, was newly approved as UOX/C(F) after re-estimating by the technical standards for IP-2(F) type package. The package is a double structure cylindrical in shape, approximately 1.2 m in diameter, 1.6 m in length, and an approximate weight of 1 300 kg. This transport container which was the first approved IP-2(F) type container in Japan has satisfied the regulatory tests required for radio-nuclide transporting packages such as the 9m drop test, thermal test (8 000°C for 30 minutes), etc.

## 6.2. RepU-Specific public acceptance

Current and potential future industrial-scale reprocessed uranium (RepU) fuel loading programmes comprise the following Member States (see Section 4.4):

- Belgium: All available RepU has been recycled;
- France: Regular reloading of physically re-enriched RepU into two NPP units;
- Japan: RepU recycling program expected to be phased in according to each utility's policy;
- Germany: Regular loading of RepU (re-enriched by blending) into five NPP units;
- Netherlands: Loading of RepU (re-enriched by blending) into one reactor started recently;
- Sweden: Small-scale loading of RepU (re-enriched by blending) into one reactor was completed;
- Switzerland: Regular loading of RepU (re-enriched by blending) into three NPP unit;
- Russia: Regular loading of RepU (re-enriched by blending) into (up to 11) RBMK units; and
- United Kingdom (UK): Recycling of "Magnox Depleted Uranium" (MDU) ceased in 1996 due to economic reasons.

However, only few protests against the particular utilization of RepU fuel assemblies took place in the past in some of these countries. (Most anti-nuclear actions were staged to try to delay or even prevent the use of MOX fuel in BWRs and PWRs.)

Nevertheless, in the recent years, political opposition against reprocessing has also grown. For example, spent fuel element transports to reprocessing facilities in France (La Hague) and the United Kingdom (Sellafield) have been, and are still the target of anti-nuclear groups. These groups often protested against the utilization of plutonium (Pu)-bearing MOX fuel, rather than RepU fuel.

# 7. CONCLUSIONS

The recycling of reprocessed uranium (RepU) is an established process through proven technologies which has been taking place since the 1970s. The technical and radiological problems associated with the handling of RepU by fuel fabricators and those loading fuel into reactors are manageable, but they require a certain minimum volume throughput to justify the investment required to minimise dose uptake by fabrication plant operators. The fuel design and reactor licensing issues are also readily manageable and are generally less than the political and public relations (PR) issues associated with deciding to load RepU into a reactor.

RepU has not, however, seen the development of a well-defined and mature market, as the two decades from the mid-1980s have seen economic factors working in favour of the use of fresh uranium and political factors working against trade in RepU between countries and reactor operators. In addition, where national policies on recycling have not been present or implemented, most reactor operators have decided that RepU should continue to be stored as a potential strategic asset rather than be committed to recycle.

The results of these technical, economic, political and policy factors have resulted in a current situation in which:

- RepU recycling has continued on a significant scale, among others also in those countries where fresh uranium supplies cannot be guaranteed;
- RepU arisings have generally exceeded recycling over the past two decades and therefore the stocks of RepU have continued to increase;
- there are no significant technical issues associated with the long term storage of RepU;
- most countries do not perceive RepU recycling as a necessary domestic capability;
- there has been little investment in the development and implementation of new technologies;
- there have been low levels of investment to maintain capacities in existing facilities and in some countries facilities have been closed and not replaced;
- there is not a recognised market price for RepU; and
- reactor operators tend to be opportunistic in their approach to the recycling of their own RepU, and only a small number of reactor operators have demonstrated a willingness to consider consuming RepU belonging to others.

Since mid-2003 prices for uranium oxide concentrate (UOC) prices have risen steadily, and at the end of March 2006 both spot and long term prices for natural uranium were above US\$ 40/lb $U_3O_8$, compared to an average of US\$ 10.50–11.00/lb $U_3O_8$ in mid-2003. This price movement has encouraged many RepU owners to reassess the opportunities for the recycling of their RepU. It can be anticipated that opportunistic purchasing of recycling services may be replaced by advance booking of available capacity and reactor operators will be willing to reconsider their capabilities to burn fuel made from RepU.

