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Impact of High Burnup Uranium Oxide and Mixed Uranium–Plutonium Oxide Water Reactor Fuel on Spent Fuel Management

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IMPACT OF
HIGH BURNUP URANIUM OXIDE AND MIXED URANIUM–PLUTONIUM OXIDE WATER REACTOR FUEL ON SPENT FUEL MANAGEMENT

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FOREWORD

The use of uranium oxide (UOX) nuclear fuel with higher enrichments and burnups has been increasing around the world as the reliability of UOX fuel increases and the economics of moving to higher burnup fuel improves. Burnup extension affects all important stages of the nuclear fuel cycle and thus concerns the entire nuclear industry. There are many aspects of switching to higher burnup UOX or MOX fuels, such as reliability, safety, sustainability and economics that decision makers should take into account prior to switching fuels or increasing the burnup limits.

The potential changes to the fuel rods and assemblies will affect the operation of the components of the back end of the fuel cycle, in particular, wet and dry storage, transportation and final disposition by either direct disposal in a repository or reprocessing. This report provides information on the impacts on spent fuel management to those countries operating LWRs and HWRs with zirconium alloy-clad UOX fuels who are considering the use of higher burnup UOX or the introduction of reprocessing and MOX fuels.

Despite intensive studies of extended burnup, there continues to be a need for additional information since the first Water Reactor Fuel Extended Burnup Study (WREBUS, started in 1988) was conducted. This report has compiled the latest available data on high burnup of UOX and MOX fuels and their potential influence on spent fuel management. The analysis in this report is focused on the back end of the fuel cycle. It compares lower burnup UOX fuel and cladding types from LWRs and HWRs having zirconium alloy based cladding and structural materials to either higher burnup UOX or MOX for the same reactors and cladding types.

This publication is a result of a technical meeting in 2001 followed by six consultants meetings. The last two meetings, when this report was finalized, were held in 2006 and 2007. The contributions of the meeting participants and assistance from other experts are appreciated. Special acknowledgement is given to R. Einziger and W. Goll for their assistance in compiling this report.

The IAEA officers responsible for this publication were Z. Lovasic and X. Zou of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

SUMMARY

There is increasing worldwide use of uranium oxide (UOX) nuclear fuel with higher enrichments and burnups as the reliability of UOX fuel increases and the economics of moving to higher burnup fuel improves. Burnup extension affects all important stages of the nuclear fuel cycle and thus concerns the entire nuclear industry. There are many aspects of switching to higher burnup UOX or MOX fuels, such as reliability, safety, and economics, that decision makers need to take into account.

The potential physical changes to the fuel rods and assemblies will affect the operation of the components of the back end of the fuel cycle. The objective of this report is to provide information on the impacts on spent fuel management to those countries operating LWRs and HWRs with zirconium alloy clad UOX fuels who are considering the use of higher burnup UOX or the introduction of reprocessing and MOX fuels.

The mechanical designs of lower burnup UOX and higher burnup UOX or MOX fuel are very similar, but some of the properties of higher burnup UOX and MOX are potentially significant. Examples of the differences in properties between lower burnup UOX and higher burnup UOX and MOX include: higher fuel rod internal pressures; higher decay heat; higher specific activity; and degraded cladding mechanical properties. Higher burnup UOX or MOX usage affects all spent fuel management components, such as wet and dry storage, transportation, reprocessing, re-fabricated fuel and final disposal.

This report briefly reviews the current fuel characteristics of UOX and MOX and the potential for characteristic changes with increased burnup. In addition, the components of the back end of the fuel cycle are discussed. Evaluation of these effects on the back end of the fuel cycle was based on the particular fuel behaviour, regulatory, safety, sustainability, or operational issues that might be impacted by the increased burnup or switch to MOX. Other than a brief mention, an economic evaluation of the switch was beyond the scope of this report. Political and strategic considerations were also not taken into account. Furthermore, the relative importance of the technical, economic, and other considerations will vary from country to country. Analysis was limited in some evaluations due to the amount of available data.

Because wet storage is associated with low temperatures, cladding degradation is expected to be low. High burnup UOX and MOX storage will increase the heat load, and potentially radioactive releases. This may require an upgrade of the pool facility with respect to heat removal and pool cleanup systems, and additional neutron poison material in the pool water or in storage racks. Re-evaluation of criticality and regulatory aspects may also be required.

In dry storage and transportation, the cask has to provide safe confinement/containment. In parallel, the decay heat has to be removed to limit temperature induced material alterations. Thus, dry storage is more sensitive to increased UOX burnup and MOX use than wet storage because of higher temperatures resulting in higher stresses on the cladding. The ability to meet applicable regulatory limits will need to be re-evaluated for higher burnup UOX and MOX on a case by case basis. Sub-criticality during transportation has to be ensured even under accident conditions, such as, for example, cask drop. Higher burnup fuel may have significantly more hydrogen in the cladding and structure and, thus, reduced ductility. Since MOX fuel has a similar design to UOX fuel, its mechanical behaviour should not be different. The result of these evaluations for storage and transportation may require a redesign of the cask heat removal and shielding systems, redesign of the structural support for the spent fuel assemblies, a decrease in the number of spent fuel assemblies that can be placed into a single storage cask, and an increased decay time in the pool prior to placement in dry storage.

Reprocessing plants are designed and licensed for maximum burnup and enrichment. Increased decay heat places an additional load on plant cooling systems. Increased neutron activity requires radiometric instruments (used to control criticality) to be recalibrated. Increased alpha activity results in increased heat generation. Increased specific activity in reprocessing phases may result in higher radionuclide discharges to the environment and may result in more HLW. These effects can be managed using blending schemes. As the burnup exceeds a certain level, a new reprocessing facility may be needed. The reprocessing of spent MOX fuel presents additional challenges due to the lower solubility of plutonium.

For fuel disposal, higher burnup UOX and MOX fuel means higher source terms of the radionuclides leading to a potentially higher release to the groundwater. Higher heat loads could exceed temperature limits in a repository. This may require significant repository operational changes to accommodate higher burnup UOX and MOX, such

as increased repository space (although the reduced volume of higher burnup UOX may counteract the need for additional space), smaller waste containers, longer decay times at the surface prior to loading into the repository, and additional shielding during spent fuel transfer from the transportation cask.

If reprocessing is the disposition method of choice the increase in discharge burnup has a significant effect on the isotopic quality of recycled fuel. Increased enrichment of the reprocessed uranium or an increased amount of plutonium in MOX fuel will be required to meet the same burnup target. Increases in shielding may be required for fuel refabrication operations.

1. INTRODUCTION

1.1. BACKGROUND

Most countries operating light water reactors (LWRs) or heavy water reactors (HWRs) use uranium oxide nuclear fuel (UOX). Historically, ^{235}U enrichment and burnup of UOX has been in the range of 3 to just under 5% w/o, and 30–45 GW·d/t U, respectively. A considerable amount of data on spent UOX properties with enrichments and burnups in this range have been collected. Systems to manage the back end of the fuel cycle, wet and dry UOX storage, transportation, reprocessing, reprocessed fuel fabrication, and disposal systems have been designed for spent UOX fuel with such properties.

There has recently been a worldwide increase in use of uranium oxide (UOX) nuclear fuel with higher enrichments and burnup as the quality of UOX fuel increases and the economics of moving to higher burnup fuel improves. Higher burnup can be obtained by various means. The most common way is to use fuel with higher enrichment but other methods, such as; subdivision of fuel to reduce rod thermal loads, reconstitution of spent fuel bundles, and reshuffling of spent fuel bundles are also used. The economy of this process is well established and is implemented by using the newer centrifuge technology for enriching uranium.

There is increasing use of mixed oxide (MOX) fuel (a combination of plutonium and uranium), the primary product of spent UOX reprocessing, as the use of reprocessing increases. There are different reasons for reprocessing spent nuclear fuel, such as; improved utilization of fuel, energy independence, sustainability, and the disposition of excess plutonium from weapons production. Thus, it is anticipated that more countries currently using moderately enriched and burned UOX fuel will consider using higher enriched and burned UOX and/or using reprocessing and MOX fuel.

In general, higher burnup UOX and MOX spent fuel assemblies have higher decay heat, higher specific activity, and degraded fuel cladding mechanical properties compared to similar UOX fuel assemblies with lower enrichment and burnup. The properties of higher burnup UOX and MOX fuels make spent fuel management more challenging. Therefore, it is important for those countries considering the use of higher burnup UOX or MOX fuels, to take into consideration the properties of these fuels prior to making the decision to use them.

1.2. OBJECTIVE

There are many aspects of switching to higher burnup UOX or MOX fuels that decision makers should take into account. The objective of this report is to provide information, to those countries operating LWRs and HWRs with zirconium alloy clad UOX fuels, on the impacts of the use of higher burnup UOX or the introduction of reprocessing and MOX fuels on spent fuel management.

1.3. SCOPE OF THE REPORT

While it is recognized that there are many reactor types other than LWRs and HWRs, and many fuel types other than zirconium alloy clad UOX that are in use today, the majority of reactors and fuel types are in these categories. Furthermore, the body of knowledge about higher burnup fuels and fuel from reprocessing is almost exclusively for reactors and oxide fuels of these types. Thus, this report only considers the impacts on spent fuel management of higher burnup UOX or MOX zirconium alloy clad spent fuel produced in LWRs or HWRs.

There is a growing amount of research and development (R&D) with the aim of closing the nuclear fuel cycle. This R&D is in the early stages. Therefore, this report is further limited in scope to just two nuclear fuel cycles: ‘once-through’ (single use of UOX fuel followed by storage, transportation, and direct disposal); and MOX fuel generated from the first reprocessing and refabrication of spent UOX fuel followed by storage, transportation, and direct disposal. Fuel cycles for both LWRs and HWRs are considered. For HWRs, the use of processed spent LWR UOX and enriched, fresh UOX fuel is also discussed.

This report is organized as follows:

- Section 1 briefly discusses reactor types, with an emphasis on LWR and HWR technology, and the nuclear fuel cycles considered in this publication;
- Section 2 discusses the current state of UOX and MOX worldwide;
- Section 3 provides information on the various fuel and cladding types and spent fuel management components (wet and dry storage; transportation; reprocessing and MOX fabrication, and disposal). It concludes with design and operation criteria, for the back end of the nuclear fuel cycle, that affect burnup and fuel type choices considered in this report;
- Section 4 provides more detailed information on the characteristics of spent fuel related to higher burnup UOX and MOX fuels;
- Analysis of the effects of using high burnup UOX and MOX fuels on spent fuel management is provided in Section 5;
- Conclusions are provided in Section 6.

1.4. REACTOR TYPES

This section provides a brief description of reactor types that are presently in use throughout the world, with emphasis on the reactor types analyzed in this report.

1.4.1. Light water reactors

The majority of reactors currently operating worldwide are LWRs. Given that the neutron absorption cross-section of hydrogen is significant, uranium based fuel for LWRs are typically enriched in ^{235}U between 3% to just below 5% w/o. All LWR reactor fuel cycles require enrichment facilities.

1.4.1.1. Pressurized light water reactors (PWRs)

PWRs operate in a pressure and temperature range such that no boiling or only nucleated boiling occurs at the fuel surface. PWRs recirculate water from the reactor through steam generators through a ‘primary loop’. The steam generator transfers heat from the high pressure and temperature water of the primary loop into a separate, lower pressure water circulation system (the ‘secondary loop’) via a series of steam generator tubes. The generated steam is then used to drive the turbines for electricity generation.

There are a multitude of different light water moderated PWR fuel types. While early PWR fuel assemblies used stainless steel for cladding, all modern PWR fuel uses zirconium alloy based cladding.

1.4.1.2. Water cooled water moderated energy reactor (WWER)

The WWER is a Russian version of a pressurized water reactor. As in a PWR, there are several versions of this design (i.e. WWER-440, Models V230, V213 or newer WWER-1000). It has a primary pressurized loop that circulates hot water through a steam generator where it boils water in the secondary loop. A pressurizer maintains pressure in the primary loop by means of electrical heating and relief valves.

1.4.1.3. Boiling water reactor (BWR)

Water entering the bottom of a BWR reactor core is heated to boiling. Unlike the PWR and WWER designs that employ primary and secondary water loops, the steam produced in BWRs is introduced directly into the turbines used for electricity generation. Due to the differences in water moderation from the bottom to the top of a BWR reactor core, BWR fuel often has higher ^{235}U enrichments toward the top of the core. Thus, there is usually a significant difference in the isotopics along the length of spent BWR fuel assemblies.

1.4.2. Pressurized heavy water reactor (PHWR)

A PHWR was originally developed by Atomic Energy of Canada Ltd (AECL) but was later further adapted and developed by India and the Republic of Korea. The CANDU type reactor uses natural uranium as fuel. Heavy water is used as a moderator and coolant in order to minimize the neutron absorption cross-section. Moderator and cooling functions are separated so that the design could be optimized.

Since natural uranium is used as fuel, only limited burnup can be achieved and frequent refuelling is required. This is accomplished 'on-line' by the use of special refuelling machines that add fresh bundles at one side of the core and remove spent fuel bundles at the other. The consequence of this design is the generation of larger quantities of spent fuel. The use of slightly enriched fuel is being contemplated.

1.4.3. Other reactor types

There are many other reactor types, some of which use fuels with increasing burnups; two examples are provided below. These other reactors are not included in the scope of this report because only a few reactors of a particular type are in commercial operation, fuels with higher burnups are not being considered for them, or no additional reactors of a particular type are planned for the future.

1.4.3.1. Advanced gas cooled reactors (AGRs)

AGRs are the second generation British gas cooled reactors, using graphite as the neutron moderator and carbon dioxide as the coolant. The fuel consists of annular UO_2 pellets placed in a stainless steel can. A series of burnup extension programs have been undertaken with AGR fuel taking the current limit to 3.7% ^{235}U initial enrichment and a burnup of 40 GW·d/t U. Spent fuel management consists of wet storage, followed by reprocessing, although plans are currently in place to develop extended wet storage for up to 80 years. As AGR reactors are only operating in the UK and the spent fuel management is special to this fuel type. AGR fuel is not considered further in this report.

1.4.3.2. Reactor Bolshoj Mochi, Kanalni (RBMK)

The RBMK is a channel type graphite moderated reactor with long fuel bundles (11m). Water comes from below and passes through the channel where steam is generated going directly to the turbine. A special characteristic of the RBMK plant is that it did not have containment.

RBMK nuclear power plants will not be constructed in the future. A decision was postponed on the management of the spent fuel. It is most probable that it will be disposed.

1.4.3.3. 'Gen III+' and 'IV' advanced reactor systems

Current reactors in operation around the world are generally considered second or third generation systems, with the first generation systems having been retired some time ago. Future reactor designs are evolving toward; 'Generation IV' reactors (Gen IV). Gen III reactors, such as the AP1000 (PWR) and the advanced boiling water reactor (ABWR), are either already in use, or will be within the next two decades. Gen III+ reactors are incremental improvements on current 'Gen III' technology to improve economics. Gen III+ reactors are all water moderated using neutrons in the thermal spectrum such that the primary fissile isotopes remain ^{235}U and $^{239/241}Pu$.

Gen IV reactors are a set of nuclear reactor designs currently being researched or in early stages of development. Most of these designs are generally not expected to be available for commercial construction before 2030, with the exception of a version of the very high temperature reactor (VHTR) called the Next Generation Nuclear Plant (NGNP). The NGNP is to be completed by the early 2020s. Research into these reactor types was officially started by the Generation IV International Forum (GIF) based on eight technology goals. The primary goals being to improve nuclear safety, improve proliferation resistance, minimize waste and natural resource utilization, and to decrease the cost to build and run such plants.

Gen IV reactors being designed and in early stages of development generally employ fast-spectrum neutrons, as opposed to water moderated reactors addressed in this report. Fast reactor designs generally use liquid sodium or lead/bismuth eutectic coolants as the primary loop with water as the secondary loop. However, there are also Gen IV designs using helium as the primary loop. Fuel types being developed for Gen IV reactors include not only oxide, but carbide, nitride, and metal forms.

Given that all of these advanced reactor designs are not yet used commercially (with the exception of the ABWR), they are not considered in this report.

1.5. NUCLEAR FUEL CYCLES

A schematic of a generic, uranium based nuclear fuel cycle is shown in Fig. 1. All of the LWR fuel cycles share the following common front end stages:

- ***Uranium mining and milling*** — Uranium ore is extracted and concentrated from underground mines, open pit mines, or through in situ leaching;
- ***Conversion*** — Uranium concentrates are converted into uranium hexafluoride, UF_6, which is solid at ambient temperature and vaporizes at moderately high temperatures;
- ***Enrichment*** — Isotopic enrichment of UF_6 is used to increase the concentration of the fissile ^{235}U (typically to an enrichment of 3% to 5% for commercial power reactors. Currently a significant quantity of enrichment service is from blended down high enriched uranium for Russian nuclear weapons.);
- ***Fuel fabrication*** — Conversion of the fluoride into uranium dioxide, UO_2, fuel pellet manufacturing, fuel rod manufacturing and assembly of the rods into bundles;
- ***Thermal reactor operation*** — UO_2 fuel is inserted into the core of a commercial nuclear power reactor for a period of about 4 to 6 years for LWRs and 13 to 15 months for HWRs. During thermal reactor operation, part of the ^{235}U fissions, releasing energy and generating fission products, while other uranium isotopes, primarily ^{238}U, are converted into other long lived actinides such as plutonium. Some of the plutonium isotopes are also fissile in a thermal reactor, but mostly just buildup in UO_2 fuel during reactor operation.

HWR fuel cycles do not require enrichment in ^{235}U. Hence, the conversion and enrichment steps above do not apply for HWRs. However, some HWR operators are considering the use of slightly enriched UOX such that some conversion and enrichment would also be required. Yet other advanced HWR schemes involve the direct use of spent LWR UOX fuel after appropriate spent LWR defueling and refabrication. While several countries use MOX fuel in LWRs, use of MOX is also being investigated for HWRs.

In Fig. 1, arrows indicate potential transportation steps. The variations in these fuel cycles occur following the initial discharge of the spent UOX fuel from a LWR or HWR. Fuel cycle schemes can be broadly categorized as follows:

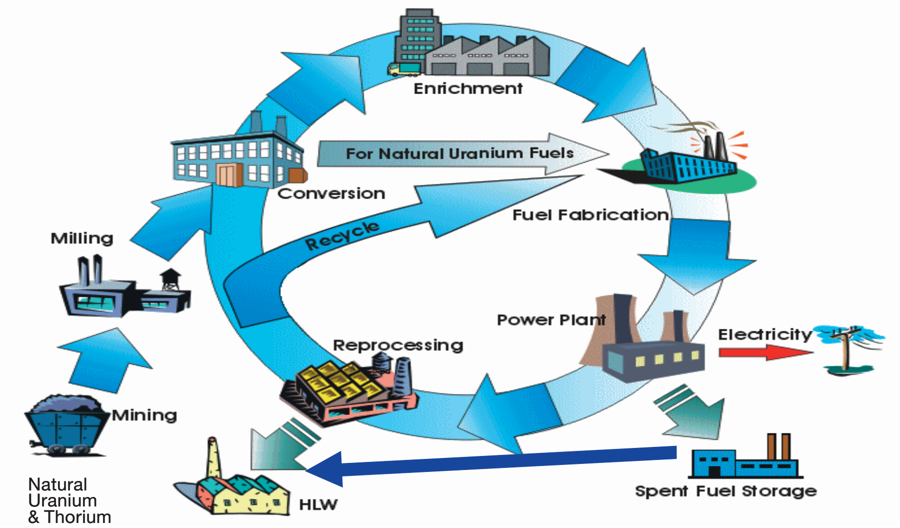


FIG. 1. Nuclear fuel cycle schematic.

— ‘Open’ cycles: ‘once-through’ (single) use of fuel followed by temporary storage and direct disposal of the spent fuel;

— ‘Partially closed’ cycles: recycle of spent fuel after reprocessing and fabrication of the spent UOX fuel into MOX, followed by single use of MOX. After irradiation of this MOX it would be temporarily stored and then directly disposed. Achieving higher burnups for once-through fuel can be considered another way to increase the use of the fissile material. Only currently available one-time reprocessing of spent UOX fuel is considered in this report;

— ‘Fully closed’ cycles: maximum use of uranium and other actinides via a combination of reactors (e.g. LWRs and FBRs), separations, fuel fabrication, and other transmutation technologies are not considered in this report.

In 2006, the NEA considered a large number of alternative fuel cycles and their effects on waste management [1]. Figure 2 shows three schemes based on current industrial technology and possible extensions. Scheme 1a is the once-through fuel cycle; scheme 1b is the ‘conventional reprocessing’ fuel cycle involving UOX reprocessing followed by once-through MOX use.

2. UOX AND MOX FUEL BURNUP

2.1. SPENT FUEL STATISTICS AND TRENDS TOWARDS USE OF HIGH BURNUP UOX AND MOX

Currently, about 10 500 tonnes heavy metal（t HM）of spent fuel are unloaded every year from nuclear power reactors worldwide (Fig. 3). This is the most important continuous growing source of civil radioactive materials generated, and thus need to be managed appropriately. This annual discharge amount is estimated to increase to some 11 500 t HM by 2010. The total amount of spent fuel cumulatively generated worldwide by the beginning of 2004 was close to 268 000 t HM of which 90 000 t HM has been reprocessed. The world commercial reprocessing capacity is around 5 550 tons per year. Projections indicate that the cumulative amount of spent fuel generated by the year 2010 may be close to 340 000 t HM with a corresponding increase in reprocessed fuel and by the year 2020, the total quantity of spent fuel generated will be approximately 445 000 t HM.

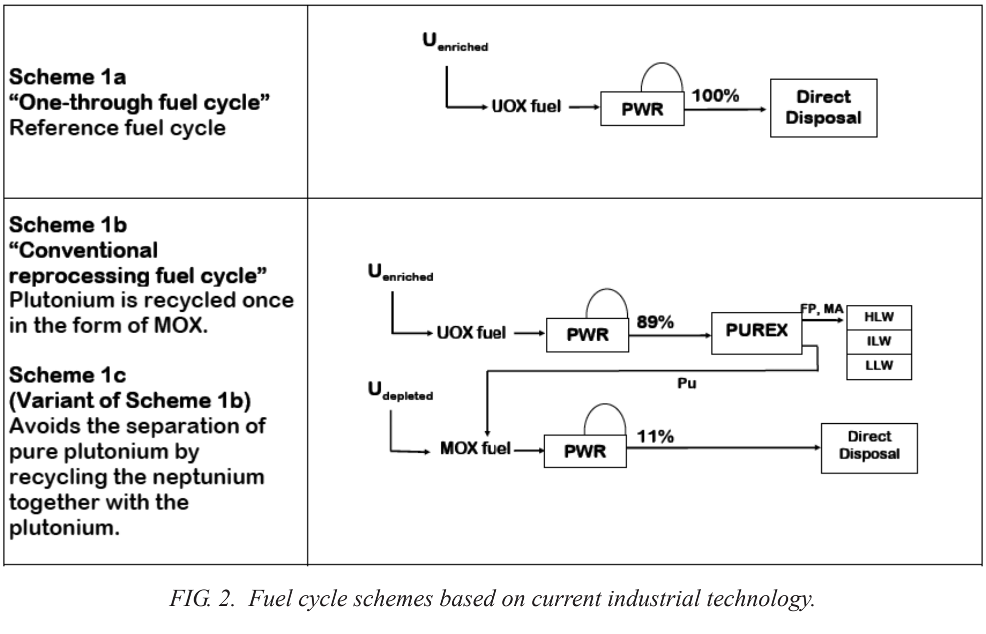


FIG. 2. Fuel cycle schemes based on current industrial technology.

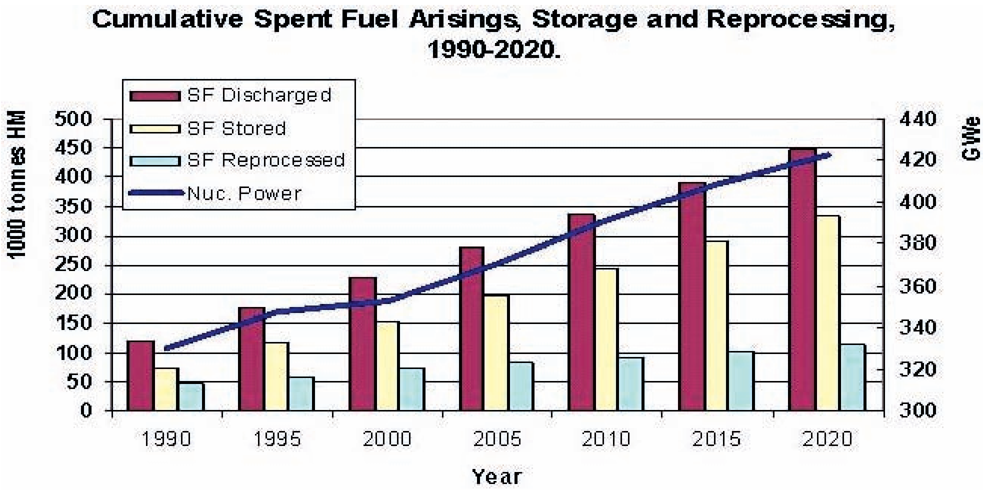


FIG. 3. *Cumulative spent fuel inventories showing reprocessing and storage shares.*

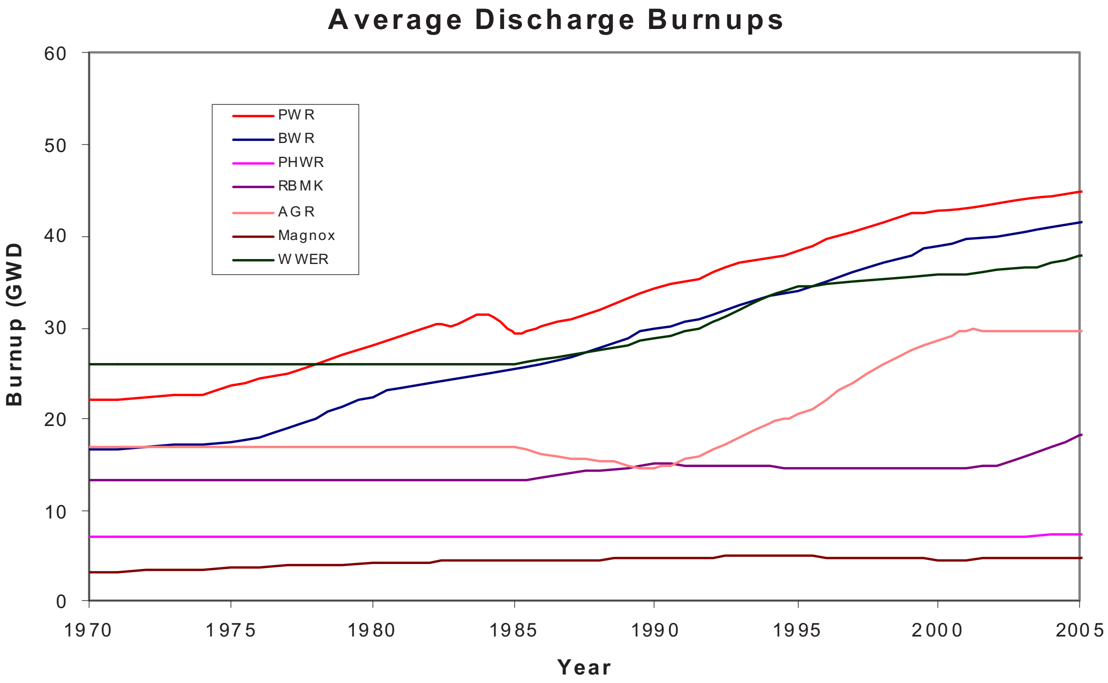


FIG. 4. *Burnup trends for various types of fuel.*

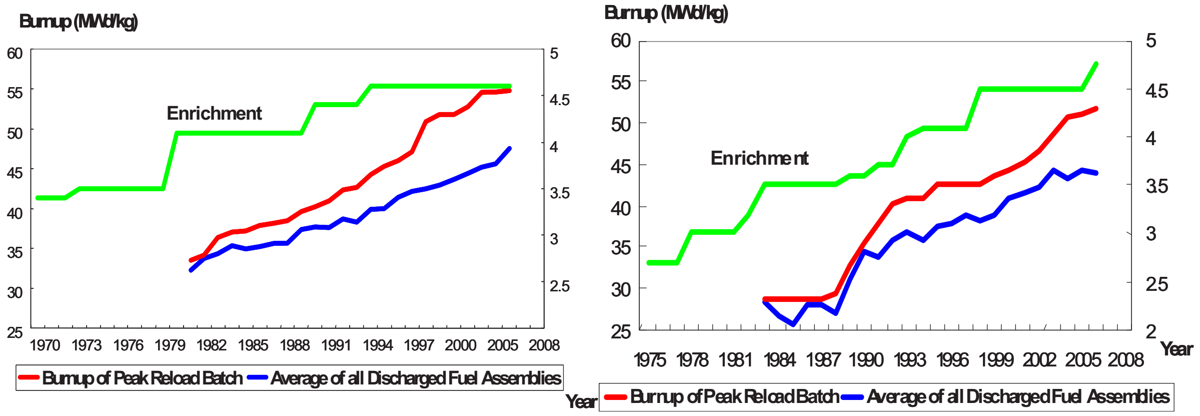
The primary driver for increased burnup is improvement of the fuel economics, but there are also implications on spent fuel management in quantities and characteristics of spent HEU fuel. There are significant quantities of spent nuclear fuel (SNF) requiring storage and any trend towards minimizing this amount is beneficial for spent fuel management.

2.1.1. UOX fuel

Figure 4 shows increased burnup trends in the world throughout years of nuclear power plants operation.

2.1.1.1. Germany

Figure 5 a and b show the increasing burnup of both PWR and BWR fuel since the early 1980s. This growth in burnup could only be achieved by increasing the enrichment of the fuel. In Germany, the enrichment is the



(a) Burnup development and enveloping enrichment for PWR fuel (Status 2006).

(b) Burnup development and enveloping enrichment for BWR fuel (Status 2006).

FIG. 5. Burnup trends in Germany for PWR and BWR reactors.

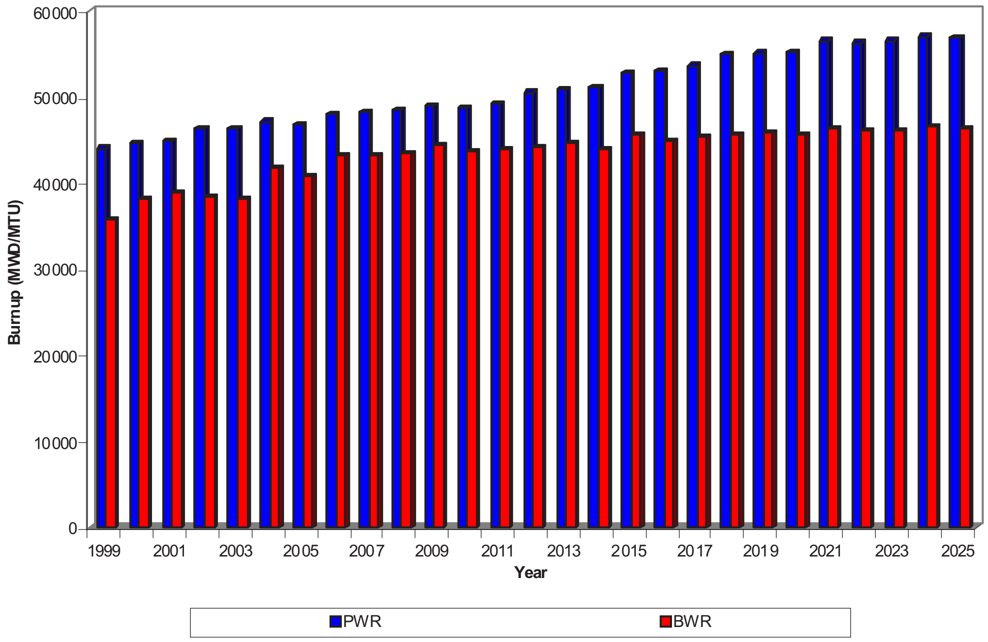


FIG. 6. Burnup trends for PWR and BWR reactors in the USA (Courtesy of Energy Resources International).

primary licensing criterion and can differ from reactor to reactor in parallel with the maximum burnup. The current maximum enrichment approaches 5 w/o, which is currently a licensing limit for handling non-irradiated fuel during manufacturing or storage. However, most of the reactors are licensed for lower enrichments between 4 and 4.4 w/o. This means only a few fuel assemblies will exceed a maximum fuel assembly burnup of 65 GW·d/t U. In the long term, the burnup may tend to stabilize at this high level.

2.1.1.2. USA

Figure 6 is a projected trend in PWR and BWR spent fuel batch-average burnup in the USA for 1999–2025. Burnup levels for spent PWR fuel are anticipated to rise to ~55 GW·d/t U; burnup levels for spent BWR fuel will likely increase to over 40 GW·d/t U. The history of burnup and enrichment in the USA follows the same trend as in Germany.

2.1.1.3. Russian Federation

The WWER-440 nuclear power plants (NPPs) were designed for 3 year cycles with a batch average burnup of ~33 GW·d/t U, (maximum assembly burnup of ~36 GW·d/t U and fuel rod average of ~41.5 GW·d/t U). Some NPPs are already using 4 year cycles with a batch average burnup of ~41–43 GW·d/t U, (maximum assembly burnup of ~47 GW·d/t U and fuel rod average of ~51 GW·d/t U). Combined cycles (4 and 5 year cycles) have been tested at the Kola NPP with a maximum burnup of ~49 GW·d/t U and fuel rod average of ~55 GW·d/t U).

Currently, the Kola NPP operates 5 year fuel cycles with a batch average burnup of ~51.5 GW·d/t U, (maximum assembly burnup of ~54 GW·d/t U, fuel rod average of ~61 GW·d/t U and maximum fuel pellet ~67 GW·d/t U).

The WWER-1000 NPPs were designed for 2 year cycles with a batch average burnup about ~29 GW·d/t U, (maximum assembly burnup of ~36 GW·d/t U and fuel rod average of 40 GW·d/t U). The majority of NPPs with WWER-1000 reactors are already using 3 year cycles with a batch average discharge burnup of ~44 GW·d/t U, (maximum assembly burnup of ~49 GW·d/t U and fuel rod local maximum of ~60 GW·d/t U).

The possibility of 5–6 year cycles for WWER-440 and 4 year cycles for WWER-1000 is under consideration. It has to be proven that the fuel rod design will allow a fuel rod local maximum burnup of more than 65 GW·d/t U [2, 3].

2.1.2. MOX fuel

2.1.2.1. France

France chose the closed fuel cycle at the very beginning of their nuclear programme. In France, the closed fuel cycle increases the nuclear power programme's sustainability by reprocessing and recycling nuclear fuel. Approximately 1100 t of irradiated fuel is unloaded every year from French nuclear reactors. After a few years of cooling down in pools at the reactor sites, irradiated fuel is shipped to the AREVA NC La Hague facility where it is stored in pools for an additional cooling period before being reprocessed. There is no dry storage of spent fuel in France. The first reprocessing facility and HLW vitrification facility was located in Marcoule (Rhone Valley). The plant was shut down in 1998.

All MOX fuel is now fabricated at the AREVA NC MELOX located in Marcoule. The capacity of the plant is currently 145 t HM/a but it could be increased to 195 t HM/a when approved by French Authorities. MOX fuel loaded to French PWR's is licensed for an average burnup of about 40 GW·d/t HM. In order to further improve the performance of rectors in operation, studies have been undertaken to further improve MOX and it's in core management. The objective is to implement parity management ensuring that MOX fuel delivers an energy equivalent to the UO_2 fuel with 3.7% enrichment in ^{235}U.

