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Plutonium Management in the Medium Term

A Review by the OECD/NEA Working Party on the Physics of Plutonium Fuels and Innovative Fuel Cycles (WPPR)

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FOREWORD

The OECD/NEA Working Party on the Physics of Plutonium Fuels and Innovative Fuel Cycles (WPPR) was established in 1993 and reports to the OECD/NEA Nuclear Science Committee. Its main activity has been to analyse benchmarks carried out to answer technical questions related to the physics of plutonium fuels. Past volumes of published work have examined the physics of plutonium-fuelled pressurised water reactors (PWRs) and boiling water reactors (BWRs), as well as the physics of metal- and oxide-fuelled fast reactors. The present report concentrates on plutonium-fuelled high-temperature reactors (HTRs).

In this report the "medium term" covers the period starting after the existing LWRs reach the end of their lives, and ending at the point at which long-term sustainable reactors (e.g. fast reactors with a closed fuel cycle) are introduced. The report intends to show that plutonium strategies in the medium term can be consistent with long-term strategies ranging from a gradual phase-out of nuclear energy to the introduction of sustainable systems. The report considers reactors such as thermal HTRs (PBMRs, GT-MHRs), LWRs with fuel and/or core design optimised specifically for plutonium utilisation (e.g. APA, CORAIL, MIX, high-moderation LWRs, high-converter LWRs). The important issue of multiple recycling is also addressed.

Kevin Hesketh (BNFL, UK), Chairman of the WPPR, took the lead in writing this report. Further contributions to the text were provided by Benoit Lance (Belgonucléaire, Belgium), Oliver Köberl (CEA Cadarache, France), Dieter Porsch and Timm Wolf (FRAMATOME ANP, Germany), Gerhard Schlosser (consultant, Germany), Hideaki Mineo, Tsutomu Okubo, Osamu Sato and Sadao Uchikawa (JAERI, Japan), Yutaka Sagayama (JNC, Japan), Toru Yamamoto (NUPEC, Japan), Rakesh Chawla, (PSI, Switzerland), Jess Gehin and Ron Ellis (ORNL, USA) and Enrico Sartori (OECD/NEA, France).

The *Physics of Plutonium Recycling* series currently comprises the following titles:

- Volume I: *Issues and Perspectives* (OECD/NEA, 1995).
- Volume II: *Plutonium Recycling in Pressurised Water Reactors* (OECD/NEA, 1995).
- Volume III: *Void Reactivity Effect in Pressurised Water Reactors* (OECD/NEA, 1995).
- Volume IV: *Fast Plutonium Burner Reactors: Beginning of Life* (OECD/NEA, 1995).
- Volume V: *Plutonium Recycling in Fast Reactors* (OECD/NEA, 1996).
- Volume VI: *Multiple Plutonium Recycling in Advanced PWRs* (OECD/NEA, 2002).
- Volume VII: *BWR MOX Benchmark – Specification and Results* (OECD/NEA, 2003).

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Abstract

The OECD/NEA Working Party on the Physics of Innovative Power Reactors (WPPR) is an expert group on reactor physics and has organised several reactor physics benchmarks aimed at improving the understanding of the neutronics modelling of innovative fuel and core designs. An important part of the WPPR's past activities has focused on plutonium recycling issues, covering the short term (conventional LWRs), the medium term (evolutionary LWRs and HTRs) and the long term (fast reactors). Some time ago it was suggested that, in view of the many proposed systems covering the medium term, it would be helpful if the WPPR were to publish a review of the technical options available.

The definition of "medium term" adopted for this review is the period from the present up to the point at which a fully sustainable fuel cycle is eventually established, which the WPPR takes as a fast reactor fuel cycle with multiple recycle. The present reality is that such fuel cycles are unlikely to be established for many years. In the intervening period, there are options for plutonium recycle which range from the recycle of plutonium in MOX assemblies in existing LWRs (as practised in many plants today) to the recycle of plutonium in innovative fuel designs in novel reactor systems. This report attempts to review the various proposals that have been made in recent years and presents a discussion of the rationale, technical attributes, capabilities and current status of each. Concepts which are no longer under active development are also included for completeness. It is intended that this report will serve a useful purpose by providing a summary description of each proposal, references for further detail and a commentary on the relative merits of the various options from a consensus of experts in the field.

1. Introduction

The decision to re-use plutonium generated in thermal reactors is a strategic decision for a utility that is closely tied with spent fuel management strategy. One option is to reprocess the spent fuel in existing reprocessing plants and immediately re-use the plutonium. Another option is to postpone re-use of the plutonium by putting irradiated fuel in interim storage. The availability of different types of reactors determines the time scales for the present, medium-term or long-term future re-use of plutonium. Current commercial reprocessing plants are all designed to separate the plutonium remaining at discharge for re-use. Historically, the rationale was to recover sufficient plutonium to enable a build-up of fast reactors, which were expected to be deployed as uranium reserves became scarce and prices rose. For a variety of reasons, but principally the low price of uranium ore, fast reactors have not yet been deployed commercially and projected time scales for doing so have been put back everywhere. Fast reactors are nevertheless still judged by many to be the most promising for long-term sustainability. Until such time as fast reactors are deployed commercially, however, the issue of how best to manage plutonium arisings from existing reprocessing plants remains. This is the subject of this review.

The majority of the spent fuel assemblies discharged from the world's 430 or so commercial nuclear reactors are currently assigned for interim storage and eventual direct disposal. Those utilities which have opted for the so-called "once-through" fuel cycle have chosen this option after considering the particular circumstances (political, economic, strategic, logistic, historic, etc.) which apply locally. With different local circumstances, other utilities' respective countries have chosen to reprocess their spent fuel.

A typical commercial thermal reactor discharges up to 28 kg of plutonium per Terawatt-hour-electrical (TWhe)[1] output, the precise figure depending on the mean discharge burn-up. A comparable-sized fast reactor core would require an initial charge of several tonnes of plutonium. Once operational, a fast reactor operating as a plutonium breeder would be self-sufficient in plutonium. The historical strategy was therefore one in which plutonium from thermal reactors would facilitate the start-up of the fast reactors that would eventually replace them. This would have unlocked the very large energy potential of ^{238}U as fertile material, allowing up to a factor 100 increase in the energy potential of uranium reserves.

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[1] 1 TW = 10^{12} W; 1 TWh = 0.1142 Gigawatt-year (GWy).

The time scale from the present to the time when sustainable large-scale fast reactor programmes are established represents the medium term, which this review paper covers. The precise extent of this interim period is indistinct, partly because of the uncertainties inevitably associated with long-term plans and partly because there are different perspectives in different countries. This report reviews the technical options available involving re-use of the separated plutonium. There are a large number of potential approaches to plutonium management in the medium term and the technical considerations are complex. This review attempts to provide an overview of the subject from the point of view of an expert technical group. The possibility of conditioning and eventual direct geological disposal of plutonium is outside the area of expertise of the WPPR and is not considered here.

2. High-level plutonium management issues

There are a number of high-level issues that need to be considered in assessing the technical options:

- Plutonium management strategies should be consistent in maintaining high standards of safety. The handling of plutonium during reprocessing operations, transport and fuel fabrication should conform to all the precautions dictated by its radioactivity and chemical toxicity. The requirements of nuclear safety in relation to the reactor physics properties of the various reactor systems are relevant, which includes guaranteeing sub-criticality in all stages of the closed fuel cycle.

- Plutonium management strategies should preferably maintain flexibility in the fuel cycle, such that future options are not foreclosed. For example, an option involving the storage of spent fuel assemblies in a retrievable form would leave open the possibility of eventual direct disposal and reprocessing and recycling the plutonium. Similarly, an option involving the recycling of plutonium in an interim generation of reactors should preferably not degrade the isotopic make-up of the plutonium to the point where it is no longer useful in fast reactors. Along the same lines, a technical option which results in plutonium being converted to an inaccessible form should also be avoided within this context (although there remains the possibility that such an option could be applied to *part* of the plutonium inventory, provided there remains sufficient plutonium which is accessible to service the expected fast reactor demand). Long-term strategies extending up to the point where fission reactors are phased out should include options to minimise the final plutonium inventory.

- Plutonium management strategies should be consistent with maintaining satisfactory standards of security and safeguards against proliferation. The assessment should be judged against effect on the overall proliferation assessment of the whole system (i.e. including all the steps in the fuel cycle). The proliferation risk assessment depends on a combination of factors, including the fissile inventory, accessibility, isotopic and chemical form, etc.

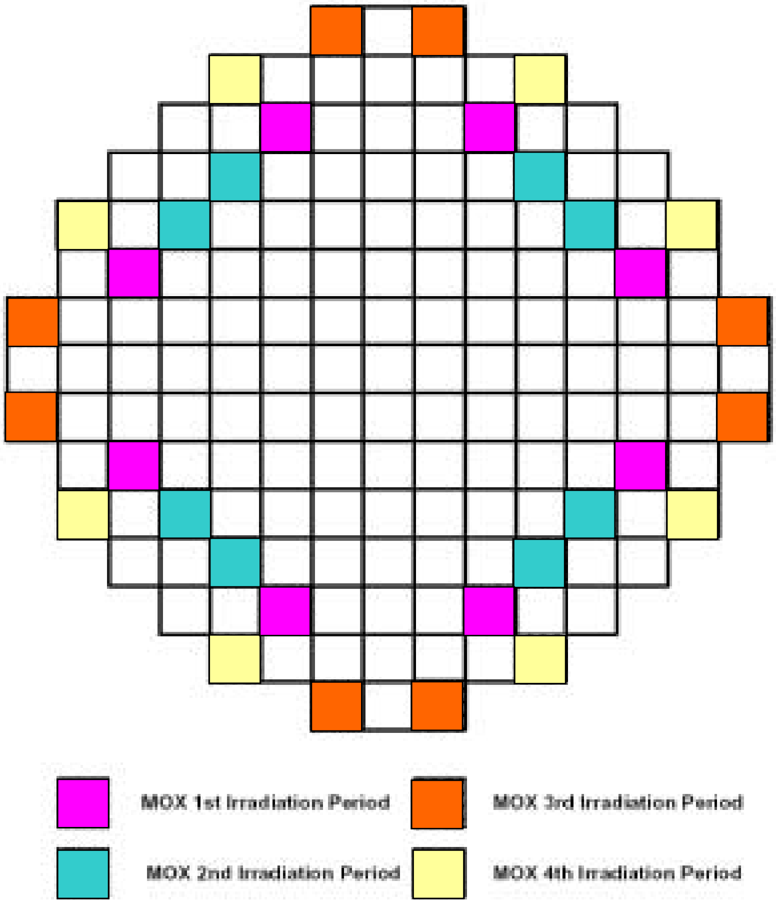
- The quantities and forms of radioactive wastes arising from each technical option are very important considerations. At the present time a quantitative discussion of waste quantities is not possible because of the difficulties of estimation, but where possible qualitative comments are made.

- Finally, a clear requirement is that the overall fuel cycle should remain economically competitive, though the economics should not be assessed in isolation, but rather as part of a Life Cycle Analysis (LCA) that accounts for all externalities.

3. Technical options

There are several technical options for plutonium management involving re-use that fall within the scope of this review. The reference case is the re-use of plutonium in existing LWRs. Partial core loading of MOX assemblies in PWRs (see Figure 1, which shows an assembly loading map for a mixed UO_2/MOX core in a PWR) is already well established on a commercial basis, with 37 reactors in Europe (two in Belgium, 22 in France, 10 in Germany and three in Switzerland) currently operating with part MOX loading and some additional reactors licensed to do so when the need arises. There is less experience of MOX usage in BWRs, with just two large BWRs in Germany currently utilising MOX fuel. Usage of MOX fuel in BWRs will probably increase in the future. There is at present no experience of MOX utilisation in VVERs, though a multilateral programme to irradiate weapons-grade plutonium in VVERs is under way. The accumulated knowledge and experience is sufficient that the use of MOX in all types of light water moderated thermal reactors can be regarded as a fully established option with low technical risk.

Figure 1. Schematic of a PWR core loading pattern with partial MOX loading



Other technical options have been investigated which may offer advantages relative to MOX (plutonium) as it is practised today. These include 100% MOX-fuelled LWRs, advanced fuel assembly designs for LWRs, high-moderation LWRs, low-moderation LWRs and high-temperature gas-cooled reactors (HTRs). Non-conventional fuel options include thorium fuels where a thorium compound is used to dilute the plutonium and inert matrix fuels (IMF), where the diluent is in the form of a material which does not undergo nuclear transformations (in particular, the diluent does not breed new fissile material). Some of these options represent small extrapolations from current technology, while others are more innovative and would require longer to develop fully.

The main objective of IMF designs is to improve the net plutonium consumption by eliminating fertile production of fresh plutonium from ^{238}U captures. A further advantage of these designs is that the production of minor actinides per unit mass of plutonium destroyed is reduced. Some of the inert matrix materials under consideration also have the advantage, in the context of plutonium disposition, of being very stable in a geological repository. It is therefore envisaged that such inert matrix fuels might be the ideal vehicle for maximising the destruction of plutonium with single recycling and effectively encapsulating the remaining plutonium in a repository, together with fission products and other transuranics.

3.1 MOX in PWRs

Partial MOX loading in LWRs is established technological practice in many PWRs. It involves the substitution of a fraction of the UO_2 assemblies by MOX assemblies with the same mechanical design. Using an identical mechanical design avoids issues of thermal-hydraulic and mechanical handling incompatibility. The plutonium concentration in the MOX assemblies is usually adjusted such that the equilibrium cycle length or "reactivity lifetime" of the MOX fuel coincides with that of the UO_2 fuel, in which case the average discharge burn-ups of the two fuel types will be approximately the same. The appropriate measure of the reactivity lifetime is the average reactivity of the UO_2 or MOX components of an equilibrium core at end-of-cycle (EOC) core conditions. There are various ways in which this lifetime average reactivity can be defined. One approach is to adjust the plutonium content of the MOX fuel such that the (core) depletion models give equivalent average discharge burn-ups for the UO_2 and MOX components. In France, mixed UO_2/MOX cores have been designed such that the average discharge burn-up of the MOX component is lower than that of the UO_2 assemblies, until high MOX burn-ups are further qualified by irradiation testing programmes. With increased MOX experience now available, this artificial licensing restriction is expected to be removed soon.

PWRs have plutonium conversion ratios of typically 0.6 to 0.7, with roughly the same value applying in both UO_2 and MOX assemblies. This implies that in a MOX assembly fresh plutonium production is insufficient to compensate for plutonium destruction (which is predominantly by fission). Therefore a MOX assembly is a net consumer of plutonium. A strategy of partial MOX loading in LWRs can result in net production or consumption of plutonium, depending on the MOX core fraction. For a MOX core fraction of about 30% (the precise value being dependent on the particular fuel management scheme and the isotopic quality of fresh plutonium in the MOX assemblies), there is a net balance between plutonium production in the UO_2 assemblies and consumption in the MOX assemblies. For higher MOX fractions the core becomes a net consumer of plutonium, while for lower fractions it is a net producer. Specific plutonium consumption is usually expressed in $kgPu^{tot}$/TWhe, the practical limits ranging between ~ -30 kg/TWhe (for zero MOX fraction) to ~ +15 kg/TWhe at 50% MOX fraction (a positive value corresponds to net destruction).