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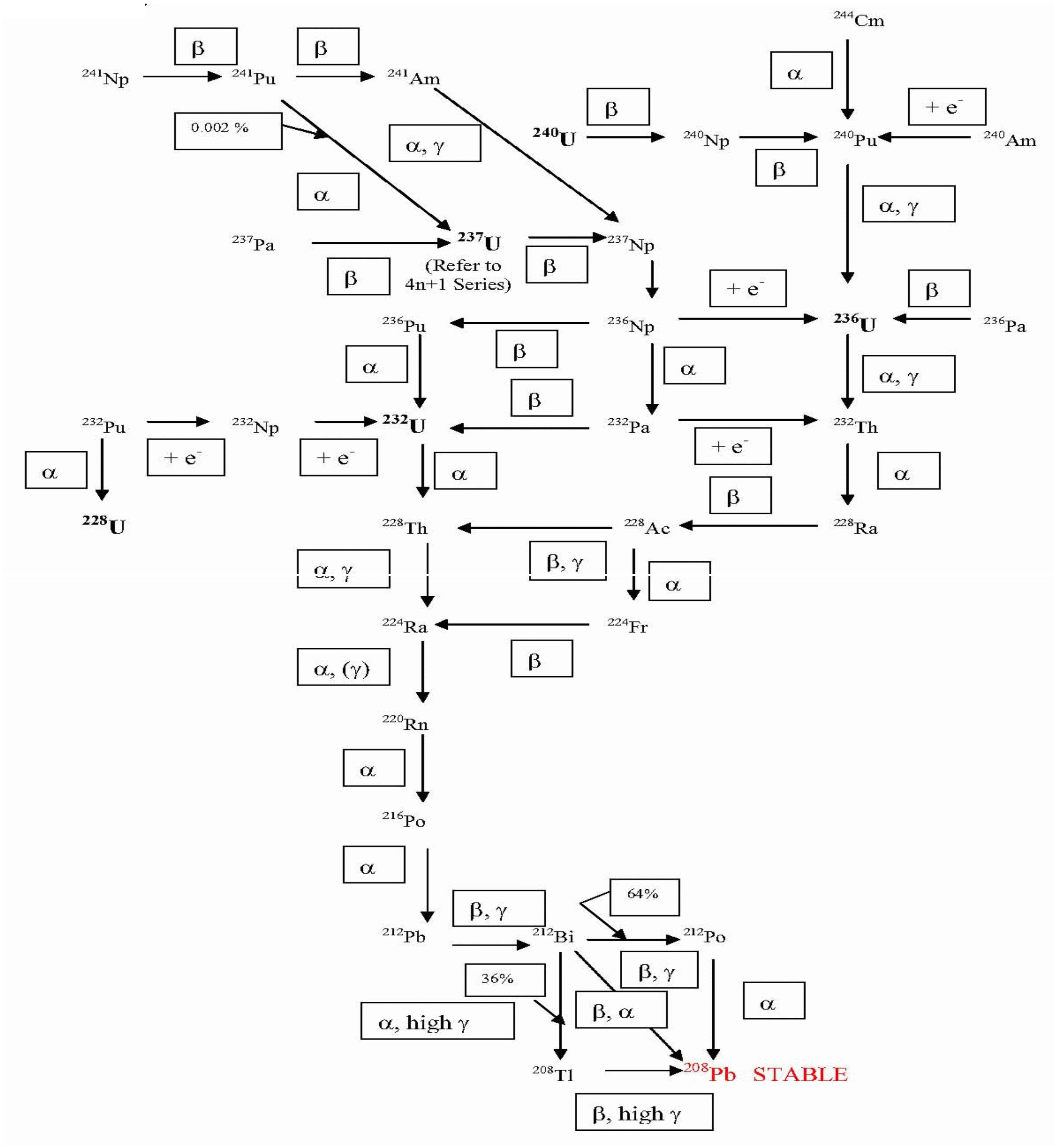