MOX fuel is stored in pools at the reactor sites and later transferred to AREVA NC La Hague Facility where it is further stored in pools until it is cooled enough for reprocessing. Spent MOX fuel assemblies are dispersed among uranium fuel assemblies to mitigate criticality and cooling requirements without penalty for storage space occupation. Typically, spent MOX fuel requires cooling periods (prior to transfer for reprocessing) that are two to six times longer than for uranium fuel.

2.1.2.2. Germany

In Germany, 11 out of 17 operating reactors are licensed for MOX fuel. The maximum share of the core of MOX fuel is 38 % for a BWR and up to 50 % for a PWR. In general, the maximum fuel rod burnup remains below 55 GW·d/t U (average batch burnup 40–50 GW·d/t U) according to licensing constraints for most of the plants. Few fuel assemblies reached maximum fuel assembly burnups of 58 GW·d/t U (BWR) and 62 GW·d/t U (PWR).

2.1.2.3. USA

The use of MOX fuel is extremely limited given that reprocessing of commercial spent fuel is not currently done in the USA. Only one commercial reactor in the USA (Catawba) is operating with MOX assemblies. This reactor is using plutonium obtained from a stockpile of US 'excess' weapons related plutonium as part of a joint

US/Russian agreement. These lead MOX assemblies were fabricated in France from the US stockpile, and then returned to the USA. A MOX fuel fabrication facility is planned for the Savannah River site in the south-eastern USA in order to convert the remaining excess plutonium stockpile to fuel to be used in commercial reactors.

3. COMPONENTS OF THE ANALYSIS

3.1. FUEL AND CLADDING TYPES

Many different types of fissile forms and cladding materials have been used in the past or are currently in use. These include oxide, carbide, metallic, and hydride forms of uranium with and without alloying components such as zirconium. These have been formed into a variety of shapes such as solid or hollow pellets, coated microparticles that are sintered into larger balls, metallic slugs or plates. Cladding is made of stainless steel, aluminum, magnesium, graphite, and a variety of alloys of zirconium. Most fuel currently used in power reactors and most likely to be taken to high burnup is UO_2 pellets in a zirconium alloy cladding. As recycling becomes more prevalent, more MOX fuel will be entering the fuel cycle for LWR fuel. Since fuels with fissile material other than UOX and MOX are only minor players in commercial fuel and claddings other than those that are zirconium based is either falling out of use or still in the experimental stage, only zirconium alloy clad oxide fuels will be considered further.

3.1.1. General description of the fuel and its irradiation

The preponderance of the fuel used in power reactors is made from sintered UO_2 right circular pellets that are either solid or hollow. The uranium is enriched up to 5% in PWR and BWR fuel and natural to be slightly enriched in HWR fuel. These pellets are enclosed in a cladding tube made from an alloy of zirconium. There is usually an initial gap between the fuel pellets and the cladding that may or may not close with irradiation. A void space is left at either the top or bottom of the pellet stack to accommodate any fission gases that are released from the fuel pellets during irradiation. Most fuel rods are filled with helium gas to aid in thermal conductivity to reduce the operating temperature of the fuel.

Older PWR and BWR fuel were clad in ZIRCALOY^{TM}-4 and ZIRCALOY^{TM}-2 respectively. The ZIRCALOY^{TM}-4 was metallurgically treated to form circumferential hydrides during irradiation while the ZIRCALOY^{TM}-2 had a random grain texture. The rods were held in arrays from the smaller 6 × 6 BWR fuel to the 17 × 17 PWR fuel. While there is some variation in length, the fuel rods are generally about 4 m long. A BWR assembly is held together with tie rods and surrounded by a solid channel. The control blades are external to the assembly resulting in assembly bow during irradiation. The PWR assemblies are held together by multiple control rod tubes that contain the control rods during irradiation.

PHWR fuel uses natural uranium oxide as fuel. MOX is not used in PHWR fuel. Presently there are 28 and 37 element fuel bundle designs[1] that are used in CANDU PHWR reactors. Both fuel designs have approximately 20 kg of fuel in each bundle and are approximately 50 cm long. A 37 element bundle showing components of CANDU fuel is shown in Fig. 7.

UO_2 pellets are placed in ZIRCALOY^{TM}-4 tubes that have ZIRCALOY^{TM} caps welded at both ends of the tube. Fuel bundles consist of fuel elements held together by welds attaching the end caps of each fuel element to two ZIRCALOY^{TM}-4 end plates. This type of fuel bundle, unique to CANDU fuel, makes each element an active component of the bundle structure and mechanically constraints each ZIRCALOY^{TM} tube due the rigid attachment of each end cap to an end plate. In comparison, PWR fuel is free to expand axially and does not have the same constraints and stresses as CANDU fuel.

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[1] In CANDU terminology, element is equivalent to fuel rod in other fuel designs.

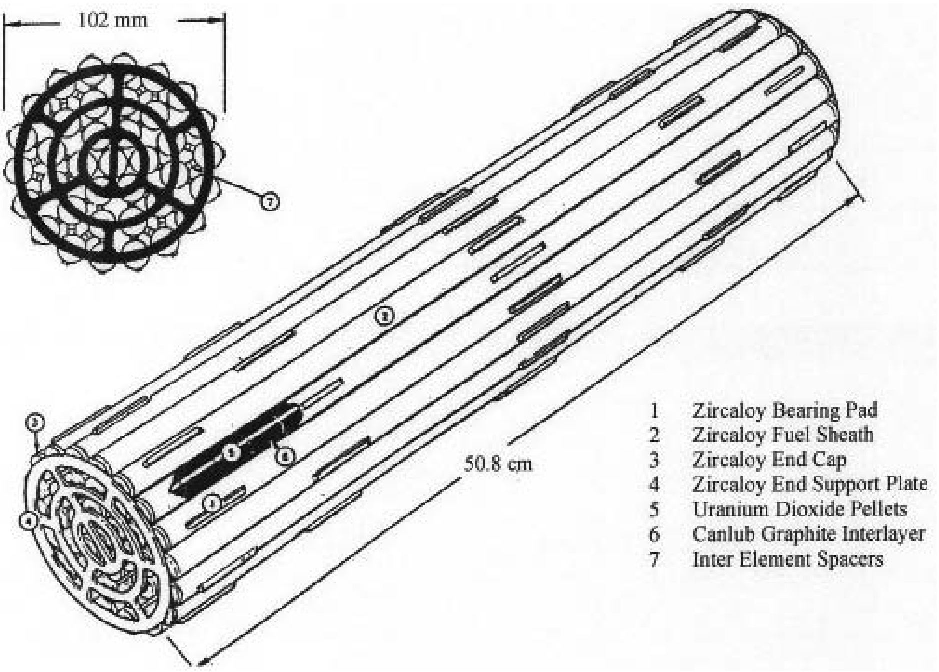

FIG. 7. CANDU 37 element fuel bundle.

ZIRCALOY$^{\text{TM}}$ spacers are attached to the surface of fuel elements by Zr-5wt% Be (5% by weight) brazing. Heating introduced during the brazing raises the temperature of the cladding near the spacers, driving the localized cladding material into the beta phase region of the ZIRCALOY$^{\text{TM}}$ phase diagram. This changes cladding mechanical properties by enhancing its ductility and decreasing its strength.

Indian HWRs also use natural uranium oxide fuel encapsulated in ZIRCALOY$^{\text{TM}}$-4 similar to the CANDU design. They use 19 and 37 element bundles. ZIRCALOY$^{\text{TM}}$ spacers are attached to the surface of the fuel element by resistance welding that has a negligible effect on mechanical properties on the cladding because of lower heat input.

Both WWER-440 and WWER-1000 reactors use fuel consisting of hollow UO_2 pellets in a cladding of Zr with 1wt% Nb. Pellet enrichment ranges between 3.6 and 5% depending on the reactor. An assembly is made up of between 66 and 84 rods. The maximum burnup of the WWER-440 fuel ranges between 42 to 65 GW·d/t U and fuel rods in the WWER-1000 can achieve burnups as high as GW·d/t U. The higher burnups are achieved by fuel manoeuvring and by introduction of optimized zirconium cladding alloys.

3.1.2. MOX fuel

MOX fuel is generally fabricated by mechanically blending PuO_2 obtained from reprocessing with natural or depleted UO_2 powder. For LWR fuels, the Pu content is typically less then 10 wt%, therefore, the fuel remains primarily UO_2. The fabrication methods generate Pu rich islands, the size and concentration of which depend in detail on the methods of fabrication [4].

Pellet and rod design of MOX fuels is generally similar to that of UO_2 fuels. The same cladding materials are used. Some small changes in terms of plenum length adjustments may be made.

The mechanical design of MOX fuel assemblies is also similar to that of conventional fuels in terms of grids, nozzles, guide tubes etc. The main difference is the neutronic design of the fuel assembly. Due to differences in the neutron absorption characteristics of U and Pu isotopes, rods of reduced enrichment are placed on the side and corners of MOX assemblies to reduce power peaking effects at the interfaces with UO_2 fuel. In addition, some assembly designs include additional water rods to improve moderation. To date, no use of integral burnable absorbers within MOX fuel pellets has been deployed on an industrial scale. If absorber rods are utilized, they tend to be UO_2-Gd fuel added to MOX assemblies.

To date, MOX fuels have been loaded in mixed cores where the majority of assemblies remain conventional UO_2 fuel. Typically in PWRs, one third of the core is MOX fuel.

3.2. SPENT FUEL MANAGEMENT COMPONENTS

3.2.1. Wet storage

All LWRs and HWRs have fuel pools in which spent fuel is initially stored. Depending on the reactor design and whether reprocessing is performed, spent fuel may stay in the pool for as short as 12 months and as long as many decades prior to removal. Spent fuel pool capacity varies significantly from plant to plant. Some spent fuel pools were designed to hold spent fuel discharges from just a few years of operation as it was assumed that shipment off the reactor site would occur fairly quickly. Other, more recent reactor designs have much larger spent fuel pools that have the capacity to store spent fuel discharged over 20 or more years of operation. As spent fuel pools have become filled, many utilities have elected to replace their lower-density spent fuel pool storage racks with higher density racks that are capable of nearly doubling the available capacity. However, spent fuel pool capacity is also governed by the ability of plant systems to eliminate decay heat and maintain required pool water quality.

To meet regulatory requirements in SNF wet storage facilities at nuclear power plants and SNF reprocessing plants in the Russian Federation, the following principle design and engineering solutions were taken for LWR and HWR wet storage pools: (1) fuel assemblies are placed in sufficient water to meet the radiation safety shielding requirements; (2) the pool is fitted with systems for water cooling and purification, water make up to maintain the required level, water pool filling and emptying, ventilation of the above water space, engineering and radiation control equipment; (3) the inner surface of the pools are lined with stainless steel; (4) leaks from under the pool lining are controlled and any leakage is captured to prevent pool water from entering neighbouring rooms or the environment; (5) fuel assemblies with damaged fuel rods are stored in tight canisters; (6) radiation monitoring is conducted both inside and outside the pool storage building.

Storage facilities are designed to take into consideration external impacts characteristic of the geographic region in which the pool is located (e.g. tornado, tsunami, earthquake events), initial events which may cause a radiation accident, and failure of equipment and personnel errors.

3.2.2. Dry storage

At many reactor sites, spent fuel storage pool capacities are being exhausted even after re-racking and therefore the operators of nuclear power plants turn to dry storage technologies to satisfy their storage needs. This technology has been implemented worldwide on a large scale since the 1980s and a multiplying number of dry storage facilities have been built in an increasing number of Member States. Dry storage is inherently simple in operation/maintenance and therefore provides a suitable option to overcome the limitations imposed by wet storage for extended storage of spent fuel. It has gained considerable popularity due to its lower cost, modularity and reduced radioactive waste generation. Spent fuel is transferred to dry storage after a prescribed cooling period under water. At the present time, almost all member states are engaged in constructing dry storage facilities to complement the existing wet stores, and are also actively pursuing a dry storage research and development program.

Almost 25 years of favourable experience exists with the dry storage of lower burnup (<45 GW·d/t U)[2] spent power reactor fuel and about 40 years with the dry storage of research reactor fuel. Dry storage experience exists with fuel from a variety of reactor types (CANDU, HWR, PWR, BWR, WWER-440, WWER-1000, RBMK, MAGNOX and the high temperature gas cooled reactor (HTGR)). Spent fuel from LWRs is typically stored in a helium atmosphere. The lower decay heat level of fuel in HWRs allows dry air to be used as the medium for storage once the temperature drops below 150°C.

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[2] This burnup limit for separating lower burnup and higher burnup fuel is a US definition and may vary between 40–50 GW·d/tU depending on the country.

Since its conception, dry storage of spent fuel has evolved into a wide variety of systems. Examples of these are concrete canisters, steel-lined concrete containers and concrete modules or vaults. A variety of storage systems have been developed to meet specific requirements of different reactor fuels and a number of designs based on these generic technologies are now available for the spent fuel containers or vaults (horizontal, vertical, etc). The technology continues to evolve in compliance with changing requirements of the market. One of the driving forces of this trend towards dry storage options (especially those of casks) is the inherent technical flexibility linked to economics. Compared to the pool facilities which need to be built at full capacity initially, the modular type dry storage facilities can be added as needed with the advantage of minimizing capital outlays. For example, concrete modules have also become popular as a competitive option, with more designs licensed and implemented. Concrete modules are merging with vault as a compact storage system which is advantageous when land availability is an issue.

Multi-purpose technologies (i.e. a single technology for storage, transportation and disposal) have also been developed, for instance in the USA and Germany. Inheriting the technology initially developed for transportation of spent fuel to reprocessing operations, several large size casks are now being marketed for storage services.

The quantities being placed into dry storage are increasing significantly. The inventory of spent fuel in dry storage in the member countries, as of 1 January 2006, was about 200 000 t HM [5].

Until such time as a final disposition path is available, fuel may need to be dry stored longer than the current 20–40 years depending on the country. Lifetime extension has been granted in the USA in part due to the favourable experience of 30 GW·d/t U fuel that was examined after 15 years in storage in the USA and other similar experience in Japan and the UK. While analyses indicate that high burnup fuel should be able to be stored for an extended duration, there is no similar experience database to support storage lifetime extension of high burnup fuel. So far, the results of the research indicate that fuel can be stored safely under the present conditions for decades. Experience has already started to accumulate in spent fuel management of higher burnup fuels.

3.2.3. Transportation

Transportation of lower burnup LWR and HWR fuel from reactor sites to independent spent fuel storage installations (ISFSIs), reprocessing plants, and research labs has been successfully conducted for many years in many countries worldwide. In addition, there have also been many successful intra site transfers. Success is gauged by no unacceptable radiation exposure to the workers or public and no release of radioactive material from the cask system. Retaining the integrity of the fuel has not been a criterion for judging success since in most cases the condition of the spent fuel after it is transported is not important because either the fuel will be reprocessed, or the cask has not yet been opened to remove the fuel.

Other than in France where reprocessing is routinely conducted, there has been limited transportation of multiple-assembly[3] high burnup BWR/PWR fuel. In those cases when high burnup fuel has been transported for non-reprocessing reasons, it is usually in the form of a few rods or assemblies so the thermal; shielding or criticality conditions are not the same as with a typical transport. The main concern with the transportation of high burnup fuel in the USA and Germany, for example, is the prevention of criticality occurring due to the loss of structural integrity of the rods as a result of the loss of ductility resulting from hydride reorientation. In addition, creep of the newer claddings, needed to reach higher burnups, is a question in Germany where no cladding breaches of any size are allowed.

The Russian Federation transports high burnup spent nuclear fuel in licensed packages. The TK-6 cask designed for dry shipments of 30 assemblies with thermal loads of 8 kW and burnup of 21–24 GW·d/t U is also used for WWER-440 spent fuel assembly (SFA) wet shipments. Wet shipments allow an increased thermal load of 15 kW. When the burnup of the WWER-440 fuel was raised to a maximum burnup of 60 GW·d/t U, shipment safety in TUK-6 was assured by placing minimal burnup spent fuel assemblies in the outer basket cells and maximal burnup spent fuel assemblies in the rest cells. The preliminary SFA cooling in the at-reactor ponds was extended up to 4–5 years, compared with the initial 3 years, in order to meet radiation and thermal levels required by the Regulations for the Safe Transport of Radioactive Materials. The minimal estimated at reactor SFA cooling time to

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[3] Above 20 assemblies in a single cask.

assure the safe shipment to the centralized storage facility in production type casks TK-13 at burnup of 55 GW·d/t U is 8–12 years depending on initial fuel enrichment in ^{235}U.

Problems with power reactor SNF shipment are treated in detail in IAEA-TECDOC-1013 entitled Implementation of Burnup Credit in Spent Fuel Management Systems [6]. It should be noted that in the use of burnup credit, long term studies of isotopic and criticality validations are needed. At the Kola NPP the FAMOS-III facility is used to determine burnup of each spent fuel assembly to be shipped.

PWR/BWR MOX is only starting to be produced in the USA in limited experimental quantities. No applications have been made to date to transport irradiated MOX in any quantity.

Shipment of experimental WWER-1000 MOX-fuel is feasible in the TUK-13 casks with either a reduced loading (3 spent fuel assemblies), mixed loading, or an increased at-reactor cooling time up to 5 years. Problems of BN-600 MOX SNF management are being evaluated. Between 2015 and 2020, shipments will be made in the TK-11 casks where options such as: reduced SFA loading, extended at-NPP cooling up to 5 years, wet shipment, and mixed MOX with uranium fuel assembly loading are possible [2]. The development of a new cask is needed to ship WWER MOX in industrial scale after 2020.

3.2.4. Reprocessing

Reprocessing of spent nuclear fuel is generally applied to recover useful resources such as fissile (^{235}U, ^{239}Pu and ^{241}Pu) and fertile isotopes (^{238}U, ^{238}Pu, ^{240}Pu and others). The combination of reprocessing and vitrification also converts radioactive wastes to a stable form for long term disposal.

Although many alternative types of reprocessing technology including aqueous and pyrochemical techniques have, or are currently being, investigated for their potential applicability to various fuel cycles, commercial scale reprocessing is currently undertaken using the Plutonium and Uranium Recovery by Extraction (PUREX) process [7].

This consists of fuel dissolution in nitric acid followed by the separation of uranium and plutonium, first from the fission products and other actinides and then, secondly, one from another using tri-n-butyl phosphate (TBP) solvent in a suitable organic diluent combined with oxidation-reduction chemical reactions. The decontamination factor of a typical PUREX process is of the order of 10^7, which implies the recovery of relatively pure uranium and plutonium products, suitable for re-use in current generation reactors but also raising potential issues of nuclear proliferation concern.

Table 1 presents a summary of the worldwide experience and capacity in PUREX reprocessing technology. Reprocessing of high burnup UO_2 and MOX fuel is currently undertaken in some of the plants listed in Table 1 using essentially the same PUREX process as has been applied to lower burn up uranium dioxide fuels for several decades. However, the trend towards higher burnup and the increasing use of MOX fuel has implications for spent fuel reprocessing operations. These will be discussed in Section 5 of this report.

An alternative recycle system that does not employ wet chemical methods is the so called DUPIC process [7]. This uses mechanical means to recover spent fuel from PWRs, where the residual enrichment is sufficient to drive an HWR. Fuel pellet material is extracted from irradiated rods and dry processed so that it can be loaded into fresh assemblies.

3.2.5. MOX and reprocessed uranium (REPU) fuel

Both REPU and MOX are based on recycled spent fuel subject to chemical purification in a reprocessing plant. As noted in Section 3.2.4, current reprocessing plants produce relatively pure U and Pu for re-use in today's reactors. The UO_3 product is now being recycled in a number of European reactors. Recycled uranium typically contains around 1% ^{235}U and significant fractions of other isotopes, such as ^{232}U, ^{233}U, ^{234}U and ^{236}U, generated during irradiation from neutron reactions or radioactive decay[4]. These isotopes have their own neutronic and activity properties and may require increased enrichment to meet the same irradiation targets as the fresh fuel equivalent. They may also impact on dose to operators in fuel enrichment and fabrication facilities or on receipt at

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[4] Natural uranium effectively consists of two isotopes, ^{238}U and ^{235}U, neither of which are significantly active. The enrichment process is used to produce fuel with the correct reactivity for the required cycle length/fuel burnup in the reactor.

TABLE 1. STATUS OF SPENT FUEL REPROCESSING

COUNTRY	FACILITY	AMOUNT	OPERATION YEAR	CAPCITY t HM (FUEL TYPE)
France	Marcoule	18 000	1958–1997	400 (GCR)
	La Hague	20 500	1967/1990–	1600 (LWR)
Germany	WAK	180	1971–1990	30 (LWR)
India	Trombay/Tarapur	60/100	1964/1974	60/100 (PHWR)
Japan	Tokai-mura	1 000	1977–2006	90 (LWR)
	Rokkasho-mura			800 (LWR)
Russian Federation	RT-1	3 500	1971–	400 (WWER-440)
UK	B205	42 000	1964–1994	1,500 (GCR)
	Thorp	4 300		900 (LWR)
USA	NFS West Valley	194	1966–1972	194 (LWR)
Euratom	Mol	105	1966–1976	105 (LWR)
TOTAL		~30 000 LWR Fuels		~3 400 for LWR Fuels

the reactor. The mix of uranium isotopes (and potential decay products) in REPU depends on the initial enrichment and burnup of the parent fuel, the cooling time before reprocessing and the REPU storage time before fabrication/irradiation. The conventional enrichment process will also act to concentrate lower mass U isotopes. An alternative method of enriching REPU is to mix with HEU from dismantled weapons.

There are a number of potential implications of higher burnup on such a fuel cycle. The activity of the spent fuel will increase with burnup and the mix of ^{235}U to Pu will change.

3.2.6. Spent fuel and HLW disposal

At present various types of geological formations are considered as possible host formations for deep disposal of spent fuel, high level radioactive waste (HLW) and other radioactive waste. The main types considered are:

- Hard rock formations, mainly granite, which are being considered by countries such as Canada, Finland, Japan, Spain, Sweden and Switzerland;
- Argillaceous formations such as clays, which are currently considered in countries such as Belgium, France, Japan, the Netherlands, Spain, Switzerland, and Germany;
- Salt layers and salt domes are being considered by Germany and the Netherlands;
- Volcanic formations including tuff and basalt; the proposed US programme for high level radioactive waste disposal focuses on a welded tuff formation at Yucca Mountain.

Repository designs developed for water bearing formations, i.e. hard rock and argillaceous formations typically have of the following engineered barriers:

- A metallic container often called canister or overpack, that has to remain intact during the initial gradient phase of the repository;
- A buffer surrounding the container, which consists in many designs of a swelling clay material, such as bentonite; the buffer has to fill possible voids and fractures and prevent advective water flow around the container;
- A backfill is used to fill transport and access galleries;
- Seals and plugs are used to isolate the disposal gallery or borehole from the transport and access galleries and from the part of the host formation that have been disturbed by the excavation of the gallery or borehole.

For repositories located in salt or unsaturated formations several of the above mentioned engineered barriers are also used, but some may not be needed.

Repository design must take into account waste characteristics such as radionuclide content, heat generation, criticality, and radiation fields. However, all repository designs use some combination of temperature limits on the engineered barriers and the rock formations. Most repository designs in formations below the water table have temperature limits at or slightly below 100°C for any part of the repository system with water present. Repository designs above the water table or in salt may have higher temperature limits. Analyses of the required decay time or disposal area required versus burnup are available (e.g. Section 4.1.1.5). The proposed Yucca Mountain disposal facility in the USA, for example, included a surface aging facility to provide the necessary decay time prior to underground emplacement so the thermal criteria are met.

3.3. GENERAL CRITERIA THAT AFFECT BURNUP AND FUEL TYPE CHOICES

This report provides a listing of the regulatory requirements, and other criteria with respect to safety, security, and non-proliferation with some limited clarifying discussion of each. The ability to meet these regulations will depend on the expected performance of the systems and the fuel. Sometimes the criteria can be met by increasing the robustness of the system to compensate for the behaviour of the fuel. It is up to the reader decide how the particular regulation for his country applies and how to rank the importance of these criteria for decision making.

3.3.1. Regulatory

The systems for any phase of the back end of the fuel cycle need to meet regulations and concerns related to safety, and operations such as retrievability of the fuel. The major safety concerns are related to criticality, containment, dose, and thermal behaviour. These are set out in regulatory documents and supplementary guidance that may vary from country to country. In the USA these are delineated in the Code of Federal Regulation, standard review plans, and interim guidance. The IAEA provides guidance for transportation. IAEA TS-R-1 Regulations for the Safe Transport of Radioactive Material are accepted worldwide. In some cases, these transportation regulations also govern dry storage in countries where storage systems must be transportable. Furthermore, there are European agreements and regulations for international transportation of dangerous goods on road (ADR) or by railway (RID). Examples of Regulatory Documents by country are given in Table 2.

As an example, consider the criteria for Russian spent fuel pools. The following, taken from the regulatory safety documents that are in force in the Russian Federation, are requirements for all spent nuclear fuel (SNF) wet storage facilities:

TABLE 2. EXAMPLES OF REGULATORY DOCUMENTS

COUNTRY ORGANIZATION	STORAGE		TRANSPORT	DISPOSITION	
	Wet	Dry		Reprocess	Disposal
IAEA			TS-R-1		
USA	10CFR 50	10CFR72	10CFR71	NA	10CFR63
Russian Federation	NP-061-05[a] NP-016-05	NP-061-05 NP-035-02[b] NP-016-05	NP-061-05 NP-053-04[c]	NP-013-99[d] NP-016-05	NP-016-05[e]
Germany	NA	AtG §6	GGBefG, ADR, RID	NA	NA

[a] NP-061-05: Safety regulations in S&T of a nuclear fuel at the places of a nuclear energy use.

[b] NP-035-02: Places of dry storage a spent nuclear fuel, safety requirements.

[c] NP-053-04: Safety regulations in radioactive materials transportation.

[d] NP-013-99: Installations of spent nuclear fuel reprocessing, safety requirements.

[e] NP-016-05: General regulations of safety provisions of nuclear fuel cycle objects.

— Individual fuel assembly location in the pool should provide nuclear safety in storage and shipment considering all possible departures from normal operation;
— Radiation safety of the personnel in the storage operation should be provided;
— The SNF condition during storage should be checked;
— SNF integrity in the storage should be guaranteed;
— SNF decay heat removal should be provided;
— Pool water should be cleaned to remove both aqueous and particulate radioactive substances entering the water from surface contaminations, from damaged fuel rods, as well as from corrosion products;
— Pool water leakage into the environment should be prevented by the storage facility design;
— Radioactive substances release into the atmosphere must be lower than permissible levels;
— SNF assemblies with defective fuel rod(s) require special isolation until decommissioning.

3.3.2. Safety criteria (e.g. criticality, confinement, dose limits, retrievability, fission gas release due to leakers during operation, thermal…)

The following criteria impact or have implications on the back end of the fuel cycle. Areas and issues that require detailed assessment are identified.

3.3.2.1. Criticality

Most regulations governing spent fuel management require that the system remain sub- critical. Usually this means a $K_{eff} < 0.95$ under normal conditions and based on the reactivity of fresh fuel (no burnup credit). Most systems are designed to be sub-critical in the as-loaded or as-handled condition. If the fuel configuration changes, criticality may occur. Pushing fuel to higher burnups would necessitate higher initial fissile content in the fresh fuel. The change in the actinide and fission product concentrations with increased burnup is discussed in Section 4.1.1. In addition, possible instability of the fuel configuration in certain situations due to the decreased ductility of the cladding resulting from increased hydrogen content, may also occur. (A discussion of the effects of high burnup on cladding ductility is presented later in this technical report). Both effects must be analyzed.

There are many available actions to assure sub-criticality (see Section 5). The choice of actions will depend on the level of risk a country is willing to assume, the burnup of the fuel, and the continuing improvement in the techniques described later.

3.3.2.2. Heat removal and thermal behavior

Temperature limits are imposed on components during all stages of the back end of the fuel cycle to prevent thermally induced degradation and maintain the effectiveness of the component; such as baskets, neutron shield and absorbers. Temperature limits are placed on the fuel cladding to prevent the formation of cladding breaches from mechanisms such as creep, and to prevent a change in the metallurgical condition due to annealing and hydride reorientation.

The heat load in spent fuel soon after irradiation is primarily due to the fission products. Much later in life it is due to the decaying actinides. In general, the higher the burnup, the higher the heat load generated and the more heat rejection capability is required.

The United States Nuclear Regulatory Commission (USNRC) currently limits the peak clad temperature to <400ºC for both storage and transportation. The United States Department of Energy (USDOE) limits the temperature of fuel in the repository to <380°C. Other countries storing LWR fuel have similar temperature limits. The maximum temperature of canister internals varies with design and materials used. The maximum allowable temperature for the storage of HWR CANDU fuel is limited to 190°C to prevent UO_2 oxidation.

3.3.2.3. Radiation protection (confinement and dose limits)

A primary requirement of any system to handle spent fuel is to protect the public from radiation exposure in any form. Exposure can be in the form of:

- Irradiation by gamma, or neutrons due to insufficient shielding in the system;
- Inhalation or ingestion of radioactive particulate that escapes from the system as a result of incomplete sealing, opening of the system due to an accident, or gross opening due to terrorist activity. Systems are designed to meet ALARA considerations.

The materials used to shield the public or seal the system are subject to irradiation degradation that might limit their effectiveness. Most countries require that this potential degradation be evaluated so the system can be engineered of materials that will remain effective throughout the lifetime of the system.

Should accidents or terrorist activities occur, the steps, and complexity of steps required to protect the pubic, and remediate the situation will depend on the potential extent of release of radioactivity and loss of shielding. Higher burnup or MOX fuel will change the radiation doses and characteristics affecting the implementation of the plan.

Higher burnup fuel will present a larger source term of radiation than lower burnup fuel of the same cooling duration. The potential affect of this higher source term on the dose rates and materials behavior will be evaluated in Section 5.

3.3.2.4. Retrievability

Retrievability requires that each individual assembly or canned assembly be retrievable, from the canister, using normal means of handling. To the extent practical, spent fuel at an ISFSI must also be retrievable for transport to reprocessing or disposal.

In the USA, the Nuclear Waste Policy Act of 1982, as amended (NWPA), requires that “each monitored retrievable storage facility shall be designed to provide for the ready retrieval of such spent fuel and waste for further processing or disposal.” In consideration of this NWPA requirement, the Commission amended Part 72 to include: “*Retrievability.* Storage systems must be designed to allow ready retrieval of spent fuel or high level radioactive waste for further processing or disposal. The Commission explained: Not only must the Monitored Retrievable Storage (MRS) be designed for removal of the spent fuel or high level radioactive waste, but an ISFSI must meet the same criteria for the stored spent fuel. The spent fuel at an ISFSI must also be retrievable for transport to either the MRS or HLW repository whenever they become available”. In 1990, Part 72 was further amended to add “To the extent practicable in the design of storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the DOE.”

Fuel that cannot be retrieved by normal means may be considered as non-standard fuel. Non-retrievability and repackaging of the non-standard fuel is an operational issue but can also be a safety problem.

3.3.3. Routine release to the environment

In addition to the possible releases of activity that may occur in fault situations covered in the safety case section above, it is also necessary to consider the routine discharges to the environment that may result from spent fuel management activities.

Active species are released into wet storage ponds during normal operations. For instance, ^{137}Cs and other species may be leached from failed rods in ponds, or loose crud rich in ^{60}Co may be dispersed into the pond water [8]. Depending on the practice at the pond, these species may be captured in ion-exchange resins for disposal as solid waste or released to the sea in the pond purge. In addition, non-active, chemically toxic species may be released from wet storage ponds in either a solid or liquid form. For example, soluble boron, corrosion inhibitors or biocides may be added to wet storage ponds.

Reprocessing plants generate active solid, liquid and gaseous wastes as a result of fuel shearing, dissolution and chemical separation processes. Due to the extensive use of nitric acid, reprocessing plants must manage volumes of NOx gases. Following extensive treatment that is generally designed to concentrate the active components; aerial and liquid wastes may be discharged to the environment at approved levels. Solid wastes are sent to engineered storage or disposal facilities.

Dry cask storage is virtually free of operational discharges, although releases during fuel drying may need to be considered. At the end of the storage period the casks would need to be disposed of, or re-used (recycled), as appropriate. Vault type stores employing a circulating coolant are subject to many of the considerations that apply to wet stores, discussed above.

Regulatory controls are applied to limit the discharges of both active and non-active species from fuel cycle facilities. The impact of storing or reprocessing higher burnup or MOX fuels on these discharges would need to be carefully considered before a change was implemented. In addition, many counties also require that environmental discharges and the technology used to control them are justified as representing the best available option for environmental protection, taking many factors, including cost into account. A change of source term, represented by higher burnup UO_2 or MOX fuels may tip the balance in these assessments towards process modifications or the installation of more efficient abatement methods.

3.3.4. Considerations in choosing spent fuel management strategies

3.3.4.1. Centralized vs. at the plant site

Currently two types of locations for spent fuel storage are distinguished. The definitions are as follows [9]:

- **At reactor (AR) spent fuel storage:** Storage of spent fuel at a reactor pool. This storage is associated with reactor operation.
- **Away from reactor (AWR) spent fuel storage:** This is a second phase storage which is away from reactor (not necessarily away from the nuclear power plant site). It may serve several reactors and both wet and dry technologies could be applied.

At present, no country in the world has an operating facility for the permanent disposal of spent nuclear fuel. While some countries, such as Finland, Sweden, and the USA may have a disposal facility in operation in the next decade or two, most countries are many decades away from permanent disposal. Furthermore, for those countries using reprocessing, there are some advantages to storing the spent fuel for many years prior to reprocessing. For example, long term spent fuel storage prior to reprocessing reduces radiation fields, thereby reducing doses to workers and degradation of reprocessing solvents.

Hence, it is usually necessary for spent fuel to be temporarily stored for a considerably long period of time prior to disposal or reprocessing. Furthermore, some spent fuel cannot be transported off-site until certain technical or other issues are addressed. Hence, there is a wide variety of interim spent fuel storage options being used worldwide. While initial spent fuel storage is at the reactor site in a spent fuel pool, longer term interim storage (perhaps over several decades) of spent heavy and light water reactor fuel may be in these same pools, or in dry storage systems either located at the reactor sites, or in some off-site storage location.

Some countries, such as Sweden and Switzerland, have constructed centralized interim storage facilities. Sweden chose large, underground spent fuel pools while in Switzerland, an above ground centralized dry storage facility was chosen. Other countries, such as France and the UK, store a considerable amount of spent fuel in pools at their reprocessing facilities such that these pools may be considered centralized interim storage.

Other countries, such as the USA, do not have centralized or have limited centralized interim storage facilities such that each utility must store its own spent fuel on site. It is becoming increasingly necessary to expand on-site storage capabilities in those countries without centralized storage facilities. While increasing burnups may be desired to minimize the number of spent fuel assemblies that need to be stored, the increased decay heat from these higher burnup spent fuel assemblies will need to be managed. All spent fuel storage systems have some combination of decay heat rate and temperature technical limits. In almost all cases, the on-site or centralized storage facilities have an administrative or legal maximum capacity.

The considerations involved in deciding whether to select centralized or on-site spent fuel storage facilities and systems is beyond the scope of this technical report, as they involve a host of societal and legal, as well as technical issues. However, the spent fuel burnup level and use of MOX fuel will be one of many important considerations.