The presence of MOX fuel batches in a PWR affects the nuclear design characteristics of the core in a complicated fashion. One important effect results from the relatively low thermal neutron flux in MOX assemblies. The higher thermal absorption cross-section in MOX assemblies lowers the thermal flux relative to the fast (> 1 eV) flux, while the fast flux itself is nearly the same as in the UO_2 component. The lower thermal flux reduces the reactivity effect of control rods and of the soluble boron used to control excess reactivity early in a burn-up cycle. The net effect is to depress the reactivity worth available from the control rods and also to reduce the magnitude of the boron reactivity coefficient. Another factor is a more negative moderator temperature coefficient resulting in higher reactivity insertion at reactor cool-down. Both affect the shutdown margins available and can limit the MOX core fraction. Most existing PWRs were designed without explicitly considering the plutonium recycling option and the number of control rod cluster locations were chosen to match the requirements of all-UO_2 cores. Although in some plants there are additional control rod locations available (limited by the number of control rod drive housings in the reactor pressure vessel head), and there is usually some additional headroom available to the shutdown margin limits, most existing PWRs are restricted to between 30 and 50% MOX core loading due in part to shutdown margin restrictions. Up to now utilities have not seen any need to raise the licensing limit to higher MOX percentages (up to 100% MOX cores). The use of MOX assemblies in first cores is a possibility with new reactors. In this case the MOX assemblies would replace the UO_2 assemblies with the highest ^{235}U-enrichment and this would help avoid heavy loading of burnable poisons.

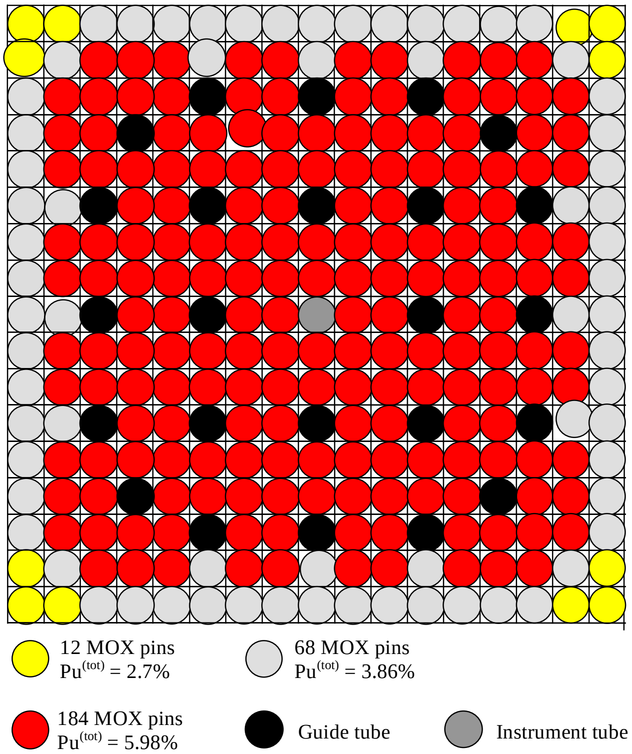
The reactivity effect of soluble boron at hot moderator conditions is typically a factor of 2 to 3 times lower in a MOX assembly than in a UO_2 assembly. This has implications for the beginning-of-cycle (BOC) boron concentrations at refuelling, cool-down and emergency shutdown. Therefore the available boron systems can limit the MOX core fraction. For emergency boron shutdown systems, the effect is that a larger mass of the mainly absorbing isotope ^{10}B may need to be injected into the core to achieve the required degree of shutdown. In a specifically designed plant, a higher boron duty could be met with larger-sized tanks with higher boron concentrations in the tanks. Retrospective modifications to existing plants are usually too expensive to be practicable, but the use of boron enriched in ^{10}B (available commercially) is an alternative approach that has been applied in some plants (using enriched boron up to ~ 30 at.% ^{10}B.)

An additional restriction on the MOX fraction may arise from the effect on the core kinetics characteristics, principally the delayed neutron fraction β_{eff}. The fissile plutonium isotopes ^{239}Pu and ^{241}Pu have smaller delayed neutron fractions than ^{235}U and depress the core-wide β_{eff}, which has implications for the transient analyses. Since ^{239}Pu has a particularly low delayed neutron fraction, this is more restrictive for MOX made with high fissile quality plutonium, such as ex-weapons plutonium.

The intermixing of UO_2 and MOX assemblies in the core has important implications for the nuclear design of the MOX assemblies and for fabrication. There is an interface effect due to the relatively low thermal neutron flux in the MOX assemblies. At the boundary between UO_2 and MOX assemblies the thermal neutron gradient sets up a current of thermal neutrons which penetrates one or two thermal diffusion lengths (~ 1 to 2 cm) into the MOX assembly. The effect is to cause localised power peaking in the outer one or two rows of fuel pins in the MOX assembly, an effect which is especially important near the corners where there is a thermal flux gradient on two sides. To avoid unacceptable local power peaking (and the risk of fuel failures), normal practice involves reducing the plutonium concentration in the outer rows of fuel pins and in the corners. The nuclear design process has to account for the reactivity effect of the various neutron poisons (boron in the moderator and fixed burnable absorbers in the fuel), which differs between UO_2 and MOX fuel and varies during the fuel burn-up cycle. Current designs of PWR MOX assemblies use three plutonium concentrations (Figure 2). While this approach gives a perfectly acceptable nuclear design, there is a cost penalty in MOX fabrication due to the need to schedule three fabrication campaigns for the different regions (which adversely affects plant throughput) and the increased quality assurance workload.

The successful licensing and operation of part MOX loadings in 37 PWRs world-wide has demonstrated that satisfactory safety cases can be constructed. Although there are fault sequences that may be adversely affected by the presence of MOX, these are balanced by other sequences where there is a beneficial effect. Modern safety cases take into account the contributions of a

Figure 2. Schematic of PWR MOX AFA 2G assembly with three different MOX regions



large number of fault sequences to the overall probability/consequence curve and the overall impact tends to be more neutral. There are no severe "cliff edge" effects with MOX which would invalidate the safety case.

The plutonium loading requirement for MOX fuel varies depending on the intended discharge burn-up and also on the isotopic quality of the plutonium. For an equilibrium condition with present day discharge burn-up of ~ 45 GWd/t, the specific plutonium loading is ~ 230 kgHM/TWhe. The plutonium content of UO_2 fuel discharged at 45 GWd/t is typically ~28 kgHM/TWhe, so that with no recycle of MOX assemblies the self-supporting MOX fraction (meaning the MOX core fraction that can be supported by recycle of plutonium generated in the UO_2 component of the *same* reactor) is approximately 0.13. Recycling the MOX assemblies can increase the self-supported MOX fraction, but only by a modest amount. The reason is that the isotopic quality of the plutonium degrades with each irradiation cycle such that the initial plutonium concentration required for burn-up equivalence increases.

Evolutionary PWR designs currently offered by reactor vendors (for example, AP600/1000, System 80+ and EPR) offer the possibility of 100% MOX loading. Relatively small modifications at the design stage, such as increasing the number of control rods and re-sizing the soluble boron system and delivery systems are effective at removing the restrictions on MOX loading that apply to the current generation of PWRs. This gives the possibility that a country or a utility might specify that a fraction of its reactors should be dedicated for MOX utilisation. These dedicated MOX PWRs would be deployed in conjunction with other PWRs fuelled with UO_2, the relative proportions of which would be determined by the available plutonium supplies and the expected future arisings. With current discharge burn-ups ~ 45 GWd/t, an equilibrium ratio of approximately seven UO_2 plants feeding one MOX plant would balance plutonium arisings and consumption.

Separating UO_2 and MOX irradiations in this way eliminates the UO_2/MOX interface power peaking effects. Therefore there would be no need for radial gradation of the plutonium concentrations and a single plutonium "enrichment" would suffice for the whole fuel assembly. The main benefit is thereby to simplify fabrication and reduce fabrication costs. There might be other benefits because core operation could be tailored specifically around MOX fuel to benefit from the behavioural characteristics of MOX, such as the slower rate of change of reactivity with burn-up, which would give the possibility of very long cycles or of higher discharge burn-ups. For reactivity control of high plutonium content reload MOX assemblies, burnable absorbers may be required. More radical benefits could be gained by adjusting the moderator/fuel ratio to the optimum for MOX (as opposed to the value ~ 2.0 of current PWR assembly designs, which is the optimum for UO_2); this is considered separately in Sections 3.6 and 3.7.

The option of dedicated MOX burners has not been researched in full and there are many details that remain to be evaluated. For example, there would be an economic penalty in operating the first cycle with 100% MOX fuel, as this implies discharge of the first batch of MOX assemblies with relatively low burn-ups (and of incurring the full MOX fabrication cost for a low burn-up). This might be mitigated by the use of a very long first cycle. An alternative might be to use mixed UO_2/MOX cores for the first few cycles until an all-MOX equilibrium is eventually reached. Another consideration is the issue of making the transition from an all-MOX core to a UO_2 core if the available plutonium was expected to be consumed within the lifetime of the reactor. In spite of such outstanding questions, dedicated MOX-burning PWRs can be regarded as an option that presents a low technical risk. This option would be well suited to the situation where a country has a requirement to utilise existing plutonium stocks. A 100% MOX-loaded core with MOX assemblies of the standard design would achieve a net plutonium consumption of approximately +60 kg/TWhe.

Isotopic degradation of the plutonium in conventional MOX assemblies restricts the number of times it can be recycled. A WPPR study [1] shows that equivalent third recycle MOX fuel would then require in excess of 12 w/o initial plutonium content in the fresh fuel, even though the scenario considered involves co-reprocessing of UO_2 and MOX assemblies in the ratio 3:1 in which they arise in the scenario (which helps reduce the degradation of plutonium isotopic quality in each recycle). This is the threshold beyond which moderator void reactivity coefficient of the MOX fuel may become positive, which would preclude its loading in the core. Utilisation of third- or later-generation recycle plutonium at less than the self-generated recycle level would be technically feasible, but this might not balance consumption and arisings.

Recycling plutonium as LWR MOX could be regarded as beneficial from the perspective of proliferation. While the overall mass of plutonium is only reduced by a modest amount (around 30% at present-day discharge burn-ups), the fissile plutonium isotopes (239 and 241) are reduced by more and the isotopic quality is significantly degraded after one recycle. MOX assemblies display a higher degree of self-protection than UO_2 assemblies, as the radiation field is more intense and persists longer.

The United States is actively pursuing a plutonium disposition programme aimed at irradiating excess weapons plutonium in existing PWRs. The aim is to irradiate surplus weapons plutonium in MOX assemblies in PWRs, thereby degrading the isotopic composition of the plutonium to LWR standard and reducing its suitability for use in weapons. Plans for facilities to convert the plutonium metal "pits" to oxide and fabricate MOX fuel assemblies are being implemented.

Compared with a once-through UO_2 fuel cycle, in which UO_2 fuel assemblies are conditioned for eventual geological disposal, MOX recycle strategies have a different profile of waste generation. A once-through cycle essentially generates principally high-level, heat-generating wastes, with relatively less significant quantities of medium- and low-activity waste from reactor operations and from other fuel cycle operations. MOX strategies are characterised by smaller volumes of high-level waste, as a result of the concentration of minor actinides and fission products in reprocessing. However, reprocessing generates relatively larger volumes of intermediate- and low-level wastes. Depending on whether spent MOX fuel is reprocessed or designated for direct disposal, there may also be spent fuel arisings with MOX fuel cycles. The relative volumes of the various waste forms generated in once-through and MOX fuel cycles vary depending on the specific details of the fuel cycle, such as the reactor and fuel design, discharge burn-up, the number of recycles and so on. Although such scenario-specific analyses have been reported in the literature, they are too detailed for this review. For our purposes, it is sufficient to note that quantitatively MOX recycle scenarios which are assumed *not* to be followed by fast reactor deployment are found to be marginally beneficial with respect to waste arisings, largely because the mass of plutonium eventually committed to geological disposal is lower. However, MOX scenarios which are followed by fast reactors entail the almost complete elimination of plutonium in geological wastes and provide large benefits with respect to waste arisings. Therefore MOX recycle satisfies a fundamental environmental principle – that MOX causes no environmental harm even if there is no transition to fast reactors. All options should satisfy this important requirement.

3.2 MOX in VVERs

The VVER is a pressurised water reactor that was developed in the former Soviet Union and is essentially the same as the standard western PWR, but differs in detail, particularly the use of a triangular pin lattice and hexagonal fuel assembly lattice in place of a square lattice geometry. There are two distinct types of VVER core. The older VVER-440 reactors use a shrouded assembly design with integral control rod, while the VVER-1000 designs use an un-shrouded design which incorporates control rod guide tubes in the same manner as western PWRs.

An important distinction between MOX usage in western PWRs is that there is currently no MOX experience in VVERs. The reason is that serious interest in the utilisation of MOX fuels in VVERs only arose relatively recently as a response to bilateral agreements on the disposition of weapons-grade plutonium. A programme of research and development is under way with the intention of establishing MOX utilisation in VVER-1000 reactors within a few years. At an early stage of the work programmes, technical studies performed in relation to bilateral US/Russian and trilateral French/German/Russian agreements have established that VVERs can burn MOX in the same way as western PWRs [2] and that most of the physics considerations are the same. However, some specific issues did arise because of the intention to use weapons-grade plutonium.

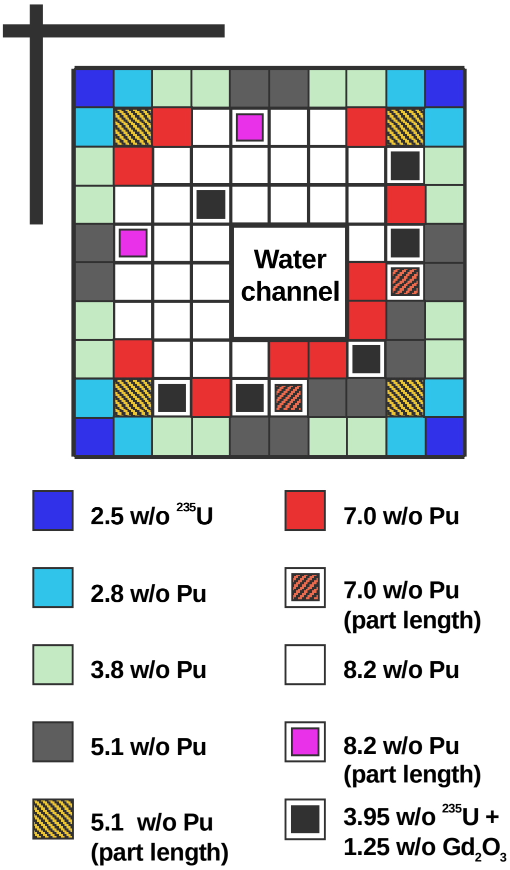
The weapons plutonium disposition programme in Russian VVERs is being pursued in parallel with that to irradiate excess weapons plutonium in US PWRs. The aim is to achieve parity between weapons plutonium disposition in the US and Russia. A Russian/US collaboration is helping to provide facilities and assist with ensuring the necessary expertise is in place, recognising that experience of MOX in VVERs lags behind that of MOX in European and Japanese PWRs.

The current designs are based on international experience with a 30% MOX core fraction in PWRs and with the MOX assemblies resident for three 12-month cycles. The MOX assemblies contain 3-4 w/o Pu. There will be three lead test assemblies inserted into Balakovo-4 in 2007, with the MOX mission starting several years later.

