International Atomic Energy Agency (IAEA) Activity on Technical Influence of High Burnup UOX and MOX Water Reactor Fuel on Spent Fuel Management - 9065

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ABSTRACT

This paper briefly reviews the results of the International Atomic Energy Agency (IAEA) project investigating the influence of high burnup and mixed-oxide (MOX) fuels, from water power reactors, on spent fuel management. These data will provide information on the impacts, regarding spent fuel management, for those countries operating light-water reactors (LWR)s and heavy-water reactors (HWR)s with zirconium alloy-clad uranium dioxide (UOX) fuels, that 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 (e.g., higher fuel rod internal pressures; higher decay heat; higher specific activity; and degraded cladding mechanical properties of higher burnup UOX and MOX spent fuels) may potentially significantly affect the behavior of the fuel after irraduation These properties are reviewed. The effects of these property changes on wet and dry storage, transportation, reprocessing, refabrication of fuel, and final disposal were evaluated, based on regulatory, safety, and operational considerations. Political and strategic considerations were not taken into account since relative importance of technical, economic and strategic considerations vary from country to country. There will also be an impact of these fuels on issues like non-proliferation, safeguards, and sustainability, but because of the complexity of factors affecting those issues, they are only briefly discussed. Data gaps were also identified during this investigation. The pros and cons of using high burnup UOX or MOX, for each applicable issue in each stage of the back end of the fuel cycle, were evaluated and are discussed.. Although, in theory, higher burnup fuel and MOX fuels mean a smaller quantity of spent fuel, the potential need for some changes in design of spent fuel storage, transportation, handling, reprocessing, refabrication, and disposal will have to be balanced with the benefits of their use.

INTRODUCTION

Most countries operating LWRs or HWRs use UOX. Historically, U-235 enrichment and burnup of UOX has been in the range of 3 to just under 5%, and 30-45 GWd/MTU, 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.

Currently about 10,500 MTHM of spent fuel are unloaded every year from nuclear power reactors worldwide. This annual discharge amount is estimated to increase to some 11,500 MTHM by 2010. The total amount of spent fuel cumulatively generated worldwide by the beginning of 2004 was close to 268,000 MTHM of which 90,000 MTHM has been reprocessed. The world commercial reprocessing

capacity is around 5,550 tons per year. Projections indicate that the cumulative amount of spent nuclear fuel (SNF) generated by the year 2010 may be close to 340,000 MTHM with a corresponding increase in reprocessed fuel. By the year 2020, the total quantity of SNF generated will be approximately 445,000 MTHM. More recently, there has been a worldwide increasing use of UOX nuclear fuel with higher enrichments and burnups as the quality and reliability of UOX fuel increases, and the economics of moving to higher burnup fuel improves. In addition, all these quantities of the generated SNF require some handling. Any trend toward minimizing this amount is beneficial for SNF management.

Higher burnup can be obtained by various means. The most common way is to use fuel with higher enrichment, but other methods (e.g., reconstitution of spent fuel bundles, reshuffling of SNF bundles) are also used. The economy of using higher enrichment is well-established and is implemented by using the newer centrifuge technology for enriching uranium.

There is increasing use of MOX fuel (a combination of plutonium and uranium), as the use of reprocessing increases. There are different reasons for reprocessing SNF such as improved utilisation 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 more highly enriched and burned UOX and/or using reprocessing and MOX fuel.

In general, higher-burnup UOX and MOX SNF assemblies have many differences in physical properties, compared to similar UOX fuel assemblies with lower enrichment and burnup that make SNF 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 on the stages of the back end of the fuel cycle before making the decision to use them. Despite intensive studies [1-3] of extended burnup, since the first Water Reactor Extended Burnup Study [(WREBUS) study started in 1988 and published in 1992 as the IAEA TRS 343 report], some detailed information is still lacking. The objective of this IAEA project was to: 1) compile data on high burnup of UOX and MOX fuels and their potential influence on SNF management, and 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 SNF management.

CHARACTERISTICS OF SPENT FUEL

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 are either falling out of use or still in the experimental stage, only zirconium alloy clad oxide fuels have been considered in this project.

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 SNF performance and assessment research (SPAR) [4]. The reader is encouraged to refer to the above source document for more detailed information. Some areas of particular interest in the handling of SNF arising from higher burnup are briefly summarized below.