3.3.4.2. Required storage time prior to transportation, reprocessing and/or disposal

The maximum decay heat rejection of wet and dry storage systems, transportation casks, reprocessing facilities, and disposal facilities are usually specified as part of the system design and regulatory approval. For higher burnup UOX and MOX, the required decay time to meet these thermal criteria may be considerable — in some cases on the order of decades or longer prior to either reprocessing or disposal. Given these potentially long decay times, it will be necessary to design systems with higher heat rejection capacities, long term integrity of heat removal, shielding, and mechanical damage resistance, and/or develop the necessary institutional controls for long term spent fuel management.

3.3.5. Proliferation resistance

The peaceful use of nuclear material in the current form potentially involves proliferation risk because of the conversion of fertile into fissile material in power reactors. The INFCE (International Nuclear Fuel Cycle Evaluation) report [10] defines proliferation as the misuse by a government of nuclear fuel cycle facilities, know-how or materials to assist in the acquisition, manufacture or storage of a nuclear weapon. The FAS Public Interest Report [11] has searched for proliferation resistant nuclear power. The statement shared by both reports is that proliferation is primarily a political and not a technical matter. The MIT study on the future of nuclear power [12] has discussed the non-proliferation problem and concluded: the global growth scenario built primarily upon the once-through thermal reactor fuel cycle would sustain an acceptable level of proliferation resistance, if combined with strong safeguards and security measures and timely implementation of long term geological isolation.

This model is based on the ‘spent fuel standard’ [13] which is considered an appropriate goal for excess weapons plutonium disposition. However, the decay of the fission products over time increases the accessibility and retrievability of the plutonium to the theft or diversion after 50 to 100 years.

Criteria for non-proliferation can be categorized into three types: institutional, material, and technology [14]. The *institutional* criterion is mainly based on the non-proliferation regime of the IAEA. GNEP (Global Nuclear Energy Partnership) [15] proposed by US DOE may also be institutional. However, it is very important to note the statement written in its section for proliferation prevention: *there is no technology “silver bullet” that can be built into an enrichment plant or reprocessing plant that can prevent a country from diverting these commercial fuel cycle facilities to non- peaceful use.*

The spent fuel standard is one of the *material* criteria for non-proliferation. The methods of using radioactivity to protect plutonium from diversion have been evaluated as pre-irradiation, spiking, and partial reprocessing [16]. Other concepts such as plutonium denaturing or heat spiking rely on an addition of ^{238}Pu denaturant to the original [17]. A non-proliferating fuel for light water reactors can be made by introducing ^{237}Np into the fuel so as to increase the ^{238}Pu/Pu ratio to 5% and above [18]. The ^{238}Pu is obtained from ^{237}Np by neutron capture in the reactor. Some of case studies [19] of recycling Transuranium (TRU) in AP600 reduce the TRU inventory and are thought to be developing proliferation-resistant fuel. The purpose of these studies is decreasing the attractiveness of nuclear materials in fuels and bulk materials (reducing the purity of fissile material in isotopic and chemical composition) [20, 21].

3.3.6. Safeguards and security

Safeguards and security has always been an important consideration in all phases of the back end of the fuel cycle. The majority of the safety and security is associated with the performance of the systems and oversight given to these systems. The fuel is the potential source term for the release of radioactivity should an event take place. Knowledge of the behaviour of the fuel during an incident is necessary to determine the magnitude of the potential release of radioactive material, types of radioactive materials released, and most expeditious methods to remediate the situation. Should the source term increase too much with higher burnup fuel, a change in the system design such as tighter seals or more shielding might be dictated.

Nuclear fuel contains radioactive material in four forms:

(1) Within the fuel matrix itself, as fission gases produced during irradiation that may remain in the fuel matrix, solid fission products, and actinides;

(2) Fission gas that has escaped to the plenum of the fuel rods;
(3) Volatile radionuclides in the fuel matrix;
(4) Crud on the outside surface of the fuel rods.

Characteristics of the fuel that will govern a release are: inventory of radioisotopes produced during irradiation, gas release from the fuel matrix to plenum during irradiation, fracture characteristics of the fuel pellet during an event, and the ductility of the cladding to resist fracture during an event. All of these characteristics change as the fuel is driven to higher burnup. The degree of change will depend on the specific type of fuel. These changes will be discussed in Section 4.

4. CHARACTERISTICS OF SPENT FUEL

A number of these aspects of high burnup fuel have been studied by member countries and have been reported in the recent and final report of a co-coordinated research project on spent fuel performance and assessment research (SPAR) [75]. The reader is encouraged to refer to the above source document for more detailed information. Some areas of particular interest in the handling of spent fuel arising from higher burnup are briefly summarized here: (1) burnup targets in various type of reactors, (2) fuel utilization parameters such as reactor operational details, fuel residence periods in the reactors and associated discharge fuel characteristics such as decay heat and neutron activities in the fuel, and (3) fuel failure rates with increased burnups.

4.1. UOX FUEL CHARACTERISTICS

The batch discharge burnups for a modern PWR core with a relatively flat radial and axial power and a profile with a peak rod average burnup of 62 GW·d/t U[5] are typically between 53–55 GW·d/t U with peak bundle average burnups around 57–59 GW·d/t U. Fuel is currently being irradiated to >50 GW·d/t U and plans are to raise the burnup limits up to ~70 to 75 GW·d/t U (rod average).

The rise in burnup limits has changed the as-irradiated characteristics of current designs of fuel and necessitated the use of newer cladding materials. Newer fuel assembly designs other than the standard PWR 17 × 17 and BWR 8 × 8 that have been extensively analyzed are also being used. Changes in a number of characteristics may occur as the fuel goes to higher burnup. These include: crud thickness, cladding oxide thickness and hydride content, radionuclide inventory and distribution, heat load, fuel grain size, fuel fragmentation, and fission gas release to the rod plenum.

Over the years, the distribution of burnup from CANDU fuel has increased from an average median of 7.9 GW·d/t U to an average maximum of 19.8 GW·d/t U. Typical burnup of CANDU fuel is considered to be 8.6 GW·d/t U with a residence time of approximately 465 full power days. Typically, CANDU fuel cladding is exposed to high energy neutron fluence in the range 6–10 × 10^{24} n/m^2. The fuel operates with a coolant temperature range of 260 and 300°C and at a pressure of 10 MPa. The average bundle power is 375 kW. Although there have been some investigations of high burnup bundles, there was no obvious tendency to increase the burnup systematically. There is no specific design for high burnup fuel. It is considered that the current CANDU fuel designs are robust enough to withstand increased burnup that is typically within the statistical distribution of design burnup of fuel coming out of pressure tubes of CANDU reactor.

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[5] Burnups in literature can be quoted in terms of batch average discharge burnup, peak bundle average burnup, rod average burnup or peak pellet (averaged radially across pellet) burnup. These different burnups can vary from plant to plant even if the peak rod average burnup is 62 GW·d/t U for each core because the core power distribution (radially and axially) can vary depending on the plant and fuel design. However, plants are flattening their core powers such that the difference between these different burnups is narrowing.

The maximum burnup of WWER-440 fuel ranges between 42 and 65 GW·d/t U while WWER-1000 fuel rods can achieve burnups as high as 55 GW·d/t U. Work is being done to raise these limits to 72 GW·d/t U. This will be achieved by fuel manoeuvring and the introduction of optimized zirconium alloy cladding.

4.1.1. Isotopes (criticality, alpha activity)

4.1.1.1. Initial enrichment and criticality

Burnup extensions in general require higher initial ^{235}U enrichment for UOX fuel. For PWRs, the required enrichment for a given burnup may be approximated using the MIT correlation [12], which is also a function of the number of fuel batches (Fig. 8). This correlation gives a little higher estimate than the Westinghouse empirical data for a constant given cycle length [22]. A maximum average discharge burnup achievable within the current 5.0 w/o fabrication limit is estimated to be approximately 65 GW·d/t U according to the latter relation. This value might be extended further by advanced core managements. Figure 8 compares these two PWR relations with the BWR data [23].

The isotopic effects of burnup extension on SF are increased levels of fission products (FPs), degraded composition of uranium isotopes and increased levels of TRU, the major part of which is Pu. These facts could impact the back end of the fuel cycle and the management of recovered material from reprocessing. For example, prolonged periods of SF storage reduce fissile ^{241}Pu and other short lived nuclides. Therefore, higher burnup levels could lead to a lower fissile content for Pu and reprocessed uranium (REPU), and increased high level radioactive waste in case of reprocessing.

Despite the increased burnout of the fissile ^{235}U isotope, high burnup fuels tend to also have higher residual enrichment levels. This is due to the higher initial ^{235}U enrichment required to achieve the high burnup levels. More attention must therefore be paid to the criticality safety of high burnup fuels. The use of the 'burnup credit' concept is becoming more widespread in optimizing the design of facilities to handle spent fuel (see a brief discussion of this topic in Section 5.1.3.3 with respect to regulatory concerns). This allows for the consideration of the effect of irradiation on reducing the level of fissile isotopes present in spent fuel compared to those in fresh fuel and also, in some cases, the increase in neutron poisoning fission products on criticality. Introduction of burnup credit can allow for more cost-efficient SF pool rack or dry storage cask designs by allowing for more realistic estimates of the reactivity of the SF considering TRU and FP compositions of SF, instead of having to assume the higher reactivity characteristic of fresh fuel. Thus it can increase storage capacity of a SF storage cask. Fuel burnup and fissile content is routinely checked in reprocessing facilities by so-called burnup monitors, which use a combination of gamma spectroscopy and active and passive neutron counting. While moderate increases in burnup have no significant implications for such equipment, different designs or calibration settings may be required for

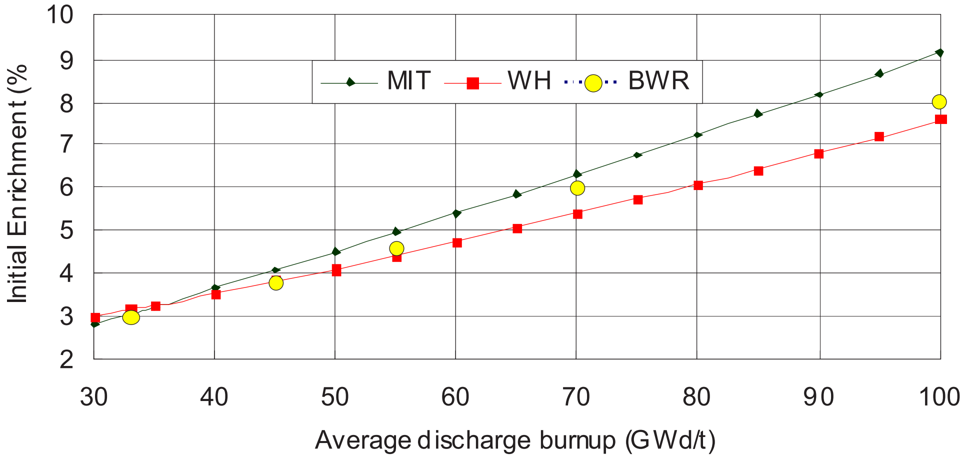


FIG. 8. Dependence of initial enrichment of UOX fuel on average discharge burnup using the MIT correlation (MIT) [12], Westinghouse (WH) [22], and BWR (BWR) data [23]. MIT relation corresponds to three-batch refuelling.

significantly increased burnup levels or for MOX fuels in order for these to respond appropriately to relatively high burnup fuel with higher levels of gamma and neutron radiation as well as residual fissile content.

The following data are taken from reference [23]. Data for 100 GW·d/t U and REPU, and recycling are taken from references [24].

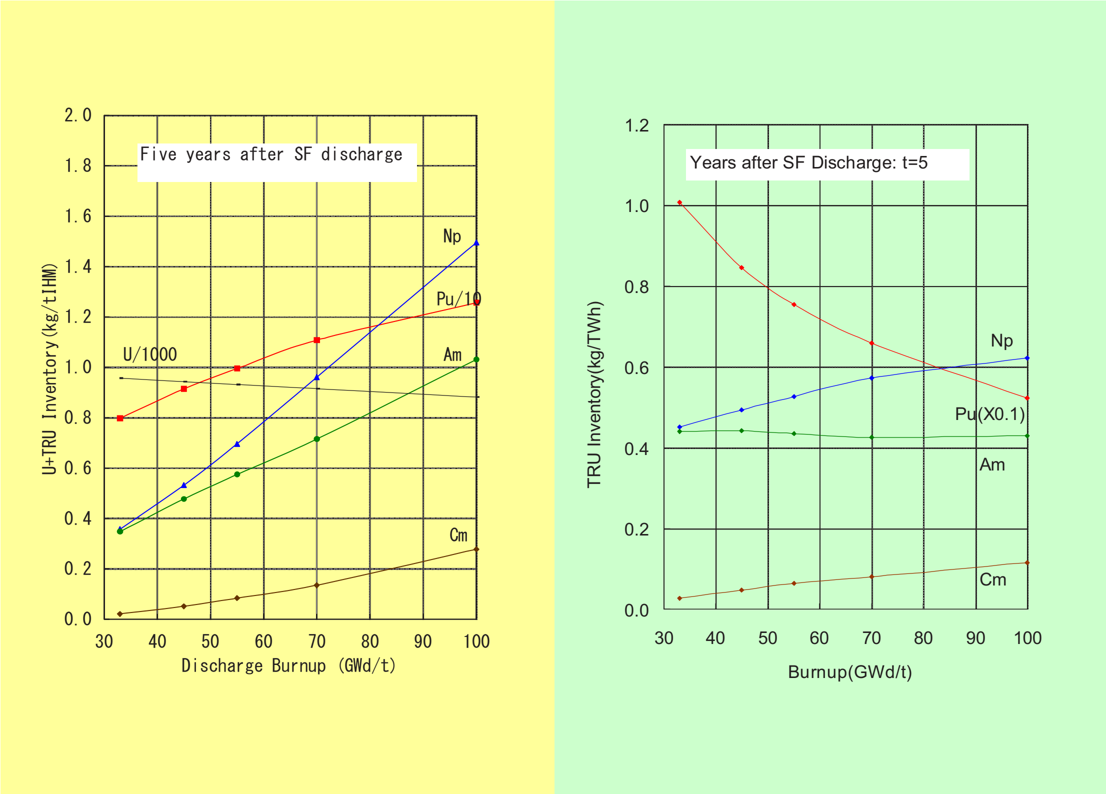
4.1.1.2. Content of transuranic elements in spent fuel

Figure 9 (a) shows the burnup dependence of the TRU inventories (kilograms per ton of initial heavy metal, kg/t IHM) in UOX-SF five years after discharge. Total uranium (^{235}U + ^{238}U) is also shown. Total uranium content decreases with burnup as the ^{235}U fissions and the ^{238}U is converted mostly by neutron capture to ^{239}Pu. Generally speaking, the amount of TRU isotopes of higher atomic mass (also known as the minor actinides (MA)), increases with burnup. However, the tendency is different for individual isotopes:

- The fraction of fissile Pu decreases with burnup;
- The fraction of Np amounts to half of the total MA. The ratio of Am to total MA has a decreasing tendency with respect to burnup;
- The ratio of MA to Pu increases with burnup.

Figure 9 (b) shows the same quantities as (a) expressed in terms of kilograms per terawatt-hour of energy generated (kg/TW·h). A different tendency from (a) is seen:

- The total heavy metal inventory per TW·h decreases with burnup. The decrease of fuel heavy metal weight per TW·h implies the possible reduction in the spent fuel storage volume per unit of energy produced;



(a) (b)

FIG. 9. Dependence of TRU inventories on discharge burnup of UOX-SF (five years after SF discharge): (a) measured per t IHM, and (b) per TW·h. Total uranium is also shown in (a). [23, 24].

— Total Pu and fissile Pu in spent fuel decreases with burnup;
— The ratio of MA to Pu increases with burnup. Total MA also increases with burnup. In particular, the fractions of Np and Cm in the MAs increase while those of Pu and Am are decreasing with burnup.

The relevance of the amount of long lived MAs to spent fuel disposal is discussed in more detail in Section 5.1.4 of this report. For some repository designs and analyses, the dose contribution from Pu and the MAs (particularly ^{237}Np and ^{241}Pu) to the individuals living downstream of the disposal facility is found to be significant. Hence, there may be an increase in estimated dose rate with the placement of higher burnup UOX into the disposal facility.

4.1.1.3. Uranium and plutonium in spent fuel

Dependence of the concentrations of uranium isotopes in the UOX-SF on discharge burnup is shown in Fig. 10, at five years after discharge. The concentrations of ^{232}U and ^{236}U increase with discharge burnup. The isotope ^{232}U is rather short lived (half-life ~70 a) and the precursor of high energy gamma emitter Tl-208, while ^{236}U is a parasitic neutron absorber with half-life ~2.34 × 10^{7} a. Both of them are absent in nature and give negative effect on uranium recycling for use in LWRs.

Figure 11 shows the dependence of concentrations of plutonium isotopes on discharge burnup at five years after discharge. Plutonium content increases more slowly with burnup as, at higher burnup levels, an increasing proportion of fission events relate to ^{239}Pu rather than ^{235}U.

Decrease of fissile ^{239}Pu and total fissile (sum of ^{239}Pu and ^{241}Pu) with discharge burnup is very evident in spite of the gradual increase of ^{241}Pu. The isotope ^{236}Pu having a short half life of 2.85 a is generated in an order of 10 ppb and alpha-decays into ^{232}U. The well known alpha emitter ^{238}Pu has a large specific radioactivity ~6.34 × 10^{11} Bq/g. Both isotopes increase gradually with respect to discharge burnup.

4.1.1.4. Specific activity as a function of UOX burnup

Figure 12 compares the specific activity of UOX-SF at 45 GW·d/t U discharge burnup with that of 100 GW·d/t U. Both figures show a similar behaviour with respect to decay time. The specific activity of the fission products is almost proportional to discharge burnup. Thus, the contribution from ^{90}Sr and ^{137}Cs is almost twice as large in 100 GW·d/t U as in 45 GW·d/t U. However, ^{238}Pu and ^{244}Cm specific activities are almost four times as large at 100 GW·d/t U. As both nuclides are alpha particle and spontaneous neutron emitters, levels of the alpha radiation (and hence decay heat) and neutron outputs will increase for higher burnup as shown in Fig. 13.

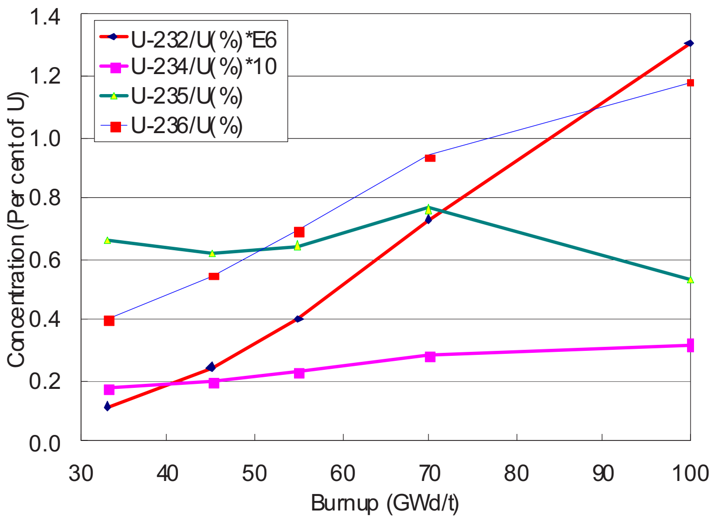


FIG. 10. Dependence of uranium isotopic concentrations on discharge burnup at five years after SF discharge. A small wiggle on the ^{235}U line is not essential as the initial enrichment for higher burnup is not fully optimized [24].

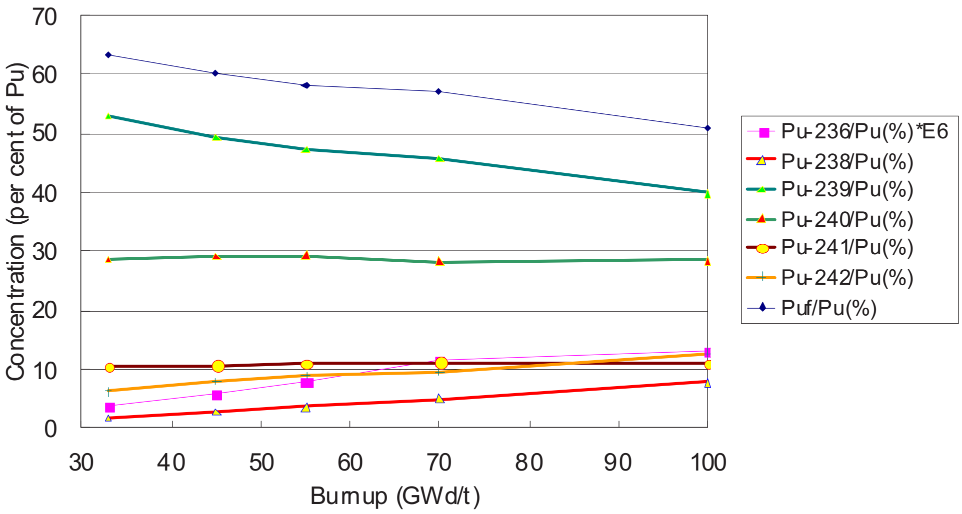


FIG. 11. Dependence of plutonium isotopic concentrations on discharge burnup at five years after SF discharges [24].

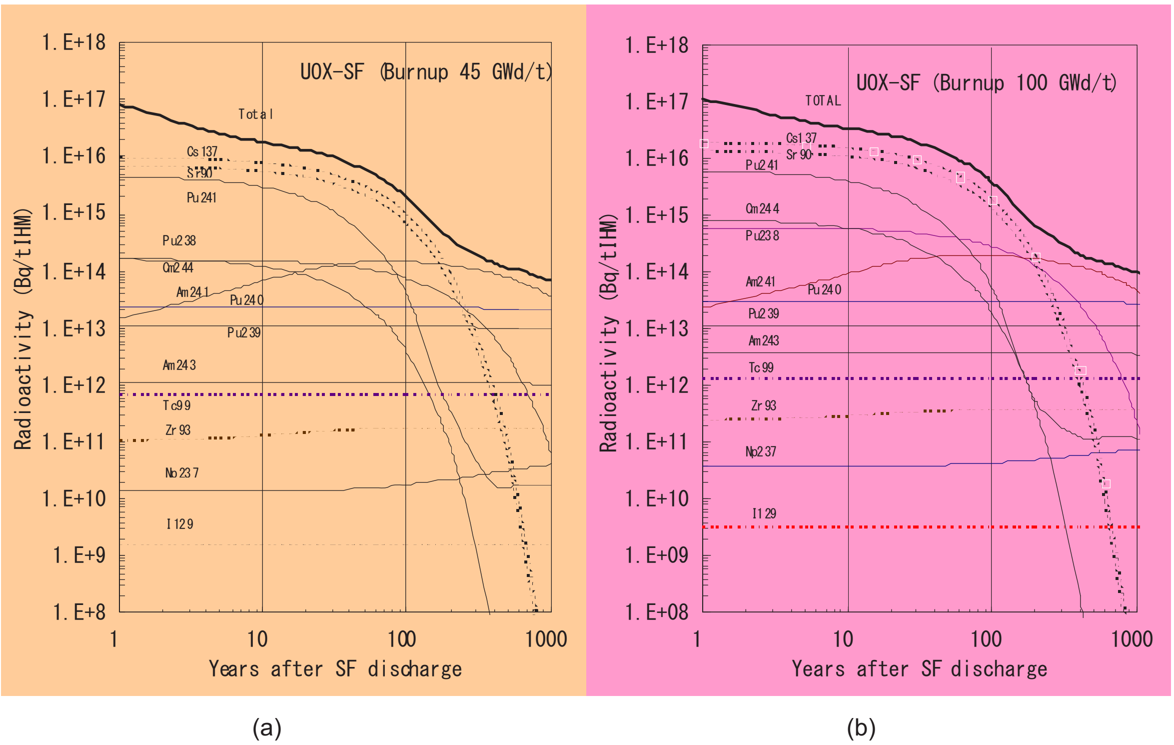
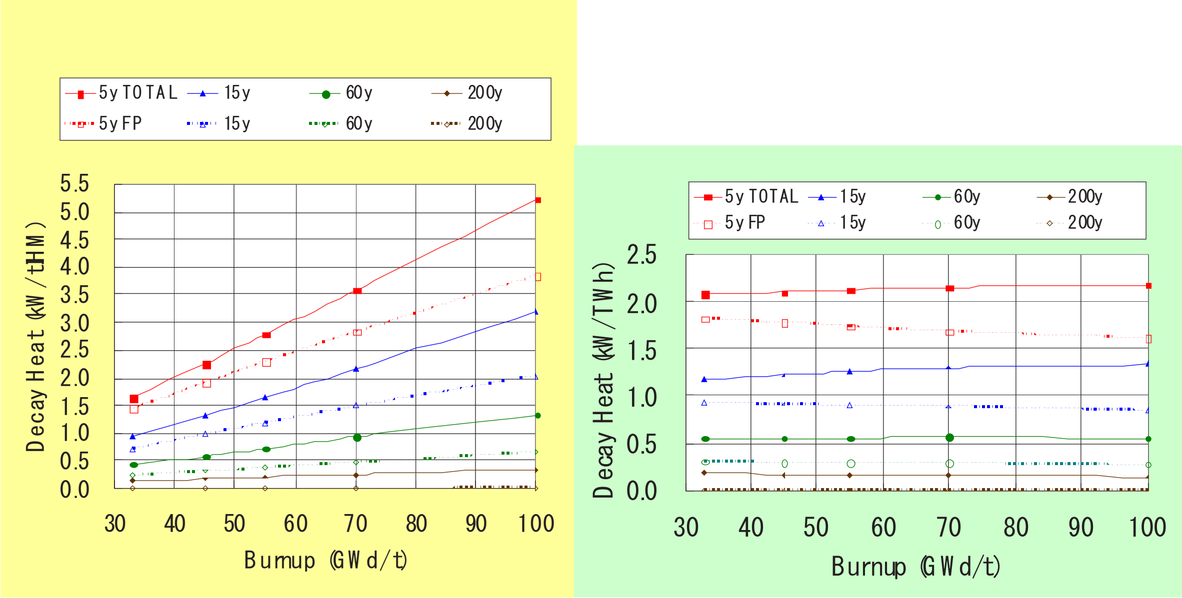


FIG. 12. Dependence of radioactivity of UOX-SF on cooling years after SF discharge: discharge burnup (a) 45 GW·d/t U, (b) 100 GW·d/t U [24].

4.1.1.5. Decay heat

Figure 13 (a) shows decay heat of UOX-SF for a ton of initial heavy metal inventory. It increases almost linearly with discharge burnup. However, the rate of increase slows down as the post-reactor decay time of UOX-SF increases. The contribution to total decay heat due to FPs is denoted by broken lines. Major fission product thermal power sources are from the beta particle-emitting radionuclides ^{90}Sr and ^{137}Cs. These two fission products have similar half-lives of around thirty years, and increase approximately linearly with burnup.

If decay heat is expressed in terms of generated energy (TW·h), its dependence on discharge burnup is found almost flat as is seen in Fig. 13 (b).



(a) (b)

FIG. 13. Dependence of decay heat on discharge burnup and UOX-SF decay time, t: (a) measured per t IHM, and (b) per TW·h. Contribution from FPs is shown by a broken line [24].

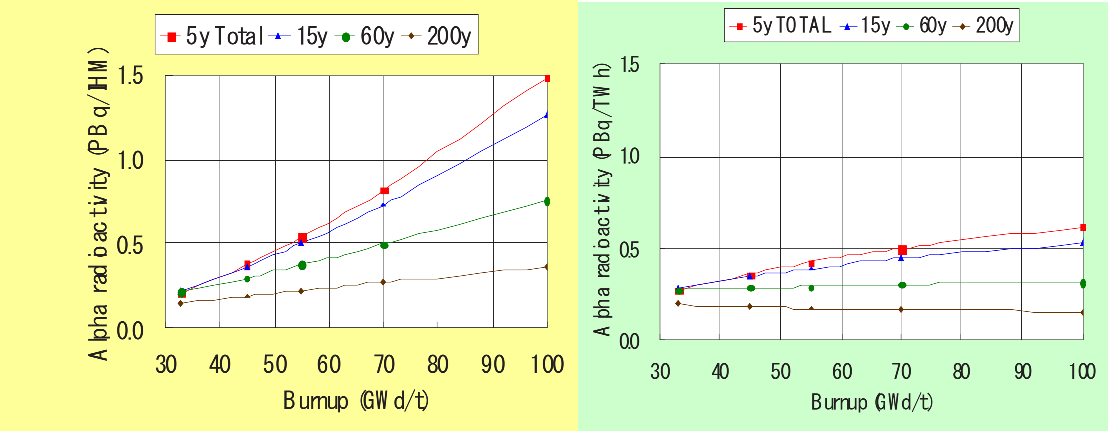
One of the most important factors related to the ability to store, transport, or dispose of spent fuel is the decay heat generated by the spent fuel. While higher burnups may result in less spent fuel assemblies generated per unit of energy produced, it is likely that the number of high burnup spent UOX or MOX assemblies that can be stored, transported, or disposed in a given cask or repository volume will be lower than for lower burnup spent UOX — unless additional decay time is used. For example, assume a particular PWR cask can hold 10 t HM of spent UOX. Furthermore, assume this cask has a total decay heat output limit of 25kW, 20kW, and 15kW for dry storage, transportation, and disposal, respectively. Using Fig. 13 (a) it can be seen that it would take approximately 5, 7, and 15 years to be cool enough to dry store, transport, and dispose, respectively, spent 10 t HM UOX fuel with a burnup of 50 GW·d/t U; it would take approximately 30, 40, and 60 years to be cool enough to dry store, transport, and dispose, respectively, spent UOX fuel with a burnup of 100 GW·d/t U.

Alternatively, spent UOX might all be transported for shorter post-reactor decay times if less spent UOX is put in the cask. For this example, spent UOX with a burnup of 50 GW·d/t U could be dry stored, transported, and disposed of after 5 years if the mass of spent UOX was limited to just 10 t HM (i.e. a fully loaded container), 8 t IHM, and 6 t IHM, respectively. Thus, if one wanted to dispose of 5-year decayed spent UOX in this type of container, nearly 1.7 times as many containers would be required for a burnup of 50 GW·d/t U. If one wanted to dispose of 100GW·d/t U spent UOX after just five years, the mass of spent fuel in the container for the purpose of dry storage, transportation, and disposal would be limited to <5, <4, and <3 t HM. In this case it would require significantly more than three times the number of containers to dispose of the same number of tones of spent fuel.

Thus, the trade-off for higher burnup spent UOX or MOX is that more storage, transportation, and disposal capacity is needed, and/or much longer decay times are needed per ton of HM. On the other hand, Fig. 14 a–b suggests that there would be little difference in required decay time or canister capacity per unit of energy produced.

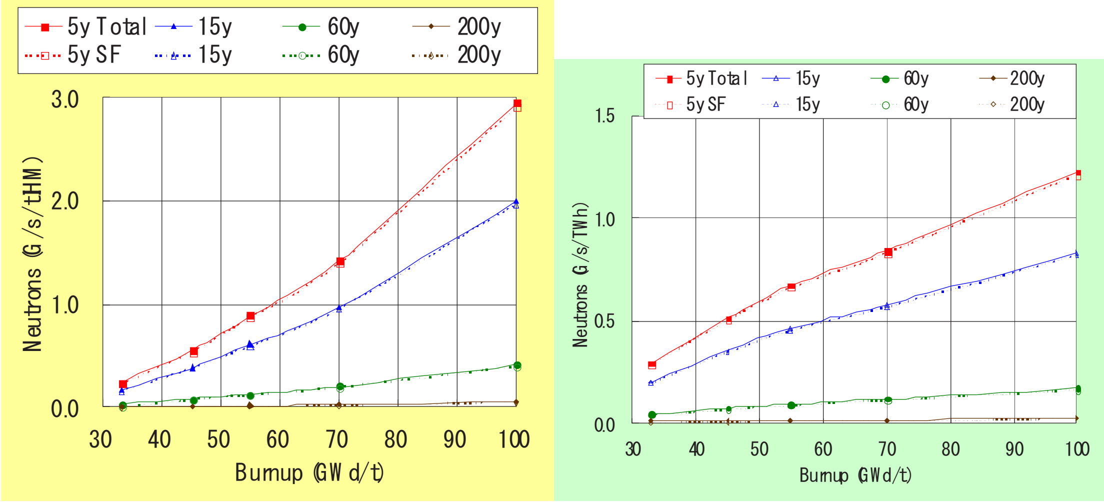
4.1.1.6. Alpha specific activity

Alpha specific activity versus burnup and decay time is shown in Fig. 14 [23]. The major nuclides contributing to alpha specific activity up to 200 years after SF discharge are ^{244}Cm (half-life: 18.11a), ^{238}Pu (87.74a) and ^{241}Am (432a).



(a) (b)

FIG. 14. Dependence of alpha radioactivity on discharge burnup and cooling year of UOX-SF: (a) measured per t IHM, and (b) per TW·h. [23].



(a) (b)

FIG. 15. Dependence of neutron emission rate (Giga neutrons/s) on discharge burnup and UOX-SF cooling year: (a) measured per t IHM, and (b) per TW·h. Broken lines correspond to spontaneous fission neutrons. Ref. [24].

4.1.1.7. Neutron specific activity

Neutron specific activity versus burnup and decay time from UOX-SF is shown in Fig. 15 [24]. Neutron sources are composed of spontaneous fission neutrons of TRU isotopes (denoted as SF in the figure) and neutrons due to (α, n) reactions on oxygen isotopes of oxide fuels. Delayed neutrons emitted by (β^-, n) decay of short lived FP are not important for SF. The spontaneous fission neutrons, especially due to Cm isotopes are paramount. However, the (α, n) contribution becomes comparable after 100 years of cooling.

4.1.1.8. High burnup effects

Higher burnup UOX fuel requires higher enrichment and generates more FP and TRU in spent fuels. The decay heat and specific activity of spent UOX fuel are a function of discharge burnup and cooling years after SF discharge. The total specific activity and decay heat is contributed almost completely by the fission products, especially ^{90}Sr (half-life 29 years) ^{90}Y and ^{137}Cs (half-life 30 years) ^{137}Ba. The half-lives of these isotopes govern the decay character of the total radioactivity until about 200 years after SF discharge irrespective of discharge burnup. After 100–200 years, specific activity and decay heat due to the actinides becomes dominant. Decay heat expressed per TW·h is almost constant with respect to burnup.

4.1.2. Cladding characteristics

The behaviour of a number of cladding properties is of concern for the handling of high burnup fuel after in-reactor irradiation. Typical irradiation induced alterations are rod growth and clad hardening. During reactor operation the cladding undergoes water corrosion with formation of hydrogen. Part of the hydrogen will be uptaken by the cladding and may contribute to further reduced ductility. Under PWR conditions corrosion layers are formed of uniform thickness, whereas under BWR conditions a thin uniform oxide layer interferes with nodular corrosion of different extent. In a BWR clad, the hydrogen concentrations reached are typically lower than those potentially reached in a PWR clad. Another feature of the cladding, which may be of concern for the handling of high burnup fuel, is crud deposition on the rod surface. Experience shows that crud can be found on the cladding in very different amounts dependent on the individual reactor. For example, crud is not an issue with WWER rods or PWRs in Germany. Crud is the only radioactive particulate that can be released without breaching the fuel rod cladding and presents a dose if the cask system is breached. The corrosion of the cladding can lead to increased driving stresses for cladding breach, while hydriding can affect the ductility which leads to cladding breach during an accident event. Irradiation growth is important since the loading on the fuel during end drop accident is affected by the size of the rod to the end cap gap. Irradiation growth is a particular concern in CANDU fuel assemblies where the rods are welded to the end caps and can cause bowing of the assembly. All CANDU fuel bundles fabricated since 1974 have a graphite layer (called CANLUB) of up to 20 µm thick on the inside surface of the Zircaloy-4 cladding. CANLUB was developed to reduce susceptibility of the cladding to fission product induced stress corrosion and embrittelment cracking, in particular during in-reactor fuel power boosts.