3.3 MOX in BWRs

In many respects the technical issues concerning MOX fuels in BWRs are similar to those which arise in PWRs and VVERs, but there are some important distinctions. These arise from the very different fuel element used in BWRs and the fact that the moderator/coolant water is a two-phase mix of steam and water. BWRs use a fuel channel element in order that un-voided moderator can be maintained in the gaps between assemblies, and they also incorporate internal water channels (Figure 3) or water rods (also containing un-voided moderator). The heterogeneity effects caused by the presence of inter-assembly water gaps and internal water channels or water rods necessitates the use of several enrichment regions even in a UO_2 assembly to reduce power peaking. BWR MOX assemblies need to use fuel rods with several different plutonium concentrations for the same reason. An important distinction between PWRs and VVERs on the one hand and BWRs on the other, is that BWRs do not use soluble boron in the moderator for reactivity control in normal operation; in BWRs the role of soluble boron for reactivity control is largely replaced by having UO_2/Gd_2O_3 (urania/gadolinia) burnable poison rods within each fuel element. The need for gadolinia burnable poison also applies to MOX fuel elements, and although it is possible in principle to incorporate gadolinia within MOX fuel rods, normal practice is to use urania/gadolinia rods (the reason being that an extensive qualification of such fuel would be needed and fabrication of fuel rods containing gadolinia requires a dedicated fabrication facility to avoid cross-contamination). The requirement to minimise within-element power peaking leads to fuel assembly designs with multiple plutonium concentrations, part length rods and urania/gadolinia rods, though the number of distinct fuel rod types is not much larger than that of UO_2 assemblies.

Figure 3. Schematic of BWR Atrium 10 fuel assembly



In many ways BWRs offer more possibilities for design optimisation in MOX assemblies than do PWRs. The reason is that each BWR fuel assembly forms an isolated coolant channel because of the fuel assembly shroud which fully encloses the assembly in the x-y plane. The large water gaps between assemblies tend to reduce the importance of the interface effects between assemblies as the moderator acts to some extent as a buffer between assemblies. Because of the shrouded design, there is always the possibility of optimising the design of MOX fuel assemblies with a different moderator to fuel ratio than that of uranium assemblies (a higher moderator to fuel ratio for a MOX assembly would make it more compatible with uranium fuel). However, such an approach would mean that a MOX assembly would be lighter than a uranium assembly, which would raise issues with regard to mechanical handling and levitation in the coolant flow.

Normal practice is that design margins are kept low in core design to avoid losses in reactivity which can be used to prolong the cycle length. The possibility of flexible gadolinia design (as is also typical in UO_2 BWR assemblies) offers some margin for shutdown reactivity, even for a large MOX fraction in the core. The requirements for the liquid boron shutdown system in BWRs are normally higher for a MOX core than for a uranium core, but for MOX fractions up to 50% the differences are usually not very large, such that an upgrade of the liquid boron shutdown system is usually not necessary.

While the presence of UO_2/Gd rods prevents a true 100% MOX core loading, it has been reported [3,4] that BWR cores are potentially capable of supporting full loadings of MOX assemblies (though these studies applied to relatively old 8×8 assembly designs with a relatively low plutonium loading). Studies of BWR full MOX cores [5,6] have been carried out for an assembly average burn-up of 40 to 45 GWd/t with 9×9 fuel assemblies. These studies show that 100% MOX core loading is feasible even for a high burn-up in terms of core performance.

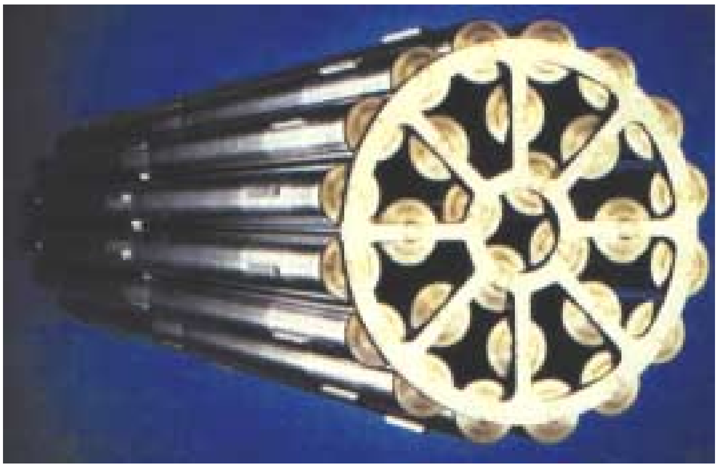
A recent OECD/NEA report [7] describes the nuclear physics issues associated with the use of MOX in BWRs in detail. The waste arisings of BWR fuel cycles are very similar to those of PWR fuel cycles and the comments regarding waste arisings in PWR fuel cycles made at the end of Section 3.1 (above) apply equally to BWRs.

3.4 Plutonium fuels in heavy-water reactors (HWRs)

Because heavy water is a very efficient moderator, heavy-water reactors (HWRs) are able to use a wide range of fuel types. As early as 1972, MOX irradiation experience was gained in the pressure vessel type heavy-water reactor MZFR at Karlsruhe in Germany. There is the possibility of using plutonium without penalising local power peaking [8].

Although there is little operational experience of utilising plutonium fuels in HWRs, a considerable amount of theoretical work has been carried out, and some small-scale experimental work in the context of plutonium disposition. HWR reactors normally use natural or slightly enriched UO_2 fuel bundles. The standard fuel bundle of a CANDU reactor comprises natural enriched fuel rods (referred to as fuel elements in CANDU parlance, see Figure 4) and is designed to attain discharge burn-ups up to 9 GWd/t with natural uranium fuel. The new Advanced CANDU Reactor concept (ACR-700) in fact has light water cooling, a much tighter pitch for the pressure/channel tubes and a smaller heavy-water moderator volume.

Figure 4. CANDU 28 element fuel bundle



For CANDU reactors AECL has developed a new bundle design, called CANFLEX, that can use low-enriched UO_2 rods (up to 2.0 w/o) or equivalent MOX rods. The CANFLEX bundle contains 43 fuel elements; there is a central element surrounded by three rings, the outer ring consisting of smaller-diameter fuel rods which reduces the tendency of the outermost ring to run at high powers because of its proximity to the moderator. The CANFLEX bundle is designed to attain peak element burn-ups up to 35 GWd/t [9]. Some small-scale experimental work of MOX fuel bundles has been carried out in the NRU test reactor in Canada [10], as part of the PARALLEX project. Under this multilateral programme, weapons-derived plutonium was irradiated in 37 element bundles. To demonstrate the irradiation behaviour of MOX in CANDU, post-irradiation examination (PIE) will be carried out as part of the programme. A longer-term option would be to use an inert matrix assembly for burning plutonium in CANDUs, the favoured material being SiC [10]. The use of an inert matrix would reduce the amount of plutonium committed to eventual geological disposal.

A joint Korean/Canadian development activity is currently pursuing the DUPIC concept [12]. DUPIC involves dry reprocessing of PWR fuel assemblies (after 10 to 20 years cooling), removal of gaseous FPs, and re-fabricating the uranium and plutonium into CANDU fuel bundles (without separating the uranium and plutonium at any point). The CANDU fuel bundles would be sent for direct disposal after irradiation. DUPIC is therefore a single recycle strategy in which the residual ^{235}U and plutonium content of PWR fuel can be re-used in CANDU. The excellent moderation characteristics of the CANDU core are such that residual fissile content of spent PWR fuel is sufficient to drive the

CANDU fuel cycle for an additional 18-21 GWd/tHM burn-up after the PWR stage. By this means, the mass of plutonium committed to final repository per unit energy output is decreased.

To date, all CANDU reactors continue to operate with a once-through fuel cycle. The volumes of high-level waste (in the form of spent fuel) are larger than for LWRs because of the low burn-up attained by the natural enriched bundles. The technical possibility of using CANDU reactors for plutonium burning could be an attractive one, as the plutonium mass would be reduced following irradiation in a CANDU. A comparison of waste arisings in a plutonium recycle scenario in a CANDU is likely to be comparable to those of LWR MOX scenarios.

3.5 Advanced plutonium fuel assembly designs for PWRs

This section considers a number of advanced fuel design concepts that have been investigated for PWRs that could be used as substitutes for conventional UO_2 and MOX assemblies. Although most of these advanced designs have been subject to detailed theoretical investigation, none has so far been carried further forward; the next stage would be to commission experimental work in research reactors to confirm the nuclear design calculations and, for some of the designs, thermal-hydraulic and fuel performance testing in commercial reactors.

In the context of medium-term plutonium utilisation it is possible to distinguish between two aspects:

1. Optimised utilisation and preservation of plutonium.

2. Destruction of plutonium.

Both aspects can be addressed with the following designs.

3.5.1 MOX/EUS

The MOX Enriched Uranium Support concept (MOX/EUS, also known as the MIX concept) is a mixed-oxide assembly design for PWRs in which the PuO_2 is only a secondary contributor to fission rate, the bulk of the fissions being from enriched UO_2. Because the PuO_2 only plays a secondary role, the nuclear performance of the assembly is close to that of a normal enriched UO_2 assembly. This contrasts with a conventional MOX assembly, where depleted or natural UO_2 is used as the carrier and PuO_2 accounts for the majority of the fissions. CEA developed the MOX/EUS design as a potential method of establishing multiple recycle of plutonium in PWRs without the technical restrictions which affect conventional MOX fuel in this situation.

The MOX/EUS assembly mechanical design is identical to that of a conventional UO_2 assembly, so that there are no mechanical or thermal-hydraulic compatibility issues. A typical application would comprise 1.3 w/o of Pu in uranium carrier enriched to 3.9 w/o. This would be equivalent in terms of lifetime reactivity to a UO_2 assembly of 4.7 w/o. Variations in the plutonium isotopic composition need to be compensated by appropriate adjustments to the enrichment of the UO_2 carrier. Also, the plutonium content needs to be limited to avoid too-extreme reactivity coefficients. These relative proportions would allow all the plutonium arisings from PWR reprocessing to be re-used. Since the nuclear characteristics of MOX/EUS are dominated by the uranium, there is relatively little sensitivity to the isotopic quality of the plutonium and the reactivity coefficients are very favourable. Moreover, multiple recycle of the plutonium is technically viable and can be achieved with low technical risk. A MOX/EUS core would achieve a significant plutonium reduction of ~ +60 kg/TWhe [13].

Recent studies [14] of plutonium multi-recycling in a park of PWRs, initially using only 4.0 w/o enriched UO_2 fuel for a final burn-up of 43.5 GWd/tHM, have considered the MOX/EUS concept. The average plutonium content of the MOX assemblies is kept constant at 8.0 w/o, in order to avoid a positive moderator void reactivity coefficient in the reactor cores. In the first plutonium recycle, the uranium in the MIX fuel can be natural enriched; for subsequent recycles, the uranium enrichment is gradually increased to compensate for the reactivity reduction due to the degradation of the isotopic quality of the plutonium. Theoretically, an equilibrium is reached after about 10 recyclings and the asymptotic fraction of MIX assemblies in the reactor park is approximately 40%, while the asymptotic ^{235}U enrichment in the MIX assemblies is 3.6 w/o.

The same concept can also be applied to the recycling of americium. The americium is contained in a number of target rods in the assemblies, with an equilibrium of approximately 50% MIX assemblies in the reactor park and an increased ^{235}U enrichment of 4.8 w/o to compensate for neutron absorption in the americium target rods.

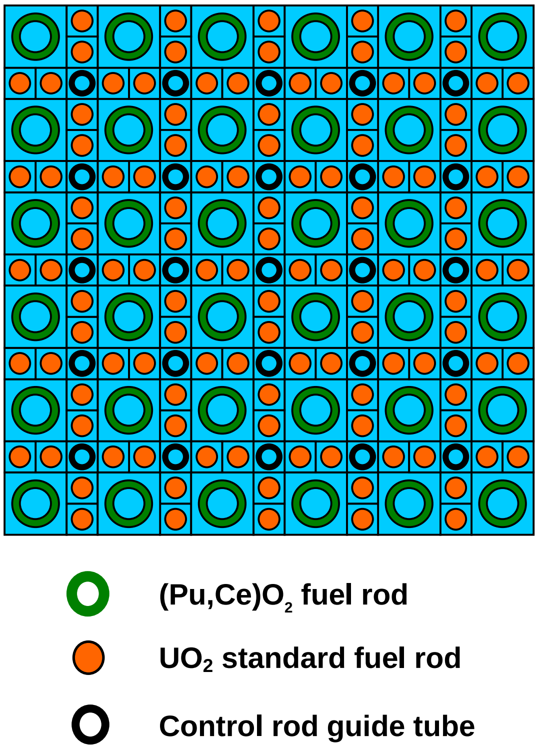
A disadvantage of MOX/EUS is that fabrication would need to be subject to the same safety standards of containment as conventional MOX fuel, while the total throughput of material would be several times higher. Consequently, higher MOX fabrication costs would be incurred and in addition to the costs of procuring uranium ore, conversion and enrichment would need to be included. Moreover, the capacity of the MOX/EUS fabrication plant would need to be two to three times larger than for a conventional MOX recycle strategy (to stabilise plutonium stocks the factor would need to be about 2.5). However, it is envisaged that as MOX fabrication technology matures, costs would decrease, possibly allowing MOX/EUS to become economically competitive. Safeguards for transport, storage and handling would have to comply with MOX standards for all or a significantly larger number of assemblies than in the standard recycling mode.

Compared with a single recycle strategy in conventional MOX assemblies, a multiple recycle MOX/EUS significantly reduces the radiotoxicity flow to the high-level waste stream, because only a small fraction of the plutonium (representing reprocessing plant losses) goes into high-level waste. At each recycle, the mass of plutonium decreases through fissions and the plutonium radiotoxicity per GWye energy output decreases progressively. In the event that a MOX/EUS strategy is eventually replaced by plutonium recycle in fast reactors, with the plutonium being used eventually to feed fast reactors, a large reduction in radiotoxicity would be realised. In the event that a fast reactor programme does not materialise, there remains a modest radiotoxicity benefit because of the plutonium which is destroyed by fission in MOX/EUS. Thus, in either eventuality MOX/EUS provides a radiotoxicity benefit compared with single recycle in conventional MOX. Volumes of low- and intermediate-level waste arising from reprocessing operations are essentially the same as for conventional MOX.

3.5.2 APA

APA is an acronym for Advanced Plutonium Assembly, developed by the CEA as an alternative to MOX fuels in PWRs [15] It is a dual design incorporating both UO_2 fuel rods and plutonium-bearing fuel rods within a single assembly. The design of the plutonium-bearing rods is specifically optimised for plutonium consumption, while the dual design avoids uranium/plutonium interface effects between assemblies, since all the assemblies in the core are identical. A schematic of the APA design can be seen in Figure 5; it is composed of 36 large-diameter plutonium-bearing rods and 120 UO_2 rods. While the UO_2 rods are completely conventional in design, the plutonium-bearing rods are very innovative. The plutonium rods contain annular plutonium $(Pu,Ce)O_2$ pellets, with a large external diameter occupying four normal fuel cells. The central annulus encloses an inner clad within which flows coolant water. This increases

Figure 5. Schematic of the PWR APA fuel assembly



the local moderator/fuel ratio closer to the reactivity optimum for plutonium. The absence of uranium in the plutonium-bearing rods decreases the rate at which fresh plutonium is created by ^{238}U fertile captures and leads to a high net plutonium consumption rate of approximately +75 kg/TWhe. Moreover, the isotopic quality of the discharged plutonium is very degraded, with a very low fraction of ^{239}Pu, which could be cited as a benefit with respect to proliferation resistance. The reactivity coefficients of the APA design are acceptable, though some optimisation might be required. The high moderation ratio that applies locally to the plutonium fuel rods has the benefit of reducing minor actinide production.