UOX Fuel

The preponderance of the fuel used in power reactors is made from sintered UO_2 circular pellets that are either solid or hollow. The uranium is enriched up to 5% in PWR and BWR fuel and natural to 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-4 and zircaloy-2, respectively. The zircaloy-4 was metallurgically treated in such a way as to form circumferential hydrides during irradiation, while the zircaloy-2 had a random grain texture. The rods were held in arrays from the smaller 6 x 6 BWR fuel to the 17 x 17 PWR fuel. Although 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. The major differences between the high and low burnup LWR fuel are :

- Fissile content is different, with implications on criticality and shielding,
- Fission product content is higher, with implications on shielding and heat transfer,
- Fission gas pressure is higher with potentially higher risk of cladding breach,
- Cladding mechanical properties are different, with potential implications for possible cladding breach, and
- Fuel rim effect is different with possible implications for radiation dose from released particulates.

Properties and behavior of the fuel, such as cladding oxidation, crud, pellet-cladding gap and cladding bowing, for example, were evaluated but found unimportant to the overall comparison of the influence on the backend of the fuel cycle when comparing low and high burnup fuel. Other properties of the fuel such as degree of pellet fracturing and fuel oxidation, were found to have only a minor influence, and are not discussed further in this paper.

Pressurized heavy water reactor (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 Canadian deuterium uranium (CANDU) PHWR reactors. Both fuel designs have approximately 20 kg of fuel in each bundle and are approximately 50 cm-long. UO_2 pellets are placed in zircaloy-4 tubes that have zircaloy 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-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 tube due to 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. Zircaloy spacers are attached to the surface of fuel elements by Zr-5wt% Be brazing. Heat 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 phase diagram. This changes cladding mechanical properties by enhancing its ductility and decreasing its strength.

Neutronics

Higher burnup UOX fuel requires higher enrichment and generates more fission product (FP) and transuranics (TRU) in SNF. The total specific activity and decay heat is contributed almost completely by the fission products, especially by Sr-90 (half-life 29 years) to Y-90 and Cs-137 (half-life 30 years) to Ba-137m. The half-lives of these isotopes govern the decay character of the total radioactivity until about 200 years after SNF discharge irrespective of discharge burnup. After 100-200 years, specific activity and

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

decay heat from the actinides becomes dominant. Decay heat expressed per TWh is almost constant with respect to burnup.

Increasing spent UOX burnup requires either longer decay times before dry storage, transportation, or disposal, or more storage/transportation/disposal container capacity. This is true even on a per-TWh basis, since the cost per TWh for extended decay time or smaller containers will increase with increasing burnup.

Initial enrichment and criticality- Burnup extensions in general require higher initial U-235 enrichment for UOX fuel. For PWRs, the required enrichment for a given burnup may be approximated using the Massachusetts Institute of Technology (MIT) correlation [5], which is also a function of the number of fuel batches. A maximum average discharge burnup achievable within the current 5.0 weight percent (w/o) fabrication limit is estimated to be approximately 65 GWd/MTU. This value might be extended further by advanced core managements. BWR data follow a similar trend. The isotopic effects of burnup extension on SNF are increased levels of 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. 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 burn-out of the fissile U-235 isotope, high burnup fuels tend to also have higher residual enrichment levels. This is due to the higher initial U-235 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. This allows for the consideration of the effect of irradiation on reducing the level of fissile isotopes present in SNF compared to those in fresh fuel, and, also, the impact of the increase in neutron poisoning fission products on criticality. Introduction of burnup credit can allow for more cost-efficient SNF pool rack or dry storage cask designs by allowing for more realistic estimates of the reactivity of the SNF considering TRU and fission product compositions of SNF, instead of having to assume the higher reactivity characteristic of fresh fuel. Thus it can increase storage capacity of an SNF storage cask.