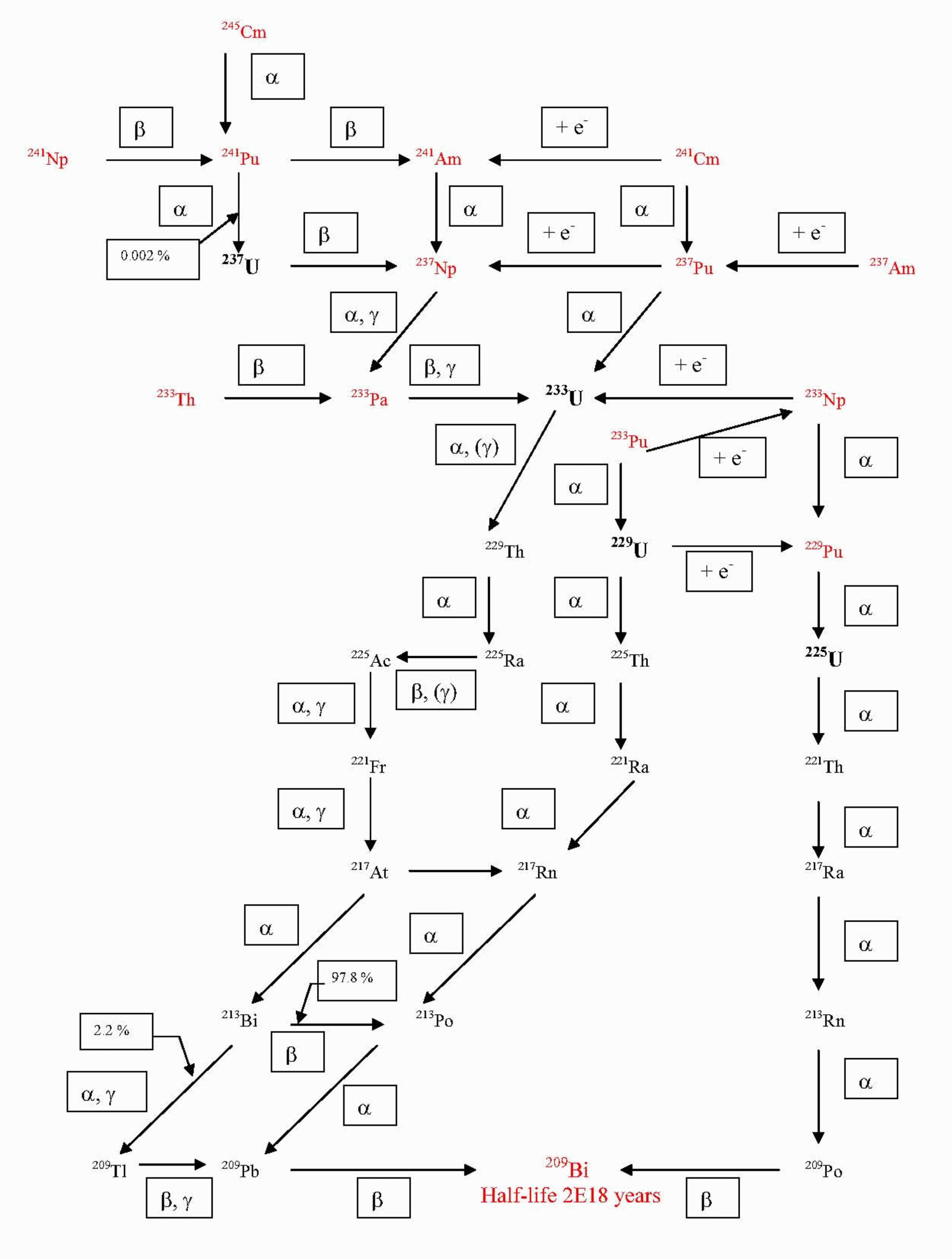
# APPENDIX I.