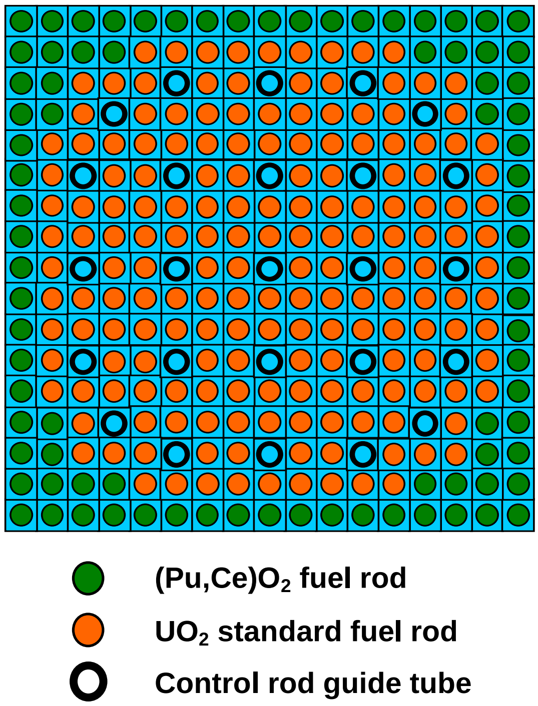
The APA design involves increased technical risk because of several factors. First of all, the mechanical design of the assembly is different from current ones; this implies that the coolant flow and heat transfer characteristics will all be different, thus requiring that new engineering correlations be established in test rigs. The mechanical design will also need proving for mechanical strength and stability, including vibrational stability. Furthermore, the positioning of the control rod guide thimbles is not compatible with the core internals of the current generation of PWRs. The principal uncertainty, however, lies in the innovative $(Pu,Ce)O_2$ pellet fuel material and the innovative geometry of the fuel. This will require extensive performance testing, including irradiation tests and lead assembly loadings that is not without technical risk. The behaviour of the novel fuel under accident conditions will also require research and development. An additional consideration is whether the inert matrix fuel is compatible with reprocessing. Commercial introduction would require fabrication lines which have to be provided. The safeguards standards for transport, storage and handling would apply for all or a significantly larger number of assemblies than in the standard recycling mode. These and other technical issues, such as mechanical strength, complex fabrication of the annular pellets, etc., were important reasons why the APA design has now been abandoned in its original form and the CEA is now moving toward the so-called APA-DUPLEX concept (Figure 6), in which the large annular $(Pu,Ce)O_2$ rods are replaced by smaller, solid $(Pu,Ce)O_2$ rods.

Compared with conventional MOX, the high plutonium consumption of APA reduces the radiotoxic inventory associated with plutonium in the active recycle loop, while the inventories of other nuclides which are important contributors to radiotoxicity are only slightly affected. For all APA scenarios, single recycle or multiple recycle, there is a modest benefit on radiotoxic inventory. The volume of low- and intermediate-level waste arising from reprocessing operations is essentially the same as for conventional MOX.

3.5.3 PLUTON

The PLUTON assembly [16] is a non-uranic plutonium assembly designed to achieve the maximum specific plutonium consumption. The plutonium is incorporated in the form of PuO_2 in an inert matrix, which eliminates fresh plutonium production through ^{238}U fertile captures. The initial concept envisages a mix of plutonia/alumina PuO_2/Al_2O_3 rods and plutonia/alumina/gadolinia $PuO_2/Al_2O_3/Gd_2O_3$ poison rods, 228 in total. To offset the deleterious impact of eliminating ^{238}U on the reactivity feedback coefficients, 36 ZrH_2 moderator rods are incorporated in the assembly. The hydrogen in the moderator rods counteracts the otherwise very deleterious moderator void effect and also improves the fuel temperature (Doppler) feedback coefficient. However, even with the moderator rods, the PLUTON assembly has an unsatisfactory void reactivity coefficient. The PLUTON assembly realises a net plutonium consumption of +134 kg/TWhe, which is essentially the theoretical upper limit. As with APA, an additional

Figure 6. Schematic of the PWR APA-DUPLEX fuel assembly

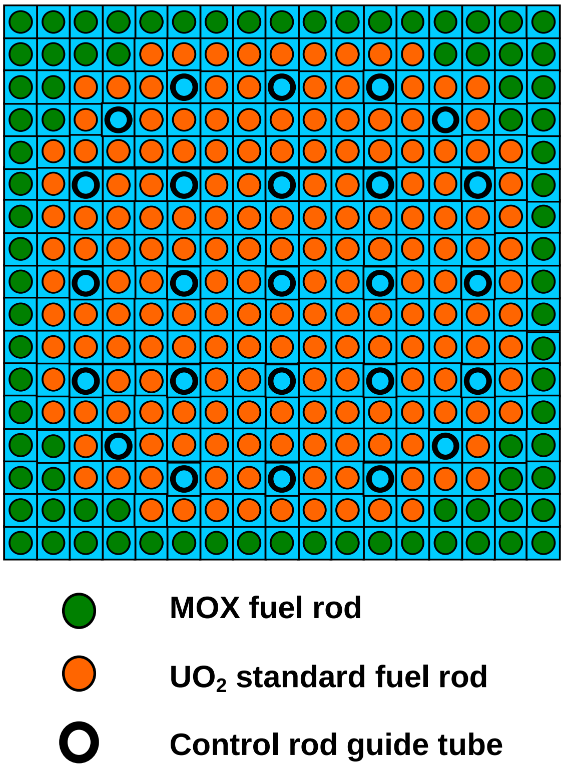


consideration is whether the inert matrix fuel is compatible with reprocessing and the high plutonium consumption means that the same comments apply to the impact on total radiotoxic inventory.

3.5.4 CORAIL

The CORAIL assembly [13] combines UO_2 and MOX rods in a conventional PWR mechanical design. Figure 7 shows a schematic of the assembly which has a central region consisting of 180 conventional enriched UO_2 rods, while the outer region comprises 84 MOX rods. The assembly layout is identical to APA-DUPLEX (Figure 6) except that the MOX rods replace the $(Pu,Ce)O_2$ rods. The UO_2 and MOX rods both have the same outer diameter, while the spacing is identical to conventional 17×17 PWR assemblies in use today. A typical application would use UO_2 rods enriched to 5.2 w/o, while the MOX rods would contain 11.3 w/o Pu^{tot}. Since the MOX rods are mostly in very close proximity to the UO_2 rods (within 1 or 2 thermal diffusion lengths), the MOX rods are to a large extent driven by the thermal flux of the UO_2 rods. This

Figure 7. Schematic of the PWR CORAIL fuel assembly



CORAIL assembly is therefore somewhat more tolerant of poor plutonium isotopic quality than conventional MOX, and allows multiple recycle. One of the advantages of the design is that it is entirely compatible with existing cores and requires no new technology to be developed. As with APA, it would be possible to load an entire core with CORAIL assemblies. The interface effect between UO_2 and MOX rods is an issue which occurs *within* the assembly and is managed by careful choice of the relative enrichments of the UO_2 and MOX rods and their positioning. In particular, only one plutonium enrichment needs to be fabricated in the MOX fabrication plant, which helps to increase plant throughputs and reduce fabrication costs. Reactivity coefficients fall within an acceptable range. Overall the CORAIL assembly displays neutral plutonium consumption; plutonium production in the UO_2 is close to balancing net plutonium consumption in the MOX rods. As for other designs, MOX safeguards standards for transport, storage and handling would apply for all or a significantly larger number of assemblies than in the standard recycling mode.

In a multiple recycle scenario CORAIL provides a significantly more reduced radiotoxicity than a single recycle with conventional MOX. Volumes of low- and intermediate-level waste arising from reprocessing operations are essentially the same as for conventional MOX.

3.5.5 Thorium-plutonium fuels

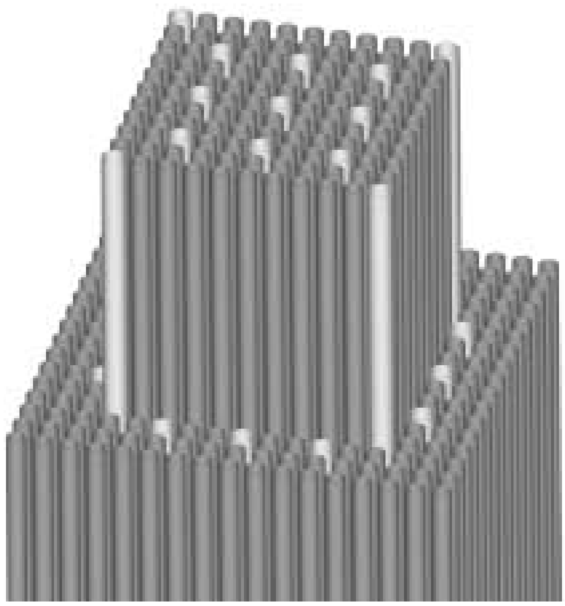
Thorium-plutonium fuel assembly designs have been developed for PWRs as an alternative means of consuming plutonium. The thorium fuel cycle uses ^{232}Th as a fertile material to breed ^{233}U, which is a very effective fissile material in a thermal spectrum. A source of neutrons is needed to start the thorium fuel cycle, and this is usually done using enriched UO_2 or PuO_2. All aspects of the thorium fuel cycle in PWR have been investigated as part of a Brazilian/German co-operation [17,18]. One approach uses a conventional PWR fuel assembly containing an admixture of enriched UO_2 and ThO_2. Another part of the study examined the impact on nuclear core design, fuel technology, thermal fuel rod design and performance predictions as well as irradiation testing and investigations into the back end of the fuel cycle. Depending on the supposed discharge burn-up, plutonium reduction rates of 75% of the initial plutonium inventory are achievable in a Th/Pu equilibrium cycle. The transfer from an all-UO_2 core to a Th/Pu equilibrium cycle and back is feasible with an adequate neutronic fuel assembly design that accounts for the interface with the adjacent uranium assemblies. The ^{233}U produced from the thorium increases the attainable burn-up for a given initial plutonium content. Another advantage is that plutonium production by fertile captures in ^{238}U is reduced, as is the production of minor actinides.

In the HEU-cycle thoria fuel may require reprocessing to realise substantial savings in fuel cycle costs. Reprocessing of spent (Th,U)O fuel is in principle feasible using the THOREX process. The inclusion of plutonium will require modifications to the reprocessing flow sheets. However, all thorium-based oxide fuels suffer from the disadvantage during reprocessing of the need to use hydrofluoric acid to obtain sufficient solubility. Once-through fuel cycles could take advantage of this high insolubility for direct final disposal.

Early irradiation experience of thorium-plutonium fuel was obtained in the LINGEN BWR (fuel was first inserted in 1970). A current EU programme "Thorium Cycle: Development Steps for PWR and ADS Application", aims to build on this early experience. This programme involves the irradiation of thorium-plutonium rodlets in a MOX assembly in the Obrigheim NPP KWO intended to reach burn-up values of up to 38 GWd/t.

A novel alternative is the RTF system [19], in which each fuel assembly is split into a seed region containing UO_2 rods at up to 20 w/o enrichment. Surrounding the seed region is a ThO_2/UO_2 blanket assembly which provides the fertile conversion to ^{233}U (Figure 8). Separating the seed and blankets allows the two components to be replaced at different time intervals, so that the total residence time of the seed matches the burn-out rate of the ^{235}U, while the blanket region has a longer residence time matched to the rate at which fertile captures takes place. A small amount of UO_2 is included in the blanket rods, in order to ensure that the ^{233}U is isotopically mixed with ^{238}U, thereby reducing the proliferation risk of ^{233}U. The mechanical design of the seed/blanket combination is compatible with existing PWR assemblies and the control rod guide thimbles are compatible with existing cores. The medium-enriched UO_2 rods in the seed region can in principle be replaced by PuO_2/UO_2 MOX rods, so that the RTR-18 design could equally well be used as a plutonium burner that, because of the small mass of ^{238}U, generates less fresh plutonium than conventional MOX assemblies and thereby achieves a higher net destruction of plutonium. A variant of the RTR-18 design for VVER-1000 reactors has also been developed. It is essentially the same apart from the use of a hexagonal lattice and of Pu-zirconium metal fuel for the driver region.

Figure 8. Perspective view of PWR RTR seed-blanket fuel assembly, with enriched UO_2 seed partially withdrawn from ThO_2 blanket (control rod guide thimbles shown in light grey)



The need for 20.0 w/o ^{235}U for the seed region, compared with 4 to 5 w/o for conventional uranium fuel assemblies, implies a higher cost per assembly for enrichment. However, this is countered by a lower mass of enriched fuel per assembly and an increased discharge burn-up relative to the initial ^{235}U loading; thus, the RTF can be claimed to be economically competitive.

Some advantages of thorium use with respect to fuel cycle and waste issues have recently been pointed out in an European Community study [20]. These are:

- The reduction in the long-lived radiotoxicity of mining waste compared with uranium is expected to be relatively small.

- The fabrication of Th/Pu-MOX fuels is comparable with U/Pu-MOX fabrication methods provided that fresh thorium (i.e. not recycled) is used and similar radiological containment and shielding arrangements are required.

- The radiotoxicity of PWR waste in the first 10 000 years is significantly reduced in the once-through thorium cycle in comparison with the once-through uranium cycle. This is because only trace quantities of elements heavier than ^{233}U are produced in the thorium blanket.

- Compared with the U/Pu cycle, the Th/Pu cycle has a higher capacity for burning plutonium. Indeed, a single plutonium recycle in Th/Pu is efficient at destroying plutonium.

The low solubility of Th/Pu fuel has advantages in terms of the stability of fuel intended for direct disposal. It is also a benefit in that it increases the proliferation resistance of the fuel cycle by making the residual plutonium harder to access. Compared with conventional MOX recycle, thorium-plutonium fuels provide significant reductions in radiotoxicity.

3.5.6 Plutonium-erbium-zirconium oxide fuels

The Paul Scherrer Institute (PSI) in Switzerland is actively investigating plutonium-erbium-zirconium oxide as IMF for the once-through (single recycle) disposition of plutonium [21,22]. The fuel material is PuO_2, as in MOX fuel. The inert carrier is yttria-stablised zirconia (a mix of ZrO_2 and Y_2O_3); its high chemical inertness, along with the poor isotopics of the discharged plutonium, provides strong justification for direct geological disposal in the context of once-through IMF deployment. Erbia (Er_2O_3) burnable poison is used to reduce the reactivity swing of the fuel (i.e. the difference in reactivity between the beginning and end of the fuel cycle) and to help compensate for the lack of adequate fuel temperature (Doppler) feedback in the absence of ^{238}U. Such fuel is able to achieve roughly twice the plutonium destruction rate of conventional MOX fuel. It is envisaged that the IMF could be virtually a direct replacement for MOX fuel within existing assembly designs, though there may need to be adaptations, such as the use of annular pellets to compensate for the relatively low thermal conductivity of the inert matrix compared with UO_2 or MOX.