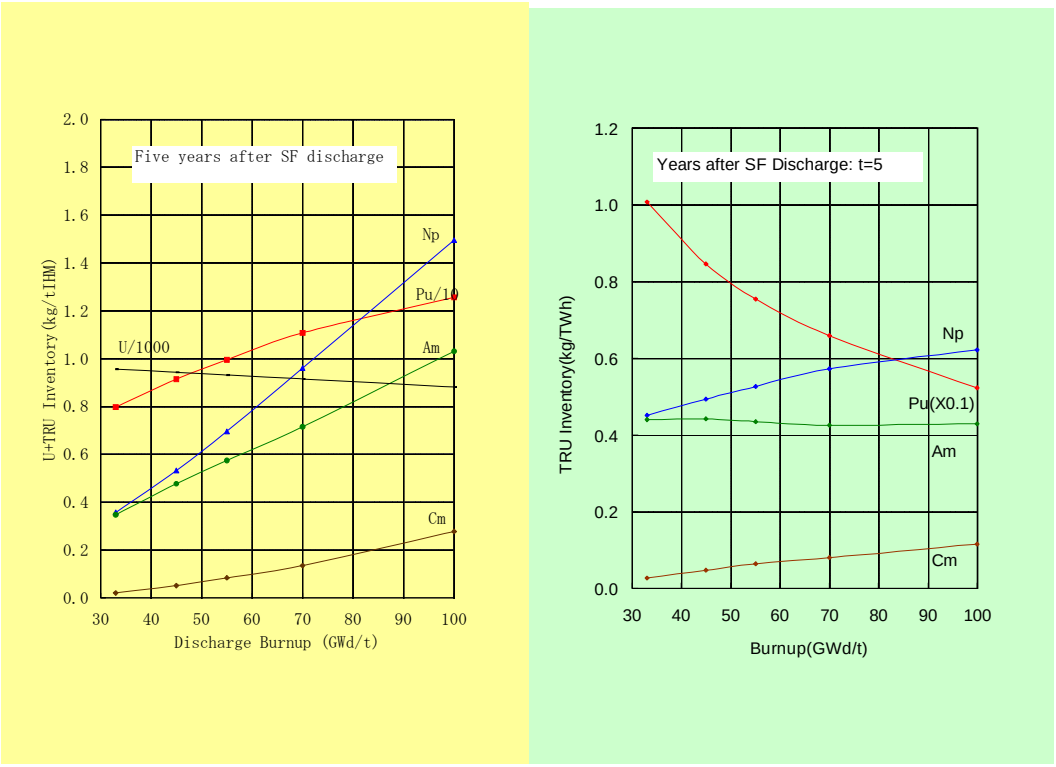
Fuel burnup and fissile content are routinely checked in reprocessing facilities by so-called burnup monitors, which use a combination of gamma spectroscopy and active and passive neutron counting. While moderate increases in burnup have no significant implications for such equipment, different designs or calibration settings may be required for 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.

Transuranic elements - Five years after discharge, the plutonium content increases more slowly with burnup at higher burnup levels. This is because an increasing proportion of fission events relate to Pu-239, rather than U-235. Decrease of fissile Pu-239 and total fissile (sum of Pu-239 and Pu-241) with discharge burnup is very evident in spite of the gradual increase of Pu-241. The isotope Pu-236 having a short half-life of 2.85 years is generated on an order of 10 ppb and alpha-decays into U-232. The well-known alpha-emitter Pu-238 has a large specific radioactivity ~6.34×10^{11} Bq/g. Both isotopes increase gradually with respect to discharge burnup.

Specific activity as function on burnup - Figure 1 shows the increase of total specific activities and specific activities of some radionuclides for high burnup fuel.

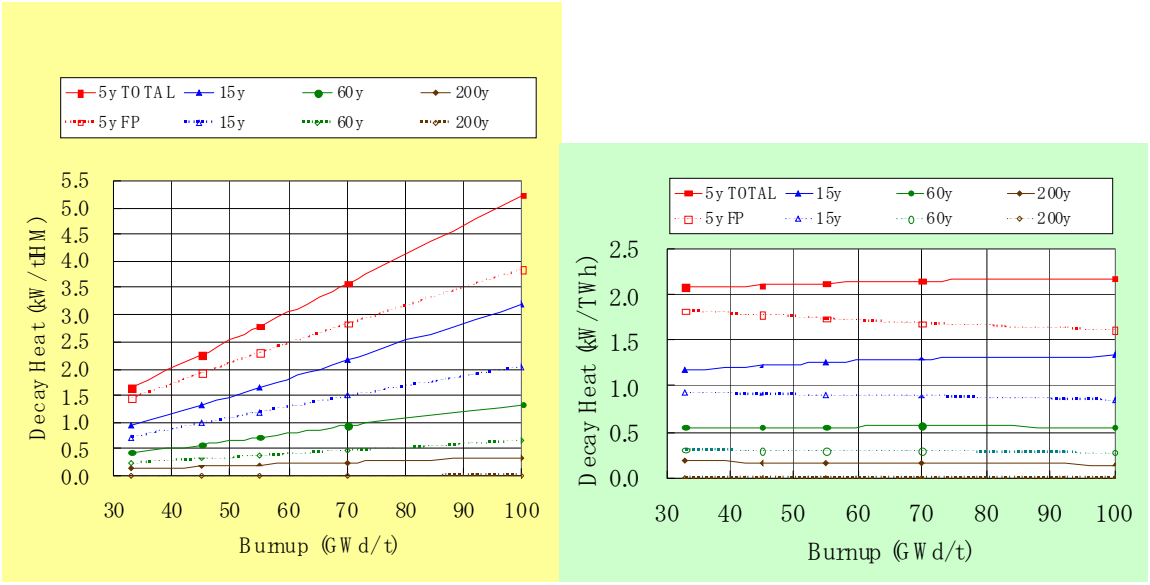
Decay heat - Decay heat increase with high burnup, as shown in Figure 2, will have an impact on SNF management activities. However, the rate of increase slows down as the post-reactor decay time of UOX-