While pellet cladding gap is not a cladding property, decrease of the gap and binding of the fuel to the cladding can change the stiffness of the rods and change their behaviour during a transport accident.

4.1.2.1. Crud

In the last 10 years, a phenomenon has been observed in plants in the USA where a distinctive crud pattern (DCP) has been observed on the outward facing side of some external rods in an assembly. It was not observed on any of the interior rods. This crud appears to be significantly thicker than the regions where there is no DCP. The number of BWR fuel cores that have experienced significant crud formation is relatively small; however, the number of PWR cores experiencing this problem has been considerably larger. The exact number of PWR cores is not known because the utilities do not report this information to USNRC unless the crud thickness is severe enough to result in power derating of the plant for a specific cycle.

The DCP[6] crud appears denser and thicker then non-DCP crud but very few quantitative measurements exist. Differential eddy current measurements [25] have indicated DCP crud thickness between 33 and >100 µm on PWR rods. This is considerably thicker than the <25µm average maximum thickness on lower burnup rods [26]. As with crud at lower burnups, the thickness varies considerably over the rod surface and is predominately Fe and ^{60}Co after 10 years. Considering that DCP crud is only the external rod faces; 13% of the rod area would have DCP crud. This is equivalent to a uniform average crud coverage of 4 to >13 µm which is in the same neighbourhood as the average maximum crud coverage on lower burnup rods [27]. As a result, the DCP does not increase the release inventory on the higher burnup rods.

Cores where the fuel has been driven harder, i.e. higher power operation for longer times, have experienced the majority of the axial-offset-anomaly (AOA) problems. More plants are driving their fuel harder than experienced a few years ago and this trend will continue in the future [28]. Vendors are currently working on fuel

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[6] The crud formation is also generally referred to as the axial offset anomaly or AOA but it was not identified as such in some of the early instances of the problem. The name AOA refers to the fact that the axial power in the core shifts from the location where significant crud formation is experienced to a lower location in the core. The power shift is the result of boron plating out with the crud at local high axial power locations that results in power suppression (boron is a neutron absorber) and the power shifts to a different axial location in the core. The crud formation has been found to be related to sub cooled boiling which is the reason for its relationship to local high power sections of the fuel.

designs that minimize AOA. The amount of crud formed is probably more dependent on the water chemistry in a particular core than the burnup of the fuel.

Crud presents a containment issue only of it spalls from the cladding during drying, transportation due to rod vibration or when an accident occurs. Some data are available on spallation during thermal events but no data are relevant to impact type events. BWR crud tends to be loosely adherent to rods and much of it spalls in wet storage. Rod heat up tests have shown that about 15% of the adherent PWR crud spalls. A small amount is in the respirable range. For lack of data, in the USA, it has been assumed that during an impact event 100% of the crud spalls [27].

Crud is (spalled from the rod) an issue only if no fuel rods breach. If a breach occurs, the dose from the crud will be overwhelmed by the contribution from the fuel.

4.1.2.2. Corrosion

The extent of cladding oxidation at higher burnup will depend on whether the standard Zircaloy or new cladding compositions are used. In PWRs, Zircaloy-4 corrosion starts to increase rapidly after 40 GW·d/t U, and becomes quite appreciable at higher burnup. High burnup PWR rods with Zircaloy-4 cladding can have oxide thicknesses in the range of 80 μm to 100 μm and higher [29]. The consequence is a thinning of the cladding wall by up to 10%. The newer cladding compositions, M5^{TM}, and ZIRLO^{TM} etc. were developed to resist corrosion during the extended irradiation [30, 31]. The corrosion of these cladding is considerably less than that of Zircaloy-4 at equivalent burnups. Even the ZIRLO^{TM} advanced cladding from Westinghouse has experienced oxide thicknesses above 80 μm in recent years in high duty plants[7] [28, 32] as the plants today are experiencing higher duties than ever before. This trend is increasing due to utilities wanting more performance from their fuel to reduce costs. In case of M5^{TM} alloy, oxide thickness remains below 40 μm up to burnups of 71 GW·d/t U [33].

As the oxide thickness increases, especially in older type cladding, it loses ductility in addition to radiation hardening, due to the absorption of a fraction of the resultant hydrogen, released during oxidation, into the outer layer of the cladding. As a result, high burnup cladding may have a loss in ductility when there is a large amount of corrosion [30–32]. The less ductile cladding will be more susceptible to burst rupture during a fire, and fracture during a drop event. Rupture and fracture testing of high burnup claddings is just in its initial stages.

Corrosion of cladding in CANDU reactors is not a problem.

4.1.2.3. Hydride characteristics

When the cladding corrodes in-reactor, a portion, (<10 up to 20%,) of the hydrogen produced by the Zircaloy/water reaction is absorbed into the cladding at the irradiation temperature. In same instances, up to 40% of the generated hydrogen has been absorbed. The average hydrogen content for low tin cladding at ~30 GW·d/t U is ~100–200 wppm (weight parts per million). As the cladding cools, and the hydrogen concentration exceeds the solubility limit, the excess hydrogen precipitates as hydrides. The PWR cladding is fabricated so the hydride precipitates in a circumferential direction. At lower burnups, hydrides in BWR cladding precipitate randomly and uniformly across the cladding. At very high burnups, radial hydrides may form in-reactor. At this concentration, the hydrogen has little effect on the mechanical properties of the cladding as long as the hydrides remain in a circumferential direction during storage. (All indications are that they remain circumferential). There is an axial hydrogen concentration gradient peaking about 2/3 the length up the rod. At higher burnup, Zircaloy-4, due to the greater cladding oxidation and the likelihood of spallation of the thicker oxide , will pick up anywhere from 500–1000 wppm hydrogen In addition, the cladding can become significantly embrittled due to localized hydriding when a large amount of oxide spallation is present. This hydrogen, which exceeds the solubility limit, tends to precipitate as a thick, brittle hydride-rich layer at the exterior surface of the cladding.

New cladding materials have improved the corrosion resistance considerably. Due to the lower corrosion and the material choice, less hydrogen is taken up. For example, the hydrogen concentration in the M5^{TM} cladding remains below 100 wppm up to very high burnups [33]. Furthermore, thinner oxides lead to less oxide spalling and, thus, localized hydriding can be avoided.

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[7] Plants with high inlet and outlet primary coolant temperatures with high LHGRs.

4.1.2.4. *Mechanical behavior*

The mechanical properties of the cladding and structural materials of the assembly will change while in-reactor due to irradiation damage and influx of hydrogen due to corrosion in the coolant. In general, the irradiation effects saturate during the first few reactor cycles and do not change significantly with additional burnup. The fuel cladding will be under stress during the back end of the fuel cycle due to the internal gas pressure, while control rod tubes will be unstressed. The materials tend to get more brittle with higher yield strength. Much of this damage may be annealed out during vacuum drying used before dry storage [34]. At 150°C the yield strength of irradiated Zircaloy-4 is ~ 800 MPa, and the ultimate strength is ~850 MPa (Mega Pascal). The yield stress decreases with temperature and is slightly lower at slower strain rates. At a fluence of ~12×10^{25} n/m^2 and 350°C the yield strength of Zircaloy-2 is ~550 MPa [35]. None of this data though accounts for any reorientation of the hydrides. Yagnik et al [36] showed that at room temperature with as little at 200 ppm H_2 that the total and uniform elongation decreased 60 to 85% with 70 ppm radial hydrides. There was no effect of radial hydrides at 300°C.

The fuel rod cladding and assembly hardware expand during irradiation caused by dislocation climb activated by neutron irradiation, and expansion of the pellets due to the swelling of the in-growth of solid fission products. This phenomena, known as irradiation growth is quasi-linear with burnup or fluence for stress relieved annealed (SRA) Zircaloy materials. Compared to Zircaloy-4, M5^{TM} due to its fully re-crystallized state exhibits a lower fuel rod growth, especially at higher burnups with pellet cladding interaction [33]. Clearance space is designed into the assembly design to accommodate this growth along with any thermal expansion of the fuel in reactor. The size of the gap between the top of the rods and the top nozzle is an important parameter in the structural analysis of the rod integrity during a drop accident. The larger the gap, the more potential damage to the rods, so higher burnup from the growth standpoint should be a benefit.

Irradiation growth is an issue in the CANDU reactors since the fuel rods are welded to the end plate. Differential growth between the inner and outer rods of an assembly can cause bow and stresses that result in fracture of the end plate/rod weld causing handling issues.

At Research Institute of Atomic Reactors (RIAR)-Institute of Atomic Reactors/SSC-State Scientific Centre/FSUE-Federal/State Unitary Enterprise (FSUE ‘SSC RIAR’) the post irradiation examination of some high burnup WWER-1000 fuel rods was carried out. It confirmed high geometrical stability of WWER-1000 alternative fuel assembly provided by the rigid frame for about 6 fuel cycles.

4.1.2.5. *Pellet-cladding gap*

The size of the pellet cladding gap is important in determining the rigidity that the pellet adds to the cladding during a drop accident and the rate of depressurization and loss of particulate when the cladding is fractured. LWR fuel is built with a small fuel pellet gap that varies from manufacturer to manufacturer. During the first few irradiation cycles, the gap closes due to pellet fracture, and creep down of the cladding due to the greater exterior coolant pressure. In PWR fuel this results in ridging at the pellet-pellet interfaces.

The gap is measured from micrographs after the fuel has cooled. As a result due to differential thermal expansion of the fuel and cladding materials it may not be the same as when thermally hot. Photomicrographs of lower burnup fuel indicate that there are cold gaps, but as can be seen in Fig. 16, the gap seems to have closed even in the cold state at higher burnups. The hot gap can be somewhat approximated by the hydraulic diameter although much of this flow area may be in the pellet cracks and not the gap. Measurement of the hydraulic diameters of sections of average and high burnup spent fuel rods show that average and high burnup spent fuel rod sections respectively have hydraulic diameters of about 50 and 35 μm [37].

4.1.3. Fuel pellet characteristics

Changes to the properties of fuel pellets at higher burnup are important since they affect the stress put on the cladding, hence the fracturing of the cladding, and provide the source term for the evaluation of potential releases during accidents and subsequent doses to the public.

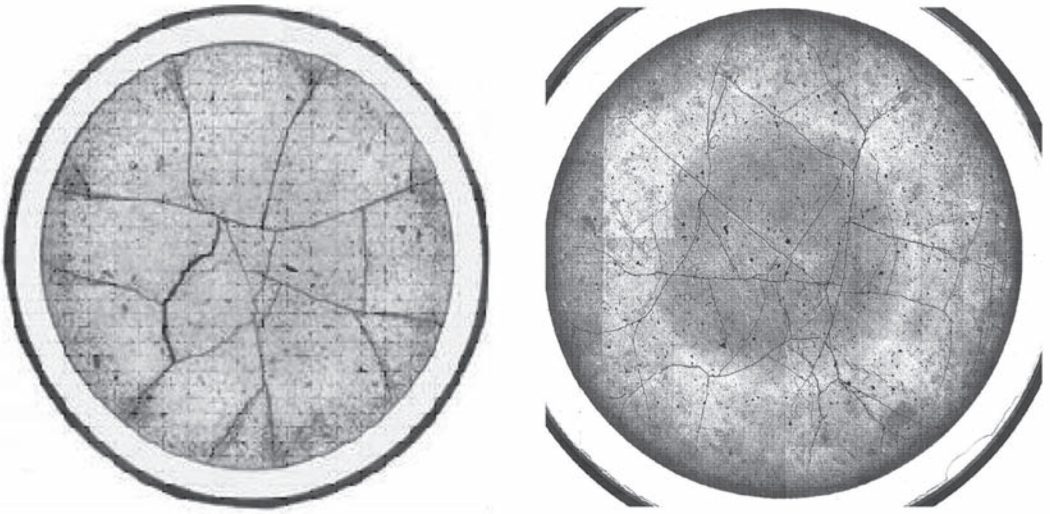
(a) (b)

FIG. 16. Cross-sections of PWR fuel at the midplane of (A) 35 GW·d/t U Surry fuel and (B) 67 GW·d/t U Robinson Fuel (courtesy ANL).

4.1.3.1. Grain size

PWR/BWR fuel, taken to higher burnup, has been manufactured with linear grain sizes mostly in the 8–12 μm range. During irradiation there is some grain growth in the central hotter region of the pellet. The extent of this growth depends on the temperature at which the fuel operated. At the same time, fission gas is produced in the grains and migrates to the grain boundaries where fission gas bubbles are formed. With continued irradiation, these bubbles line up on the grain boundaries and bubbles become interlinked releasing the fission gas and volatile products to the fuel cladding gap. This is observed to occur within the central portion of the fuel such that the radius at which inter-linkage occurs increases with increasing burnup for a constant temperature profile. Thus fission product gases could be more readily released in case of an accident or cladding breach. This is one of the principle reasons for the potentially increased fission gas release with burnup.

Today's tendency is to further increase the grain size to extend the diffusion length of the fission gas inside the grain and, thus, to reduce its release. In case of chromium doped fuel, grain sizes of up to 45 μm may be reached.

4.1.3.2. Fission gas release

An issue of significance for out of reactor operations such as storage and reprocessing is the increase in fission gas releases with burnup.

Figure 17 summarizes fractional fission gas release as a function of rod power. A near three-fold increase in PWR rod power from 130 W/cm to 320 W/cm would increase fractional fission gas release from about 0.1% to 20%. Fractional fission gas releases show a parabolic increase, from about 5% to 25% for burnup increases of 20 to 100 GW·d/t U (Fig. 18). The large increase in fission gas release after 80 GW·d/t U is from an accelerated release from the intermediate region of the pellet $r/r_o = 0.4$ to 0.7 due to grain re-crystallization. The fission gas in the rim region of the pellet migrates from the grains into the porosity. This is shown clearly by EPMA traces [38].

The release of fission product volatiles such as iodine and caesium may also be important. As the burnup increases above 60 GW·d/t U the Cs migrates to the pellet-pellet interfaces. The release fraction tracks that of the Xe closely. The Cs migrates from the hotter grains interior to the pellet. The small rim grains almost completely retain their Cs, although some may be released at very high burnup [39]. The iodine and caesium are generally found on the fuel surface and open cracks when there is significant fission gas release (FGR) observed but they are not generally volatile at room temperature but do become volatile at higher temperatures. Based on in-reactor release values for xenon and iodine it is known that their release is very similar and the ratio of D_I/D_{Xe} is

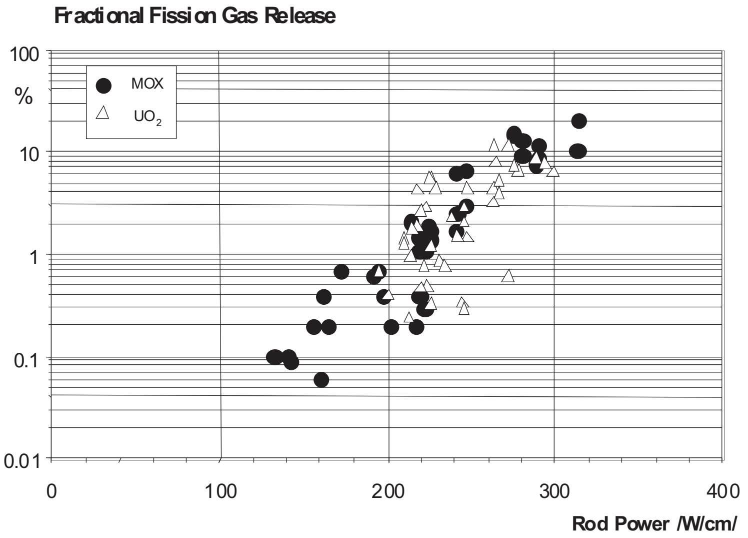

FIG. 17. Fission gas release of MOX and UO2 fuel relative to the average rod power of the release relevant cycle (rod burnup <40 GW·d/t U).

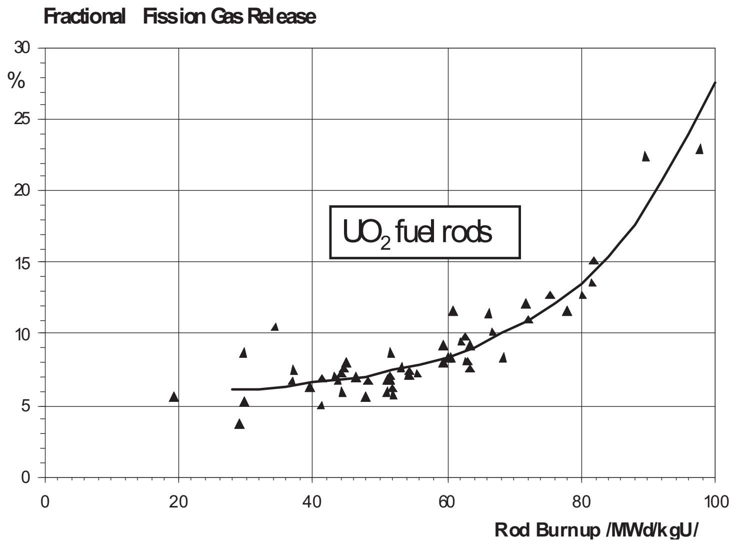

FIG. 18. Fission gas release from UO_2 shows an accelerated increase at high burnup.

approximately 1.0 [40, 41]. Similarly, the ratio of caesium releases to xenon release is also approximately 1.0 [42] (see Table 3)[8]. Therefore, the fraction of iodine and caesium on the surface of the fuel available for release is similar to that for FGR.

Fission gas release for WWER is shown in Fig. 19 [43]. Beginning from a burnup of 45–50 GW·d/t U, the FGR release is intensified. The increased gas release can be, mainly, caused by restructuring of uranium dioxide at formation of the rim-layer. The data obtained from the significant amount of the examined fuel rods suggest that the maximum gas release value from WWER fuel does not exceed 4.5%.

4.1.3.3. Rim

At about 40 GW·d/t U average pellet burnup, a rim region starts to form on the outer radius of the PWR/BWR pellet. The rim is characterized by a much higher porosity, formation of many small grains in the submicron range

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[8] This information is currently being included in a new standard (ANS 5.4) for calculating the release of radioactive isotopes as a result of steady state in-reactor operation.

TABLE 3. Cs AND Xe RELEASE AT HIGH BURNUP [42]

PELLET BURNUP (GW·d/t U)	Cs RELEASE (%)	ROD BURNUP (GW·d/t U)	Xe RELEASE (%)*
102	26.3	98	26.0
90	19.9	90	19.0
83	11.0	78	12.5
67	8.2	63	9.0
65	8.0	60	8.2
50	8.0	45	7.0

* The release values for Xe are taken from the curve in Fig.18.

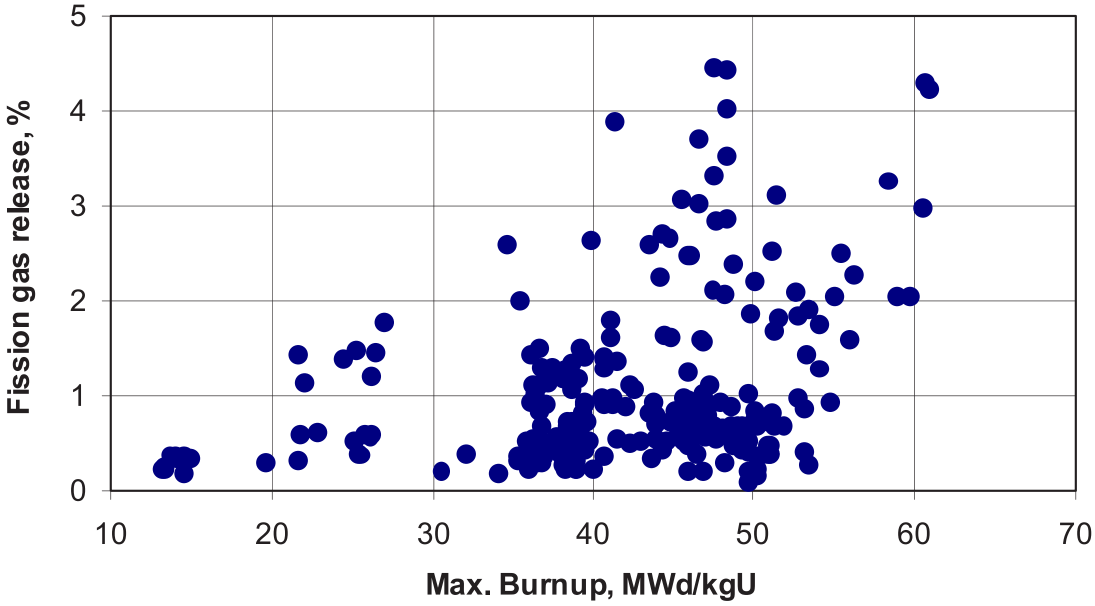

FIG. 19. WWER fission gas release.

[26, 39, 44, 45] (compare Figs 20, 21), and a higher retention of the noble gases. Each of these parameters has been used as a measure of the rim width, and as such there is a large uncertainty in the width of this region.

Total rim porosity ranges between 10 and 40%, but most researchers today report 15 to 20% [46]. This compares to about 6% in the bulk material (Fig. 22). There is significantly more porosity in the rim above 40 GW·d/t U [46], which could make the fuel in this region more friable, compared to the bulk of the fuel (see Fig. 23). There is also much more closed porosity (Fig. 24) containing gas [47] that could tear the fuel apart further when heated in a fire. For example, there have been reactivity insertion accidents (RIAs) that achieved fuel temperatures above 1100°C and pushed fuel out of the fuel rod after the cladding fractures due to the high pressure bubbles on the grain boundaries [48]. These closed pores are intragranular within 400 μm of the pellet edge and intergranular further inward. Most of the micrographs and SEM fracture surfaces show large bubbles between sub-grain boundaries (several sub-grains surrounding the bubble) The typical pore in the rim region is about 1 μm [45].

The inside of these closed bubbles tend to be lined with many submicron-size fuel particles in the 10–200 nm range [45], which would also be released when these bubbles burst. Small grains between 20–400 nm are also found between the larger grains at the inner transition zone between the rim and the fuel outside of the rim [46]. The amount of small grains increases locally with burnup. At the higher burnup rate fission products deposits occur where the fuel cracks meet the cladding (see Fig. 25) and the fuel in general swells due to the creation of solid

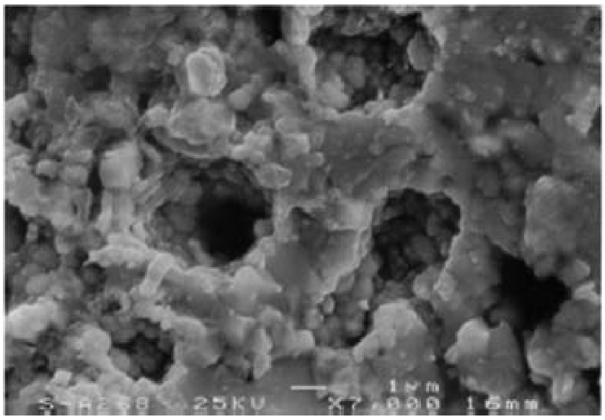

*FIG. 20. SEM micrograph of the rim region ($r/r_o=1$) of fuel with 98 GW·d/t U average burnup showing grains in the submicron range and faceted pores (J. Spino et al., J Nuc Mater **354** (2006) p 66).*

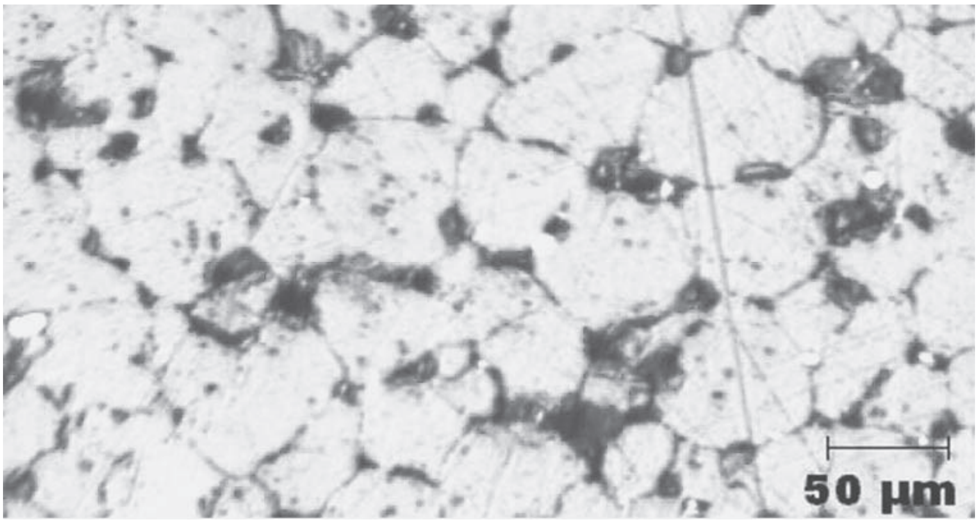

FIG. 21. Etched fuel, center of pellet, H.B. Robinson 67 GW·d/t U (courtesy ANL).

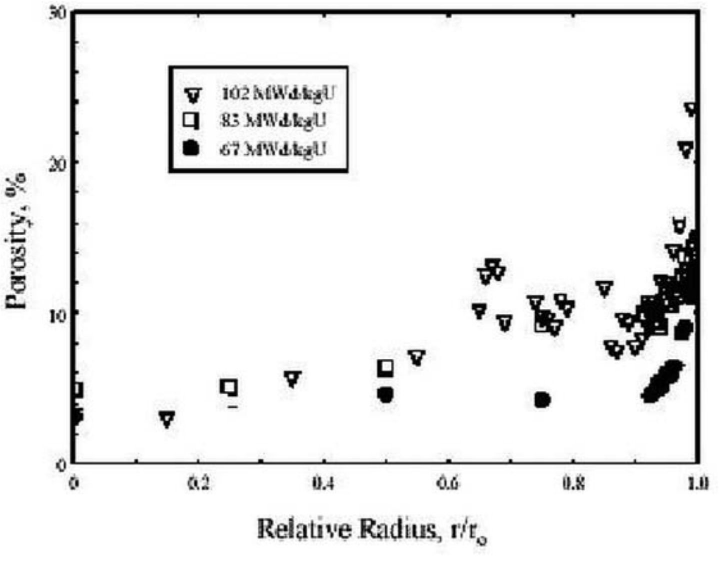

FIG. 22. Effect of pellet burnup on the radial distribution of porosity.

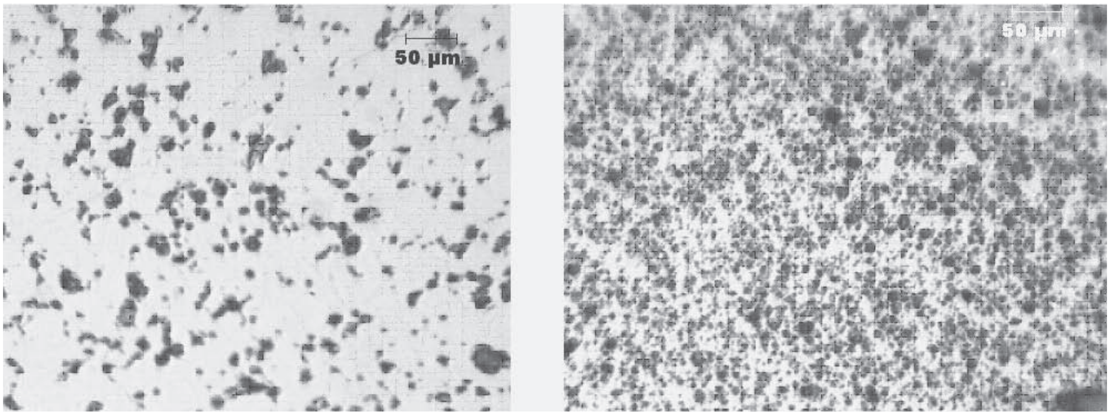

FIG. 23. Comparison of porosity at rim and centre of H.B. Robinson fuel at 67 GW·d/t U (courtesy ANL).

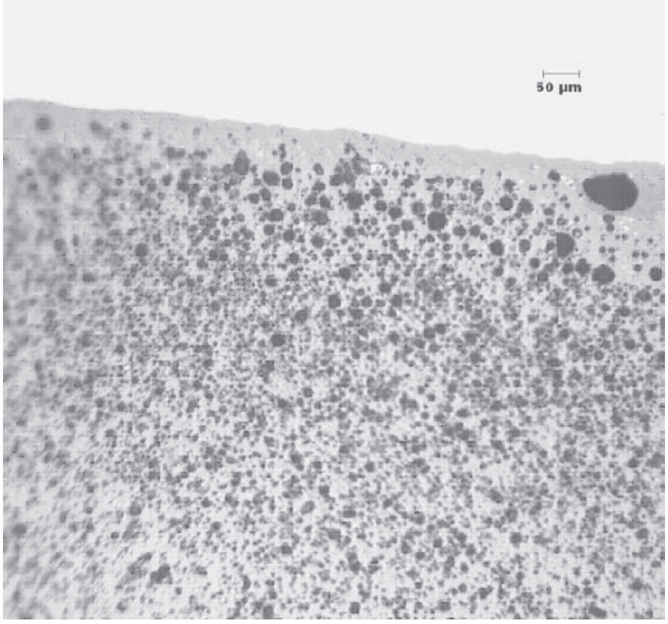

FIG. 24. Unetched rim of H.B. Robinson fuel at 67 GW·d/t U showing extensive porosity (courtesy ANL).

fission products. This may reduce the amount of delayed particulate discharged from the rod caused by the blow down of the rod gases due to a more convoluted path.

The grain size changes within high burnup fuel as you proceed from the central portion to the outer rim of the fuel. The major portion of high burnup fuel will have a grain size similar to (unchanged from) the as-fabricated grain size of approximately 10 µm typical of commercial fuel. The central portion of the fuel may have some grain growth (up to a factor of 2)[9]. The rim portion of high burnup fuel will have much higher burnups than the pellet average and forms restructured fine sub-grains at pellet average burnups > 40 GW·d/t U. The sub-grain sizes are generally between 0.1 µm to 0.3 µm [39.49–51]. As the burnup of the rim increases the original as-fabricated grain boundaries begins to disappear as the sub-grain structure becomes dominant. This restructured rim is not present in the older fuel where rod or bundle burnups did not exceed 33 GW·d/t U.

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[9] If rod powers were high early in life, i.e. at burnups <20 GW·d/t U, but this will be ignored because it is difficult to estimate the amount of fuel with significant grain growth.

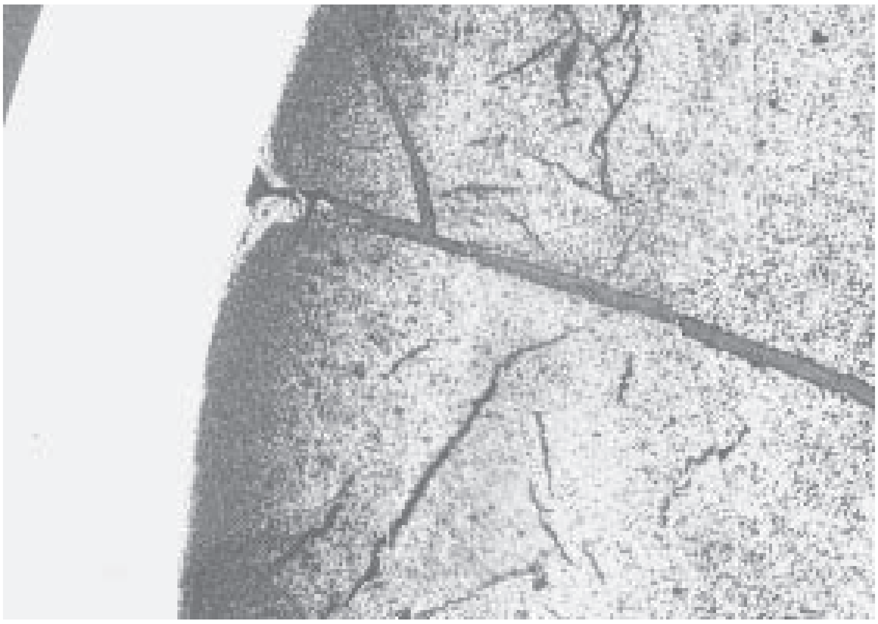


FIG. 25. Pellet-cladding interface of Robinson Fuel at 67 GW·d/t U showing small radial micro cracks in the rim region of the fuel and deposits of fission products at the large crack cladding interface (courtesy ANL).

The size of the restructured rim increases exponentially with increasing burnup and the data presented by Manzel on measure rim thickness versus pellet average burnup up to 100 GW·d/t U [39]. (See Fig. 26) is considered to be the best available data. A correlation based on the Manzel data has been developed for use in the FRAPCON-3 [52] fuel performance code and is given as:

t = 1.439e-6*(burnup) ^ (4.427)

where

t = the rim thickness where the grains are fully transformed (microns)
burnup = the pellet-average burnup in GW·d/t U.

The largest rim observed, in very high burnup fuel beyond current expected high burnup irradiation ranges, was 25% of the fuel pellet radius [39] (44% of the fuel volume). Predictions of a 1 mm thick rim at 75 GW·d/t U have been made [53].

4.1.3.4. Degree of fracturing

Due to thermal gradients, fuel pellets tend to fragment early in life. The number of fragments is usually determined from examination of the cross-sectional and longitudinal micrographs. This can range from 20 to 50 large fragments. No systematic study has been conducted of the fragment size distribution to see if it is different for BWR and PWR fuel, or if it changes with burnup. Example cross-sections of cracked fuel at 35 and 67 GW·d/t U are given in Fig. 16 for comparison purposes. Although this is only one example, there appears to be more cracks in the higher burnup fuel. Many fuel fines form at the pellet-pellet interfaces as the edge of the pellets are ground away when the pellet dish is compressed. At the higher burnup, additional small fuel cracks are seen in the fuel region immediately interior to the rim (see Fig. 26, 26a). This may lead to more fragmentation when the fuel is stressed. It is expected that there is similar fracturing in both CANDU and WWER fuel. The degree of fracturing will depend on a variety of parameters such as ramp rate and operating temperature.