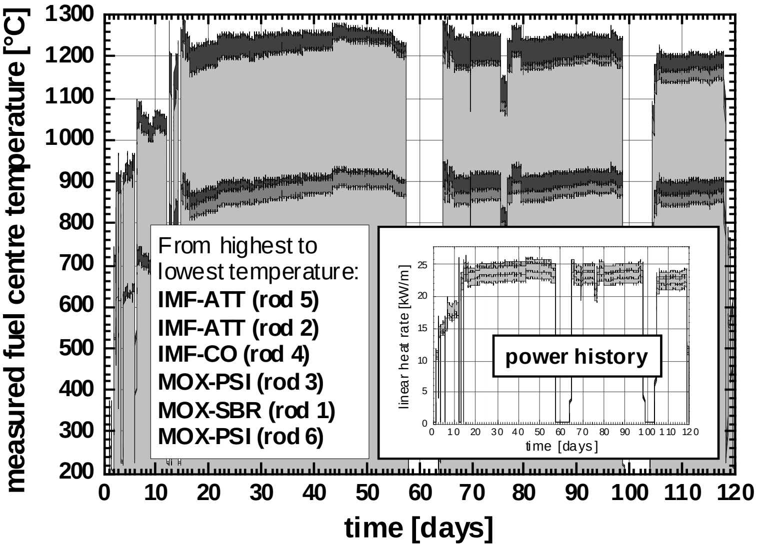
Analytical studies have shown that a 1/8 core loading with the IMF would be sufficient to achieve self-generated recycle. Compared with conventional part-MOX loaded cores, the shutdown margin would be increased. For cores with large fractions of IMF assemblies, e.g. 100% Pu fuel loadings, however, other safety-related parameters such as the effective delayed neutron fraction and Doppler coefficient would be less favourable than for the corresponding MOX case.

In order to validate the encouraging results obtained in the analytical studies, two types of experimental investigations have recently been initiated employing plutonium-erbium-zirconium oxide IMF rodlets fabricated at PSI. These are (a) neutronics measurements in the PROTEUS critical facility for validating the reactor physics methods and data applied, and (b) irradiation testing in the OECD Halden test reactor in Norway. Results obtained to date in the former case have indicated that similar accuracies as for MOX fuel can be expected in neutronics calculations for the fresh IMF.

The irradiation testing at Halden is scheduled to last until 2005. Results from the first irradiation cycle, carried out at moderate power, have indicated quite satisfactory behaviour for the IMF, as illustrated in Figure 9. The figure shows that the centre temperatures are all quite stable, the higher values for the IMF reflecting its lower thermal conductivity. Valuable data related to fission product gas release mechanisms are expected from the full-power irradiation cycles currently under way. These are considered to be crucial for the planning of a possible test for the IMF in an actual nuclear power plant environment.

It should be noted that, because of the high plutonium destruction rate, inert matrix fuels achieve a significant reduction in radiotoxicity relative to conventional MOX. Volumes of low- and intermediate-level waste arising from reprocessing operations are essentially the same as for conventional MOX.

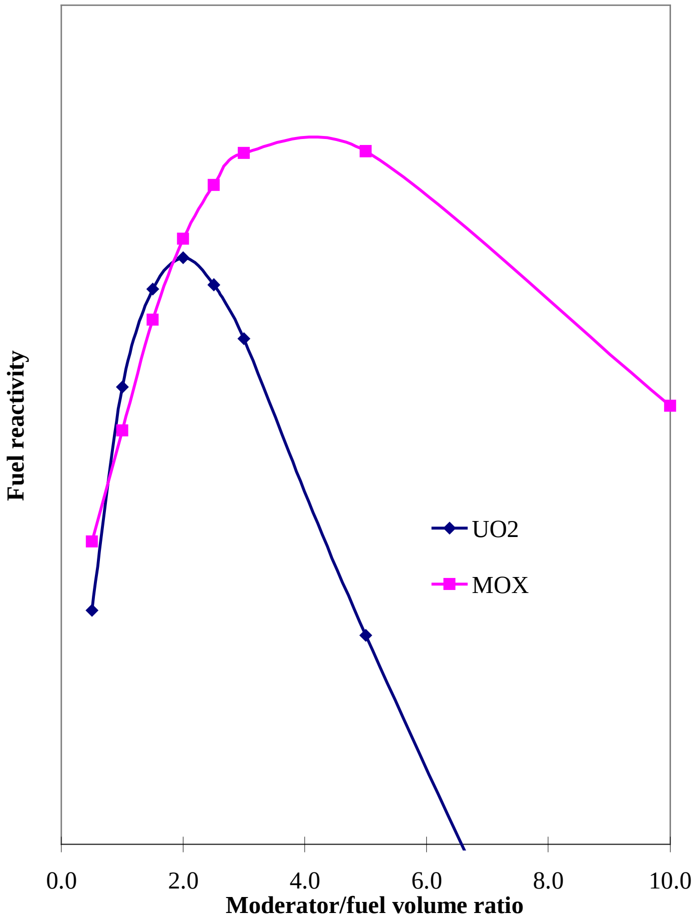
Figure 9. Measured fuel centreline temperatures for Pu-Er-Zr oxide (IMF) and MOX fuel rodlets vs. time during the first cycle of the irradiation testing at Halden



3.6 High-moderation LWRs

The current generation of PWR assemblies is designed to have a moderator/fuel volume ratio a little less than 2:1, which is optimum for maximum reactivity with UO_2 fuel (Figure 10). A UO_2 assembly is under-moderated and increasing the ratio improves the thermal utilisation of the fuel. However, there is an opposing effect due to absorption in the moderator, which increases as the fuel/moderator ratio increases, because the relative probability of a neutron finding itself in the moderator increases when there is a greater volume of moderator in the system. Absorption in the moderator thus reduces the number of neutrons available to cause fissions. At around 2:1 these two effects are in balance for uranium fuel and reactivity is an optimum. The optimum moderator/fuel ratio corresponds to the minimum enrichment requirement, which is an economic benefit since both the separative work units and the number of kg of natural uranium ore per kg of enriched fuel are minimised. In practice, for reasons related to maintaining a negative moderator temperature coefficient, actual assembly designs choose a point slightly on the under-moderated side of the optimum. In this way, a reduction in moderation due to heat-up of the moderator reduces reactivity, as required to maintain negative feedbacks.

Figure 10. Schematic variation of k-infinity versus moderator/fuel ratio for PWR UO_2 and MOX assemblies



For MOX fuels the optimum balance between absorption and moderation is very different, due to the much-higher thermal absorption of plutonium compared with uranium. The optimum is nearer 4:1 (Figure 10), which corresponds to a larger lattice spacing for the same rod diameter or a smaller rod diameter for the same lattice spacing (alternatively, the same could be achieved with hollow pellets, stabilised by a low absorbing and moderation material). A conventional MOX assembly, with a moderator/fuel ratio near 2.0, is therefore considerably under-moderated for plutonium, and this is part of the reason why relatively high plutonium enrichments are sometimes required in current MOX designs. The result is a relatively large thermal cross-section that is balanced by a low thermal flux, which in turn causes the reaction rate of the plutonium to be low. Thus the rate at which the plutonium isotopes fission and undergo capture due to thermal neutrons is depressed, while at the same time the number of neutrons in the resonance region is virtually the same as in a UO_2 assembly. This leads to a relatively high rate of fertile neutron captures on ^{238}U compared with a low plutonium reaction rate, giving a low net plutonium consumption.

A MOX assembly designed for a moderator/fuel ratio closer to the 4:1 optimum would provide considerably improved nuclear design performance. The initial plutonium concentration is reduced, leading to an increase in the thermal flux and a higher reaction rate. This in turn reduces the quantity of plutonium in the discharged fuel and leads to a higher net plutonium consumption. Furthermore, the higher thermal flux increases the reactivity worth of soluble boron (improving the boron reactivity coefficient) and also increases the effectiveness of control rods and burnable poison rods. A major disadvantage of this approach is that it is incompatible with existing reactor designs and would only be possible in a new reactor core. Moreover, such a core would need to be dedicated completely to MOX fuel usage as the moderator/fuel ratio would be incompatible with the use of UO_2 fuel, being so far removed from the UO_2 optimum. Therefore a highly moderated LWR would need to be dedicated for its entire lifetime to use with MOX fuel. Such a reactor might conceivably be built as a minor component of a fleet of LWRs in which MOX burning was concentrated in dedicated plutonium burning reactors. An advantage is that only one plutonium concentration would be needed within the reload fuel assemblies, because UO_2/MOX interface effects do not apply. Also, the poor discharge plutonium quality could be regarded as a proliferation resistance benefit. An additional benefit is that the low initial plutonium content increases the MOX/UO_2 ratio in an equilibrium scenario, which implies that the energy content in the plutonium recovered from UO_2 fuel is utilised more fully. There is also a benefit from reduced minor actinide production.

The nuclear design of a high moderator/fuel ratio PWR has been minutely investigated [1], with particular emphasis on the potential role of such a reactor in a scenario where the plutonium is recycled several times. The study confirmed that many of the nuclear design parameters are indeed improved in such a system, but also highlighted that the multiple recycle capability is not significantly better than conventional MOX designs. The reason for this is that in the highly moderated core the plutonium fissile fraction is degraded in each irradiation more quickly than in the conventional MOX design such that the initial plutonium loading reaches impractical levels as soon as the second or third recycle.

Detailed studies of high-moderation PWRs and BWRs have been carried out by Japanese researchers [23-25] including multiple recycle studies [26,27], so that a very good understanding of the technical characteristics of such systems has been built up. The appendix provides a summary of the studies for reference.

Since high-moderation PWRs and BWRs achieve a high plutonium destruction rate, they show a beneficial reduction in plutonium radiotoxicity. This is especially true of multiple recycle scenarios where the only source of plutonium in high-level waste is from reprocessing losses. Volumes of low- and intermediate-level waste arising from reprocessing operations are essentially the same as for conventional MOX.

3.7 Low-moderation LWRs

Based on an idea of Edlund [28] various institutions have, in the 1980s, investigated the possibility of modifying the core of a conventional pressurised water reactor to obtain higher conversion ratios without changing any of the other main system components. The aim was to achieve higher burn-up and lower consumption of fissile material. This can be realised by using plutonium fuel along with steel cladding tubes and by reducing the moderator fraction in the core, which shifts the neutron spectrum towards higher energies. Parametric studies showed a clear relationship between burn-up and conversion ratio, depending on the moderator-to-fuel volume ratio. Very narrow lattices with moderator-to-fuel ratios of about 0.5 would be very effective at conserving fissile material, as the conversion ratio is high.

A NEA CRP burn-up benchmark for high-conversion light water reactor lattices [29] eventually demonstrated acceptable accuracy of the calculational methods. The concept and design studies included contributions from Belgium, Canada (with heavy water as coolant), France, Germany (including experimental work on critical heat flux and emergency core cooling), India, Japan (BWR and PWR systems, including experimental work on critical experiments, critical heat flux and emergency core cooling), Russia and Switzerland (critical experiments at PROTEUS in co-operation with Germany). The technical aspects of high converter reactors were discussed in detail at the Technical Committee Meeting held in Nuremberg, 26-29 March 1990 [30]. A consensus was reached that for physical/thermal hydraulic reasons the very tight lattice designs originally envisaged would need to be relaxed, perhaps to values approaching that of conventional MOX assemblies, but with a hexagonal lattice. With such a more modest approach, the physical/thermal-hydraulic questions appear more tractable.

The rationale for high converter lattices (as is the case in Japan and some other countries) is to view plutonium as an important strategic resource that should be conserved for eventual use in fast reactors. Having recognised that commercial fast reactor deployment is now delayed, low-moderation (or high-converter) LWRs are seen in Japan as a promising option for the next step in plutonium recycling after partial MOX loading in LWRs. By taking advantage of LWR technology, they would contribute to the early establishment of a multiple-recycle system for plutonium, and thereby to a substantial reduction in the consumption of natural uranium. Therefore in Japan the concept and design studies were restarted at the end of 1990s to attain conversion ratios significantly above 1.0 and to effectively act as plutonium breeders for plutonium multiple recycling. In this new series of studies, the high-converter reactor system is named the Reduced-moderation Water Reactor (RMWR) [31,32] and the research activities include experimental work on critical heat flux as described in Section 3.7.1.

In spite of the development effort that would be required to support low-moderation LWRs, the concept would benefit considerably from existing LWR design and operating experience, much of which would still be applicable and directly transferable. The concept would therefore be well suited to deployment in the medium term. An advantage of low-moderation LWRs would be that the specific plutonium inventory of the reactor core would be relatively large, so that a small number of such reactors could recycle all the plutonium from a large-scale programme of UO_2-fuelled LWRs. With a neutron spectrum similar to fast reactors, such reactors would also better preserve the plutonium isotopic quality, possibly improving the prospects for multiple recycle.

In low-moderation LWRs, multiple recycling might be possible using advanced fuel reprocessing schemes with low decontamination factors. Waste arisings, particularly minor actinides, would be considerably reduced and this would be an environmental benefit.

3.7.1 Reduced-moderation water reactor (RMWR)

RMWR aims at achieving a high conversion ratio (greater than 1.0) with MOX fuel, based on well-understood water-cooled reactor technology. Such a high conversion ratio can be attained by reducing the moderation of neutrons, i.e. reducing the water fraction in the core using very narrow lattices with moderator-to-fuel ratios much less than 0.5. Another important design target for the RMWR is to achieve a negative void reactivity coefficient. This is one of the important characteristics of LWRs, especially from the safety point of view.

However, ensuring a negative void reactivity coefficient and high conversion ratios at the same time is difficult, and this represents a trade-off that needs to be made in the nuclear design.

Several basic design concepts satisfying both these main design targets mentioned above have been proposed [33,34] which include both BWR and PWR-type systems. The common design characteristics are a tight-lattice fuel rod configuration and a short core. The former is to attain the high conversion ratio and the latter is to ensure a negative void reactivity coefficient. Additionally, the axial, i.e. upper, lower or internal, or the radial blankets made of the depleted UO_2 are also introduced by necessity for both purposes mentioned above. Although both the BWR and PWR-type concepts can attain the design goals, the BWR-type concept can achieve higher core performance. This is because it has an advantage in being able to utilise void in the water to reduce the water fraction in the core without reducing the spacing between rods.

As a specific example, a core concept of a large-scale BWR-type RMWR is shown in Figures 11 and 12. The main core parameters are summarised in Table 1. Axial blankets with depleted uranium are also introduced to increase the conversion ratio and to reduce the void reactivity coefficients. In the present design, the MOX core layer is shortened to 195 mm and 205 mm high for the upper and lower MOX regions, respectively. An internal blanket region of 295 mm high is positioned between two MOX layers. With the upper and lower blankets, the total core region has five-layer structure with a total height of 1.105 m.

As shown in Figure 12, the pitch of the assembly is 228 mm. The diameter of the fuel rods and the gap between rods are 13.7 mm and 1.3 mm, respectively. The fuel assembly consists of five kinds of fuel rods with different Pu contents to reduce the radial power peaking factor. The average fissile Pu content in the MOX part is 18 w/o, and the average fissile Pu content in the core region including the internal blanket part is 10.4 w/o. The fuel assembly consists of five kinds of fuel rods with different Pu contents to reduce the radial power peaking factor. Y-shaped control rod with a follower structure are introduced for every three fuel assemblies.