SF increases. The contribution to total decay heat from fission products is denoted by broken (dotted) lines. Major fission-product thermal-power sources are from the beta particle-emitting radionuclides Sr-90 and Cs-137. These two fission products have similar half-lives of around 30 years, and their contents increases approximately linearly with burnup.



(a) *(b)*

Fig. 1. Dependence of radioactivity of UOX-SF on cooling years after SF discharge: discharge burnup (a) 45 GWd/MTU, (b) 100 GWd/MTU [4].



(a) (b)

Fig. 2. Dependence of decay heat on discharge burnup and UOX-SF decay time, t: (a) measured per tIHM, and (b) per TWh. Contribution from Fission Products s is shown by a broken (dotted) line [6].

Fission Gas Release

Fission gas release from the pellet to the rod plenum increases with burnup. Fractional fission gas releases show a parabolic increase, from about 5 to 25% for burnup increases of 20 to 100 GWd/MTU. The large increase in fission gas release after 80 GWd/MTU 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. The internal gas pressure in the rod, of which the fission gas is a major component, at high burnup, is a major driver for fuel rod-related degradation in the back end of the fuel cycle.

Mechanical properties of the Cladding

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 from 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 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. At 150°C the yield strength of irradiated Zircaloy-4 is ~ 800 MPa, and the ultimate strength is ~850 MPa. The yield stress decreases with temperature and is slightly lower at slower strain rates. At a fluence of ~12 x 10^{25} n/m^2 and 350°C the yield strength of Zircaloy-2 is ~550 MPa. [7]. None of these data though accounts for any reorientation of the hydrides.

Approximately 20% of the hydrogen produced during corrosion of the Zircaloy in-reactor is diffused into the rods. The mechanical properties of the zircaloy will change due to the influx of hydrogen [8]. At low burnup, the maximum hydrogen is ~ 150 wppm. This can increase to up to 700 – 800 wppm at higher

burnups, depending on the particular zirconium alloy composition. Excess hydrogen above the solubility limit will precipitate as hydrides, usually in a circumferential orientation. Yagnik, et al [8] 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 wppm radial hydrides. There was no effect of radial hydrides at 300°C. Up to ~1000 wppm, circumferential hydrides have little effect on the mechanical properties of zirconium alloys. The propensity for hydride reorientation to a degrading radial orientation at high burnup is still under study.

Pellet Rim

At about 40 GWd/MTU 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, 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. The size of the restructured rim increases exponentially with increasing burnup and is about 100-200 μm wide at about 100 GWd/MTU localized burnup [9].

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). 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 more than 40 GWd/MTU. The sub-grain sizes are generally between 0.1 μm to 0.3 μm [9]. 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 GWd/MTU.

Total porosity in the rim ranges have been reported between 10 and 40%, but most researchers' today report 15 to 20%. This compares to about 6% in the bulk material. There is significantly more closed porosity containing gas in the rim above 40 GWd/MTU [10, 11], compared to the bulk of the fuel that could tear the fuel apart further when heated in a fire.

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 total 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 depends in detail on the methods of fabrication [12].

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 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 used, 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, 1/3 of the core is MOX fuel.

Spent MOX Isotopics, Decay Heat and Radioactivity

Five years after SNF discharge, the fissile plutonium Pu-239 is the main isotopic component in the UOX-SNF with its value decreasing with discharge burnup. On the other hand, a significant decrease of Pu-239 and an increase of Pu-240 is seen in the MOX-SNF. Thus, the isotopic content of the plutonium is degraded on the subsequent recycle because of the presence of increasing proportion of the higher mass isotopes.