Little is known about the fracture behaviour of the fine grained rim region formed at high burnup. Anecdotal evidence from the preparation of ceramographic mounts indicates that the region is rather friable. On the other

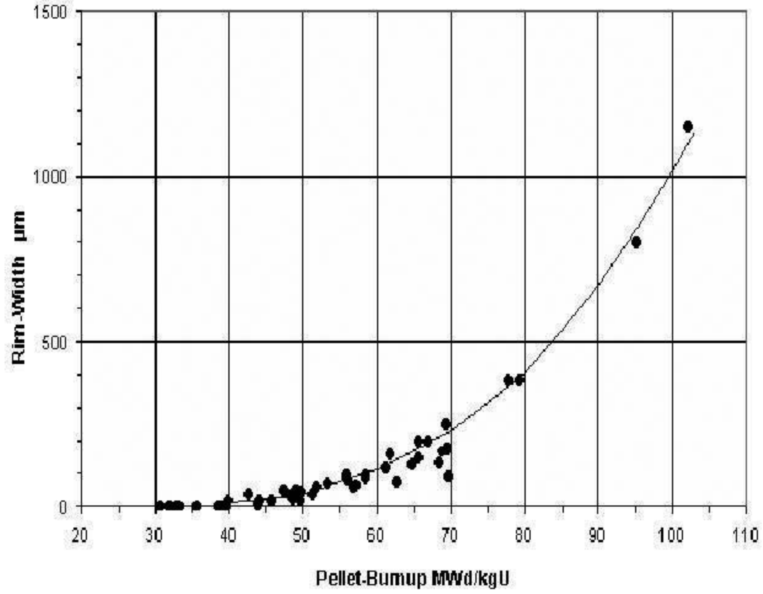


FIG. 26. Development of the pellet rim structure with burnup. The width of the rim has been determined by optical microscopy [39].

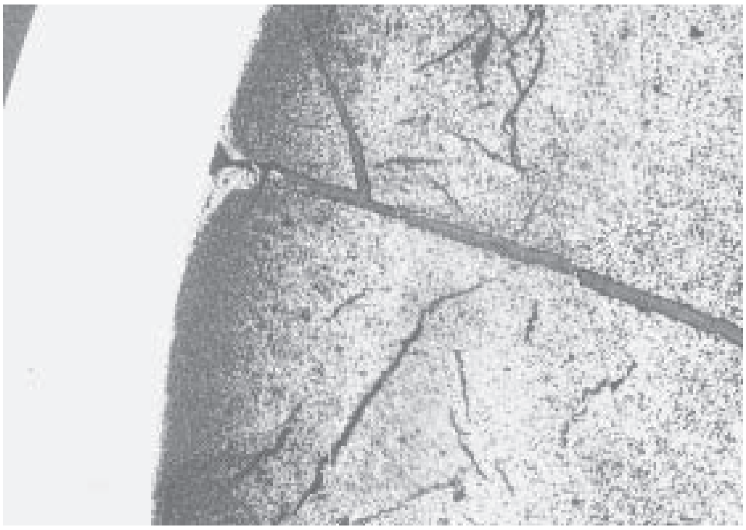


FIG. 26a. Pellet-cladding interface of Robinson Fuel at 67 GW·d/t U showing small radial micro cracks in the rim region of the fuel and deposits of fission products at the large crack cladding interface (courtesy ANL).

hand, microhardness measurements on the rim have led Goll [36] to conclude that compared with normal fuel microstructure that the rim has a greater resistance to crack propagation and, hence, micro fracture.

4.1.3.5. Oxidation

Irradiated UO_2, exposed to an oxidizing atmosphere, will eventually oxidize to U_3O_8. The oxidation time is an Arrhenius function of temperature. The grain boundaries of irradiated fuel pellets are highly populated with voids and gas bubbles. Initially, the grain boundaries are oxidized to U_4O_9 resulting in a slight matrix shrinkage and further opening of the pellet structure. Oxidation then proceeds into the grain until there is complete transformation of the grains to U_4O_9 [54]. The grains remain in this phase for a temperature dependent duration until the fuel resumes oxidizing to the U_3O_8 state. The transformation to U_3O_8 occurs with ~33% lattice expansion that breaks the ceramic fragment structure into grain sized particles. The mechanism of oxidation in irradiated fuel appears to be different than in unirradiated fuel [55]. In unirradiated fuel, the U_3O_7 is formed and oxidation proceeds from the fragment surface and not down the grain boundaries. This mechanistic change occurs at or below ~10 GW·d/t U. Expansion of the fuel, when it transforms to U_3O_8, induces a circumferential stress in the cladding. As a result the initial crack starts to propagate along the rod. Axial propagation, spiral propagation and a combination of the modes that result in circumferential splitting have been observed in PWR rods [56].

The oxidation data base was developed mostly in the 1980s in the USA, Canada, the UK, and Germany. Some later work was done in Japan, and most recently work is on-going by the French primarily on MOX fuel. Limited work on cladding splitting was done in the early 1980's by the USA [56–58] and Canada [59, 60].

No oxidation or splitting studies have been conducted on spent fuel rods with burnup greater than 45 GW·d/t U. Between 30 and 45 GW·d/t U, the data seem to indicate that there is a decrease in the oxidation rate due to the presence of certain actinides and fission products that are burned into the fuel. There is no reason that this trend should not continue at higher burnups. As indicated earlier, the oxidation process is a grain boundary effect. Higher burnup fuel (>55 GW·d/t U) forms an external rim on the pellets that consists of very fine grains (1 micron vs. 10 micron). While the rate of the oxidation may decrease with burnup, the total amount of fuel that is oxidized may increase due to a much greater intergranular surface area in the rim region. Hanson's analysis [61] of data from the NRC whole-rod tests [57] in which defect propagation was observed to occur earlier at the defects at the lower burnup end of the rod supports a burnup effect.

4.1.4. Fraction of cladding that is 'damaged' (pinhole leaks or hairline cracks) or 'failed'

Damaged fuel, for the purpose of this discussion, is being defined as fuel that has suffered cladding breaches in reactor, or mechanical damage to the assembly structure while being handled. It does not imply that these assemblies are or should be considered damaged for steps in the backend of the fuel cycle where the requirements are considerably different than in reactor.

The latest breach statistics for USA PWR/BWRs given by Yang et al [62] are shown in Fig. 27.

As can be seen, the number of defective assemblies has been decreasing yearly. This is in spite of the average discharge burnup increasing.

In addition, as shown in Fig. 28, the major causes of rod damage is crud, fretting, debris, manufacturing, etc, all of which are unrelated to the burnup of the fuel. In addition, one would not expect burnup to influence the number of handling damage incidents.

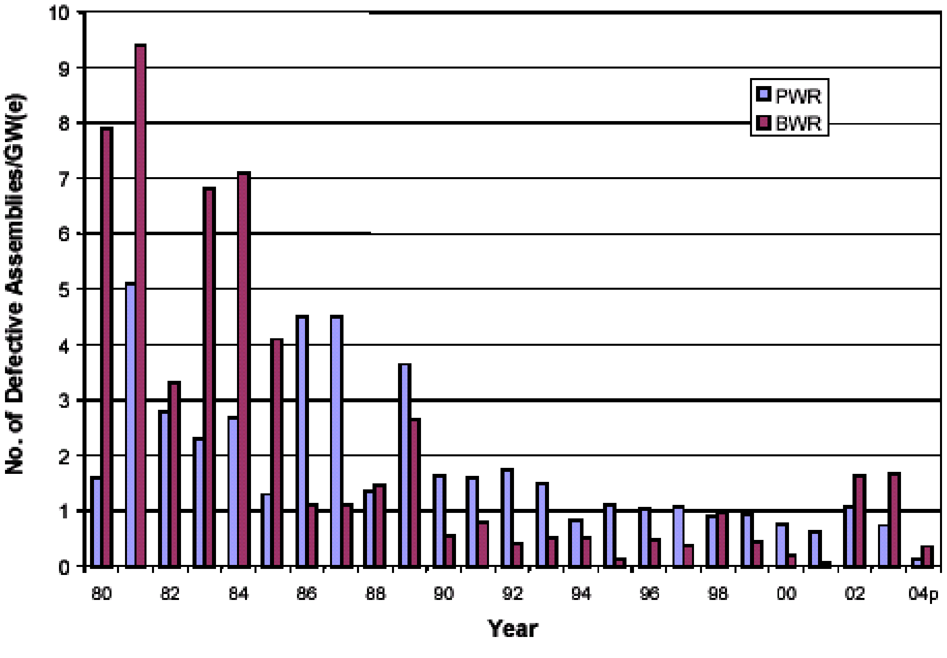


FIG. 27. Trend in the USA in fuel failure rates.

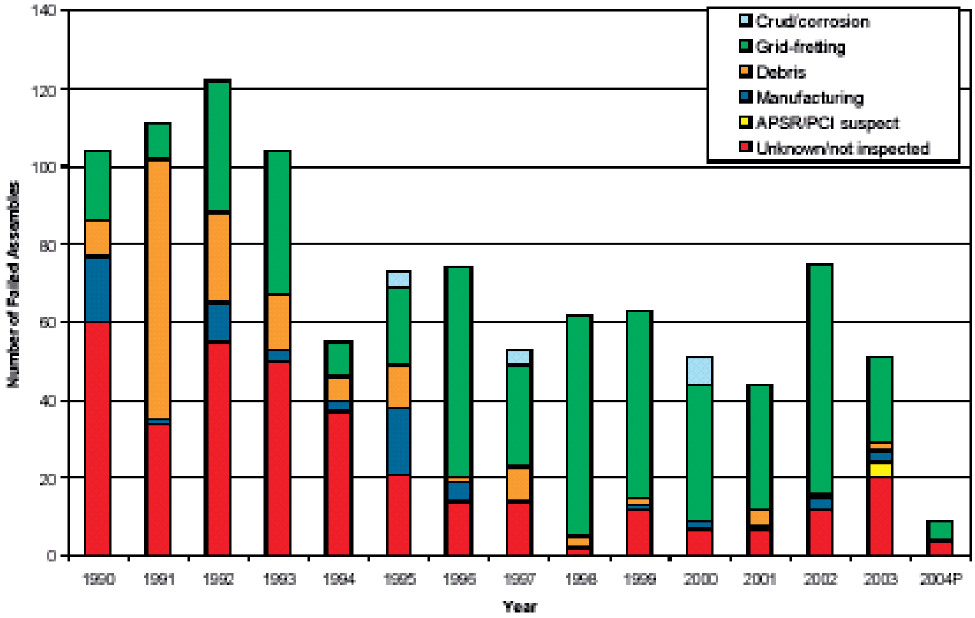


FIG. 28. Trend in US PWR failure root causes (2004 results are incomplete) [62].

Post radiation examination (PIE) results and history of operation are used for evaluation of used fuel characteristics. It was estimated so far that only a small percentage of bundles may have conditions that may cause loss of integrity during storage (0.02–3.6 %). The wide range of percentage of bundles with a potential loss of integrity is due to uncertainty in how representative the inspected fuel bundles are of the general population.

4.1.5. Overall assembly mechanical properties

To increase in-reactor performance, the vendors have new assembly and rod designs. BWR assemblies are going to utilize an increased number of rods per assembly, and assemblies with large central stems such as the Water Cross, and 9 × 9 and 10 × 10 fuel are being used [63]. All PWR assemblies have kept their rod array unchanged until today. However, a major development in PWR assemblies is new designs of the grid spacers to improve flow and decrease fretting failures.

Unlike cladding which is under a hoop stress that might reorient the hydrides to a radial direction, as discussed in an earlier section, control rod tubes are only under residual stress from their formation. As a result, reorientation is not expected and reduction of mechanical properties of the control rod tubes at higher burnup should not be a concern.

4.1.5.1. Bowing

Due to differential local fluence and temperature in the reactor, rods on one side of assembly grow more then the other side. As a result the assembly assumes either a top to bottom ‘C’ or ‘S’ shape curvature referred to as ‘bow’. The bow tends to increase with burnup but is limited by movement of the assembly within the core for continued reactor operation. The usual sign of bow in the reactor is difficulty in insertion of either the control rods in a PWR or the control blades in BWRs. There are usually problems before the bow reaches 0.25” measured at the point of maximum defection [64]. Bow can be reversed or minimized by rotating the assemblies or moving them to other slots in the reactor core. Excessive bow will cause problems in storage and transportation if the assemblies are too curved to fit in the basket slots. The bow of the assembly must be accounted for when determining the potential rod damage during a drop accident. Since it will be no larger than the basket slot, higher burnup does not affect the breach results.

4.1.5.2. *Irradiation growth*

Irradiation growth is expected to be an issue in CANDU fuel when the fuel is welded to the end plates. Bowing of the assembly in-reactor can lead to stress on the weld and subsequent weld failure.

4.1.5.3. *Grid behavior under accidents*

The stability of the PWR/BWR assembly following accident drops depends on the integrity of the grid spacers. If the grid spacers collapse, there is the potential to change the rod spacing. Should water or other moderator ingress occur, criticality might be achieved.

The grid spacers become brittle as they are irradiated and incur more radiation damage. This damage will increase with increased burnup. Tests under dynamic lateral loads, not representative of drop event accident condition on lower burnup grids show the grids will deform and crack.

4.2. COMPARISON OF LWR MOX TO UOX FUEL CHARACTERISTICS

The main fissile material of the fresh MOX fuel is fissile plutonium (Puf: ^{239}Pu and ^{241}Pu), whose content in total Pu depends on (a) discharge burnup (see Fig. 29), (b) cooling years of the original UOX-SF and (c) storage years of Pu after reprocessing. The lead and lag time concept in the fuel cycle is very important because of the rather short half-life (14.35 a) of ^{241}Pu, which beta decays to ^{241}Am. Therefore we have three parameters (a), (b) and (c) to define available Pu. We do not consider MOX fuel made of a surplus weapons material, which is under the ongoing US–Russian Pu disposition programme, although the MOX fuel characteristics is essentially the same as that made of civil (i.e. reactor grade) Pu.

The conventional MOX fuel is Pu oxide mixed in the UOX carrier material, which may be enrichment tails, natural uranium, or reprocessed uranium (REPU). The MIX or MOX/EUS (enriched uranium support) concept [65] whose UOX matrix is enriched uranium may be excluded here because MOX is only a secondary contributor for fission rate. In the following, we do not refer to unconventional fuel types either, which are uranium free oxide such as thorium or other type of chemical form such as nitrides or carbides. The ^{235}U content of the UOX matrix has also to be taken into account to determine the Puf content in the fresh MOX fuel with the expected burnup. An 'equivalent ^{239}Pu' formula for a given LWR MOX fuel is in Annex G of Ref. [66]. The changes in the reactivity worth of the Pu vector are analyzed by applying reactivity equivalence relations to MOX systems [67].

Figure 29 compares the total fissile (the sum of ^{235}U, ^{239}Pu and ^{241}Pu) contents of various fresh MOX with those of fresh UOX fuels (solid curve) for corresponding discharge burnup. These data are taken from references [24] and [68]. The total fissile of MOX fuel is always larger than that of UOX fuel because of larger neutron absorption cross-sections compared to uranium isotopes, increased concentration of parasitic neutron absorbers ^{242}Pu due to higher burnup and β decay of ^{241}Pu to ^{241}Am for longer cooling years of UOX-SF.

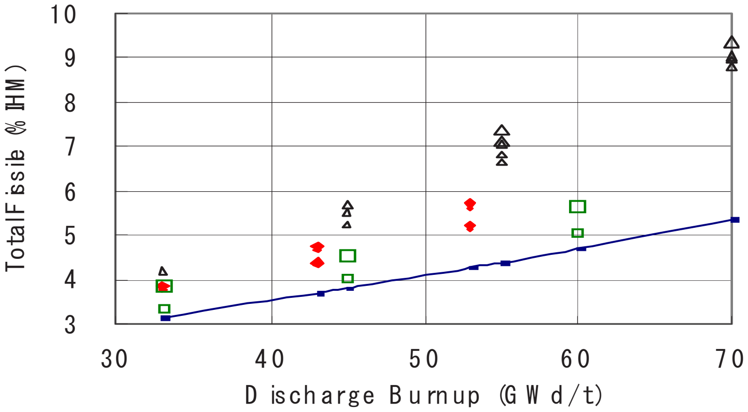


FIG. 29. Dependence of total initial fissile content of MOX fuel on average discharge burnup.

The solid curve is for UOX fuel. Triangles correspond to Ref. [24], squares to Ref. [68], and red diamonds to Ref. [66]. The size of each symbol approximately represents the cooling years of the original UOX-SF which supplies Pu.

The following analysis refers only to data from Ref. [24]. Each MOX fuel, for the expected discharge burnup, is made of isotopic Pu given in Fig. 30. The matrix for MOX fuel is 0.2% enrichment tails. The necessary fissile Pu contents in the initial heavy metal (IHM) are summarized [24] for the calculated cases in Table 4.

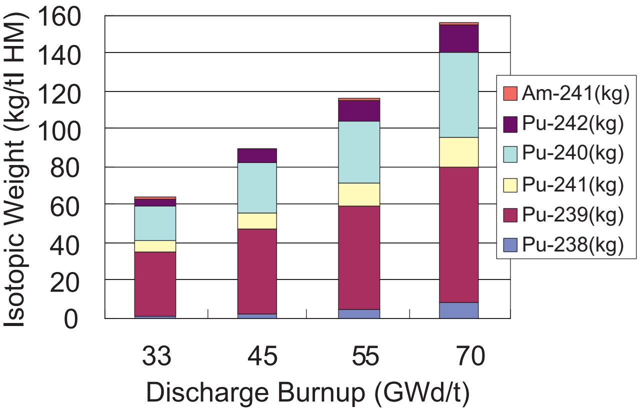


Fig. 30. Plutonium composition of the fresh MOX fuels for expected discharge burnup. Composition is given at two years after plutonium separation. This figure is prepared from the data reported in Ref. 24.

TABLE 4. FISSILE Pu CONTENT FOR EACH FRESH MOX FUEL AND ITS ORIGINAL UOX FUEL [66]

Source material for Pu				MOX burnup (GW·d/t U) and Puf (% IHM)[2]			
UOX fuel	Initial enrichment (%)	Discharge burnup (GW·d/t U)	UOX-SF storage years[1]	33	45	55	70
U33	3.0	33	5 y	3.97 (M33)	5.07		
			15 y				
			60 y				
U45	3.8	45	5 y		5.31 (M45)	6.46	
			15 y		5.50		
			60 y			6.95	
U55	4.6	55	5 y			6.64 (M55)	8.64
			15 y			6.88	
			60 y			7.22	
U70	6.0	70	5 y				8.78 (M70)
			15 y				8.88
			60 y				9.19

[1] Lead time for MOX fuel fabrication: 2 years.

[2] The content of fissile plutonium (Puf) in the initial heavy metal (IHM) of MOX fuel is shown for the expected discharge burnup of MOX fuel. Carrier material for Pu in MOX fuel is enrichment waste uranium of 0.2% tails assay. Symbols such as M55 in the parentheses denotes MOX fuel for 55 GW·d/t U discharge burnup.

The data in Table 4 indicate that the necessary fissile plutonium for a MOX fuel with an expected discharge burnup depends not only on the discharge burnup of an original UOX-SF but also on its cooling years. In the following only four cases (M33, M45, M55 and M70) corresponding to the 'standard case (UOX-SF is reprocessed after five years cooling, then recovered plutonium is fabricated into MOX fuel and loaded into a LWR after two years lead time from the reprocessing)' will be analyzed further. These four cases are also reported in Ref. [23].

Figure 30 [24] shows the plutonium composition of the fresh MOX fuels for expected discharge burnups. The β decay product ^{241}Am from ^{241}Pu is included in the total plutonium. Plutonium recovered from UOX-SF is recycled to fabricate MOX fuel of the same discharge burnup.

4.2.1. Spent MOX isotopics, decay heat and radioactivity

Based on data reported in Ref. [24], Fig. 31 shows the dependence of plutonium isotopic components on discharge burnup at five years after SF discharge. Fissile plutonium ^{239}Pu is the maximum of the whole isotopic components in the UOX-SF with its value decreasing with discharge burnup. On the other hand, a significant decrease of ^{239}Pu and an increase of ^{240}Pu is seen in the MOX-SF. Thus the isotopic content of the plutonium is degraded on the subsequent recycle due to the presence of an increasing proportion of the higher isotopes.

Figure 32a shows the burnup dependence of the TRU inventories (kg/t IHM) in the MOX-SF five years after discharge. Generally speaking, the amount of higher atomic mass TRU increases with burnup. Inventories of plutonium, americium and curium isotopes in MOX-SF are almost ten times larger than those of UOX-SF. In contrast to the UOX-SF where neptunium inventory is larger than americium (see Fig. 9). The Np inventory is much smaller than curium in the MOX-SF. The smaller initial inventory of ^{235}U results in the reduction of ^{237}Np formation by the process:

$$^{235}_{92}U(n,\gamma) \rightarrow {}^{236}_{92}U(n,\gamma) \rightarrow {}^{237}_{92}U(\beta^-) \rightarrow {}^{237}_{93}Np$$

in the MOX fuel.

When these quantities are expressed in terms of kg/TW·h, burnup dependence of TRU inventory becomes almost flat (Fig. 32b).

Dependence of the MOX-SF radioactivity in Fig. 33 is similar to what is shown in Fig. 12. The largest contribution to MOX-SF radioactivity is due to ^{241}Pu until it is replaced first by ^{137}Cs after around 60 years of

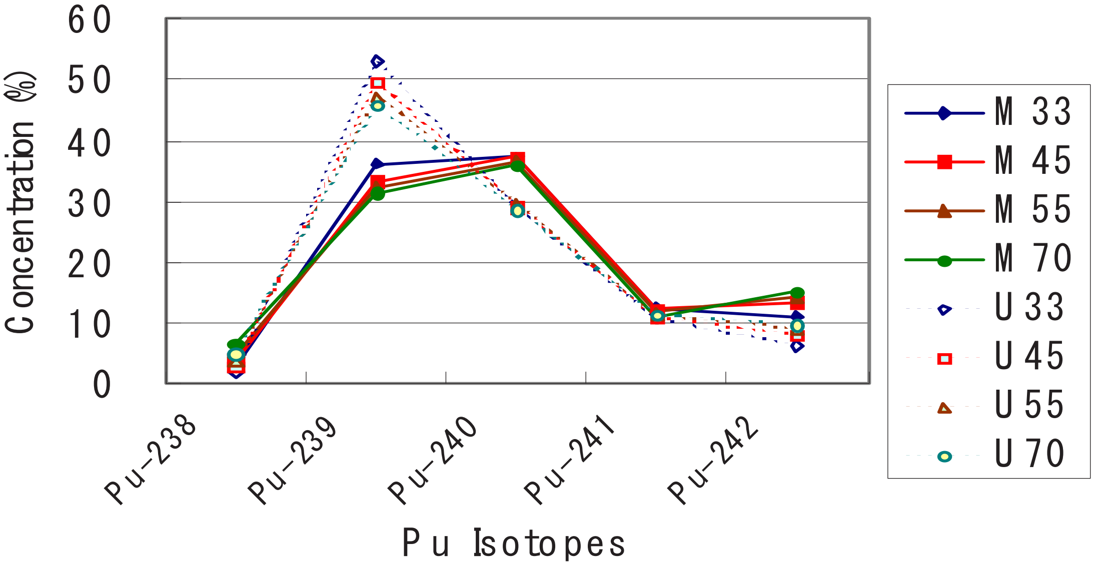
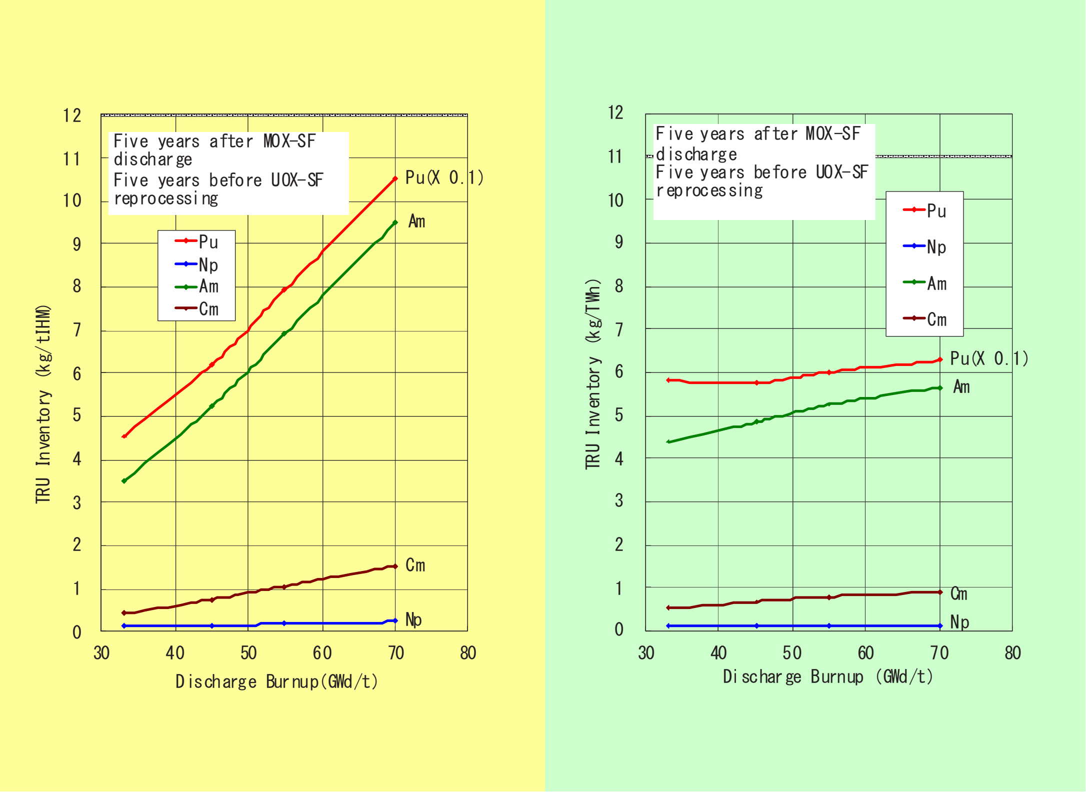


FIG. 31. Relative concentration of plutonium isotopes of UOX (broken lines) and MOX (solid lines) spent fuels for each discharge burnup at five years after SF discharge.



(a)

(b)

FIG. 32. Dependence of TRU inventories on discharge burnup of UOX-SF (five years after SF discharge): (a) measured per t IHM, and (b) per TW·h [25].

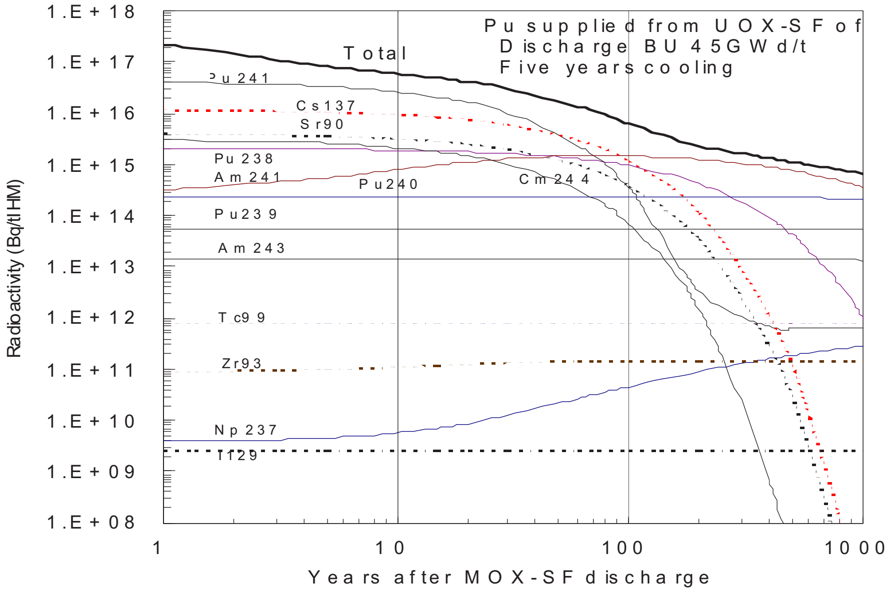
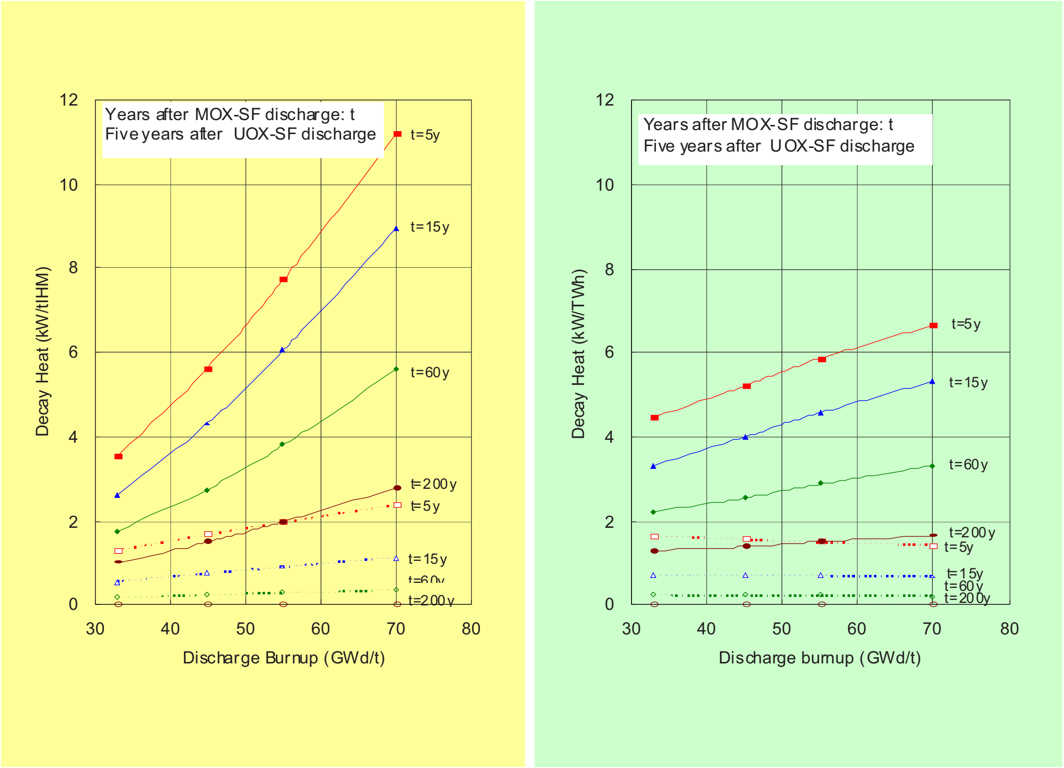


FIG. 33. Dependence of radioactivity of MOX-SF on cooling years after SF discharge: discharge burnup 55 GW·d/t U [24].

cooling and then by ^{241}Am just before one hundred years of cooling. On the other hand, the largest contribution to radioactivity in UOX-SF is due to fission products ^{137}Cs and ^{90}Sr during two hundred years as is seen in Fig. 33. It is to be noted that the cumulative fission yield of ^{90}Sr is smaller than that of ^{137}Cs especially for thermal neutron fission of ^{239}Pu compared to ^{235}U [24].



(a) (b)

FIG. 34. Dependence of decay heat on discharge burnup and MOX-SF cooling year t: (a) measured per t IHM, and (b) per TW·h. Contribution from FP is shown by a broken line. Parameter is cooling year after MOX-SF discharge [24].

Figure 34a shows decay heat of MOX-SF for 1 tonne of initial heavy metal inventory. It increases quickly with discharge burnup. However, the rate of increase slows down as cooling years of MOX-SF increases. The contribution of decay heat due to fission product (FP) is denoted by broken lines. Main contributor to decay heat of MOX-SF is ^{244}Cm and ^{238}Pu in contrast to UOX-SF where fission products ^{137}Cs and ^{90}Sr are the largest.

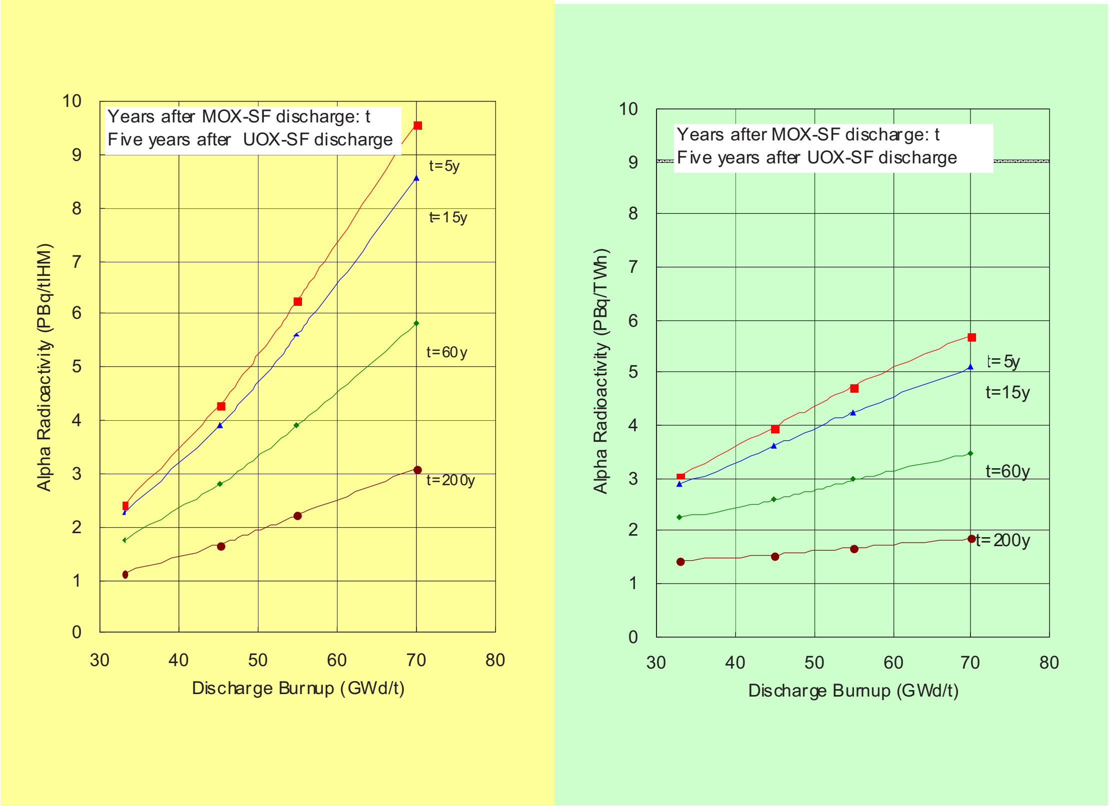
If decay heat is expressed in terms of generated energy (TW·h), its dependence on discharge burnup is found to be milder as is seen in Fig. 34 b.

Total alpha radioactivity is shown in Fig. 35 [24] as a function of discharge burnup with a parameter of cooling years after MOX-SF discharge. It is almost ten times larger in MOX-SF than that of UOX-SF (Fig. 14) corresponding to the TRU inventory of MOX-SF [23, 24] and UOX-SF (Fig. 9).

Emission of neutrons from MOX-SF is shown in Fig. 36 [24]. Total neutron sources indicated by solid lines are mainly of spontaneous fission neutrons of TRU isotopes (broken lines in the figure) Contribution due to (α, n) reactions on oxygen isotopes of oxide fuels is very small compared to spontaneous fission.

4.2.2. Fission gas and helium release

As with UO_2 fuels, discussed in Section 4.1.3, MOX generates fission gases during irradiation, with a bias towards Xe isotopes [52]. Studies to date suggest that the mechanisms of fission gas swelling and release in modern MOX fuels with a relatively homogeneous distribution of Pu within the pellet are qualitatively similar to those operating in UO_2 fuel (which is unsurprising given that MOX is ~ 90% UO_2) and as a function of temperature, the releases are similar [67–71]. However, there are a number of factors which act to increase the fission gas release of MOX fuels compared with UO_2 as a function of burnup.