The core average void fraction achieved is 70%. The conversion ratio of fissile Pu is 1.05. The void reactivity coefficient is negative, with a value of -0.5×10^{-4}dk/k/%void. The electric output of 1 356 MWe is accomplished as in the case of ABWR. The core can be cooled by natural circulation because the core pressure drop is as low as 0.04 MPa due to the low flow velocity and short

Figure 11. Core design of large-scale reduced-moderation water reactor (RMWR)

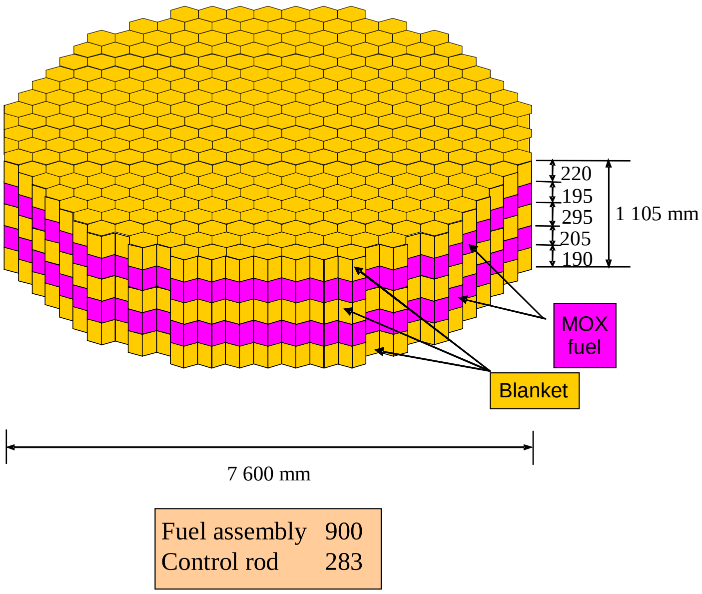


Figure 12. Fuel assembly design of large-scale reduced-moderation water reactor (RMWR)

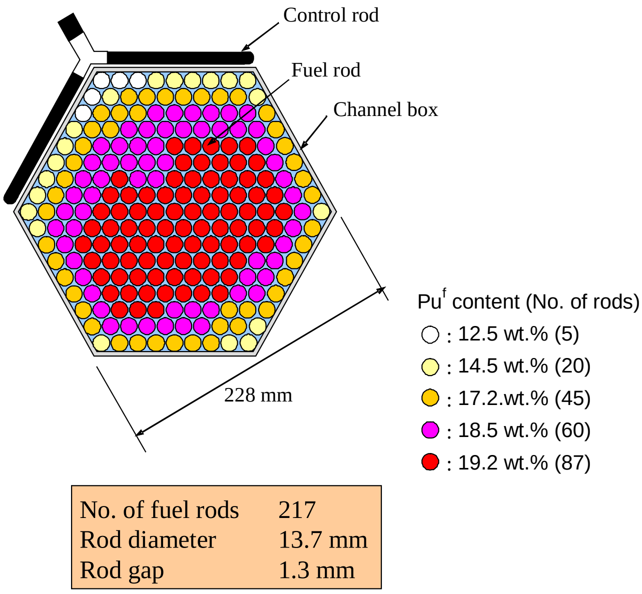


Table 1. Main characteristics of BWR-type reduced-moderation water reactor (RMWR)

Item	Units	Value
Electrical power	MWe	1356
Core diameter	m	7.6
Core average burn-up	GWd/tHM	60
Core height	m	0.695*
Core average void fraction	%	70
Core average Pu(fissile) fraction	%	10.4
Pu(fissile) conversion ratio	–	1.05
Void reactivity coefficient	dk/k/% void	-0.5×10^{-4}
Operational cycle length	Months	24

* Additionally, there are upper and lower blankets of height 0.22 and 0.19 m, respectively.

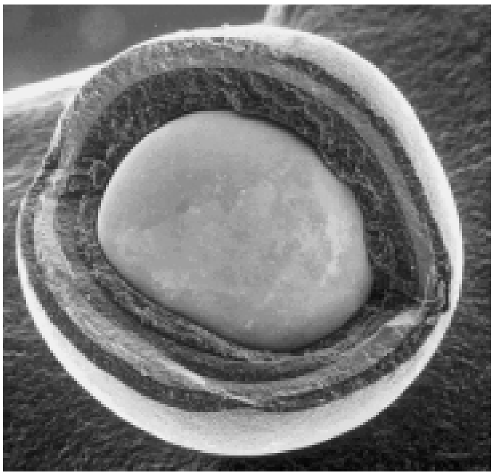
core height. Therefore, the reactor internal pumps used in the ABWR can be eliminated in the present design. The number of fuel assemblies is 900 and each assembly has 217 fuel rods. The core outer diameter is 7.6 m. The core average discharge burn-up including the internal blanket is 60 GWd/t and the operation cycle length is 24 months. The maximum linear power density is 55 kW/m and the minimum critical power ratio (MCPR) is evaluated to be 1.3 under normal operational conditions, which is acceptable.

For RMWR utilisation, multiple recycling of plutonium is the essential basis for the MOX spent fuel reprocessing fuel cycle. On this point, the fuel cycle proposed for FBRs might also be adopted for the RMWR. The core performance of the RMWR in multiple recycling situations with advanced fuel reprocessing schemes with relatively low decontamination factors have been investigated and confirmed [33]. In the multiple recycle case, an advanced reprocessing scheme is used with low decontamination factors in which all the minor actinides and up to 20% of the fission products are assumed to be recycled with plutonium. Only the residual fission products need to be treated as waste. Another possibility is the deployment of RMWR in combination with an accelerator-driven system for waste transmutation. In this case, a different multiple recycle scheme with relatively high decontamination factors would also be acceptable from the environmental point of view. Regarding the critical heat flux in the very tight lattice core, an experimental investigation has been performed and the core coolability has been confirmed [34]. Also, critical experiments for the tight-lattice MOX core are under way at JAERI’s FCA facility and MOX fuel irradiation behaviour has been investigated with a fuel analysis code [35].

3.8 High-temperature reactors

In recent years there has been a revival of interest in high-temperature reactors (HTRs). These were developed up to the mid-1980s but interest faded in Europe because they were then perceived as not being competitive with LWRs. Since that time technological developments, in particular the possibility of implementing a direct helium thermal-dynamic cycle, have led to a reappraisal. Two HTR concepts are currently being pursued, the gas turbine modular helium reactor (GT-MHR) [36] and the pebble bed modular reactor (PBMR) [37]. Both systems are based on a TRISO-coated particle fuel design (Figure 13) in which small spherical fuel particles are embedded in a graphite matrix. The coolant is helium gas. In the GT-MHR the fuel/graphite matrix is in the form of hexagonal blocks, while in PBMR the matrix is in the form of 6-cm diameter spheres. The fuel particles consist of a central kernel of UO_2 or UO_2/PuO_2 encapsulated by layers of SiC and pyrolytic carbon that act as a barrier to the escape of fission products. The complete absence of metallic fuel components allows very high operating temperatures (for high thermal efficiency) and also leads to inherent safety and excellent passive safety characteristics, since fuel damage and fission product release does not occur even in the most severe accident scenario. There remain some outstanding safety issues with HTRs, such as demonstrating that there is adequate protection against graphite fires. It is believed that a satisfactory safety case for graphite fires and other safety issues can be constructed and it is expected that this will be demonstrated as part of the development programmes for GT-MHR and PBMR.

Figure 13. TRISO-coated fuel particle for high-temperature reactor (HTRs)



HTRs have the flexibility to use uranium or plutonium fuel, and with the latter can achieve a high rate of plutonium consumption of approximately 100 kg/TWhe [38]. The spent fuel management strategy for both GT-MHR and PBMR, as currently envisaged, will involve interim spent fuel storage and eventual geological disposal, and there are no immediate plans to develop a reprocessing strategy. Reprocessing of TRISO fuel necessitates additional stages to crush the SiC coating and to separate the fuel material from the graphite matrix. Process development on TRISO fuel reprocessing was carried out in the 1970s and early 1980s but was abandoned because of reduced interest in HTRs and was never fully developed. Consequently, the reprocessing of TRISO fuel is, at the present time, a theoretical possibility that would require development.

A possible role for HTRs in the medium term would be to act as a sink for plutonium which is excess to requirements. The lack of a demonstrated reprocessing route could be considered an advantage in this case, because it could be argued that this is beneficial for proliferation resistance. If it were intended that plutonium incorporated in spent fuel from HTRs should eventually become available to fuel the initial cores of fast reactors, a reprocessing route would need to be developed. Such a scenario in the medium term might fit well with a strategy in which HTR technology was eventually adopted for the fast reactors to follow, since one of the long-term options currently of interest is high-temperature gas-cooled fast reactors (HT-GCFRs) which would use particle fuel similar in concept to TRISO. If HT-GCFRs were selected for deployment, it would be necessary to implement a reprocessing route for sustainability. In this case the same technology could conceivably be used to recover plutonium from spent HTR fuel if it were needed to start the HT-GCFR programme. In summary, therefore, the HTRs might find a role in the medium term as consumers of excess plutonium from LWRs, while in the longer term they might fit well in a transition to HT-GCFRs.

The waste arising from HTRs is in a very different form than that of normal spent fuel, consisting either of fuel spheres (for PBMR) or fuel compacts (for GT-MHR). The fuel spheres or fuel compacts are mostly comprised of graphite, with only a very small proportion of the volume taken up with the fuel microspheres. The microspheres, if they perform as designed, will retain almost all the activity of fission products and heavy nuclides. The final disposal route for fuel spheres or fuel compacts has yet to be finalised and it is not yet known what conditioning, if any, would be required prior to geological disposal. It is likely that the fuel microspheres could be regarded as a stable matrix for geological disposal.

3.9 *Fast reactors*

In the abstract and introduction to this report the view was taken that only fast reactors can satisfy the need for a fully sustainable nuclear system. This has long been the view in Europe, Japan, Russia and elsewhere, and now with the emergence of the Generation IV Initiative is becoming internationally accepted; only fast reactors operating with multiple recycle breeding fuel cycles will be able to meet the world's increasing energy demand in the very long term. Although it is clear that it will be many years before this becomes a reality, it is important not to overlook the *potential* role of fast reactors within the context of plutonium management in the medium term considered by this report.

The fact is that fast reactor research and development world-wide has already achieved a very high stage of maturity and the technical knowledge does exist to build fast reactors today. Therefore fast reactors need to be included in this report because they represent a viable technical option for plutonium management. Indeed, in the past ten years or so a large amount of research work has been carried out into the potential role of fast reactors in plutonium management, notably as part of the CAPRA-CADRA research collaboration in Europe. CAPRA-CADRA was created in response to the 1991 French law which mandated a 15-year research programme to investigate the technical options available for the nuclear fuel cycle in France.

When it was first set up, one specific task under CAPRA-CADRA was to explore the potential of modified EFR cores as plutonium burners. This work is now considered complete, and more recent work has focused on the potential role of gas-cooled fast reactors for plutonium management. Early during the CAPRA-CADRA collaboration, the emphasis was on establishing EFR variant cores that would maximise plutonium consumption. More recently, the emphasis has shifted towards cores which have smaller plutonium consumption rates and which also have the flexibility to operate as breeding cores. This section briefly summarises the fast reactor core studies that have been investigated as part of CAPRA-CADRA and highlights how they could potentially contribute to plutonium management.

The CAPRA-CADRA collaboration has investigated more than 30 EFR core variants and has built up a good understanding of the technical capabilities and limitations of sodium-cooled fast reactors with respect to plutonium management. Each core variant has been the subject of detailed nuclear design modelling. The scope of the nuclear design analysis included establishing satisfactory fuel management schemes. The nuclear design models were then used to calculate the important core parameters, including peak linear ratings, reactivity coefficients, control rod reactivity worths and core kinetics parameters. For sodium cores the sodium void coefficient and the Doppler coefficient require particular attention and in several cases special mitigating measures were needed to ensure a satisfactory combination of void and Doppler coefficient.

Not all these variants are relevant to this report (some of the core variants looked at the potential to destroy minor actinides) and this section will be limited to presenting a summary overview. Under the CAPRA-CADRA collaboration, a core designated CAPRA 04/94 became accepted as the reference plutonium-burning core. There were several variants of this reference core, but these will not be distinguished here because the only differences were small fine tunings of the core design. The CAPRA 04/94 core is a derivative of the EFR breeder core in which both the radial and axial breeder blankets were removed (to reduce the plutonium conversion ratio) and other details of the fuel design were modified. The principal modification was to increase the initial plutonium content of the fuel from ~ 20 w/o in EFR to ~ 40-45 w/o. This had the effect of reducing the ^{238}U mass in the core and further reducing the plutonium conversion ratio. The change to such a high initial plutonium fraction necessitated the introduction of steel diluent rods within the fuel assemblies and also of steel diluent assembles (to avid too-high a fissile density in the core). In some moderators diluent materials ($^{11}B_4C$) were used to soften the neutron spectrum.

The various CAPRA 04/94 variants have a plutonium destruction rate in the region of 60 to 70 kg/TWhe, comparable to some of the advanced thermal reactors options discussed earlier. More important is the fact that the cores have a large initial plutonium mass of approximately 10 tHM for a 1 450 GWe plant. To start up such a core requires an initial plutonium mass of approximately 10 tHM and approximately a further 10 tHM to sustain the reactor through the first few years of operation until the first batches of fuel are reprocessed and the second-recycle plutonium becomes available. Thereafter, the reactor will continue operating with a mass of approximately 20 tHM of plutonium in the active fuel cycle, supplemented with 60 to 70 kg/TWhe of plutonium from store or recovered from thermal reactors (equivalent to 600 to 700 kg/year for a single CAPRA 04/94 core at 75% load factor). In equilibrium a single CAPRA 04/94 core could consume the plutonium recovered from approximately 3 LWRs with 1 GWye output each. A relatively small number of CAPRA 04/94 reactors would be able to remove hundreds of tonnes of plutonium from store.

Some early CAPRA-CADRA work examined the feasibility of plutonium nitride fuel containing no uranium. The core studies for this variant provided a plutonium consumption of 125 kg/TWhe, which coincides with the theoretical maximum. However, various technical difficulties were identified which led to a decision not to pursue this option further. No further CAPRA-CADRA studies on sodium-cooled fast reactors exist, as the emphasis has changed to gas-cooled fast reactors. It is considered that the sodium-cooled studies have effectively answered all the relevant technical questions for sodium-cooled reactors in sufficient detail for CAPRA-CADRA to respond adequately to the 1991 law. The feasibility of sodium-cooled reactors as plutonium burners is considered to have been established.