## 4N SERIES (THORIUM CHAIN)



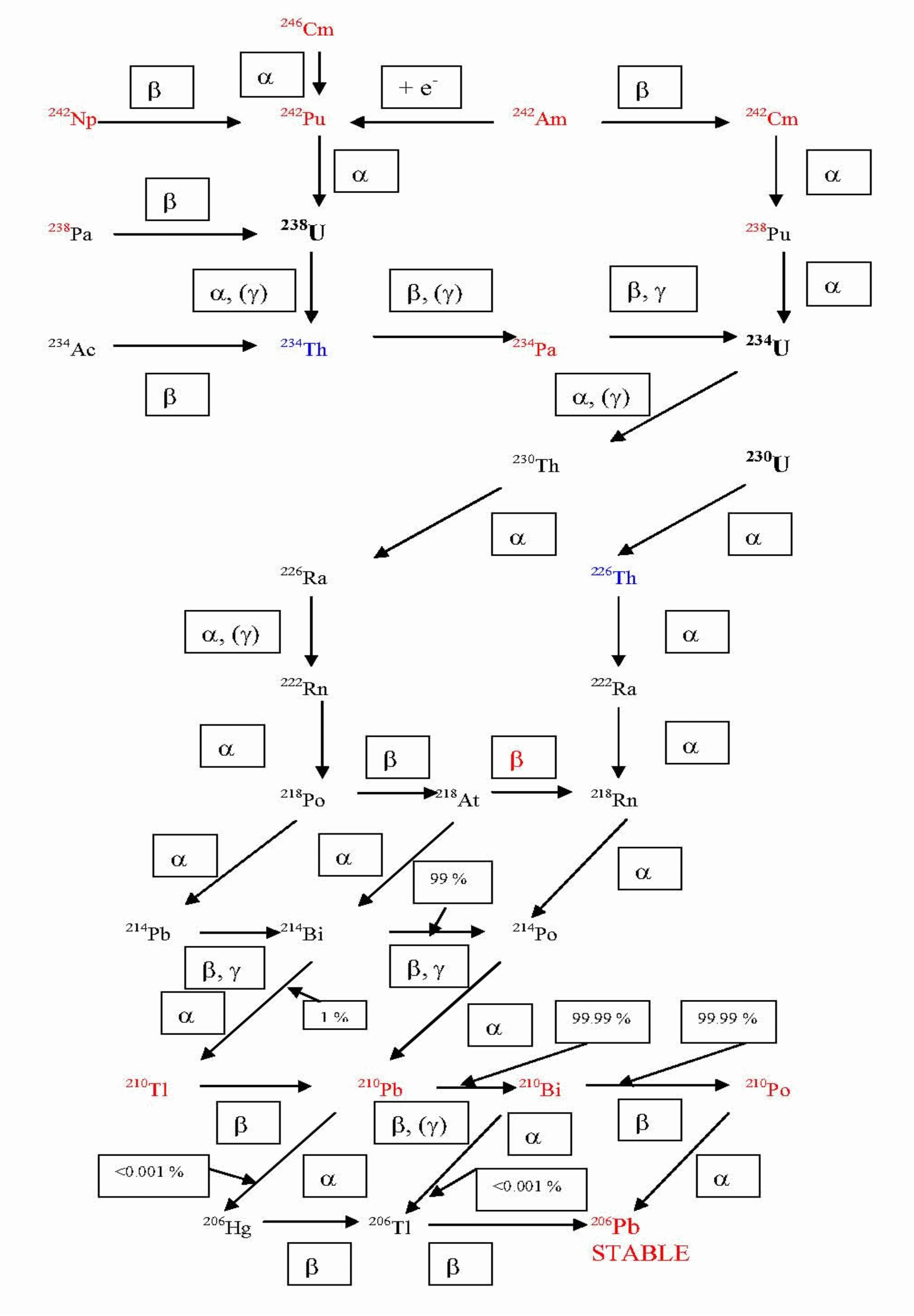
# APPENDIX II

## 4N+1 SERIES (NEPTUNIUM CHAIN)



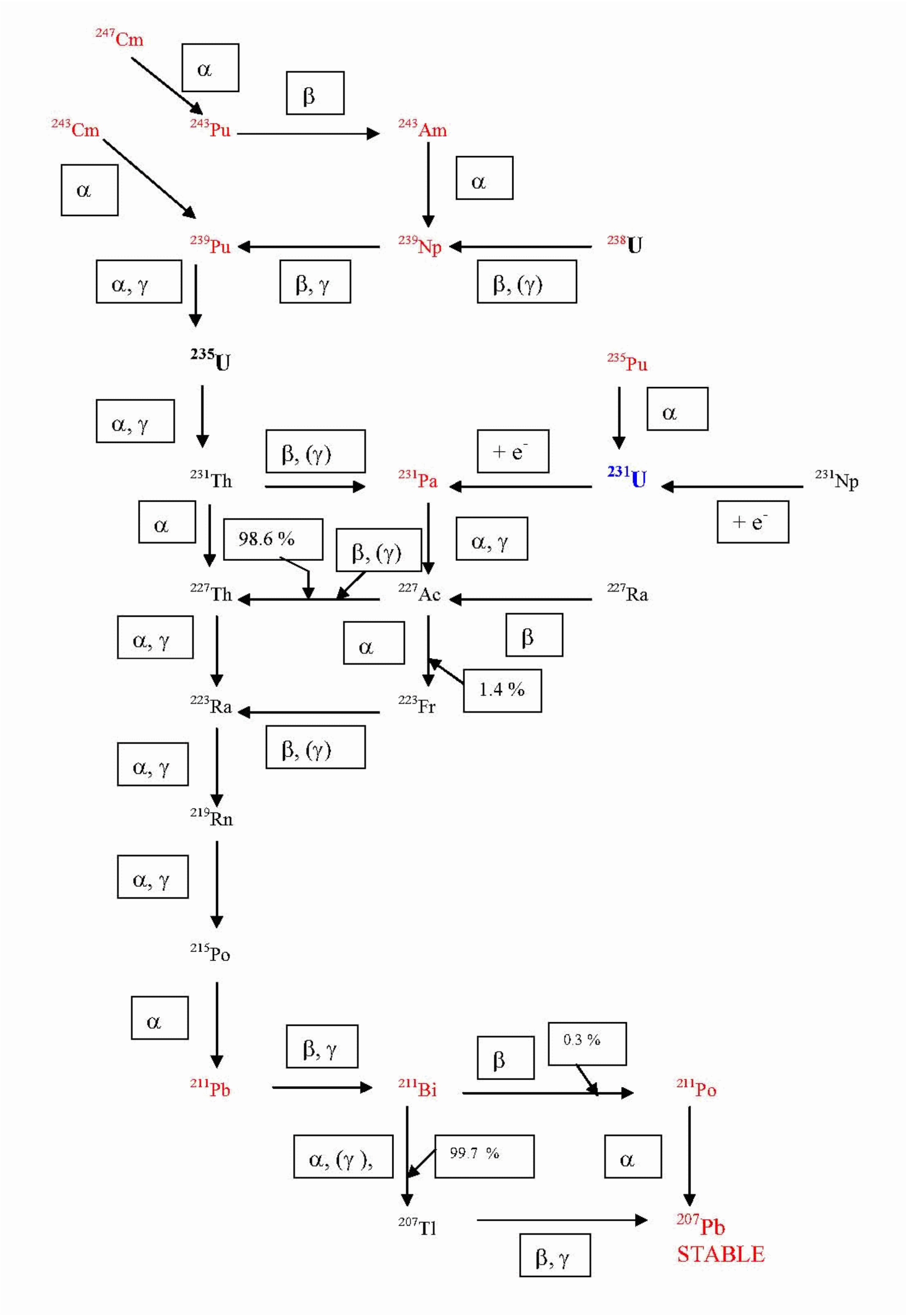
# APPENDIX III

## 4N+2 SERIES (RADIUM CHAIN)



# APPENDIX IV

## 4N+3 SERIES (ACTINIUM CHAIN)



# APPENDIX V
# UO3 POWDER SPECIFICATIONS - COMPARATIVE TABLE, REPU UO$_3$ POWDER ANALYSES

| **Item** | **Rep U UO$_3$ Powder Typical Specification A** | **Rep U UO$_3$ Powder Analyses** | |
|---|---|---|---|
| | | **Maximum** | **Typical** |
| **UF6 Content** | | | |
| **U235 Content wt% wrt U** | =<1.6 wt% wrt U | 1.0 wt% wrt U | 0.85 wt% wrt U |
| **Transuranic alpha activity kBq/kgU Total** | | | |
| Cm-242 | | | |
| Cm-243/244 | | | |
| Np-237 | Np=<6.0 KBq/Kg U | 1.0 KBq/Kg U | 0.5 KBq/Kg U |
| Pu-236 | | | |