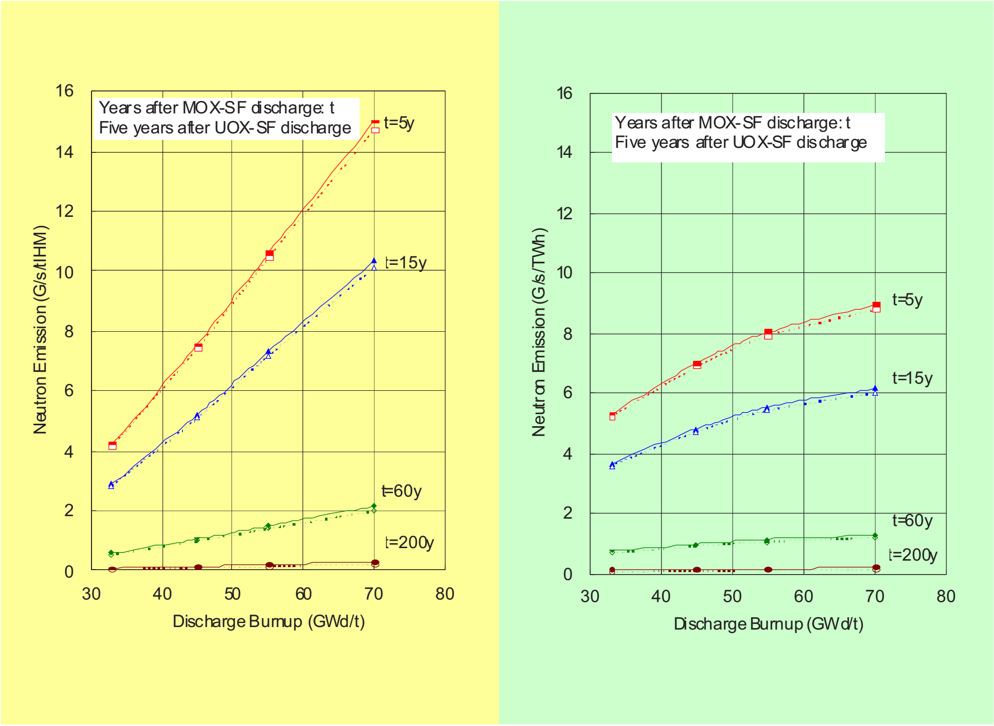
(a) (b)

FIG. 35. Dependence of alpha radioactivity on discharge burnup and cooling year of MOX-SF: (a) measured per t IHM, and (b) per TW·h [24].

- Fissile ^{241}Pu is generated in ^{240}Pu during irradiation. This mechanism operates in UO_2 fuels, but is exaggerated in MOX due to the pre-existence of ^{240}Pu in the fuel at fabrication. This leads to higher late in life powers and a flatter reactivity/burnup curve compared with UO_2 fuels at a time when the fission gas inventory has developed and the temperature of the release threshold has reduced. Note that this mechanism depends on the isotopic quality of the Pu and will vary according to the source fuel reprocessed and its burnup.
- The thermal conductivity of MOX fuel is lower than that of UO_2. For a given power rating, the centerline temperature is higher than that of UO_2 fuel.

Therefore, MOX fuels have a tendency to higher fission gas releases than UO_2 fuels of an equivalent burnup. This is illustrated in Fig. 37 which shows FGR for a range of MOX and UO_2 fuels as a function of burnup.

All irradiated fuel contains α active isotopes which decay to generate helium nuclei embedded within the pellet structure. However, the as-fabricated content of Pu isotopes and ^{241}Am in MOX fuels allows for more rapid formation of ^{242}Cm, which decays with a half-life of 163 days and is the main source of helium generation relevant to irradiation and initial storage. As with the fission gas release, this effect depends on the isotopic quality of the Pu. The distribution of this helium within the fuel rod is less clear. Typically some of the initial He fill gas is soluble in the fuel pellets and at lower burnups there may be less helium in the free space of the rods than that added during manufacture. It is known that large helium releases can occur during irradiation of BWR fuel rods with a low initial fill gas pressure. On the other hand, PWR rods, with relatively high initial pressures demonstrate little if any release of helium gas up to moderate burnups, ~40 GW·d/t U [71, 72]. Figure 38 shows the results of measurements of the helium content in the free space of MOX rods, relative to the initial fill gas content as a function of burnup. The evolution of helium generation and release as MOX fuels progress towards higher burnups is a subject of current research.



(a) (b)

FIG. 36. Dependence of neutron emission rate (Giga neutrons/s) on discharge burnup and MOX-SF cooling year: (a) measured per t IHM, and (b) per TW·h. Solid lines and broken lines correspond to total and spontaneous fission neutrons, respectively. This corresponds to cases reported in [23] but the figure itself is found in [24].

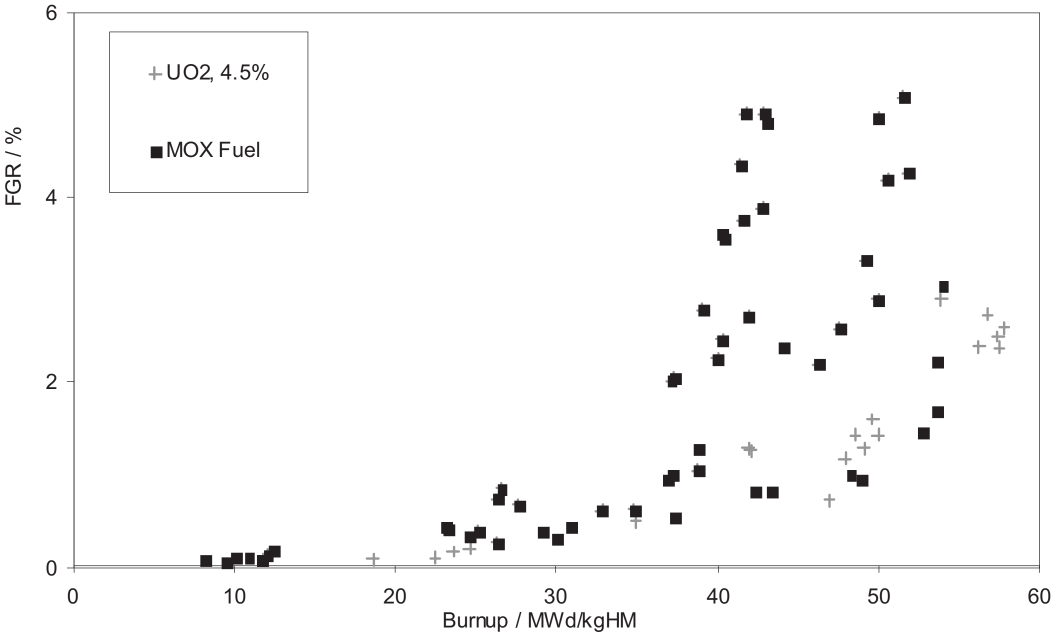


FIG. 37. FGR vs burnup for MOX and UO_2 fuel.

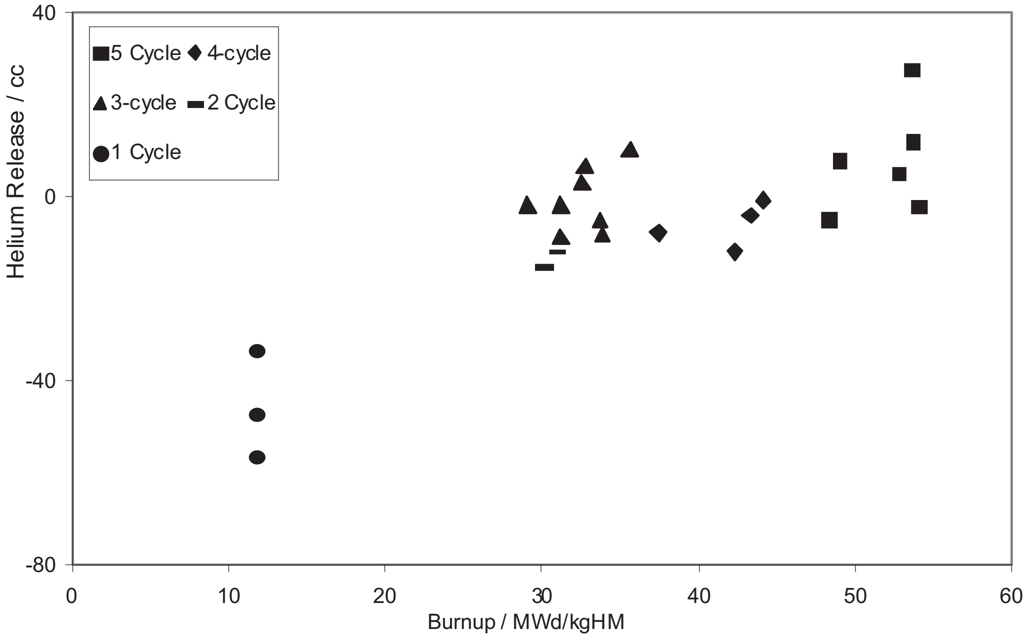


FIG. 38. Helium release from PWR MOX fuel as a function of burnup.

Due to the increased fission gas release and the potential for additional contributions from helium, the end of life pressures inside MOX rods tend to be higher than those of UO_2 fuels at a given burnup.

4.2.3. Comparison of high BU UOX and spent MOX microstructure

As discussed in Section 3.1.2, a MOX fuel pellet is primarily UO_2 and therefore exhibits many of the features described in Section 4.1.3. The main difference is the distribution of plutonium within the pellet structure and the presence of Pu-rich islands which result from mixing of UO_2 and PuO_2 feed powders during manufacture [4].

As Pu is the fissile element, these islands result in an uneven burnup distribution within the pellet and local concentrations of fission products, including gases and minor actinide species. Where the local temperature is too low to allow significant migration of fission products, a porous microstructure, similar to that of the 'Rim Effect' develops in the Pu-rich islands. These 'high burnup' islands can clearly be seen in optical micrographs and have an appearance similar to that shown in Fig. 20 ('Rim of H B Robinson Fuel') [70, 71, 73]. The wider significance of the uneven burnup distribution is less clear. As previously noted, the increase in fission gas release from MOX fuel as a function of burnup can largely be explained by the factors of power rating and thermal conductivity at the burnups experienced to date.

MOX fuels also have a tendency to form insoluble microscopic fission product precipitates rich in five metal particles (Mo, Ru, Rh, Pd and Tc) due to differences in the local chemistry within the fuel pellet. These may be observed in optical micrographs [70, 71, 73]. These particles appear to have little impact on reactor fuel performance, but can be of significance during fuel reprocessing [74].

4.2.4. Cladding and assembly performance of MOX fuel

MOX fuels utilize the same cladding materials and essentially the same mechanical rod and assembly designs as UO_2 fuels, therefore, the behaviour of rod cladding and assembly components is effectively the same as that of UO_2 fuel described earlier [70, 74]. A notable exception is the oxidation of the clad bore (clad inner surface).

The oxidation states of uranium favour the retention of free oxygen within the UO_2 fuel pellet. In contrast, Pu may exist in lower oxidation states than PuO_2, therefore free oxygen, released by the fission process is able to migrate to the inside surface of the fuel cladding once the pellets and cladding come into contact. Typical bore oxide thicknesses in MOX fuel at today's burnups are in the range 10–15 μm and therefore significantly less than the oxide thickness formed on the outer surface of the rod due to corrosion in the coolant [70]. They are also not subject to the same concerns in terms of spalling and resultant local hydride accumulation.

5. ANALYSIS OF EFFECTS OF HBU UOX AND MOX ON SPENT FUEL MANAGEMENT

5.1. HIGH BURNUP UOX

An increased burnup, although mainly motivated by improved fuel utilization, achieves the reduction of the volume of spent fuel and thereby the cost for spent fuel management. The former is the direct result of the fact that the volume of spent fuel is inversely proportional to the discharge burnup of fuels. However, extended burnup implies the generation of more fission products because of increased fission processes and more transuranic isotopes through accompanying neutron capture processes. Therefore a larger amount of radioactivity would be confined within the smaller volume somewhat off-setting the advantages derived by smaller volumes. This may or may not have an economic advantage. An example is the instance for those countries where the waste management fee is charged on the amount of electricity production, and not on the amount of waste generated, in which case the saving of reduced waste production does not serve as the economic incentive for higher burnup.

Fuel is designed for optimum performance in the reactor where the utilities produce revenue, and to minimize, even eliminate any fuel rod failures. Therefore the characteristics of the high burnup fuel as it is taken out of the reactor are a given when analyzing the effect on the back end of the fuel cycle by using high burnup fuel. The characteristics of the high burnup fuel can affect the mechanisms by which the fuel can breach, and the directly effect the ability of the fuel to meet the regulatory requirements.

The regulatory and operational concerns for the back end of the fuel cycle include: criticality, heat rejection, radiation shielding, containment/confinement, retrievability, and operations/construction. These concerns may differ in different parts of the backend of the fuel cycle as shown in Table 5.

In order to analyze the effects of high burnup fuel in these areas, the materials behaviour, especially fuel cladding breach, hydrogen generation and materials compatibility, need to be evaluated under both normal, and accident conditions. While the characteristics of the spent fuel are fixed by its design and reactor operations, the materials behaviour in the back end of the fuel cycle can be modified by changing the conditions such as cover gas, temperature, handling stress etc.

Due to relatively low temperatures and short insertion times of CANDU fuel, neutron fluence-induced effects most likely do not have to be considered. Material effects, such as corrosion or hydrogen uptake also are unlikely. Furthermore, a maximum burnup of 12 GW·d/t U reached under those less demanding conditions should not result in elevated amounts of fission gas release. Therefore, high burnup material issues of LWR fuel largely cover HWR fuel.

The following section will present the analysis of the materials behaviour during each phase of the back end of the fuel cycle and the impact expected in meeting each of the regulatory concerns.

TABLE 5. WHERE HIGH BURNUP AFFECTS THE BACK END FUEL CYCLE

	Wet storage	Dry storage	Transportation	Disposal	Reprocessing
Criticality		L	L	L	L
Heat rejection	L,	L	L	L	L
Shielding	L,H	L, H	L,H	L,H	L,H
Containment/confinement	L, H	L,H	L,H	L,H	L,H
Retrievability	L, H	L,H	L,H	L,H	L,H
Operations/construction		L,H	L,H		L,H

L = LWR fuel, H = HWR fuel.

5.1.1. Wet storage

5.1.1.1. Fuels behaviour

Cladding corrosion is the factor of most interest for primary barrier or containment purposes in wet storage. However, retention of fuel assembly structure integrity is the more important factor when retrieval is taken into consideration.

Reduction in clad thickness due to corrosion and mechanisms that will induce localized attack/accelerated corrosion are the main concerns during wet storage. High burnup LWR cladding usually exits the reactor with an oxide coating up to 100 microns and anywhere from 50 to 700 wppm (parts permillion by weight) hydrogen, depending on the cladding type, in the outer environs of the cladding. Lower burnup cladding usually has below 100 wppm hydrogen. This hydrogen is in the form of circumferential hydrides. There may be a layer of crud on top of the oxide layer. The oxide layer formed during 3+ years of reactor operations at ~350ºC. The oxidation decreases as the temperature is dropped and at wet pool operating temperatures of ~40ºC little additional oxide is expected to be formed over a 100 year period, and little additional hydrogen due to the corrosion process will be introduced into the cladding. Therefore, due to the low temperatures involved in wet storage, cladding corrosion is not likely to result in fuel integrity issues for at least 100 years of wet storage.

In order to redistribute within the cladding, the hydrogen must go back into solution, or be placed under a large thermal gradient. At 40ºC the solubility limit of hydrogen in Zircaloy is in the 1 or 2 wppm level, therefore redistribution can only occur under a large thermal gradient. Hydrogen migrates down the thermal gradient, i.e. it moves from hot to cold areas. The cold area is on the outer wall of the cladding where the hydrogen is already located, so no migration is expected. For wet storage, however, low temperatures and small thermal gradients across the cladding are expected.

In addition to the cladding behaviour, sloughing of crud from the outer fuel rod surface is a concern. Crud is formed due to the disposition of primary system corrosion products on the cladding during reactor operations. Its characteristics vary for PWR and BWR crud, but there is no indication that the thickness of the crud has any burnup dependence. Therefore the crud should not be an issue in wet storage for high burnup fuel from a particular reactor if it was not a concern for low burnup fuel from that same reactor.

Cladding failure can occur due to excessive hoop stress in the rod due to internal pressurization. The fission gas released from the fuel pellets to the fuel cladding gap will increase as much as ten-fold for high burnup fuel over lower burnup fuel. In addition, the stress will increase in higher burnup fuel due to a thinner cladding due to higher oxidation wastage that occurred in reactor. The much lower temperatures in a pool compared to in-reactor will drop the stress considerably and mitigate stress-rupture failure.

5.1.1.2. Environmental limitations

The principal environmental burden during wet storage is due to the number of failed rods and the resultant leaching of activity. As shown in Section 4.1.4, rod failure rates have generally been reduced in recent years, despite increases in burnup, and there is no reason currently to expect significant increases in activity release from higher burnup fuels.

Another potential environmental burden is crud and release of loosely bound crud to the pond or containment systems. As discussed in Section 4.1.2.1, while crud formation is not directly related to higher burnups, some modern fuel cycle strategies may encourage crud formation and any crud generated in-reactor will impact on fuel management during storage and disposal/reprocessing.

5.1.1.3. Regulatory and safety concerns

Designs for spent fuel pools for higher burnup UOX and MOX fuel should take into account the impact of various fuel design and core management changes on criticality control, decay heat rejection, confinement and containment, retrievability, and shielding. Each of these potential issues is discussed below.

Criticality

If the initial enrichment of fuel assemblies of a full core off-load exceeds 5%, the requirement for criticality safety of the full-core reserve may have to be reanalyzed with the possible need for reconfiguration or use of neutron absorber, or even additional storage space to assure sub criticality. On the other hand, the fission product and actinide content of the fuel will be higher at the higher burnup, so if burnup credit is allowed, then criticality in the pool may not be a concern.

Heat rejection

As spent fuel increases in burnup, or as MOX spent fuel is introduced into the pools, the additional decay heat and potentially larger number of damaged fuel assemblies will need to be factored into maximum pool capacity estimates.

Confinement/containment

As indicated above, no additional cladding breaches are expected to occur in the pool under normal storage conditions. Therefore two cases are postulated. The first is when no damaged fuel is put in the pool, unless it is hermetically encapsulated, and the second is if fuel with breached cladding can be placed directly in the pool. In both cases crud from undamaged rods has been shown in the past to slough off the cladding that must be handled by the water cleanup system. Since there is no evidence that high burnup fuel has a higher crud level than lower burnup fuel, this phenomenon should not place an increased burden on the pool.

Actinides and fission products may be emitted from damaged fuel. In the first case, when all damaged fuel is hermetically sealed, and there is no mechanism for future rod breaches, no fission product or actinide load will be available to disperse into the pool water other than tramp elements trapped in the crud; so damaged, high burnup fuel puts no additional burden on the systems. In the second case, those rods that have breaches can release both fission products, and actinides from the rod due to dissolution of the spent fuel pellet by the pool water. Repository studies have shown that for lower burnup fuel, the pellet dissolution is a congruent process; i.e. the release of elements contained within a pellet is proportional to the amount of material dissolved. This has not been confirmed for higher burnup fuel containing a rim region that is rich in plutonium, and has a very small grain size. Since dissolution is proportional to surface area and the surface area is much larger in the rim region, one would expect for some period of time that dissolution in higher burnup fuel would be faster than lower burnup fuel. Since higher burnup fuel has a higher actinide, fission product, and fission gas content than lower burnup fuel, even if the dissolution rate was the same, one would expect more radionuclides to be released into the pool water. If there is also a high dissolution rate this will be exacerbated. Fortunately the dissolution rate of LWR fuel is quite low so that though the higher burnup fuel generates more radioisotopes than the lower burnup fuel it may not be a problem.

In the event of an accident such as a loss of pool water or rod breakage due to a handling accident, there would be more radionuclides available for release from high burnup fuel than lower burnup fuel.

Retrievability

Since there is no thermal cycling that will allow hydride reorientation, data [36] show that the mechanical properties of the cladding should not decrease at room temperature with the higher hydrogen content. Therefore, one would expect that the retrievability of the fuel, which is a handling issue, would not be affected by the burnup of the fuel.

Shielding

The source term for dose will increase with the higher burnup due to the higher radionuclide content. Dose can come directly from the fuel that is shielded by about 14 ft of water, from radionuclides that are released into the pool water due to crud sloughing and pellet dissolution; and fission gas released from the pellet due to pellet dissolution. This latter source will diffuse through the water and be directly available to the work space above the pool. Radiation levels in the pool area may increase. Analyses will be required to ensure that dose rates are kept within allowable limits, or modifications to the pool area or administrative controls established to preclude excessive exposure to plant personnel.

Facilities operation

Reactors are required to be able to unload the complete core into the pool if necessary. This is full core reserve. Fuel that operated at a higher burnup would require fewer assemblies to produce the same level of power so fewer assemblies would have to be accommodated in the pool. Space in the racks for a full core reserve could be maintained longer. On the other hand, due to their increased heat load, higher burnup fuel assemblies would need to be left in the pool for a longer time before they could be transferred to dry storage (see Section 4.1.1.5). This would occupy slots for a longer time than lower burnup fuel. Analysis would have to be conducted to determine which effect dominates, and to determine if full core reserve in an existing pool is jeopardized by increasing the fuel burnup. These same effects would need to be considered when designing a new pool for high burnup fuel.

Re-racking is a simple method that has been extensively used at many nuclear power stations requiring storage capacity extension. The sizes of the slots in a re-rack are the same as the original size, just closer together. Since the size of the assembly is independent of the burnup, the size of the re-rack slots is not a consideration. Most re-racking has been done using poison plates based on criticality considerations for lower burnup fuel. It would have to be determined if these same racks could be used with the higher burnup fuel, or if some administrative plan would have to be used to mix high and low burnup fuel, or new racks designed for high burnup fuel would have to be installed. High burnup and MOX fuel may be able to take burnup credit, thus allowing racks designed for lower burnup fuel to be used for higher burnup fuel.

A main consideration is additional maintenance costs as the pool is loaded with high burnup fuel. If breached fuel is allowed to be stored directly in the pool, there may be an additional release of fuel particulate to the water and fission gas to the above pool atmosphere. To maintain radiation levels at acceptable levels, additional water and air purification systems may need to be installed.

5.1.2. Dry storage

Some Member States have begun loading high burnup fuel into dry storage systems where it has resided for a number of years. Dry storage power reactor fuel to 62 GW·d/t U has become a mature technology and its safety has been satisfactorily demonstrated. The additional heat load and radioisotope inventory of high burnup fuel do create additional burdens on the storage systems. The fuel must be cooled for a longer period of time in order to meet the system heat rejection capabilities and keep the fuel within storage temperature limits. This puts additional burden on the pool capacity. Shielding must be increased or cask capacity lowered to meet the surface dose limits. Tighter sealing requirements may be necessary due to the higher source term in the event of an accident. If MOX fuel is to be stored, the additional stress on the cladding that might result in rod cladding breach, due to the build up of decay helium, need be considered.

5.1.2.1. Fuels behaviour

In most cases, PWR/BWR fuel will be subjected to ‘normal’ conditions during storage. The fuel initially will be dried to remove the majority of the cask moisture, then covered with either a He or N_2 inert atmosphere. For low to moderate burnup spent UOX fuels, the temperature, governed by the decay heat and loading pattern, will be well below the design limit of the cask, for instance, no higher than 400°C and will decrease with time. The radiation field will be $<10^5$ R/hr (γ) with a neutron flux of $\sim10^4$ to 10^6 n/cm^2·s, which is substantially less than in-reactor fluxes. Higher burnup spent UOX or spent MOX will have higher decay heat and radiation fields challenging the cask design with regard to shielding and heat removal capacity.

In rare instances during the storage period, ‘off-normal’ events might occur due to a seal leak, or an over-temperature excursion due to a blocked vent. Air might get into the cask. If per chance, the cask was mistakenly miss-filled with air, the condition is also off-normal.

Even rarer are accidents such as cask drop, cask tip-over, airflow blockage, and fire that could lead to fuel damage, or leakage of the confinement boundary. Some natural phenomena events may include flood, tornado, earthquake, burial under debris, lighting, tsunami, and hurricane.

Under all the above situations, there are mechanisms that may change the condition of the fuel rod, increase the radiation source term in case of a cladding breach, or compromise the ability to meet the requirements and functions stated earlier in this report. If off-normal/accident conditions exist, the storage conditions of the spent nuclear fuel (SNF) could change due to either air or water ingress, or excessive temperature excursion. In addition, the possibility of fuel damage by mechanical trauma also may occur. If there is air ingress, oxidation of the Zircaloy cladding and fuel may occur. These potential changes and mechanisms are shown in Table 6. The mechanisms that may lead to these changes are discussed briefly. These mechanisms must be analyzed to determine if high burnup fuel sufficiently degrade to the point where the regulatory concerns are not met.

Inert medium, such as helium, are employed for LWR fuel storage. Hence, the oxidation of UO_2 is not expected to cause problems as long as oxygen in the storage medium is restricted to negligible levels.

Clad creep and hydride reorientation during storage and vacuum drying that might change the mechanical properties of the cladding and effect subsequent cladding behaviour during transportation are considered to be the most critical phenomena for the LWR fuel storage, especially for higher burnup fuel and MOX. Creep in storage is a self-limiting mechanism due to the expanding internal rod volume and dropping temperatures reducing the hoop stress. In MOX fuel the effect of He production, which increases the rod stress and the creep strain, will depend on both the storage temperatures and duration, and would have to be analyzed to determine if this was an issue. Fuel integrity under normal storage conditions can be maintained if the creep strain of the Zircaloy clad is restricted to about 1%. To limit the effect of hydrogen reorientation, the hydrogen concentration is restricted to about <250 ppm (parts permillion) in the USA, whereas in Germany the circumferential cladding stress is restricted to 120 MPa (million Pa). Based on this, the peak-clad temperature of clad during storage is limited to about 350–400°C.

Hydride reorientation occurs when the circumferential hydrides go into solution during the heatup for drying, and then re-precipitates under an excessive hoop stress. At lower burnups, there are insufficient hydrides available to reorient and change the mechanical properties and due to the lower fission gas release during irradiation the hoop stress will be too low to cause reorientation of those hydrides that do go into solution. At the higher burnups, the stress is higher due to the higher fission gas release in reactor, and more hydrogen, depending on the hydrogen solubility and the cladding type, is available for reorientation. It is unclear at this time whether the stress threshold is low enough to be exceeded at high burnup, how much reorientation is needed to significantly change the mechanical properties, and how the cooling rate, and total hydride content will affect the length of the precipitated hydrides. It is currently thought that a temperature of 400ºC might limit the reorientation to an acceptable level but

TABLE 6. POTENTIAL LWR ROD CHANGES AND MECHANISMS DURING DRY STORAGE

Condition	Fuel	Cladding
Normal	Increased stress from fission gases release (FGR), He generation*	Breach (creep, hydride reorientation)
		Change in properties (annealing, hydrogen migration)
		Crud contamination (spallation)
Off-normal	Particle size reduction (oxidation)	Wall thinning (oxidation)
	Gas release (oxidation)	Excessive creep
	Decreased cover gas conductivity	Mechanical property change (annealing)
Accident		
Impact	Fuel fracture	Breach
	Gas release (oxidation)	Oxidation (corrosion)
	Particulate formation (oxidation)	Crud contamination (spallation)
Fire	Stress increase due to thermal expansion; strength decrease	Breach (stress rupture)
	Gas release and particulate formation due to oxidation	Change in mechanical properties (annealing, hydride reorientation)

* for MOX fuel only

that has not been confirmed yet. Since it does not affect the behaviour of the fuel in storage, storage of high burnup fuel up to 400ºC is allowed in the USA. This might be an issue in countries that require transportability (Section 5.1.3) prior to storage.

Investigations performed to date [75, 76] identified the components of CANDU fuel that may be affected during long term storage (50 years or more). It is currently believed that the welding of the endplates to the fuel rod end caps may be, in long term affected by hydrogen effects due to stresses on those components. Nevertheless, the investigation of other potential mechanisms continues.

The basis for the derivation of temperature limit in an air atmosphere for CANDU fuel is the oxidation of UO_2 in a defective fuel. The UO_2 oxidation rate is restricted to extremely low levels to ensure safe storage by limiting the maximum cladding temperature to 150°C. The presence of moisture in the storage medium can result in the formation of very low density hydrated uranium oxides that has about 160% lower density than UO_2. Hence, moisture is almost completely removed from the storage internals and dry air of about (–20°C to –40°C) dew point is used as storage medium. The storage container is sealed to prevent ingress of outside air and moisture. The presence of actinides and fission products in MOX and high burnup fuel should mitigate both the effects.

5.1.2.2. Regulatory and safety concerns

Designs for spent fuel dry storage systems for higher burnup UOX and MOX fuel should take into account the impact of various fuel design and core management changes on criticality control, decay heat rejection, confinement and containment, retrievability, and shielding. Each of these potential issues is discussed below.

Criticality

All systems must be configured to have a $K_{eff} < 0.95$. Under this current constraint, criticality is not a concern during dry storage accidents in the USA, since it is assumed that there is no way for a moderator to enter the storage cask, design basis accidents are not severe enough to reconfigure the fuel, and the fuel is assumed to be unburned. It would be a lesser concern at high burnup where the fuel is less reactive due to the presence of more fission products and actinides.

Heat removal

In many storage systems, the limiting parameter is the peak temperature of the fuel in the basket that contains the stored assemblies. Higher decay heat that leads to higher basket temperatures and higher fuel rod temperatures must be controlled by limiting the combined heat generation of all of the assemblies in the dry storage cask. This can be done by an optimized loading of the cask. Optimized cask loading schemes take into account the available fuel assemblies, with specific neutron and gamma emissions, as well as the results from statistical performance calculations. Therefore systems are designed to handle a maximum heat load. High burnup fuel that has been cooled for the same duration in the pool as lower burnup fuel before dry storage will have a higher heat load.

In order to meet the heat load limits for a particular system, either:

- The high burnup and lower burnup fuel need to be administratively mixed. For example, after reactor in-service, CANDU fuel typically spends ten years in storage pools under controlled chemistry conditions. Afterwards, the fuel is loaded into modules that are placed in dry storage containers. During this process operations staff may consider mixing higher burnup fuel with the low burnup fuel evenly in the modules and in the dry storage containers to avoid local concentrations of higher heat from high burnup bundles, Refs [75, 76];
- Slots may need to be left unfilled, if all high burnup fuel is loaded;
- The system may be designed for better heat transfer. For new cask development, materials in the baskets can be optimized with regard to heat conduction. In addition, heat conducting material can be used to conduct the heat to the cask surface where fins can provide a better cooling;

- The high burnup fuel may need to be left to cool for a longer period of time in the pool before transfer to dry storage, or the discharges of higher burnup MOX spent nuclear fuel will require longer buffer storage in pools somewhat off-setting cooling requirements in the casks. The mixed loading of UOX and MOX fuel assemblies in the casks is an alternative solution;
- Data may be developed to show that the maximum allowable fuel temperature can be raised without causing cladding breach.

Confinement/containment

The ability to meet confinement/containment regulations (A_1 and A_2 values) is a function of the allowable gas leakage rate from the container, operative deposition mechanisms for particulate within the cask, and both the volume and size distribution of particulate and gas released from the fuel rods. The maximum gas leakage rates are set by regulation and are independent of the burnup of the container contents. Likewise the deposition mechanisms for particulate are independent of burnup.

The releasable radioactive material can be in the form of crud, gas, volatiles and particulate. As indicated in a previous section there is no information indicating that the crud level increases with burnup, hence crud release should be independent of burnup.

The release of gas and volatiles is only a concern during an accident[10] or drying since both of these constituents will have been released from breached rods prior to cask loading. The volume of the gas and volatiles that are releasable should track each other except during an accident where the fuel temperature rises above 600ºC, and RuO_4 becomes volatile. The amount of gas and volatiles available for release will depend on (a) the number of rods that fracture during and accident; (b) the amount of fission gas released to the gap during reactor operation; and (c) the amount of fuel that fractures during an accident and releases gas trapped on the grain boundaries.

The integrity of the rod cladding during an accident will depend on the ductility of the cladding. Low burnup fuel has a long record of storage and transport without rod failure indicating it has good ductility. This is supported by low hydrogen content that remains in the circumferential orientation during storage. Higher burnup cladding may have, as discussed above, significantly more hydrogen that may reorient to a radial direction and may lower the ductility of the cladding making it more susceptible to breach during a cask tip over or drop. There is considerable controversy over the effect of the hydrides in the cladding, so at this time it is impossible to say with certainty that higher burnup rods would or would not breach during a storage accident. The effects of reorientated hydrides on the creep behavior of the cladding during storage have not been resolved.

Once a rod breaches, the gases and volatiles in the gap are immediately available for release. The amount of gas and volatiles in high burnup fuel that can be released from a breached rod to the container and subsequently to the atmosphere can be as much as a factor of ten greater than in lower burnup fuel. In addition, gas can be released from any pellets that fracture during an accident. Once again there is considerable controversy on the question whether the presence of the small grained rim in high burnup fuel fractures differently than the larger grains in lower burnup fuel. If they fracture in the same manner, the additional gas release from the higher burnup fuel should again be about a factor of ten higher than the lower burnup fuel. If there is more fracture in the rim region, there could be considerably more gas and volatiles released from the rod to the container from this mechanism.

Fuel particulate released to the cask from the rod comes from two sources: (a) fuel fractured while in reactor due to thermal gradients, and (b) fuel fractured during the accident event. Fuel fractured in reactor has been measured for lower burnup fuel [77]. No measurements have been made for higher burnup fuel, but it is probably similar to lower burnup fuel since thermal fracturing takes place during early irradiation cycles when the rim has not formed. As indicated above, fracture of the fuel and the resultant particle size distribution is still an open question. Should the rim fracture excessively it could skew the size distribution towards the respirable range since the rim consists of submicron size grains.

Under normal conditions burnup should not affect containment issues. During an accident, burnup should not affect crud releases. If any rods breach, the potential gas and volatile release per rod could be an order of magnitude

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[10] Releases during vacuum drying are not a concern, since volatile fission products will be removed during the drying prior cask draining process. Crud also is removed during this draining process. Since the vacuum drying process is essentially a distillation, fuel particulates volatiles in the remaining water will remain in the cask.

higher for high burnup rods. A greater number of high burnup rods might breach. Depending on the fracture properties of the cladding and the pellet rim region more particulate and more respirable particulate may be releasable from high burnup rods. This is still an open question. In the USA, storage of high burnup fuel is allowed with no significant containment issues.

Retrievability

Under normal and off-normal conditions, retrievability of fuel, other than possibly more shielding being provided when working with high burnup fuel, should not be an issue. After an accident, retrievability might be operationally more complicated for high burnup fuel if high burnup cladding fractures and lower burnup cladding does not. As explained above, this is still an open question.

Shielding

Higher radiation levels can be addressed by increasing the shielding in the storage system, but basket temperatures must be controlled by limiting the combined heat generation of all of the assemblies in the container.

5.1.2.3. Facilities operation

Facility operations should not be affected by the burnup of the fuel from the handling aspect. There may be procedural changes to ensure that the exposure of personnel is kept at minimum.

5.1.3. Transportation

5.1.3.1. Fuels behaviour

Since dry transportation is nothing more than moving dry storage, the degradation mechanisms evaluated above for dry storage also apply to dry transportation. The impulses that the fuel receives during transportation may be higher than in dry storage.

5.1.3.2. Potential releases to the environment

For transportation, there is no routine discharge to the environment.