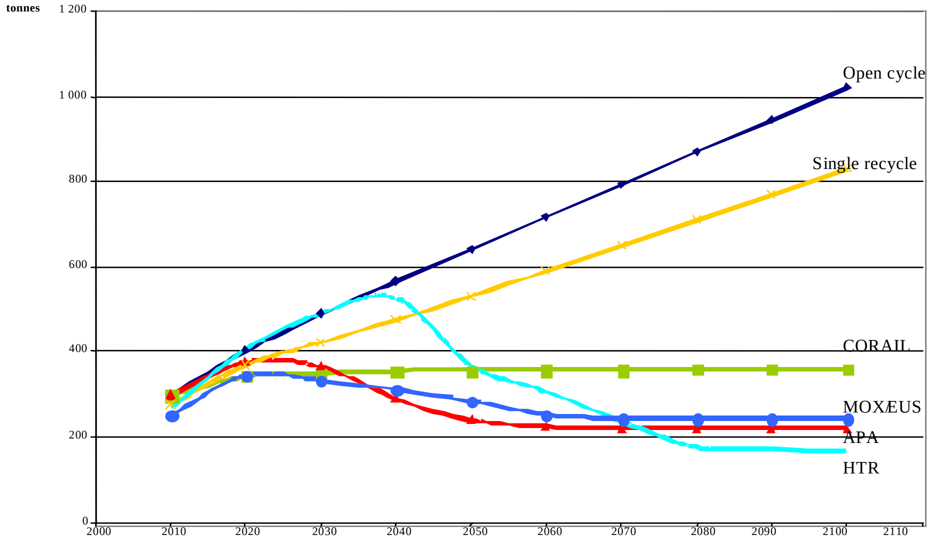
More recent work under CAPRA-CADRA has concentrated on gas-cooled fast reactors. Although the gas-cooled fast reactor work is not yet complete, it is clear that gas-cooled fast reactors can achieve plutonium consumption rates very similar to those of CAPRA 04/94. Total plutonium inventories in the gas-cooled cores examined to date tend to be higher than those of CAPRA 04/94, so start-up has an even more pronounced effect on plutonium stocks. Recognising the eventual importance of a fully self-sustainable fuel cycle, the current CAPRA-CADRA studies place more emphasis on breeding systems with a conversion ratio close to 1. Such breeders essentially require the same start-up plutonium feed, but once in equilibrium they become self-sufficient in plutonium.

3.10 Summary

Table 2 provides a summary of the main parameters of the PWR plutonium management options (MOX/EUS, CORAIL, APA and PLUTON) that originated in France. The table also provides the corresponding values for UO_2 and conventional MOX fuel cycles for comparison. For each of the systems, the table lists details of the fuel cycle parameters assumed, the ^{235}U enrichments and plutonium concentrations required, details of the number of fuel rods per assembly and the mass of plutonium per assembly. Also shown are fuel cycle parameters related to the consumption or production of plutonium and minor actinides and, for the concepts originating at the CEA, data relating to the deployment of each option in a 60 GWye park of EPRs. The CEA has evaluated the potential impact of each of these options within the 60 GWye park. The results are shown in Figure 14.

The open cycle (UO_2) case corresponds to the uppermost (blue) curve. The total inventory of plutonium accumulates at a rate dependent on the output of the park. The single-recycle MOX case also accumulates plutonium steadily, though at a lower rate. The other options are all able to stabilise the total

Figure 14. Evolution of total plutonium inventory with time for different scenarios in the French nuclear park



plutonium inventory at various levels between 200-400 tonnes. This demonstrates how effective these advanced options are at preventing the steady accumulation of the total plutonium inventory.

Table 3 gives a qualitative summary of the various systems considered earlier that are not covered by Table 2. It indicates the status of the various systems, the potential for plutonium utilisation or destruction and gives an indication of the economic implications.

4. Plutonium management strategies

This section discusses in turn all the medium-term plutonium management strategies that have been proposed. It is difficult to assess the suitability of these strategies in anything other than a qualitative way. The reason for this is that the precise requirements that should be met in the longer term are not clear. Thus in the long term it is envisaged that there will be a reactor fleet consisting of a mix of thermal reactors and fast reactors, possibly including a small fraction of minor actinide transmuting systems. The precise proportions of each depends on the eventual choice of reactor systems and their operating regime, all of which are not known at this stage. Different fast reactor designs could in

Table 2. Main characteristics of various PWR fuel options

		UO_2 (open cycle)	PWR MOX (single recycle)	MOX/EUS equilib.**	CORAIL equilib.	APA equilib.	PLUTON
Burn-up	GWd/t	60	60	60	60	48	
Fraction of core discharged per cycle	–	1/4	1/4	1/4	1/4	1/5	1/3
Fuel cycle length	efpd	440	440	440	440	280	325
Enrichment of ^{235}U	w/o	4.7	0.25	3.9	5.2	3.7	0
Amount of plutonium	w/o	-	10.5	12	11.3		12.5*
Number of UO_2 pins	–	264	–	–	180	120	–
Number of MOX or Pu pins	–	-	264	264	84	36	228
Mass of Pu per assembly	kg	0	53	62	18.6	33.3	28.772
Suitable for multi-recycle		No	Limited				–
Plutonium balance (positive indicates net generation)	kg/TWhe	26	-70	-58	-7	-68	136
Fissile fraction of plutonium from multi-recycle	% Pufiss	–	–	48	47	39	
Production of minor actinides (positive indicates net generation)	kg/TWhe						
Np		1.8	0.4	1.1	1.6	0.8	–
Am		1.6	14.1	17.5	6.1	11.5	6.1
Cm		0.3	2.9	3.9	1.7	5.0	5.9
Total		3.7	17.4	22.5	9.4	17.3	12.0
Natural uranium requirement per reactor type	t/TWhe	19.4	0	14.2	15.2	8.8	
Enrichment requirement	kSWU/TWhe	14.6	0	10.1	11.8	6.1	
Core management established		Reference	Yes	Yes	Yes	Short cycles	
Number of Pu pins per assembly	kg/TWhe	–	3.8	4.6	10.5	0.5	
Equilibrium scenario – reactor park with 100% EPR and stabilised plutonium inventory:							
Fraction of reactors to load	%	–	–	31	79	28	
Number of rods to reprocess each year	–	–	–	1.31×10^5	1.06×10^5	2.0×10^4	
Mass of MOX (or plutonium) to fabricate per year	tHM/y	–	–	253	205	18.4	
Mass of assemblies with plutonium to treat	tHM/y	–	–	253	645	148	
Mass of plutonium in stabilised inventory	tHM/y	–	–	407	310	221	

* 87.5% Al_2O_3 ** Standard moderation ratio

Table 3. Qualitative summary table for various systems

System	Status	Potential for Pu utilisation/ Pu destruction	Economic aspects
LWR with high moderation	• Reactor at reduced power needed.	• Possibility of using Pu with low Pu_{fiss}-content. • Mainly for Pu destruction.	• Disadvantage: cost of investment for the dedicated plant.
LWR with low moderation	• Theoretical development needed. • Testing needed. • New or (highly) modified LWR.	• Restores Pu. • Ideal partner for fast breeder cycle.	• Expensive interim step for a new plant and for the fuel cycle.
HTR	• Development, testing and test reactor needed.	• Pu utilisation limited. • As reprocessing technology not developed, mainly Pu destruction.	• New reactor line to be established.
Alternative FAs for LWRs	• Developments needed. • Testing in existing reactors (LWR and VVER) possible.	• Most proposals aimed at Pu destruction.	• Possible small scale technology in exchange to MOX.
HWR	• New plants needed.	• Improved conversion factor.	• More expensive fuel cycle as fissile contents are lower.
FB	• New plants needed.	• Single proposal to use existing Pu quantities for a long range utilisation of fissile nuclear energy.	• Economic competitiveness needs to be improved.
Th fuel cycle	• In status of (initial) reactor testing. • New type of fuel cycle plants needed. • Difficult fuel fabrication: γ-activity via reuse of U and Th. • Difficult reprocessing to reuse ^{233}U: low solubility.	• Exchange Pu by ^{233}U. • Open fuel cycle destroys Pu and wastes ^{233}U.	• Some disadvantage by more expensive fuel cycle.

principle vary considerably with respect to the mass of plutonium required for their initial cores. With both the initial mass of plutonium per fast reactor and the number of fast reactors in the park unknown, it is not possible to define quantitative targets for the medium term considered here, such as the mass of plutonium that needs to be available at a particular date, or any required flexibility in plutonium inventory to satisfy reload requirements.

Table 4 is a decision table that attempts to present the high-level issues associated with the decision to opt for the direct disposal, interim storage or reprocessing options. This presents a concise picture of the advantages of each option and the research and their respective development requirements.

Table 4. Decision table concerning Pu utilisation: reactor types and fuel assembly alternatives

Strategic option	Final (direct) disposal	Interim storage	Reprocessing
Pu utilisation	• No.	• Postponed.	• Implemented.
Advantages	• No access to Pu. • Early encapsulation of reactivity. • Unlimited (though expensive) resources of U.	• Postpone decisions. • Time to establish: final storage pond and/or additional reprocessing capacity.	• Long time sustainable fission power with limited U resources, preferably leading to fast breeder cycle.
Research and development requirement	• Only evolutionary development needed to reduce waste arisings and improve fuel utilisation: PWR, VVER, BWR, HWR & HTGR.	• Development of (new) reactor systems advisable to reduce waste arisings and improve fuel utilisation.	• New concepts need development and testing: FAs for existing reactors: MOX/EUS, APA, PLUTON, CORAIL, Pu/Er/Zr fuel; Th fuels (with Pu); high/low moderation LWR; HTR; fast breeder systems.

4.1 Spent fuel interim storage with deferred reprocessing

In this strategy spent fuel LWR fuel is simply put in interim storage in the reactor ponds, in at-reactor dry storage modules or perhaps wet or dry storage away from reactor and reprocessing is deferred until the plutonium is required for the fast reactor programme. This is effectively the strategy that applies to the majority of LWR fuel discharges, since most utilities world-wide have a policy of interim storage to be followed eventually by geological disposal. Such a policy does, however, leave open the option of eventual reprocessing for succeeding reactor generations which can use or destroy fissile material stocks.

The principal advantages of this strategy are that interim storage costs are relatively low and that the mass of plutonium is conserved (with losses only due to decay of ^{241}Pu). The plutonium is also stored within the protective radiation field of spent fuel assemblies, which may be considered a proliferation resistance benefit. The main disadvantages of this strategy are that the energy content of the plutonium is unused during the interim storage period and that a large mass of fuel would need to be reprocessed to start a fast reactor programme; the plutonium content of spent LWR fuel is typically in the region of 10 kg/tHM, while the initial plutonium charge of a large fast reactor might be 15 t, which implies that reprocessing 1 500 tHM of LWR fuel is required to support one fast reactor. A decision to go ahead with a programme of fast reactors utilising plutonium would therefore be dependent on the availability of a large LWR reprocessing capacity. If such capacity were not available commercially, there would be a need to build an indigenous plant which would involve possibly very long time scales for planning, design, construction and commissioning. Because of these logistical difficulties, this strategy would not be the first choice for maintaining strategic independence.

4.2 Prompt reprocessing with no recycle

Prompt reprocessing of LWR fuel with no recycle preserves the mass of plutonium (^{241}Pu decay excepted) and also generates separated plutonium which is available immediately if it is required for a fast reactor programme. The principal disadvantage is that a large mass of separated PuO_2 is accumulated that needs to be stored, with all the aspects connected to safeguards and proliferation. Storage of plutonium is penalised by the inevitable loss of ^{241}Pu and the associated build-up of ^{241}Am, increasing helium content (from accumulated α-decays), hydrolytic hydrogen build-up in wet storage and increased proliferation risk.

4.3 Prompt reprocessing with single recycle in LWRs

Single recycle implies that plutonium is recycled one time only and the irradiated MOX assemblies either subject to interim storage or reprocessed without recycle of the recovered uranium. A strategy of prompt reprocessing of LWR fuel followed by a single recycle in LWRs in the form of MOX, APA, PLUTON or CORAIL assemblies has the advantage that the energy content of the plutonium is at least partly utilised. (The MOX/EUS option is not considered here because it is specifically intended for a multiple-recycle strategy.) The strategy would avoid, at least in the short term, any issues that might arise in reprocessing plutonium assemblies with a low or no admixture of uranium assemblies. Furthermore, with each of these options it is technically feasible to balance plutonium arisings and consumption such that the separated plutonium is limited to the mass of plutonium active in the fabrication plant and of plutonium in completed MOX assemblies awaiting loading in the reactor.

With respect to plutonium balance, there is little to choose between the fuel designs except that those options requiring more development work to implement on an industrial scale are more likely to result in a temporary imbalance of plutonium arisings and consumption. This comment applies especially to the APA and PLUTON, for which time would be required to develop and demonstrate the novel fuel pellet design. Indeed, these designs cannot be considered for immediate deployment and would need to be introduced following an earlier phase of recycle in conventional MOX assemblies.

A single recycle strategy with conventional MOX concentrates the plutonium in a smaller spent fuel mass, since the plutonium content of MOX assemblies at discharge is typically in the region of 40-50 kg/tHM at present discharge burn-ups. In order to recover 15 t of plutonium to start a single fast reactor, it would be sufficient to reprocess in the region of 300-400 tHM of MOX fuel (compared with ~ 1 500 tHM of UO_2 fuel). A possible approach would be to anticipate the introduction of fast reactors by building the fast reactor reprocessing plant in advance of the commissioning of the first group of new fast reactors and using this plant to reprocess the LWR MOX fuel. A fast reactor reprocessing plant would be designed to handle fuel which is considerably more active than LWR MOX and might well be suited for this purpose with the addition of a head-end plant designed to handle LWR MOX assemblies. The CORAIL assembly could also perform this role, although the total mass to be reprocessed would be larger, as there are both MOX and UO_2 rods in the same assembly and the overall concentration of plutonium is correspondingly smaller.

The APA and PLUTON assemblies would be less suited to the role of acting as a source of plutonium for fast reactor start-up, because the inert matrix containing the bulk of the plutonium in the APA assembly and all the plutonium in PLUTON would necessitate the development of a novel reprocessing method. Concepts like APA and PLUTON may therefore fit better with a scenario where there is a surplus of plutonium from the thermal reactor programme and these assemblies effectively are a vehicle for immobilising the excess.

4.4 Multiple recycle in LWRs

A multiple recycle strategy for LWR MOX fuel minimises the cumulative mass of plutonium produced, while increasing the energy recovered from the plutonium. It also paves the way for a multiple recycle in fast reactors to follow. However, there is a difficulty with the multiple-recycle strategy in LWRs that does not apply to the same degree in fast reactors. This is the issue of the build-up of the even mass plutonium isotopes, which will not fission directly in the thermal spectrum. Their build-up degrades the isotopic quality of the plutonium leading to a requirement for impractically high initial plutonium contents as soon as the third recycle [1]. The build-up of the higher plutonium isotopes also contributes towards increased levels of the other minor actinides, particularly Am and Cm, with implications for radiotoxicity. An additional consideration is that the reprocessing of MOX assemblies is technically more demanding than for UO_2 assemblies, which may result in an economic penalty. Also, multiple recycle further reduces the mass of plutonium available for start-up of a fast reactor programme.

The MOX/EUS concept was developed specifically in order to avoid the technical constraints that apply to multiple recycle of conventional MOX. The predominant role of enriched UO_2 in the assembly ensures that the plutonium plays only a minor role in the nuclear design performance and that isotopic degradation does not become a limiting factor. Separate MOX reprocessing batches could be used to extract quantities of plutonium with a low fraction of thermally fissile isotopes from the fuel cycle.

4.5 Prompt reprocessing with recycle in high-moderation LWRs

A strategy of prompt reprocessing of LWR fuel followed by recycle in high-moderation LWRs has been demonstrated to be technically feasible [7,22] and could form an alternative to any of the options considered in Section 4.3. The principal difference is that dedicated plutonium-burning cores (only possible with operational restrictions in existing reactors) would be needed for this purpose and this implies some development effort would be required to demonstrate those fuel and core performance aspects that are not covered by standard LWR operational experience. Compared with conventional MOX recycle, the principal logistical difference is that the initial plutonium content of the fuel is lower than for conventional MOX, so that a higher capacity fraction (and therefore a higher fuel fabrication rate) is required for a given plutonium consumption rate. Also, the plutonium content of the discharged fuel is lower in quantity and quality than for conventional MOX, so that the mass potentially available for fast reactors is smaller.