Generally speaking, 5 years after discharge the amount of higher-atomic-mass TRU increases with burnup. Inventories of Pu, Am, and Ce isotopes in MOX-SNF are almost 10 times larger than those of UOX-SNF. The Np inventory is much smaller than curium in the MOX-SNF. The smaller initial inventory of U-235 results in the reduction of Np-237 formation in the MOX fuel. When these quantities are expressed in terms of kg/TWh, burnup dependence of TRU inventory becomes almost flat.

Comparison of UOX and MOX Fuels

The main fissile material of the fresh MOX fuel is fissile plutonium [(Puf): Pu-239 and Pu-241], whose total Pu content depends on: (a) discharge burnup (see Figure 3); (b) cooling years of the original UOX-SF, and (c) storage years of Pu after reprocessing. Lead and lag time concept in the fuel cycle is very important because of rather short half-life (14.35 y) of Pu-241, which beta-decays to Am-241. 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 U.S.-Russian Pu disposition program, although the MOX fuel characteristics are essentially the same as those 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 REPU. The MIX [MOX/EUS (Enriched Uranium Support) concept] [13] 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 type of fuels either, which are uranium-free oxide such as thorium or other types of chemical form such as nitrides or carbides.

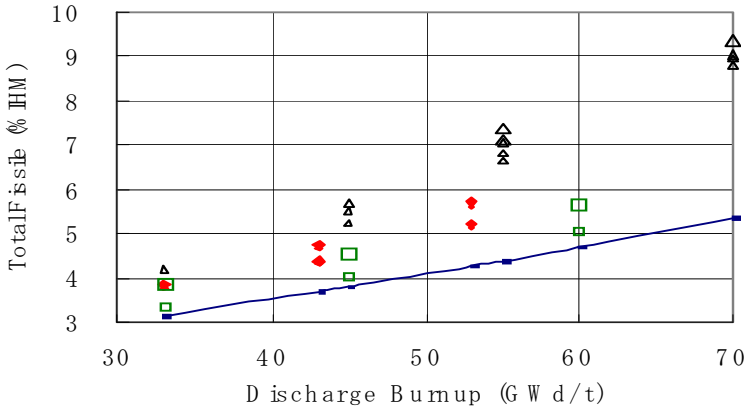


Fig. 3. Dependence of total initial fissile content of MOX fuel on average discharge burnup; The solid curve is for the UOX and various symbols correspond to various sources of data.

The Figure 4 shows specific radioactivity of MOX fuel.

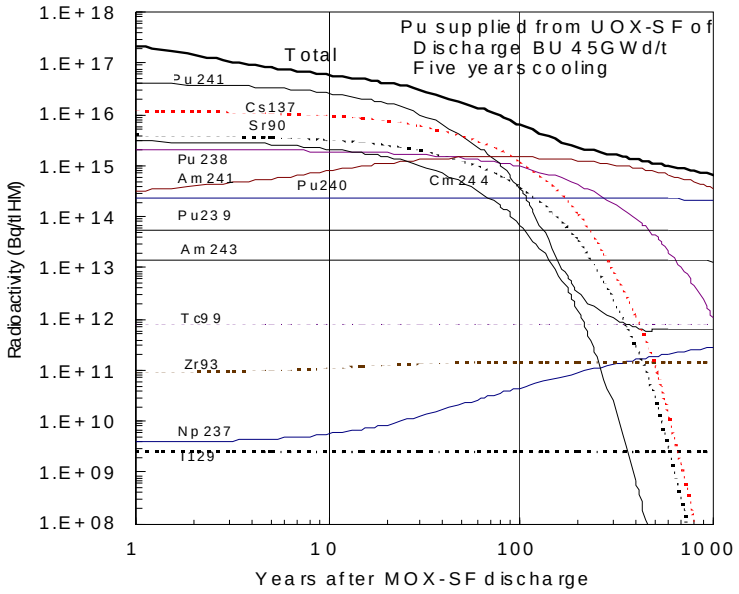


Fig. 4. Dependence of radioactivity of MOX-SF on cooling years after SF discharge: discharge burnup 55 GWd/t [6].

ANALYSIS

The components of spent fuel management that were analyzed were wet and dry spent fuel storage, transportation, reprocessing, and disposal. Extensive experience with wet and dry storage has been very positive, indicating mature technologies. Although the same is valid for SNF transportation, there is still no disposal of SNF from power reactors in operations, yet. There are several SNF disposal projects that have studied safety and environmental aspects. Experiences related to reprocessing and MOX fuels are limited to a number of countries that pursue fuel recycling strategy (i.e., France, Japan, Russia, United Kingdom) and to various magnitude some others.