5.1.3.3. Regulatory and safety concerns

Criticality

In the USA, a container must remain subcritical when the container is fully moderated and the fuel has reconfigured to the most reactive, physically possible condition. Usually the fuel is considered fresh when calculating criticality but the issue would be a lesser concern at high burnup where the fuel is less reactive due to the presence of more fission products and actinides. Currently though, different countries allow burnup credit only to varying degrees. Some only allow for actinides, some for actinides and fission products, and others not at all.

Two other ways of assuring sub-criticality are moderator exclusion building the container so that under the most limiting accident the container will not allow the ingress of moderator, and addition of poison materials. The welds to close the container are independent of the contents so moderator exclusion should be independent of the fuel burnup. Less neutron poison should be required for higher burnup fuel due to the lower reactivity.

A third way to assure sub-criticality is to assure that the fuel configuration, assumed during the original as loaded criticality calculations, does not change during a transport accident. In the USA, the most limiting transport accident is the regulatory 9 meter side drop. As explained above, the ductility of the high burnup cladding after vacuum drying is questionable. Therefore, the ability to assure the rods will not breach and fuel distribution change is in question. There is no evidence that lower burnup fuel changes configuration during transport since the cladding maintains its ductility. Higher burnup may have criticality issues if an accident occurs during transport.

Heat removal

See section on heat removal during dry storage.

Confinement/containment

All the arguments concerning containment and confinement for dry storage accidents will also hold for transport accident except to a larger degree, since the potential impact during a transport accident are greater.

In addition, due to vibration and falls from a truck or train, which are not considered accidents but potentially normally expected events, fuel may fracture. Higher burnup fuel would have a greater source term under normal conditions than lower burnup fuel.

Retrievability

Mechanical integrity of the fuel assembly structure must be maintained during handling and transport operations. Any method used to retrieve lower burnup fuel from a cask would be suitable for high burnup fuel. More shielding might be required. Although transport casks are designed to accommodate fuel assemblies with up to 100% defected pins, the individual transport authorities, the utilities and the receiving facilities are unlikely to approve such shipments on dose grounds when the cask must be unloaded. This dose will be higher for high burnup fuel. In the majority of cases, known leaking fuel assemblies are required to be conditioned prior to shipment, or there will be a limit set by the appropriate transport authority on an acceptable level of activity within the cask body.

Due to the uncertainty in the ductility of the high burnup cladding as explained earlier, more high burnup rods than low burnup rods may fracture under any accident

Shielding

Higher radiation levels for the high burnup fuel assemblies have to be addressed by increasing the shielding in the cask system. This can be accomplished in different ways depending on the system. The γ radiation in existing cask designs can be lowered by the loading configuration of the fuel. Fuel assemblies with lower burnup can be loaded in the outer ring, while those with higher burnup can be loaded in the centre taking advantage of self-shielding of radiation within the casks. For new cask designs, the wall thickness can be increased or additional material with a high specific weight can be used like lead or depleted uranium. For higher neutron radiation it is necessary to increase the neutron absorbing or moderating material.

5.1.3.4. Facilities operation

High burnup fuel should have no impact on facilities operations since the assemblies are physically the same and all the handling will be conducted remotely.

5.1.4. Repository

5.1.4.1. Fuels behaviour

As discussed in Section 4.1.1 and 4.2.1, higher burnup spent UOX and MOX fuels have higher radionuclide source terms than lower burnup spent UOX fuel. Assuming direct disposal of spent UOX or MOX, the rate of radionuclide release from the spent fuel following exposure of the spent fuel to groundwater depends on the particular chemical properties of each radioactive element. In a reducing environment typical of proposed repositories below the groundwater table, solubilities of most actinide species, along with many fission products is fairly low. The release rate for solubility limited species is likely to be independent of the amount present, hence independent of burnup or fuel type (UOX versus MOX). Some species such as iodine and selenium are highly soluble. Technetium is also soluble in oxidizing environments. For these more soluble species, the release rate will likely be proportional to the source term and therefore would be released at a greater rate for high burnup UOX and MOX spent fuels. The radionuclides that contribute the most to long term dose to the critical group depend to a great extent on the waste type. For example, long term dose rates from HLW are often dominated by dose contributions from ^{79}Se, ^{126}Sn, and ^{135}Cs; long term dose rates from spent UOX and MOX are often dominated by ^{129}I, ^{99}Tc, and (for oxidizing environments) ^{237}Np/^{229}Th. This is largely due to the fact that iodine and technetium are mostly lost during the reprocessing and HLW glass production stages. Differences in the lifetime of the waste form (spent fuel versus HLW glass) also contribute to differences in peak dose rates, although to a lesser extent than the radionuclide content of each waste form.

5.1.4.2. Regulatory and safety concerns

The primary considerations for meeting applicable disposal regulations and safety requirements are criticality, heat rejection, confinement/containment, retrievability, and shielding. Each of these is briefly discussed below.

Criticality

Factors involved in assessing the probability and consequences of a potential criticality over the very long time frames involved in geologic disposal are the same as those discussed earlier for storage and transportation. The relative amounts of fissile material in spent UOX and MOX fuel for lower and higher burnups has been discussed earlier in this report. The two additional mechanisms governing the possibility of a criticality event during disposal are the removal of the neutron poison material, and collapse of the container internal structure, leading to different fuel geometry. The neutron poison material within the disposal container itself may be leached (dissolved) away by the groundwater. In such a case, it may be possible, that groundwater could fill the degraded container, and provide sufficient moderation to cause a criticality event. Also, collapse of the waste container internals may occur, which can lead to either increased or decreased risk of criticality.

Given the large uncertainties involved with predicting the very long term evolution of the waste container and spent fuel, some countries will use burnup credit arguments to accommodate the potential removal of neutron poison materials. Since the dissolution properties of higher burnup spent UOX and MOX are less well known, it would be more difficult to make burnup credit arguments for disposal. Hence, additional R&D may be required prior to application to high burnup UOX and MOX.

Heat rejection

All repository designs employ various temperature limits on one or more components in the repository. Hence, decay heat management is a critical part of repository design. Since high burnup UOX and MOX generate more decay heat, they will need to be managed differently than lower burnup UOX. Section 4.1.1.5 provided an example of the effect of higher decay heat on spent fuel management.

Confinement/containment

Waste disposal operations involve similar confinement and containment considerations as that for spent fuel storage and transportation. In addition, some disposal operations may involve the transfer of spent fuel from a transportation specific container to a disposal specific container in air. In such a case, the decay heat of higher burnup UOX and MOX, combined with the presence of oxygen, could lead to a higher possibility of cladding degradation, and UO_2 conversion to U_3O_8, which is more easily dispersed due to its small particle size. For such a disposal facility design and operation, calculations would have to be made to determine the risk of spent fuel degradation during the transfer operations in air.

Retrievability and shielding

It is common for there to be specific requirements on the length of time that spent fuel that is emplaced in a disposal facility must be retrievable. Hence, retrieval operations need to factor in the appropriate amount of shielding, and disposal container management to meet this requirement. The introduction of higher burnup UOX and MOX may require the retrieval operations and necessary shielding to be redesigned.

5.1.5. Reprocessing

Existing reprocessing plants are generally designed and licensed for processing of fuels up to a specified maximum burnup and/or enrichment and a minimum fuel cooling time. While fuels to be reprocessed remain within these originally envisaged limits, no significant impact is anticipated from the processing of fuels having a higher burnup than usually encountered. However, in the case of fuel with a burnup higher than originally considered in the design and development of existing plants, the effects of processing such fuel must be carefully considered.

At the Thorp Plant in the UK, the standard safety case is based on PWR fuel with 4% w/o ^{235}U initial enrichment irradiated to 40 GW·d/t U and cooled for a minimum of 5 years. A number of high burnup trials campaigns have been performed under a modified safety case on fuels with peak assembly burnups up to 49.3 GW·d/t U and campaign average burnups of up to 42.6 GW·d/t U with no noticeable negative impact on plant performance. Further trials are planned for the future with assemblies up to 50 GW·d/t U burnup and an increase in the initial enrichment limit to 4.5% w/o ^{235}U.

It is generally the average inventory of a vessel in reprocessing plants that is important rather than the most challenging fuel assembly feed. This allows blending schemes to be employed where fuels outside the current safety case can be mixed with other assemblies or liquors to provide a process feed within the operating envelope. Such schemes are clearly limited to the availability of benign fuel for blending purposes and must also respect contractual arrangements with different customers and the requirements to maintain accurate nuclear material accounts.

Burnup and/or enrichment limit extension activities are underway in a number of reprocessing plants around the world to cater for the increase in discharge burnups from reactors. The impact of burnup extension of BWR fuel beyond 45 GW·d/t U on the fuel cycle in Japan has also been assessed [7, 78]. In so doing, the increase in accumulated activity, decay heat and neutron generation rates were determined and the following list of potential issues raised: increased shielding in line with enhanced gamma and neutron emission rates, increased filtration capacity, increased potential for blockages due to higher levels of insoluble residues and solvent degradation products, increased frequency of solvent regeneration due to enhanced radiolysis, increased disposal capacity of radioactive waste disposal facilities and storage capacity of solidified high level waste.

In France, the reference fuel parameters for the UP2-800 and UP3 plants have been increased from the initial values of 3.25% w/o ^{235}U PWR fuel, irradiated to 33 GW·d/t U, then cooled for 3 years up to initial enrichments of 3.7% w/o ^{235}U and burnups of 45 GW·d/t U, cooled for 4 years [7].

When making safety or environmental justifications for reprocessing higher burnup fuels a number of criteria, described below, must be studied.

5.1.5.1. Fuel handling prior to dissolution

As shown in Fig. 13, the decay heat of high burnup assemblies can be substantially greater than that of lower burnup fuel. The cooling systems in the fuel receipt ponds and those serving the process vessels must be capable of the extra duty imposed, both in normal operation and possible fault conditions.

Some plants employ radiometric instruments to verify the burnup, cooling and enrichment parameters of feed fuel prior to shear. These may need to be recalibrated for the increased neutron and γ emissions from higher burnup fuels.

5.1.5.2. Dissolution

Irradiated fuel tends to dissolve more quickly than unirradiated material under the same conditions due to both physical changes in the fuel (fragmentation through thermal cycling, induced mircroporosity etc.) and chemical changes (some fission products have catalytic effects on the rate of dissolution). However, the changes brought about by current burnup increases are not significant and in practice no changes to fuel dissolution behaviour are apparent.

Increased levels of insoluble fission products will arise increasing the required duty of the clarification system and potentially increasing the frequency of wash-outs of the equipment and vessels involved. The increase in insoluble fission products occurs at a rate slightly greater than linear with respect to burnup due primarily to the increased significance of plutonium fission at higher burnups and the slight shift in the fission spectrum of plutonium relative to that of uranium. This leads to the formation of increased amounts of five metal particles, which together with zirconium and molybdenum oxide phases, constitute the principal active components of insoluble residues. The laboratory scale dissolution behaviour of high burnup UOX fuel irradiated in commercial LWRs has been studied and analysis of the residue from the dissolution process reported in the literature [79, 80].

5.1.5.3. Solvent extraction

Current increases in UO_2 fuel burnups present only a moderately increased challenge to the solvent extraction process in terms of Pu concentration and α decay energy. The fission products and minor actinides represent only a few percent of the total fuel mass and thus relatively large proportional changes have little effect, in absolute mass terms. No significant changes to solvent extraction process chemistry are required, although the increased solvent radiation damage will have a slight effect on the level of solvent degradation products present. Neutron monitors used to ensure the safety of the process during operation may need to be recalibrated to allow for the increased neutron emissions from higher burnup feeds.

Increased feeds of reducing agent to the uranium-plutonium separation contactor may be needed in order to accommodate the higher plutonium concentration. Increased levels of stabilizing agents, e.g. hydrazine, may be required to prevent oxidation of species such as U(IV) and Pu(III). Feed liquors to the solvent extraction stage have undergone significant blending within the head end processes and the challenge presented is closer to that of the campaign average than the highest burnt assembly.

5.1.5.4. Product recovery

The process chemistry of uranium product recovery is essentially unchanged when considering the processing of high burnup UO_2 fuels. Increasing burnup does however lead to increased levels of anthropogenic uranium isotopes as shown in Fig. 10 and this may lead to increased shielding requirements. Assuming that element specific decontamination factors remain constant then it might also be expected that as burnups increase, the low levels of fission product contamination present in reprocessed uranium may increase in line with their increased levels in higher burnup spent fuel, although this has yet to be seen at the burnup levels in current reprocessing. Similar arguments apply to the recovery of plutonium when processing high burnup UO_2 fuels.

The storage of separated Pu also needs to be considered. As shown in Fig. 11, the fractions of ^{238}Pu and ^{241}Pu in spent fuel increases as a function of burnup. Once separated, the heat generation and α activity of Pu increases both due to decay of ^{238}Pu and in-growth of ^{241}Am with consequent impact on radiolysis and heat generation of product cans.

5.1.5.5. Effect on corrosion of process equipment

The reliability of construction materials is a significant issue in the operation of reprocessing plants. Several fission products and transuranic elements having oxidizing characteristics, amongst them Np, Pu, Ru and Pd, are thought to be significant in influencing rates of corrosion, particularly in equipment such as evaporators subject to relatively high temperatures. As fission product and transuranic element concentrations increase with burnup rates of corrosion may be expected to increase and these must be managed appropriately.

5.1.5.6. Waste treatment and environmental discharges

The principal impact of increased burnup on reprocessing waste management comes from the additional amounts of fission products and minor actinides which are generated during irradiation. To a first approximation, the amount (mass) of raw, unprocessed high level waste generated can be considered to scale linearly with increased burnup, the mass of raw waste therefore being higher per ton of fuel processed, but essentially constant as a function of energy production. In addition, it should be noted that gadolinium, included within burnable poison rods or added to the dissolver in the reprocessing plant to protect against criticality faults may also increase due to higher initial fuel enrichments, or changes in fuel management practices. This gadolinium is ultimately routed with the HLW and presents an additional volume of in-active material to be managed.

Providing these raw wastes can be blended with those from lower burnup fuels, there is little impact on the volume of vitrified waste product. However, it should be noted that although the amount of waste scales approximately linearly with burnup, the ^{244}Cm content and neutron activity tends to increase exponentially, as shown in Fig. 15. Therefore, as burnups increase, if the waste cannot be blended, the amount which can be incorporated into each vitrified product container is reduced to remain within the agreed parameter levels for acceptance of HLW in future repositories. To illustrate this point, a typical HLW container will contain maximum of 100 kg of waste oxides (kgWO). A blend of liquid HLW is chosen to match this mass as closely as possible within the acceptance limits to minimise the volume to be stored/transported. Table 7 shows how an increase in fuel burnup for a typical PWR reduces the mass of waste oxide per container, thereby increasing the volume of treated waste per tonne of fuel reprocessed. It has to be noted that there is a limitation on loading of fission product (FP) and activates in a canister for glass waste.

In the future, R&D or changes to fuel cycle practice will be required to improve glass loadings for vitrified HLW to militate against the potential for a large increase in waste volumes from high burnup fuels. However, for now, existing plants have sufficient stocks of blend material to optimize waste loading.

Higher levels of γ, especially neutron shielding may also need to be installed in the waste plants to handle the the increased activity of high burnup fuel.

There are two main fuel related intermediate level waste (ILW) streams from the reprocessing plant.

Hulls and centrifuge cake. The hulls contain the cladding and assembly components that remain after dissolution. The volume of hulls depends on the assembly volume and does not change with burnup, however, the hulls contain a small amount of undissolved fuel and insoluble fission product deposits and clearly, as fuel burnups increase, so does the activity of the hulls. At the burnup levels processed to date, no challenge to agreed ILW parameters has been encountered. The centrifuge cake contains the microscopic suspended solids from the dissolution process, including insoluble fission products (IFP) and fragments of cladding. As shown in [79, 80], the mass of IFPs tend to increase with burnup. The main heat/activity source in this stream is ^{106}Ru, which has a short half-life, ~1 year. Therefore challenges from increased burnup in this process stream can be managed through appropriate short increases in cooling time of solid low level waste (LLW) and plutonium containing material (PCM) arising from reprocessing plants tend to be independent of fuel type or throughput.

Reprocessing plants also have aerial and liquid effluent streams, which are generally subject to some form of regulatory control. As with the product streams discussed above, providing the process decontamination factor (DF) remains constant as the burnup is increased, discharges of active species will generally increase in direct proportion to the burnup increase. The proportions of some species, e.g. ^{85}Kr will tend to change as more fission occurs in Pu isotopes at higher burnups. By scheduling high and lower burnup fuels for processing over the regulatory time period, one year, it is possible to manage these discharges within authorized limits.

It is clear that the amount of solid, liquid and gaseous wastes from reprocessing plants increase per ton of higher burnup fuel processed. However, this needs to be set against the increase in efficiency of the fuel cycle by proceeding to higher burnups, with additional power generation from a fixed amount of fuel. For this reason, waste volumes are often considered per unit of energy production when making decisions as to the potential benefits of increased burnups and spent fuel management strategies.

TABLE 7. CHANG OF VOLUME OF TREATED REPROCESSING WASTE WITH BURNUP OF NUCLEAR FUEL

Burnup (GW·d/t U)	Mass of waste oxides per container (kgWO)	Equivalent mass of fuel reprocessed per container (t U)
25	100	3
40	88.6	1.82
60	49	0.71

5.1.6. Refabricated fuel REPU and MOX

5.1.6.1. REPU

There are several recycling modes of REPU in LWRs. Re-enrichment of REPU is most effective in its utilization and commercially successful, especially in France and Belgium. Blending down of high enriched uranium (HEU) recovered from dismantled nuclear weapons with REPU to make low enriched uranium (LEU) is also reported [81]. This section will discuss only the re-enrichment mode.

As shown in Fig. 10, the fraction of two U isotopes (^{232}U and ^{236}U) which are not included originally in natural U tends to increase with fuel burnup. Therefore, the activity of REPU tends to increase with the burnup of the parent fuel. The α-decaying isotope ^{232}U (half-life: 70 a) also increases during spent fuel (SF) cooling through the α-decay of ^{236}Pu (half-life: 2.85 a) and attains its maximum concentration at about 14 years after discharge. As soon as REPU is recovered from the UOX-SF, ^{232}U decays into ^{228}Th and finally the high intensity (2.61 MeV) γ emitting daughter ^{208}Tl is spawned as the last chain of 4n radioactive decay series of ^{232}Th. The radioactivity of ^{208}Tl becomes maximum at about ten years after separation of REPU.

On the other hand, as the amount of the parasitic neutron absorber ^{236}U increases with burnup, the ^{235}U enrichment of the product must be raised additionally by $\Lambda x_5 = K*x_6$, to meet the same fuel management goals, further concentrating the minor U isotopes. Here, x_5 and x_6 are respective concentration of ^{235}U and ^{236}U in re-enriched fuel. The value of K is approximately around 0.25 [24, 82]. Initial uranium isotopic concentrations of re-enriched REPU are compared with those of enriched natural uranium in [24, 82]. Re-enriched REPU fuel is prepared from the REPU recovered from the UOX-SF of the same burnup shown in Fig. 39.

The practical consequences of this increase are a possible need for increased shielding during manufacture, transport and reactor handling. In addition, as initial enrichments increase to meet higher burnup targets they approach the 5% limit applied in many reactor licenses. Clearly REPU will approach this limit faster due to the presence of ^{236}U isotope as is seen in Fig. 39 [24].

Over enrichment of ^{235}U and presence of ^{236}U in the fresh re-enriched REPU fuel, increases the contents of ^{237}Np and hence ^{238}Pu in the SF more than three times larger than those of enriched natural U fuel. Therefore the decay heat of irradiated re-enriched REPU fuel would become appreciably larger than that of irradiated enriched natural U fuel.

5.1.6.2. MOX fuel

The impact on plutonium recycle of moving to higher UO_2 fuel discharge burnups is that the proportion of ^{239}Pu, the primary fissile isotope in MOX fuel, reduces with increasing burnup of the parent UOX reprocessed,

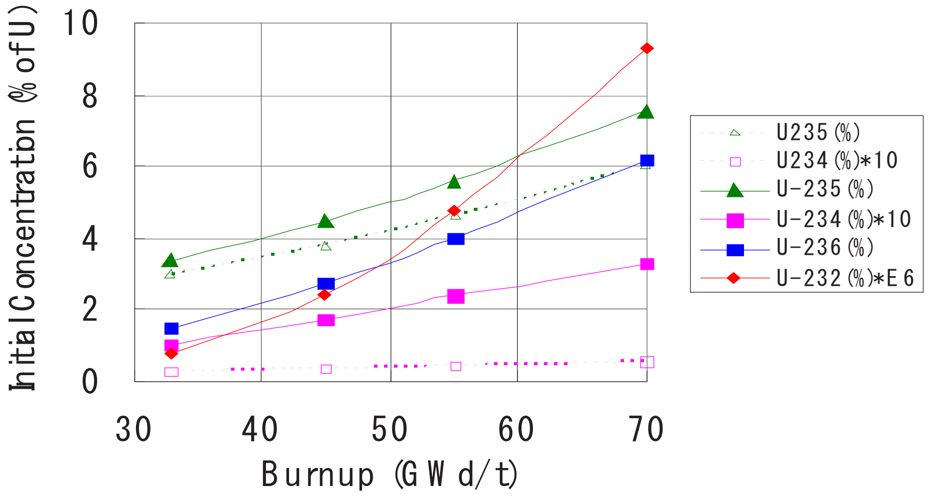


FIG. 39. Uranium vectors of fresh enriched natural uranium (broken lines) and re-enriched REPU (solid lines) for expected discharge burnup [24].

while the fraction of absorbing isotopes increases, as shown in Fig. 10. Hence for a given burnup target the necessary Pu enrichment is increased. In addition, the fraction of ^{241}Pu also increases. While this is fissile, its short half-life means that in general, the amount of ^{241}Am in the fuel also increases shown in Fig. 12. The increase in ^{238}Pu, ^{240}Pu and ^{241}Pu with burnup of the reprocessed fuel, means that the heat and activity of the derived Pu increases, with consequent implications for shielding and cooling requirements for assemblies in storage/transportation prior to irradiation.

Pu isotopes are all fissile to some extent in the fast neutron spectrum. Therefore, the void coefficient of the fuel will depend on the isotopic mix of the plutonium, hence the burnup of the parent fuel. As Pu enrichments are increased to meet higher burnup targets and the fissile fraction drops due to higher reprocessed fuel burnups, the void coefficient of the MOX may tend to be positive. This is clearly unacceptable in LWRs and therefore, imposes limits on the maximum enrichment of the fuel, or the number of times Pu can be recycled. These limits are generally taken to be around 12 wt % Pu and one Pu recycle, but depend on the details of the fuel cycle considered. Blending strategies may be used to manage within these limits. For instance, Pu cans may be mixed within the MOX fabrication plant, or fuel assemblies blended in the reprocessing plant.

5.1.6.3. Comparison of characteristics of re-enriched REPU and MOX fuels with enriched natural uranium fuel

Figure 40 shows radioactivity of SF of enriched natural uranium (UOX) enriched REPU (ERU) and MOX fuels at five years after SF discharge. The radioactivity of ERU-SF is slightly larger than that of UOX-SF, but still much smaller than that of MOX-SF. The effect of higher burnup is clearly seen for each type of spent fuels. Almost similar relationship is seen also in the decay heat shown in Fig. 41.

Neutron emission of the ERU-SF is smaller than UOX-SF, as is seen in Fig. 42 because of the smaller generation of americium and curium isotopes compared with UOX-SF.

5.2. DIFFERENCES BETWEEN THE EFFECTS OF HIGH BURNUP UOX AND MOX ON SPENT FUEL MANAGEMENT

5.2.1. Spent MOX storage

Due to the mechanical designs of MOX elements being identical to those of UOX fuel, MOX may be stored in the same facilities and often alongside UOX assemblies. The primary challenges from spent MOX relative to

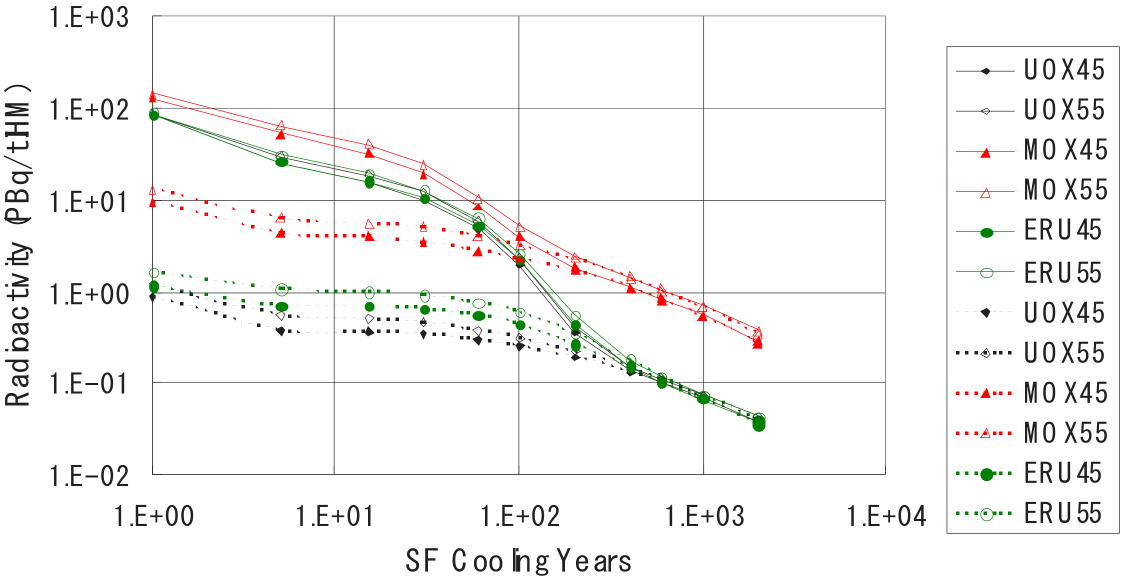


FIG. 40. Radioactivity (solid line) and α-radioactivity (broken line) of SF of UOX, MOX and ERU at five years after SF discharge [24].

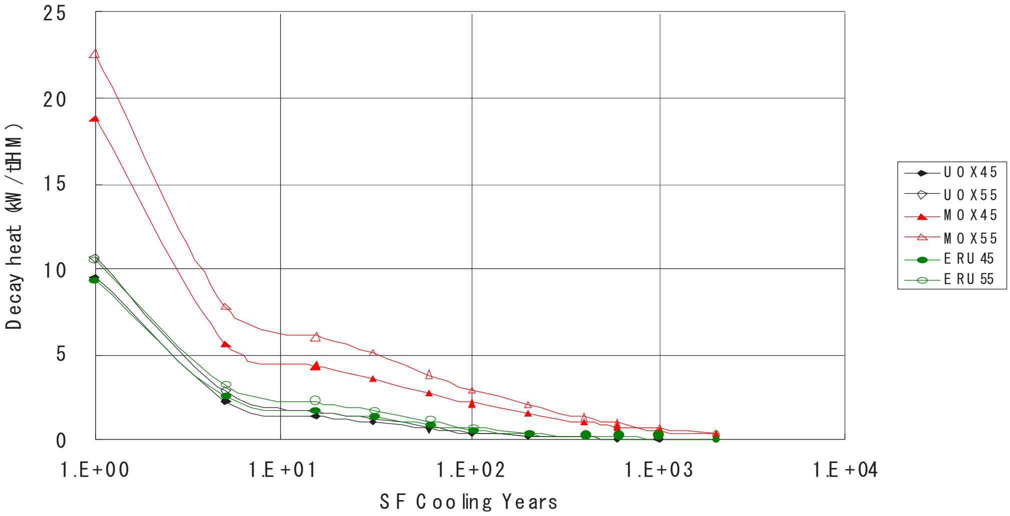


FIG. 41. Decay heat of SF of UOX, MOX and ERU at five years after SF discharge [24].

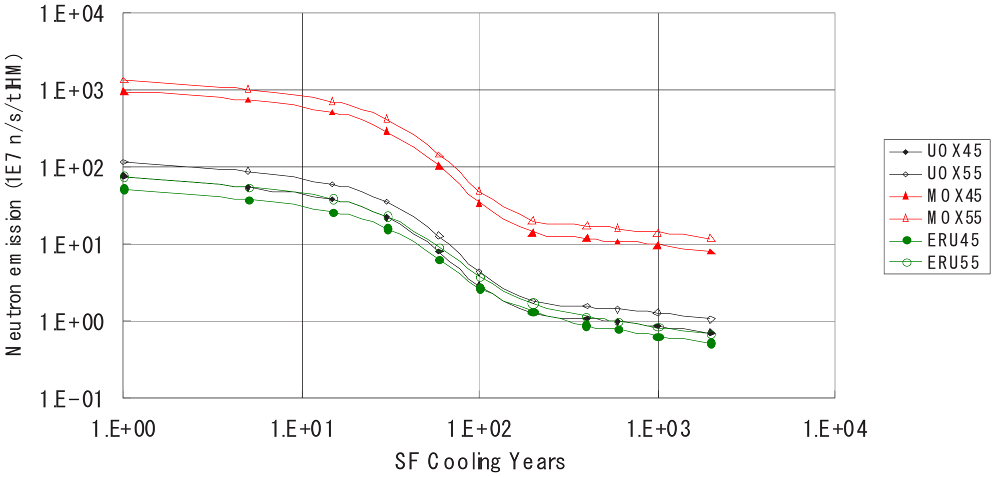


FIG. 42. Neutron emission of SF of UOX, MOX and ERU at five years after SF discharge [24].

UOX fuels are decay heat and neutron activity, due to minor actinide content, as shown in Figs 41 and 42. The increase in ^{244}Cm content requires additional neutron shielding, while the increased decay heat must be removed by the cooling systems.

Dry storage of spent MOX may be more challenging. As noted in Section 4.2.3, the end of life pressure in MOX fuels tends to be higher than that of UOX of equivalent burnup. In combination with increased decay heat, the driving force for cladding creep is potentially higher than that for HBU UOX. This may be compounded by a need to add additional neutron shielding to the storage facility.

Two strategies may be considered to mitigate this problem. Increasing the pond storage time prior to a move to dry storage may allow the decay heat to drop to acceptable levels, however, as shown in Fig. 41, the rate of decrease is relatively slow and therefore, this strategy may not release sufficient pond space in time. In addition, MOX elements may be mixed with UOX fuel in the same storage facility or cask to balance decay heat and neutron generation, with fuel assemblies being used as part of the shielding design.

To date, it is generally more convenient for utilities to load storage casks with lower burnup UOX fuels and retain MOX assemblies in the pond, however, some MOX elements have been loaded to casks and there is little evidence of difficulties with this.

5.2.2. MOX fuel reprocessing

Section 5.1.3 presented the impact of higher burnup UOX fuels on reprocessing plants. Many of the issues discussed are also applicable when reprocessing MOX fuels, modified for the differing actinide and fission product contents. The discussion below presents the effect of spent MOX fuel reprocessing on the factors considered in fuel handling prior to dissolution and heat removal.

MOX elements have the same mechanical design as UOX fuel and present no new physical handling problems. However, as shown in Fig. 41, the decay heat of MOX fuel decreases more slowly than that of UOX fuel due to increased minor actinide content. Therefore, MOX fuels may need to be cooled for longer prior to reprocessing, or more likely, blended with UOX fuels to allow the heat loads to be managed within the plant.

Radiometric instruments will need to be recalibrated when MOX fuel is fed to the reprocessing plant, however, in this regard, the increased passive neutron emissions of MOX fuel may be used as a differentiator from UOX fuel and used to control the feed order and blending strategy.

5.2.2.1. Dissolution

There are two key differences in the dissolution behaviour of MOX fuels compared with UOX. MOX fuels generate higher levels of insoluble fission products due to Pu fissions and differences in fuel rod chemistry [74, 79]. As discussed in Sections 3.1.2 and 4.2.3, the manufacturing processes used to fabricate MOX fuels result in Pu-rich islands within the pellet structure. While these islands are depleted by irradiation they retain a high Pu concentration which may make them difficult to dissolve [83]. Many modern MOX fabrication methods limit the local Pu concentration to ~30 wt %, the point at which PuO_2 dissolution becomes more challenging [84]. On dissolution any undissolved Pu rich particles are released into the product liquor and it is important to demonstrate that over a period of time PuO_2 particles do not accumulate within process vessels to produce a criticality hazard.

The size range of both the insoluble fission products and the PuO_2 particles mean they are generally captured by the centrifuge step within the reprocessing plant. Therefore, the centrifuge cake from MOX fuel will contain both increased IFPs and Insoluble Pu solids.

5.2.2.2. Solvent extraction

As shown in Fig. 40, the α activity of MOX fuel is much higher than that of UOX fuel of equivalent burnup. Therefore, the energy passed to the solvent, if processing MOX fuels is significantly increased. The higher plutonium concentrations present when reprocessing MOX fuel may also lead to increased rates of parasitic and autocatalytic reactions and increase the risk of plutonium recycle within contactors and the accumulation of unacceptably high plutonium concentrations in particular stages. Increased feeds of reducing agent to the uranium-plutonium separation contactor are required in order to accommodate the higher plutonium concentration and increased levels of stabilizing agents; e.g. hydrazine, may be required to prevent oxidation of species such as U(IV) and Pu(III).

Again, a relatively simple way of overcoming issues related to the reprocessing of MOX fuel in the current generation of reprocessing plants is to adopt a blending strategy in which MOX fuel is mixed with a larger amount, perhaps three or four times as much, uranium dioxide fuel in the early stages of reprocessing, prior to solvent extraction. Alternatively, different solvent contactors with faster throughput rates that minimize residence time may be considered.

5.2.2.3. Product recovery

The process chemistry of product recovery is essentially unchanged when considering the liquors derived from MOX fuels or blended feeds. The key limit tends to be the throughput of the Pu powder production plant. Typical industrial reprocessing plants have designs suited to total daily throughputs of approximately 5 tons of UOX fuel and plutonium contents of 1% or 50 kg Pu per day. The amount of plutonium present in a typical spent MOX fuels is typically several weight percent and therefore redesign of plutonium recovery operations (precipitation, drying, calcining and canning) may be necessary to accommodate the increased amounts of plutonium present.

As discussed in Section 5.1.6, the α activity of separated Pu derived from MOX reprocessing is increased relative to that of standard UO_2 and similar considerations regarding radiolysis and heat during storage apply.

5.2.2.4. Waste treatment

As with UOX fuel, the amount of fission product waste in MOX increases linearly with burnup, however, the amount of Pu and minor actinides is higher. Within the HLW area, the main challenge is ^{244}Cm and neutron activity, leading to potential reductions in waste incorporation rates in glass containers if the liquors cannot be blended with less onerous material [85]. For ILW, the main challenge is in the centrifuge cake stream as noted above due to increases in IFPs and the presence of PuO_2 rich solids [83].

As with UOX fuels, providing process decontamination factors (DFs) remain the same, the amounts of active species released to the environment generally depend on the fuel burnup, however, the precise nature of the discharge will be modified to reflect the fission products produced by Pu fissions. For instance, aerial discharges of ^{85}Kr may be reduced, while ^{129}I releases could increase. The main challenge to acceptable discharges if reprocessing MOX fuel is maintaining efficient operation in the solvent extraction processes as previously discussed.

Many of the management strategies described above depend on the blending of MOX with UOX fuels to reduce the overall activity (particularly α and neutrons) of the process liquors. While the amount of low burnup UOX in fuel ponds is reducing over time, UOX blend stock will continue to be available while the practice of 1/3 MOX core loadings is maintained in PWR reactors.