| Pu-239/240 | =<25.0 KBq/Kg U<br>for Total incl. Np. | 10 KBq/Kg U | 3 KBq/Kg U |
| Pu-238 / Am-241 | | | |
| **Other U-isotopes ppm wrt U** | | | |
| U-232 | **To be reported only** | 1.7 E – 3 ppm | 1.0 E – 3 ppm |
| U-234 | **To be reported only** | 190 ppm | 160 ppm |
| U-236 | **To be reported only** | 4000 ppm | 3000 ppm |
| **Non-Uranic gamma activity kBq/kgU** | | | |
| Ru 106 | | 17 KBq/Kg U | 8 KBq/Kg U |
| Ru 103 | | | |
| Zr 95 | | | |
| Nb 95 | =<524 KBq/Kg U | | |
| Cs 137 | | | 40 KBq/Kg U |
| Ce 144 | | 120 KBq/KgU | |
| Sb 125 | | | |
| Pb 212 | | | |
| **Physical Properties % wt powder retained** | | | |
| Mesh Size 5600 | **=<15 % wt/wt Retn.** | | |
| Mesh Size 2000 | **=<30 % wt/wt Retn.** | | 20 % wt/wt Retn. |
| Mesh Size 355 | | | 70 % wt/wt Retn. |
| Mesh Size 53 | | | |
| Mesh Size 2.5 | | | |