4.6 Prompt reprocessing with recycle in low-moderation LWRs

This strategy is similar to recycle in high-moderation LWRs in that dedicated reactor cores are required and similar comments apply. The principal difference from the point of view of medium-term plutonium management strategy is that the capacity requirement is reduced because the initial plutonium concentration is higher and the total mass of plutonium is essentially conserved. From a slightly longer perspective, this strategy is significantly different from others in that low-moderation LWRs would possibly be applied as an alternative to fast reactors for building a plutonium breeding cycle by achieving net conversion ratios (i.e. conversion ratio after subtracting fissile plutonium lost during recycling processes) nearing 1.0.

4.7 Phase-out of nuclear

For completeness it is necessary to consider plutonium management in the event of a country deciding to phase out nuclear power. In this case a desirable end-point would be to have all separated plutonium re-incorporated in irradiated fuel. The option considered in Section 4.1 of interim storage with deferred reprocessing is compatible with this goal. The various recycle options considered in Section 4 are all compatible in principle with this objective, although the detailed logistics of plutonium utilisation will require very careful consideration for each specific case. There would be very little to distinguish between the various options considered in this scenario. For a strategy in which plutonium is separated but not recycled, however, a phase-out of nuclear completely removes the possibility of incorporating the plutonium in irradiated fuel. Inert matrix fuels could have a very important role in a phase-out scenario, as they provide a means of maximising the incorporation of separated plutonium, while minimising the mass of plutonium eventually committed to geological disposal and potentially providing a stable and self-protecting incorporation matrix for the plutonium.

5. Discussion

This review has highlighted the complexity of the question of plutonium management, both in the medium term considered here and in the long term. There are very many potential options that have been considered, each of which has its advantages and disadvantages and there is no single approach which is optimal for all circumstances. Many of the options considered make use of technology that is already fully mature and can be considered to be fully understood technically. Some of the options, however, rely on new technology that is not demonstrated and will require research and development. For these cases there are technological uncertainties that may require extensive time scales to fully resolve.

A feature that is common to all the options is that the time scales involved in reaching an equilibrium situation tend to be very lengthy. A decision to implement a particular technical approach needs to be followed by a period of time for development and demonstration if it involves any new technological development. Then follows a possibly lengthy time scale for implementation at commercial scale, followed by a lengthy operational period before the eventual equilibrium is established. Another common feature is that many of the options allow a degree of flexibility in the fuel cycle. None of the options considered are irreversible, though for some of them reversing policy may prove more difficult than for others. For example, a decision to implement plutonium recycle in HTRs or in one of the inert matrix options would be more committing because of the need to develop and implement a commercial scale reprocessing route that does not exist at present. There clearly needs to be flexibility, because it will be many years before the choice as to what reactor systems will be built in the long term will be clear and the precise plutonium requirements for their initial cores are known. However, the lead times for introducing a first group of new fast reactors will be long, perhaps 20 years or more, and this would be sufficient time to make any adjustments that may be needed to the medium-term strategy.

It is important to note that recent fast reactor studies [39] have demonstrated that there is considerable flexibility in the fast reactor fuel cycle for operation as plutonium burners or as breeders; relatively minor modifications such as the addition of breeder blankets are sufficient to achieve this flexibility. Any choices that are made in the medium term that result in a stock of plutonium in excess of that required for fast reactors to follow can be compensated for relatively easily by such adjustments. Recently, some studies [40-42] have illustrated the potential of fast reactors. They also highlight the importance of the closed fuel cycle. In the 21st century, the expanding world population and motives to improve standards of living in the world will require ever-increasing amounts of energy with electricity representing a large share. Taking into consideration the environmental burden, a sustainable, recycling-based society must be sought for as an alternative to the old scheme of mass production, mass consumption and mass waste.

6. Conclusions

This review has highlighted the many technical options available in principle that are designed to address the issue of plutonium management in the medium term. The majority of the options considered here are aimed at LWRs, reflecting the present dominance of LWRs and the likelihood that it will persist over the medium term considered here. Which, if any, of these options are eventually adopted by nuclear utilities will depend on a large number of factors, including political, strategic, logistic, environmental, sustainability, economic and others. The option of single recycle as MOX in LWRs is already soundly established on a commercial scale and presents very low technical risk to utilities (essentially the same technical risk as UO_2 fuel in a once-through cycle) and requires the minimum of future development to support. Other options, especially those involving non-oxide fuels, represent a higher technical risk, and will require extended research and development programmes to establish on a commercial scale. However, many of these more advanced options have the potential to deliver significant benefits over the single MOX recycle approach. As issues such as the sustainability of the nuclear fuel cycle gain increasing prominence with utilities, it is likely that research and development efforts will concentrate on a small number of the technical options which will be carried over into commercial applications within the medium time scale considered in this report.

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Appendix

SUMMARY OF DETAILED JAPANESE STUDIES ON HIGH-MODERATION PWRS AND BWRS

This appendix provides a summary of detailed Japanese studies of high-moderation PWRs and BWRs [23-25]. The high-moderation full MOX PWR has been studied for moderator/fuel volume ratios varying from 2.4 to 2.9 [that is, corresponding to hydrogen to heavy metal atomic ratio (H/HM) of 5 to 6] and the high-moderation full MOX BWR for a moderator/fuel volume ratio from 3.7 to 4.4 (corresponding to an H/HM of 5.9 to 7.0) based on the Japanese APWR and a ABWR cores, respectively. The core design parameters are given in Table A.1, and the assembly design and the core performance are given in Table A.2 for PWR and Table A.3 for BWR. Smaller fuel pin diameters were adopted for the high-moderation PWR assemblies. In one variant of the high-moderation BWR (Assembly-1), eight additional water rods displace fuel rods. In a second variant (Assembly-2) a smaller fuel rod diameter was adopted. Maintaining the same core thermal power of the reference ALWRs gives reduced thermal margins in these high-moderation LWRs. However, the core analysis showed that they still meet the criteria of the thermal margins (see Tables A.4 and A.5).

A study of multi-recycling of Pu [26,27] has also been carried out for these high-moderation LWR designs assuming the core parameters of Table A.1. The basic difficulty is that plutonium multi-recycling in LWRs causes progressive degradation of the plutonium vector. Taken in isolation, it is difficult to overcome this degradation in a theoretical self-generated recycle scenario, where all the plutonium originating form a particular reactor is assumed to be consumed in the same reactor. However, this is an artificial scenario that is unlikely to arise in practice. Recognising the reality that in practice UO_2 cores and MOX cores will actually co-exist, it is therefore probable that plutonium will be available from the combined reprocessing of both spent UO_2 fuel and spent MOX fuel. Therefore a more pragmatic and realistic approach would be to assume that the two types of Pu vectors are co-mixed during recycle. In each recycle generation the assumption is made that any deficit of the amount and isotopic quality of plutonium in the spent MOX fuel is made up by using plutonium from spent

UO_2 fuel for the next MOX core cycle (see Figure A.1). After repeating this process five times, the plutonium vector almost reaches an equilibrium. Core designs with this equilibrium plutonium vector were analysed and the core characteristics evaluated. Tables A.4 and A.5 show the core design and performance for the high-moderation PWR and BWR. These illustrate that high-moderation cores can usefully limit the increase in plutonium enrichment needed in each successive recycle generation, making multiple recycle feasible.

Table A.1. Basic core design parameters for Japanese high-moderation MOX studies [23-25]

	APWR	ABWR
Rated thermal power	4 100 MW	3 926 MW
Effective core height	3.66 m	3.71 m
Operation cycle length	15.5 EFPM	15 EFPM
Number of fuel assemblies	257	872
Fuel assembly type	17 × 17	9 × 9
Number of control rods	77	205
Maximum burn-up	55 GWd/t	55 GWd/t

Table A.2. Specifications of high-moderation MOX assembly and core performance (PWR) – Japanese studies [23-25]

	Reference 17 × 17 assembly	High-moderation assembly-1	High-moderation assembly-2
Volume ratio (H/HM*)	2.0 (4.0)	2.4 (5.0)	2.9 (6.0)
Fuel pin diameter	9.5 mm	8.8 mm	8.4 mm
Assembly lattice	17 × 17	17 × 17 (same pitch)	17 × 17 (same pitch)
Number of thimbles	25	25	25
Fissile Pu enrichment**	7.2 w/o	5.8 w/o	5.1 w/o
Matrix	Depleted uranium	Depleted uranium	Depleted uranium
Burnable absorber	–	–	–
Refuel assembly	88	108	120
Cycle burn-up	16.5 GWd/t	19.0 GWd/t	21.3 GWd/t
Max. assembly burn-up	52.7 GWd/t	53.5 GWd/t	53.7 GWd/t
Control cluster (Ag-In-Cd)	97	85	77

* Atomic ratio of hydrogen to heavy metal.

** Enrichment is uniform in the assembly.

Table A3. Specifications of high-moderation MOX assembly and core performance in BWR – Japanese studies [23-25]

	Reference 9 × 9 assembly	**High-moderation assembly-1**	**High-moderation assembly-2**
Volume ratio (H/HM*)	3.1 (4.9)	3.7 (5.9)	4.4 (7.0)
Add water rods	0	8	0
Fuel pin diameter	11.2 mm	11.2 mm	9.8 mm
Assembly lattice	9 × 9	9 × 9	9 × 9
Assembly av.		(same pitch)	(same pitch)
Fissile enrichment**	5.0 wt.%	4.6 wt.%	4.7 wt.%
Fissile Pu enrichment	3.6 wt.%	3.3 wt.%	3.5 wt.%
Matrix	Depleted uranium	Depleted uranium	Depleted uranium
Gadolinia rod			
Gadolinia conc.	2.5/1.5*** w/o	3.0/2.0*** w/o	3.3/2.3*** w/o
Matrix	Enriched uranium	Enriched uranium	Enriched uranium
Refuel assembly	232	264	300
Cycle burn-up	12.0	13.6	15.5
Av. assembly burn-up	45.1 GWd/t	44.9 GWd/t	45.1 GWd/t
Max. assembly burn-up	49.7 GWd/t	47.3 GWd/t	52.5 GWd/t

* Atomic ratio of hydrogen to heavy metal.

** Assembly-averaged concentration of (^{235}U + ^{239}Pu + ^{241}Pu).

*** Axial zoning of gadolinia concentration (lower/upper).

Table A.4. Core design and performance of Pu multi-recycling in PWR – Japanese studies [26,27]

Core type	Reference core		High-moderation core (high-moderation assembly-2)	
Items	**1st recycle**	**5th recycle**	**1st recycle**	**5th recycle**
Volume ratio (H/HM)	2.0 (4.0)		2.9 (6.0)	
Fuel pin diameter (mm)	9.5 (base)		8.4 (88%)	
Reload assem. (no.of batch)	88 (2.9)		120 (2.1)	
Loaded heavy metal (U + Pu)	118 t		91 t	
Operation cycle length	15.5 EFPM		15.5 EFPM	
Cycle burn-up (GWd/t)	16.5		21.3	
Fissile Pu enrichment	7.2 wt.%	9.0 wt.%	5.1 wt.%	6.8 wt.%
Ave. discharge exposure (GWd/t)	48.2	48.2	45.6	45.7
Max. assembly exposure (GWd/t)	52.7 [55.0]	53.0 [55.0]	53.7 [55.0]	53.4 [55.0]
Max. operating boron conc.	2 600 ppm	2 900 ppm	2 200 ppm	2 200 ppm
Shutdown margin (%Δk)	2.2 [1.6*]	2.2 [1.6*]	2.3 [1.6*]	2.2 [1.6*]
Heat flux peaking factor (F_Q)	1.87 [2.32*]	1.88 [2.32*]	1.88 [2.32*]	1.89 [2.32*]
Nuclear enthalpy rise hot channel factor ($F_{\Delta H}^N$)	1.44 [1.6*]	1.48 [1.6*]	1.46 [1.6*]	1.46 [1.6*]
Moderator temperature coeff.icient (pcm/^{o}C)	-74 to -16	-64 to -15	-70 to -2	-69 to-8
Doppler coefficient (pcm/°C)	-4.1 to -2.6	-4.2 to -2.6	-4.0 to -2.5	-4.1to -2.6
Delayed neutron fraction (%)	0.36 to 0.38	0.36 to 0.38	0.34 to 0.38	0.36 to 0.38

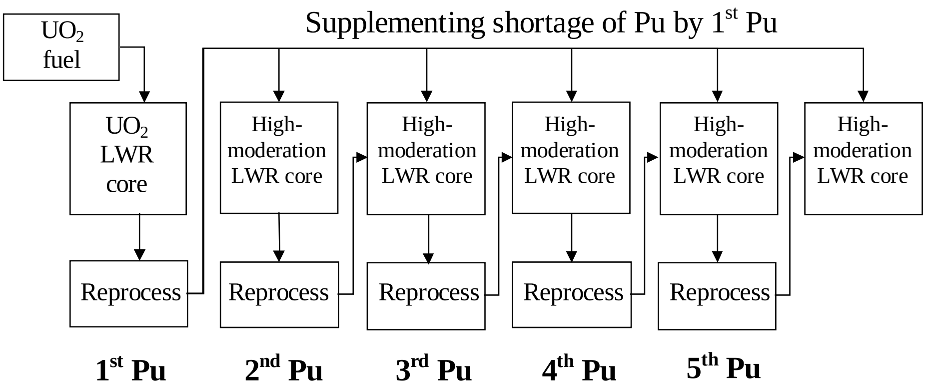
[ ] Limiting value.
* Limiting value of conventional four-loop plant.

Table A.5. Core design and performance of Pu multi-recycling (BWR) – Japanese studies [26,27]

Core type / Items	Reference core		High-moderation core (high-moderation assembly-2)	
	1st recycle	**5th recycle**	**1st recycle**	**5th recycle**
Volume ratio (H/HM)	3.1 (4.9)		4.4 (7.0)	
Fuel pin diameter (mm)	11.2 (base)		9.8 (88%)	
Reload assem. (no. of batch)	232 (3.8)		300 (2.9)	
Loaded heavy metal (U+Pu)	149 t		115 t	
Operation cycle length	15 EFPM		15 EFPM	
Cycle burn-up (GWd/t)	12.0		15.5	
Fissile Pu [Pu(fiss)] enrichment	3.6 w/o	4.9 w/o	3.5 w/o	4.8 w/o
Ave. discharge exposure (GWd/t)	45.1	45.1	45.1	45.1
Max. assembly exposure(GWd/t)	49.7 [55.0]	50.4 [55.0]	52.5 [55.0]	53.6 [55.0]
Cold shutdown margin (%Δk)	3.1 [1.0]	3.8 [1.0]	4.9 [1.0]	3.8 [1.0]
MLHGR (kW/m)	37.5 [44]	35.0 [44]	36.9 [44]	36.2 [44]
MCPR	1.45 [1.3]	1.54 [1.3]	1.31 [1.3]	1.40 [1.3]
Void coefficient (Δk/k/%V)	-9.5×10^{-4}	-7.4×10^{-4}	-9.1×10^{-4}	-8.5×10^{-4}
Doppler coefficient (Δk/k/°C)	-1.01×10^{-5}	-1.04×10^{-5}	-0.84×10^{-5}	-0.88×10^{-5}
Delayed neutron fraction (%)	0.45	0.45	0.44	0.44

[ ] Limiting value.

Figure A.1. Concept of Pu multi-recycling for high-moderation LWRs



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