As with a move to higher burnup UOX fuel, spent MOX is more challenging to current reprocessing plants, per ton of fuel processed. However, by recycling the Pu from spent UO_2 for irradiation, the need for further enrichment activities is reduced and Pu that would have been committed to a repository is reduced to shorter lived fission products. Therefore, as with high burnup UOX it is important to consider the advantages and disadvantages of MOX fuel utilization within the overall context of the fuel cycle and total energy generation rather than a narrow focus of management of current spent fuel.

Industrial MOX fuel reprocessing activities have recently started in France [86]. Around 10 t HM of MOX fuel irradiated to ~35 GW·d/t U were reprocessed in the UP2-800 plant. After modifications to the dissolution cycle, the fraction of insoluble Pu was limited to just over 0.01% of the Pu in the spent fuel. Prior to feeding to the extraction cycles, the MOX process liquor was diluted with uranium nitrate recycled from the uranium product stream in the plant. This blend was then processed through solvent extraction without incident.

5.2.3. Spent MOX transportation

The conditions of MOX spent fuel transportation would be very similar to ones described for high burnup UOX fuel transportation. Due to higher concentration of Pu and therefore higher emission of neutrons, there may be some changes in transportation conditions depending on criticality and neutron shielding calculations.

5.2.4. Spent MOX disposal

The impact of increased decay heat and ^{129}I production will have to be evaluated for MOX fuel disposal.

5.3. PROLIFERATION RESISTANCE

The proliferation attractiveness of high burnup UOX and MOX spent fuel from water reactor is clearly decreased. This is shown in the Chapter 4 of The Characteristics of Spent Fuel. The main formation chain of each plutonium isotope in the fuel is as follows:

$$^{238}_{92}U \rightarrow {}^{239}_{92}U \Rightarrow {}^{239}_{93}Np \Rightarrow {}^{239}_{94}Pu \rightarrow {}^{240}_{94}Pu \rightarrow {}^{241}_{94}Pu \rightarrow {}^{242}_{94}Pu$$

Therefore, higher burnup in UOX fuels increases higher plutonium isotopes and already existing plutonium isotopes in MOX fuels accelerate higher plutonium isotopes further. The grades of plutonium material are classified into super; weapons, fuel, and reactor according to the content of the fertile isotope ^{240}Pu are shown in Table 8.

The classification of the grade of plutonium is based on the high spontaneous fission neutron yield of ^{240}Pu is shown in Table 9.

Fast neutrons released by the spontaneous fission of the doubly even isotopes ^{240}Pu and ^{242}Pu can initiate a premature chain reaction in the plutonium when the implosion of nuclear warhead material takes place. This is the 'fizzle' of the weapon, reducing its explosive yield [20].

The spent fuel of the enriched reprocessed uranium (REPU) contains more ^{237}Np and hence more ^{238}Pu than the enriched natural uranium due to the presence of ^{236}U in the fresh fuel. Therefore it is more proliferation resistant.

It is felt that the degradation of the quality of recycled uranium could require additional protective measures because of the presence of the high energy gamma ray emitter ^{232}U in the front end facilities where the material is treated before being reused in a reactor (enrichment facilities, manufacturing chains devoted to the use of REPU, transport containers). However no need for significant additional modification of such facilities is anticipated for the envisaged burnup extension frame. The situation could become little more critical in case of the multiple recycling of the same material [82]; however this is clearly not a short or medium term issue.

5.4. ECONOMICS OF SPENT FUEL MANAGEMENT

The amount of impact of HBU and MOX on the economics of spent fuel management is dependant on the type of fuel cycle chosen (i.e. open fuel cycle, closed fuel cycle, partially closed fuel cycle or other advanced fuel

TABLE 8. GRADE OF PLUTONIUM [20]

Grade of Pu	Pu-240 (%)
Super	< 3
Weapons	< 7
Fuel	7~18
Reactor	> 18

TABLE 9. VARIOUS PROPERTIES OF PLUTONIUM ISOTOPES (AND ^{241}Am) [21]

Isotope	Half-life[a]	Spontaneous fission neutrons	Decay heat
	years	$(\text{gm-sec})^{-1}$	watts kg^{-1}
Pu-238	87.7	$2.6\ 10^{3}$	560
Pu-239	24,100	$22\ 10^{-3}$	1.9
Pu-240	6,560	$0.91\ 10^{3}$	6.8
Pu-241	14.4	$49\ 10^{-3}$	4.2
Pu-242	376,000	$1.7\ 10^{3}$	0.1
Am-241	430	1.2	114

[a] By α-decay, except ^{241}Pu, which is by β-decay to ^{241}Am.

cycles that may be applied in future (e.g. pyrochemical reprocessing, partitioning and transmutation etc.). The positive impact on economics may come from:

- Lower mass and volume of fuel per generated energy will likely lower the spent fuel management cost;
- Burnup credit of HBU fuel will have some positive impact on spent fuel criticality management costs;
- Partitioning and transmutation of actinides and other fission products from waste streams would reduce waste volumes for storage and disposal.

Increasing the spent fuel management cost may happen because of the following:

- Higher storage costs due to higher decay heat, higher actinide content, higher neutron emissions, additional shielding requirements and higher operation and maintenance (O&M) costs;
- The reprocessing plant (PUREX) may require technical upgrading to reduce degradation of organic solvents and dissolution problem associated with MOX fuel;
- The plutonium separated from HBU and MOX fuels will have isotopics giving higher neutron emissions, gamma doses and heat output. These will affect the shielding and other engineering requirements of recycled fuel fabrication plant. The plutonium content required to be added in the fuel will also increase because of unfavorable isotopics compared to plutonium generated from low burnup fuel;
- The separated uranium from a reprocessing plant (if used as feed material for enrichment and fuel fabrication) will have higher ^{232}U and ^{236}U content requiring additional shielding;
- Future recycling of actinides and other fission products (pyrochemical processing) may reduce waste storage and transportation costs; however fuel fabrication cost component may increase.

6. CONCLUSIONS

The use of high burnup UOX and MOX fuel is a worldwide strategy to optimize the economy of the front end of the fuel cycle (fuel production) and reactor operations. However, high burnup UOX and MOX use will also affect spent fuel management. The analysis in this report is focused on the back end of the fuel cycle, comparing lower burnup UOX fuel and cladding types from LWRs and HWRs having zirconium alloy based cladding and structural materials to either higher burnup UOX or MOX for the same reactors and cladding types. It has to be noted that the mechanical designs of lower burnup UOX and higher burnup UOX or MOX fuel are very similar, but some of the properties of higher burnup UOX and MOX are potentially different.

Examples of differences in properties between lower burnup UOX and higher burnup UOX and MOX include: higher fuel rod internal pressures; higher decay heat; higher specific activity; and degraded cladding mechanical properties. Evaluation of these effects on the back end of the fuel cycle was based on regulatory and safety criteria, and operational and economic considerations. Political and strategic considerations were not taken into account in this document. Furthermore, the relative importance of the technical, economic, and other considerations will vary from country to country. Higher burnup UOX or MOX usage affects all spent fuel management components, such as wet and dry storage, transportation, reprocessing, re-fabricated fuel and final disposal.

The evaluation in this publication spans a wide range of technical and economic issues. The factors that need to be considered with respect to the use of higher burnup UOX or MOX were identified. Analysis was limited in some evaluations due to the amount of available data.

Analysis of the different spent fuel management components revealed:

Wet storage: Because wet storage is associated with low temperatures, the clad material property degradation is expected to be low. High burnup UOX and MOX storage will increase the heat load, and potentially radioactive releases. This may require an upgrade of the pool facility with respect to heat removal, pool cleanup systems, and additional neutron poison material in the pool water or in storage racks. Re-evaluation of criticality and regulatory aspects may also be required.

Dry storage: In dry storage, the cask has to provide safe confinement/containment and, in parallel, the decay heat has to be removed to limit temperature induced material alterations. This means, dry storage is more sensitive to increased UOX burnup and MOX use than wet storage because of higher temperatures and, consequently, higher stresses on the cladding. The ability to meet applicable regulatory limits will need to be re-evaluated for higher burnup UOX and MOX. The result of these evaluations may require, for example, a redesign of the cask heat removal and shielding systems, a decrease in the number of spent fuel assemblies that can be placed into a single storage cask, and an increased decay time in the pool prior to placement in dry storage.

Transportation: Sub-criticality has to be assured even under accident conditions, such as a cask drop. Higher burnup fuel may lead to significantly more hydrogen in the cladding and structure and, thus, reduced ductility. These considerations will require additional evaluation for higher burnup UOX and MOX fuels. Since MOX fuel has a similar design to UOX fuel, its mechanical behavior should not be different. The result of these evaluations may require: (a) a redesign of the transportation cask heat removal and shielding systems, (b) redesign of the structural support for the spent fuel assemblies, or (c) additional cooling time prior to transport.

Reprocessing: Reprocessing plants are designed and licensed for maximum conditions of burnup and enrichment. Presently, the burnup limits are between 40 and 55 GW·d/t U. Extensions are planned to meet even higher burnups. Increased decay heat places additional duty on plant cooling systems. Increased neutron activity requires radiometric instruments (used to control criticality) to be recalibrated. Increased alpha activity results in increased radiolysis and product heat generation. Increased specific activity results in higher discharges to the environment and into HLW. These effects can be managed using blending schemes. As the burnup exceeds some level, a new reprocessing facility may be needed. The reprocessing of spent MOX fuel presents additional challenges due to insoluble plutonium.

Repository: In a repository, higher burnup UOX and MOX fuel means higher source terms of the radionuclides leading to a potentially higher release to the groundwater and to higher storage temperatures. Temperature limits on repository systems (spent fuel, waste container, backfill, near-field rock) can be quite limiting. These temperature limits may require significant repository redesign or operational changes to include higher burnup UOX and MOX, such as: increased repository space (although the reduced volume of higher burnup UOX may counteract the need for additional space), smaller waste containers, longer decay times at the surface prior to loading into the repository, and additional shielding during spent fuel transfer from the transportation cask.

REPU and MOX: An increase in discharge burnup has a significant effect on the isotopic quality of recycled fuel. Therefore, increased enrichment of REPU or an increased amount of plutonium in MOX fuel is required to meet the same burnup target. Increases in shielding may be required for REPU and MOX fuel fabrication operations.

Economics: The use of higher burnup UOX and MOX fuels will reduce the mass and volume of fuel material handled in the front end and back end fuel cycle, which may reduce the cost. On the other hand, there will likely be an increased need for longer storage of spent fuel due to higher decay heat and radioactivity. This may involve some changes in the design of wet and dry spent fuel storage, transportation, reprocessing, refabrication, and disposal systems. Furthermore, given the large variations in the price of uranium and the large uncertainty in the cost of reprocessing, it is difficult to make a decision whether or not to reprocess based purely on economics. The cost–benefit evaluation requires analysis and optimization that includes not only the major benefits in cost of reactor operation, but also the cost for the back end of the fuel cycle.

Non-proliferation: High burnup UOX, REPU and MOX fuels are more proliferation-resistant because of the higher specific activity of each of these fuel types.

Safeguards: The fuel has four potential forms of radioactivity which could be released, and the ability of the rod to release these during an accident or terrorist attack depends on the material properties of the fuel. As these properties change with burnup, it may be necessary to place additional institutional controls on higher burnup UOX and MOX.

Sustainability: For the purpose of this report, ‘sustainability’ is defined as conservation of natural uranium resources and waste minimization. To achieve the ultimate goal of long term sustainability of nuclear power, use of fast reactors must be achieved. However, current high burnup trends and mono-recycling of plutonium in LWRs are interim steps for the above goal until the introduction of fast breeder systems.

Appendix I

EFFECT OF DECAY TIME ON REPROCESSING, WASTE GENERATION, AND REQUIRED UOX TO GENERATE AN EQUIVALENT AMOUNT OF MOX

Recycling of plutonium and reprocessed uranium in light water reactors can save up to 20–30% of the uranium ore requirements compared with the once-through cycle [87].

This appendix analyzes the effect of first generation high burnup UOX fuel on the uranium saving and the mass balance between the first and the second generation of the LWR fuel cycle.

Annual consumption of enriched uranium for a gross electricity of Q GW(e) LWR at an equilibrium cycle amounts to

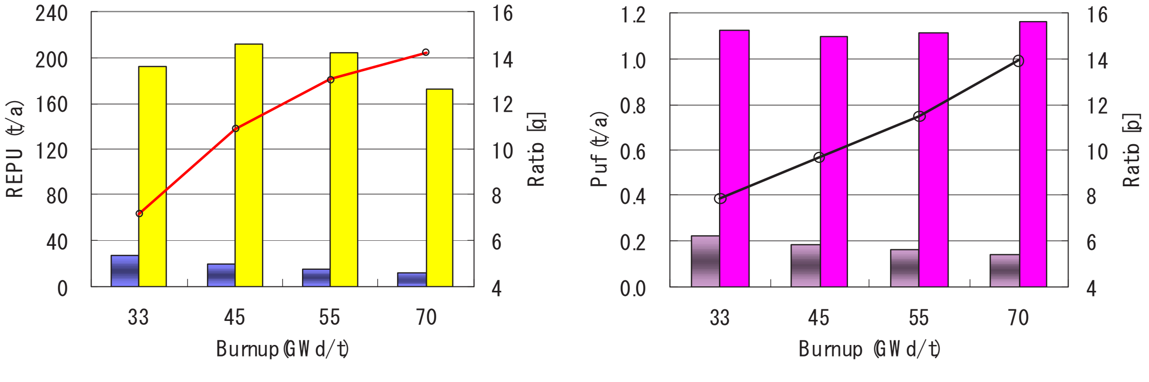
$$M = \frac{Q \cdot CF \cdot 365}{\eta \cdot B}$$

Here B is burnup, CF is plant capacity factor and η is thermal efficiency. We assume CF=0.86 and $\eta = 0.34$ for all cases in the following analysis. The quantity M is also equivalent to the annual SF production. Although burnup extension reduces SF production, it degrades the contents of REPU and plutonium recovered from SF due to the increase in the relative concentration of isotopes of higher mass number, as is discussed in Section 4.1.1.

Figure 43 (a) shows the burnup dependence of the ratio [p] of the requirement to generation of fissile plutonium (Puf). As similar quantity is shown in (b) for REPU. Spent UOX fuel is discharged after five years' cooling, then reprocessed. Recyclable material is recovered and fabricated into MOX or enriched REPU fuel to replace UOX fuel made of enriched natural uranium for the same burnup.

The enrichment of REPU is determined in a manner to compensate for the negative reactivity of ^{236}U for each burnup (see Fig. 39). The lead time between UOX-SF reprocessing and the fuel loading of the second generation (MOX or enriched REPU) fuel is assumed to be two years. The reprocessing and fabrication losses are all neglected.

Figure 44 shows UOX fuel saving vs. burnup. The saving effect is composed of Pu recycling (lower part, 1/p) and REPU recycling (upper part, 1/q), each of which represents MOX fuel and enriched REPU fabricated from one ton of UOX spent fuel. The total value (1/p + 1/q) amounts to the recycling effect already stated in Ref. A.1. Therefore higher burnup decreases the fuel saving effect. The effect of decay time on the fissile plutonium requirements for MOX fuel is summarized in Table 4 (main text).



(a) (b)

FIG. 43 (a) Annual Puf generation in UOX-SF (left rod) and annual Puf requirement (right rod) for MOX fuel of a LWR of 1 GW(e) gross. The curve shows the ratio [p] of the requirement to generation for each burnup. (b) Annual REPU generation in UOX-SF (left rod) and annual REPU requirement (right rod) for enriched REPU fuel of a LWR of 1 GW(e) gross. The curve represents these ratios [88].

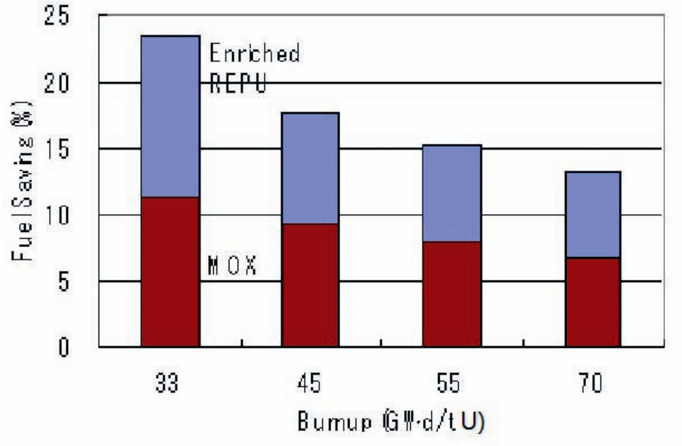


FIG. 44. Enriched uranium saving for each burnup by Pu and REPU recycle.

Appendix II

DISCUSSION OF IMPACT ON SPENT FUEL STORAGE ECONOMICS

HBU and MOX will have some impact on the economics of spent fuel management. Although it is not intended here to try to quantitatively express this impact, some general analysis of possible impact of HBU and MOX is given.

There are three basic categories of costs associated with life cycle of storage facilities for use in storage of spent fuel; i.e. capital–related costs, operating (and maintenance) costs, and the cost of decontamination and decommissioning at the end of the useful life. In addition to the storage, the cost of spent fuel transportation from the storage site may have to be taken into consideration. Within each category, there are different components which should be well distinguished in order to make a rational composition to cost estimation and analysis.

The factors and cost components are the same as for presently used estimates for spent fuel storage costs.

Costs are closely linked with the schedule for a project implementation associated with the lifetime of a facility.

A project for spent fuel storage involves a series of phases for a life cycle of facility that can be summarized as shown in Table 10.

Cost categories and components

The following is the list of typical cost categories:

- Preparatory costs: like project management, design and engineering and preliminary licensing;
- Capital–related costs: these costs depend on the type of fuel and the type of storage selected;
- Operating and maintenance costs: these costs cover salaries, materials and supplies, utilities, annual license charges, overhead and administrative expenses;
- Decommissioning costs: these costs vary widely for various types of storages;
- Transportation costs: these costs depend on the location and transportation related technologies;
- Technical contingencies: aside from the usual project contingencies, it is good to allow some contingency for unforeseen technical issues, in particular if it is for a fuel with different characteristics than those previously used.

TABLE 10. COST CATEGORIES AND COMPONENTS

CATEGORY OF COSTS	PROJECT PHASE	REMARK
CAPITAL	Project definitions	Alternatives are evaluated to select the best option A plan for project implementation established.
	Design engineering	Facility designed Investment plan established
	Regulatory approval	Safety analysis documents are prepared License issued by authority for the facility
	Construction	The storage facility is built
O&M	Spent fuel loading	Spent fuel is placed in storage in the facility Procurement of dry storage casks/modules
	Storage only	Monitoring and protection of the stored spent fuel
	Unloading	Spent fuel removed from storage Transferred to a transportation cask (if applicable) Shipping of spent fuel to off-site destination
D&D	Decontamination and decommissioning	The fuel storage facility is decontaminated and dismantled Site restored to its original condition.

Factors affecting storage economics

There are various technical and non-technical factors which are involved in the economic performance of spent fuel storage systems that are reviewed for qualitative or quantitative analysis of the economics of storage systems.

Facility design for the given requirements bears a very important implication on the economics as well as on the safety of the facility to be built. This will include the following examples:

- In terms of capital–related costs, pools are always more expensive but provide a number of advantages: storage temperatures are considerably lower than the temperature ranges encountered in dry storage and pools provide a large margin for high burnup and high enrichment fuels, including MOX fuels that dry storage systems cannot yet provide. Pool storage is also convenient in assuring safeguards and for carrying out fuel inspection during storage. However, in most cases, metal and concrete casks seem to be the currently preferred storage systems because of their lower capital–related costs, their modularity and ease of making additions and their passivity (i.e. independence from the need of active cooling systems such as for a pool). Casks designed for both storage and transport also limit direct handling of the fuel to the initial loading phase.
- All of these design costs are generally incurred up front prior to the commencement of storage, and probably over a period of two or so years. While it is possible to modularize the design of a pool so that it can be built in increments, such as at the CLAB facility in Sweden, the advantage of delaying the cost of building modules (after the initial one) is diminished by the higher cost of building a module building next to one that is in operation. Moreover, the capacity increments for a pool are large compared to the capacity of individual storage casks, which reduces its flexibility to acquire storage capacity only as it is needed.

Technical factors affecting storage costs

The key technical parameters to be considered in the safety of spent fuel storage include:

- Thermal load of spent fuel;
- Burnup of spent fuel (the credit of burnup is often used for wet type storage facilities); partial burnup credit in dry storage is also allowed in some countries, with full burnup credit being considered in the future);
- Criticality calculation (loading pattern of reactor pool); use of neutron poisons and flux traps in dry storage systems);
- Radionuclide inventory of spent fuel (currently at least safeguards relevant radionuclides are calculated in terms of mass and activity);
- Structural characteristics of the pool, dry storage system, and spent fuel to withstand accidents (e.g. assembly drops during pool transfer or dry storage cask loading, dry storage cask tipover).

These parameters have significant implications in the design and operation of storage facilities and therefore on the economics of spent fuel storage, just as it affects spent fuel management options.

Types of nuclear fuel and characteristics of spent fuel

The characteristics that must be considered are: fuel type(s) (geometry, weight and enrichment); burnup (minimum to maximum range); cooling time (decay heat); radionuclide inventory; and the physical state of the clad (clad failure and external contamination).

The applications of technical options in spent fuel management are dependent on the reactor and the fuel cycle, which in turn is dependent on the type and design of fuel being adopted.

Although the most frequently used fuel type in current use for the majority of commercial nuclear power today is the LWR fuel, there are several other fuel types in commercial use such as PHWR, GCR, RBMK, etc.

While pre-irradiation characteristics of nuclear fuel are determined by design specifications, post-irradiation characteristics are determined by irradiation during reactor operation and post-irradiation after discharge, from which data needed for design and operation of storage facility is derived.

Spent fuel burnup

Spent fuel burnup is a crucial parameter to be taken into account for spent fuel management facilities in general and storage facilities in particular.

The reactor operators' spent fuel data will be used for any further calculation needed in subsequent steps of spent fuel management.

Burnup credit

The reactivity reduction of spent fuel in comparison with the fresh fuel enrichment can be taken as credit to the safety margin obtainable associated with fuel burnup, which means the analysis conservatism can be reduced without compromise of criticality safety margin. Fresh fuel assumptions have been used as to be the baseline of regulatory conditions for safety in spent fuel management such as storage and transportation casks by many Member States.

The major motivation for burnup credit in safety criticality applications is based on economic considerations, with additional benefits contributing to worker and public health and safety and resources conservation by reducing considerably cost and worker dose of operations required for spent fuel management, and the reduced number of spent fuel cask shipments.

Currently, burnup credit applications are being used or considered by a number of countries.

High burnup

The trends toward higher burnup fuel for UOX fuel and use of MOX fuel have already initiated implications for the development, design and operation of storage systems. The issues arising from high burnup fuel are being considered in the regulatory requirements in some Member States. Otherwise, the high burnup fuel is required to be enclosed by approved baskets to ensure the potential degradation will not pose technical problems (with respect to redistribution of material during storage or subsequent transportation).

The acceptance criteria for high burnup (Zircaloy-clad) fuel storage in dry casks can be summarized as follows for defect fuel:

- More than 1% of the rods in an assembly have peak cladding oxide thickness greater than 80 micrometers
- More than 3% of the rods in an assembly have peak cladding oxide thickness greater than 70 micrometers
- Assemblies containing fuel rods with oxide that has become detached or spalled from the cladding

The licensees are required to take measurements of the cladding oxide thickness and/or conduct special computer analysis for any high burnup fuel that needed to be loaded into dry cask storage to demonstrate compliance with the guidance.

The impact of HBU and MOX spent fuel management on economics can be briefly discussed in qualitative terms.

The increase in spent fuel management cost may be expected in the following components.

- Additional effort may be required for the design of the storage system which will have to take into account higher heat source and radiation source term of HBU and MOX fuel;
- Additional effort may be required in licensing HBU and MOX fuel storage to provide safety assessments;
- The design of the storage systems, either wet or dry, has to take into account the need for more effective shielding and heat transfer. This will affect the capital cost for materials and storage construction;
- Spent fuel handling system would have to be adequately designed for higher radiation source term and heat source;
- Transportation containers and vehicles have to be either re-designed or special conditions may apply for using existing transportation containers;
- Most of spent fuel is typically stored in wet storage for cool of period before it is placed into containers/casks for dry storage. This cool of period in wet storage may be longer to reach the conditions for manipulation and transport. Operation and maintenance costs are typically higher for wet storage;

- Additional cost may be incurred in research to demonstrate the spent fuel integrity of HBU and MOX fuel over long term storage period (like criteria in the USA described above).

The decrease in cost may be affected by the following factors:

- Lower mass and volume of fuel per unit of generated energy will likely have an impact on lowering the storage costs per energy produced. This will be probably become the major cost–benefit of HBU fuel or recycled fuel because of the large impact of the size of the spent storage facility on costs;
- Burnup credit of HBU fuel will have some positive impact on spent fuel management costs.

The presented discussion indicates a need for cost–benefit analysis of each specific case and circumstances. Although the cost of storing HBU and MOX spent fuel is higher, there may be overall benefit in using less fuel for the same amount of energy generated as well as increasing the operational time for refueling unless it is on line as in case of CANDU fuel.

Reprocessing/recycle/refabrication

The economics of reprocessing, recycle, and refabrication are strongly dependent on the assumptions made for a wide variety of issues such as: the scale of the facility capacities and capacity factors; whether R&D costs to achieve commercial operation are fully included; discount rates; whether the plant is the first or the n^th^ of a kind; local regulations and safety requirements; the relative cost of fresh uranium ore; and whether the facility will be constructed and operated by the government, private industry, or a combination of the two.

Uncertainties in the price of reprocessing are large, as are uncertainties in the future price of uranium. Hence, there is a very large uncertainty in the “breakeven” uranium price above which it becomes economic to reprocess. Figure 45 provides just one example of many about the relative uncertainties in reprocessing costs and the effect those uncertainties have on the economics of reprocessing. For a given breakeven uranium price, the full uncertainty range on reprocessing costs is a factor of two to three per kilogram of separated heavy metal (HM). Figure 45 shows the historical range in uranium prices, which spans nearly an order of magnitude.

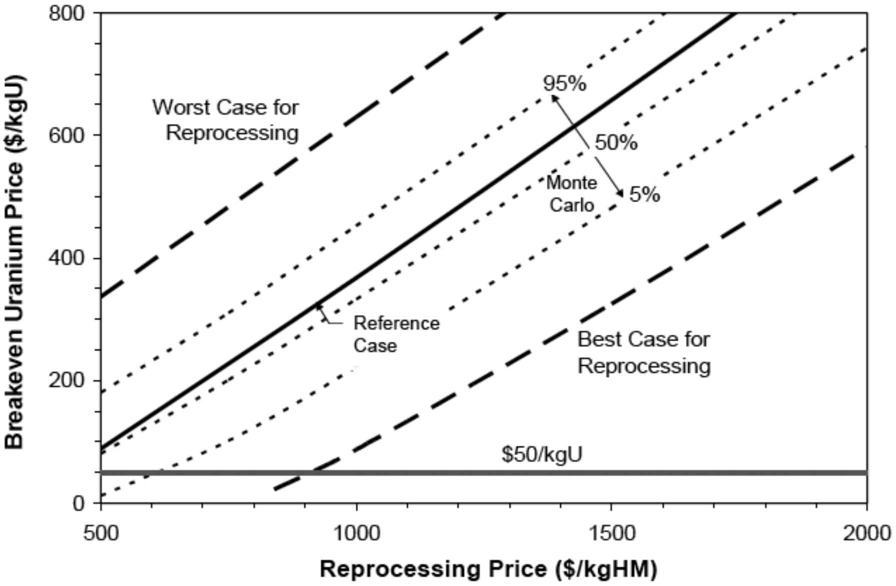


FIG. 45. Example of a study of the uncertainty of the required ‘breakeven’ price of uranium above which it becomes economic to reprocess spent LWR fuel (from Matthew Bunn, Massachusetts Institute of Technology, USA).

Thus, with the current state of development of reprocessing, it is difficult to make a decision whether or not to reprocess based purely on economics. For example, a recent US study regarding the economics of reprocessing and recycle reached the following conclusions[11]:

- Near term US adoption of spent fuel processing would incur a substantial cost penalty. In addition, to reap the major benefit possible to uranium conservation and/or the major reduction possible to required repository capacity, processing would have to be accompanied by deployment of fast reactor plants. But demonstration fast reactor plants to date have mostly proved expensive and unreliable, which aggravates processing's economic handicap;
- However, decisions on a possible second repository will not really be necessary until at least mid-century, so there are decades available to see whether an escalating uranium ore price will create an incentive to adopt processing and/or whether engineering development can reduce the costs of the processing scenario. All the existing spent fuel will, of course, be accessible for processing should that be the decision;
- The nation needs a broad consensus on which processing/fast-reactor technology combination is the best choice to take through as far as a demonstration. Developing and demonstrating an acceptable, affordable and reliable fast reactor appears likely to control the overall schedule and should receive appropriate development programme emphasis.

Disposal

Figure 46 provides a schematic of the relative cost components for a recycling strategy involving reprocessing of spent UOX and the single use of MOX in LWRs. This particular schematic assumes that spent MOX fuel will not be disposed of; hence this fuel cycle cannot be said to be "closed". However, it is illustrative of the costs involved.

An example of the uncertainty in cost estimates is provided in Fig. 46. This figure provides insight into the relative impacts on the choice of fuel cycles[12]. A similar cost estimate comparison could be done for the relative cost impacts of using higher burnup UOX.

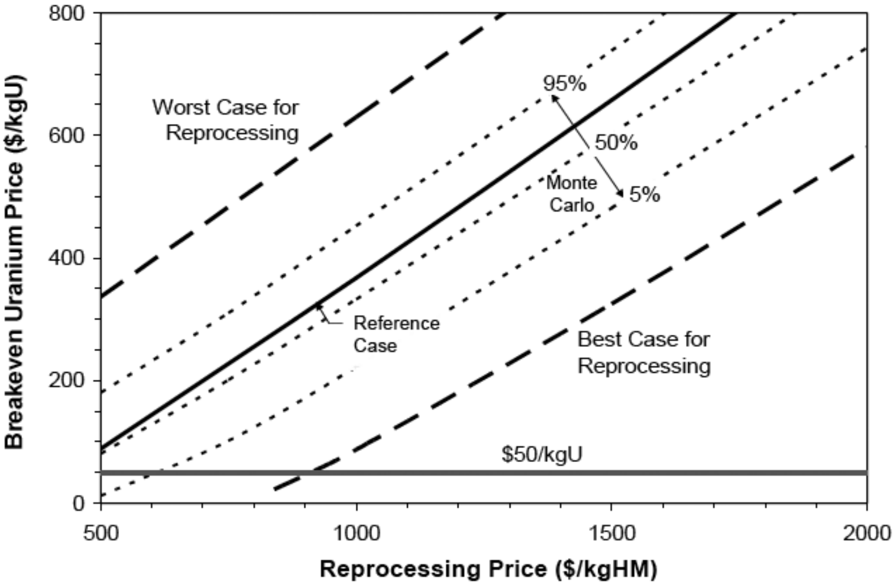


FIG. 46. Cost components of a recycling strategy (BCG, 2006).

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[11] EPRI, 2006. *An Updated Perspective on the US Nuclear Fuel Cycle,* Report 1013442, Electric Power Research Institute, Palo Alto CA, USA, June 2006.

[12] The discount rate range used is quite low. Discount ranges of ~3% to upwards of 8–10% are more commonly used. A higher discount rate range would tend to skew the results in favor of a once-through cycle. Furthermore, if spent MOX is destined for direct disposal in a repository, repository costs for MOX would have to be added to the right side of the upper uncertainty bar.

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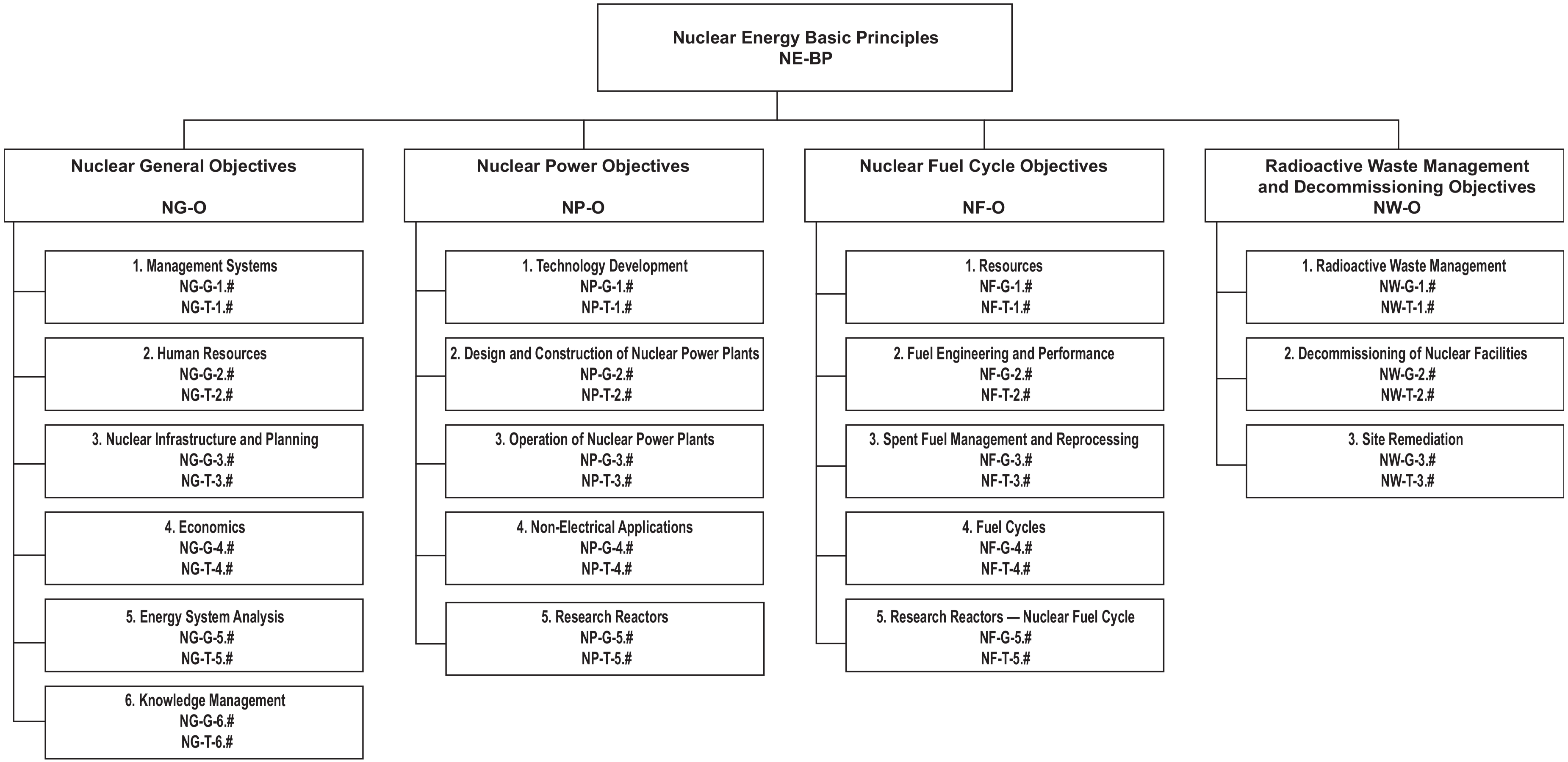

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