Regulatory and operational concerns for the back end of the fuel cycle include: criticality, heat removal, 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 1 below. In addition, economics, and non-proliferation were also considered, but not analyzed in detail.

Table 1: Where High Burnup Affects the Backend of the Fuel Cycle

	Wet Storage	Dry Storage	Transportation	Disposal	Reprocessing
Criticality		L	L,W	L, W	L,W
Heat removal	L, W	L	L, W	L, W	L, W

Shielding	L,H, W	L, H, W	L,H, W	L,H, W	L,H, W
Containment/Confinement	L?, H?,W	L,H, W	L,H, W	L,H, W	L,H, W
Retrievability	L, H,W	L,H, W	L,H, W	L,H, W	L,H, W
Operations/construction		L,H, W	L,H, W	L	L,H, W

L = LWR fuel, H = HWR fuel, W = WWER fuel.

The ability to meet these regulations and demands 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 behavior of the fuel. It is up to the reader to decide how the particular regulation for his country applies and how to rank the importance of these criteria for decision-making.

Fuel is designed for optimum performance, in the reactor, where the utilities produce revenue, and to minimize, even eliminat,e 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. 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. While the characteristics of the SNF are fixed by its design and reactor operations, the materials behavior in the back end of the fuel cycle can be modified by changing the conditions such as cover gas, temperature, handling stress, etc.

The characteristics of the high burnup fuel can affect the mechanisms by which the fuel can breach, and directly affect the ability of the fuel to meet the regulatory requirements. To analyse the effects of high burnup fuel in these areas the materials behavior especially fuel-cladding breach, hydrogen generation and materials compatibility, need to be evaluated under both normal, and accident conditions.

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 GWd/MTU reached under those reduced demanding conditions should not result in elevated amounts of fission gas release. Therefore, high burnup material issues of LWR fuel largely cover HWR fuel.

Following are the general results of the analysis of the effects of the fuel properties on each phase of the back end of the fuel cycle and the impact expected in meeting each of the regulatory concerns. 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 that 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: 1) a redesign of the cask heat removal and shielding systems, 2) a decrease in the

number of spent fuel assemblies that can be placed into a single storage cask, and 3) 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 have 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: 1) a redesign of the transportation cask heat removal and shielding systems, 2) redesign of the structural support for the spent fuel assemblies, or 3) 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 40 to 55 GWd/MTU. 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 high-level waste (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 lower solubility of Pu.

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 disposal 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: 1) increased repository space (although the reduced volume of higher burnup UOX may counteract the need for additional space), 2) smaller waste containers, 3) longer decay times at the surface prior to loading into the repository, and 4) 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 SNF 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 to reprocess, based purely on economics. The cost-benefit evaluation requires analysis and optimization that include 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 tend to be more proliferation-resistant, because of the higher specific activity of each of these fuel types and because of less favorable fuel isotopics for proliferation (less Pu-239 and more Pu-240 and Pu-238).

CONCLUSIONS

Although experience has shown the safe feasibility of using high burnup and MOX fuels in the reactor, it appeares that longer cycles, higher radiation and heat load may enhance some characteristics of SNF that could require additional attention and specific investigation in SNF management.

An evaluation of the back end of the fuel cycle, based on regulatory and safety criteria, and operational considerations, was conducted. While important, detailed economic considerations were beyond the scope of this work and only minimally touched on. Political and strategic considerations were not taken into account in this document.

Cladding and fuel pellet characteristics, and the pellet-cladding gap, were all discussed and analyzed. Fuel pellet characteristics like fission product and actinide production, fission-gas release, rim formation, and potential increase in cladding fracture were determined to be the most influential fuel characteristics when evaluating the effects on the back end of the fuel cycle. Potential impact of high burnup on crud deposition, corrosion of cladding, hydride characteristics, and grain size were also discussed, but were found to only have secondary effects.

Potential effects of use of high burnup UOX or MOX on wet and dry storage, transportation, repository disposal, reprocessing, REPU and MOX fabrication, and both economics and non-proliferation, during the back end of the fuel cycle, were evaluated. Analysis was limited, in some evaluations due to the amount of available data. Higher burnup UOX or MOX usage affects all SNF management components, with pros and cons related to each step. Due to the relative importance of the technical, economic, and other considerations from country to country, there can be no definitive recommendation on whether to use or not use higher burnup UOX or MOX based on SNF management issues. Each individual country will have to read the above analysis and consider how evaluate the relative importance of the various criteria for its own individual situation.

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 SNF management. The conclusions reached in this paper are 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.

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