| **Chemical Composition** | | | |
| Moisture Content % wt/wt wrt powder sample | =<0.5 % wt/wt | 0.6 % wt/wt | 0.3 % wt/wt |
| Nitrate ion Content % wt/wt wrt UO$_3$ | =<1.0 % wt/wt | 0.7 % wt/wt | 0.55 % wt/wt |
| Sulphate ion content micro g/gU | =<1200 micro g/gU**<br>**Normal | 1300 ppm | 1100 ppm |
| Tc 99 ppm wrt UO$_3$ | =<0.5 micro g/gU | 0.07 ppm | 0.04 ppm |
| **Chemical Composition Total hydrocarbon, chlorocarbon, partially substituted hydrocarbon** | | | |
| **Other Chemical Impurities (in ppm wrt U)** | | | |
| As | =<3 ppm wrt U | 0.25 ppm | 0.02 ppm |
| Sb | =<1 ppm wrt U | 0.2 ppm | 0.04 ppm |
| B | =<1 ppm wrt U | 1 ppm | 0.2 ppm |
| F | | | |
| Br | =<5 ppm wrt U | 4 ppm | 1 ppm |
| Cl | =<100 ppm wrt U | 15 ppm | 5 ppm |
| Cr | =<50 ppm wrt U | 20 ppm | 15 ppm |

| Item | Rep U $UO_3$ Powder Typical Specification A | Rep U $UO_3$ Powder Analyses | |
|---|---|---|---|
| | | Maximum | Typical |
| | Also included in Non-Volatile Fluorides Total =<300 ppm wrt U | | |
| Fe | Also included in Non-Volatile Fluorides Total =<300 ppm wrt U | 140 ppm | 80 ppm |
| Mo | =<2.5 ppm wrt U | 0.4 ppm | 0.3 ppm |
| Ni | Also included in Non-Volatile Fluorides Total =<300 ppm wrt U | 25 ppm | 20 ppm |
| Nb | =<1 ppm wrt U | 0.1 ppm | 0.05 ppm |
| P | =<75 ppm wrt U | 60 ppm | 45 ppm |
| Ru | =<1 ppm wrt U | 0.2 ppm | Nil |
| Si | =<100 ppm wrt U | 25 ppm | 5 ppm |
| Ta | =<1 ppm wrt U | 0.4 ppm | Nil |
| Th | | | |
| Ti | =<1 ppm wrt U | 0.6 ppm | 0.1 ppm |
| W | =<1.4 ppm wrt U | 0.4 ppm | <0.05 ppm |
| V | =<1.4 ppm wrt U | 0.4 ppm | <0.1 ppm |
| **Alkaline and Alkaline Earth Metals** | | | |
| Na | **To be reported only** | 5 ppm | 2 ppm |
| K | **To be reported only** | 2.5 ppm | 2 ppm |
| Ca | **To be reported only** | 5 ppm | 2 ppm |
| Mg | **To be reported only** | 1 ppm | 0.5 ppm |
| **Non-Volatile Fluorides** | | | |
| Al | **=<300 ppm wrt U** | 150 ppm wrt U | 100 ppm wrt U |
| Ba | | | |
| Be | | | |
| Bi | | | |
| Cd | | | |
| Ca | | | |
| Cr | | | |
| Cu | | | |
| Fe | | | |
| Pb | | | |
| Li | | | |
| Mg | | | |
| Mn | | | |
| Ni | | | |
| K | | | |
| Ag | | | |
| Na | | | |
| Sr | | | |
| Th | | | |
| Sn | | | |
| Zn | | | |
| Zr | | | |
| **Non-Volatile Fluorides** | | | |
| **Total** | =<300 ppm wrt U | 150 ppm wrt U | 100 ppm wrt U |

## LIST OF ABBREVIATIONS

| | |
|---|---|
| ADU | ammonium diuranate |
| AFA | advanced fuel assembly |
| AFC | advanced fuel cycle |
| AFFF | advanced fuel fabrication facility (Tarapur, India) |
| AGR | advanced gas-cooled reactor |
| ALARA | as low as reasonably achievable |
| ALI | annual limit of intake |
| ASTM | American Society for Testing and Materials |
| AVLIS | atomic vapor laser isotope separation |
| BWR | boiling water reactor |
| CANDU | Canadian Deuterium-uranium reactor |
| DU | depleted uranium (tails material) |
| EFPD | equivalent full power days |
| ENU | enriched natural uranium |
| ERU | enriched reprocessed uranium |
| EUREX | enriched uranium extraction |
| FBR | fast breeder reactor |
| GCR | gas-cooled reactor |
| HEU | highly-enriched uranium |
| HWR | heavy water reactor |
| INB | installation nucleaire de base (basic nuclear installation) |
| ISO | International Standards Organization |
| KARP | Kalpakkam reprocessing plant |
| LEU | low enriched uranium |
| LSA | low specific activity |
| LWR | light water reactor |
| MA | minor actinides |
| MELOX | mélange d’oxyde d’uranium et d’oxyde de plutonium |
| MLIS | molecular laser isotope separation |
| MOX | mixed (uranium-plutonium) oxid |
| MTR | material test reactor |
| NFS | nuclear fuel services, Inc. (USA) |
| NPT | non-proliferation of nuclear weapons |
| PFBR | plutonium fast breeder reactor |

| | |
|---|---|
| PHWR | pressurized heavy water reactor |
| PUREX | plutonium and uranium recovery by extraction |
| PWR | pressurized water reactor |
| RBMK | light water cooled, graphite moderated reactor (Russian Federation) |
| SEU | slightly enriched uranium |
| SF | spent fuel |
| SILEX | separation of isotopes by LASER excitation |
| Thorp | thermal oxide reprocessing plant |
| UFFF | uranium fuel fabrication facility |
| UNH | uranyl nitrate hexahydrate |
| UNL | uranyl nitrate liquor |
| UOC | uranium oxide concentrate |
| UOX | uranium oxide |
| VHLW | vitrified high level waste |
| viz. | videlicit (lat.) = namely |
| VVER | Vodo-Vodyanoi Energetichesky Reaktor (Water-Cooled and Water-Moderated Reactor) |
| WAK | Wiederaufarbeitungsanlage Karlsruhe (Germany) |
| WNA | World Nuclear Association |

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## Consultants Meetings

Vienna, Austria: December 2003 and June 2004

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† Deceased.