DISCRETE ISSUES IN MANAGING HIGH BURNUP, MOX AND FAST NEUTRON REACTOR SPENT FUEL

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IAEA CONSULTANCY ON TECHNICAL INFLUENCE OF HIGH BURNUP UOX AND MOX LWR FUEL ON SPENT FUEL MANAGEMENT

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Abstract

This paper 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 SFM. These data will provide information on the impacts, regarding SFM, for those countries operating light-water reactors (LWR) 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 irradiation. 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. The advantages and drawbacks 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.

1. INTRODUCTION

Most countries operating LWRs or HWRs use UOX. A considerable amount of property data on spent UOX with enrichments in the range of 3 to <5%, and burnups in the range of 30-45 GWd/MTU 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 ~10,500 MtHM/y of spent fuel are unloaded from nuclear power reactors worldwide. This 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 ~268,000 MtHM/y of which 90,000 MtHM has been reprocessed. The world commercial reprocessing capacity is ~5150 MtHM/y. Projections indicate that the cumulative amount of spent nuclear fuel (SNF) generated by the year 2010 may be ~340,000 MtHM with a corresponding increase in reprocessed fuel.

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. There are many reasons for reprocessing SNF such as better utilisation of natural resources, 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. All these quantities of the generated SNF require some handling, so any trend toward minimizing this amount is beneficial for SNF management.

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 2) 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.

2. CHARACTERISTICS OF SPENT 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 effect of these fuel properties on the stages of the back end of the fuel cycle before making the decision to use them.

Most fuel currently used in power reactors 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 [1–3] and have been reported in the recent and final report of a co-coordinated research project on SNF performance and assessment research (SPAR) [4] and the TECDOC from this Consultancy currently in printing. 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.

2.1. UOX Fuel

The preponderance of the fuel used in power reactors is made from sintered UO_2 circular pellets that are hollow in some designs. 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 × 6 BWR fuel to the 17 × 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 with implications on criticality and shielding;
- Fission product content with implications on shielding and heat transfer;
- Fission gas pressure with potentially higher risk of cladding breach;
- Cladding mechanical properties with potential implications for possible cladding breach, and
- Fuel rim effect with possible implications for radiation dose from released particulates.

Canadian deuterium uranium (CANDU) PHWR reactors pressurized heavy water reactor (PHWR) fuel uses natural uranium oxide as fuel in 28- and 37-element fuel bundle designs[1] are used in. Both fuel designs 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 makes each element an active component of the bundle structure and mechanically constrains each Zircaloy tube due to the rigid attachment of each end cap to an end plate. 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. MOX is not used in PHWR fuel.

2.1.1. 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 from the beta particle-emitting radionuclides Sr-90 and Cs-137. These two fission products have similar half-lives of ~30 years. Their contents increase approximately linearly with burnup, but decreases as the post-reactor decay time of UOX-SF increases. The half-lives of these isotopes govern the decay character of the total radioactivity until ~200 years after discharge irrespective of burnup. After 100–200 years, specific activity and decay heat from the actinides dominants.

Five years after discharge, the plutonium content increases more slowly with burnup at higher burnup levels, because an increasing proportion of fission events relate to Pu-239, rather than U-235. Decrease of fissile Pu-239 and total fissile (Pu-239 + Pu-241) with discharge burnup is very evident in spite of the gradual increase of Pu-241. The isotope Pu-236 (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 discharge burnup.

Burnup extensions in general require higher initial U-235 enrichment for UOX fuel. A maximum average discharge burnup achievable within the current 5.0 weight percent (w/o) fabrication limit is estimated to be ~65 GWd/MTU. This 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.

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

2.1.2. Fuel pellet and cladding mechanical properties

The fractional fission gas release from the pellet to the rod plenum shows a parabolic increase, from ~5–25% for burnup increases from 20–100 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.. 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.

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 the as-fabricated grain size of ~10 μm, typical of commercial fuel. At ~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. The size of the restructured rim increases exponentially with increasing burnup and is ~100-200 μm wide at ~100 GWd/MTU localized burnup [5]. The rim portion of high burnup fuel will have much higher burnups than the pellet average and forms restructured fine sub-grains between 0.1 μm to 0.3 μm [5] at pellet average burnups more than 40 GWd/MTU. As the burnup of the rim increases the original as-fabricated grain boundaries begins to disappear as the sub-grain structure becomes dominant. Total porosity in the rim ranges is usually between 15–20%, compared to ~6% in the bulk material.

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. Approximately 20% of the H_2 produced during corrosion of the Zircaloy in-reactor is diffused into the rods, and precipitates as circumferential hydrides. The mechanical properties of the Zircaloy will change due to the influx of hydrogen [6]. 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. 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.

2.2. MOX fuel

Conventional MOX fuel is PuO_2 mixed in the UOX carrier material,. The MIX [MOX/EUS (Enriched Uranium Support) concept whose UOX matrix is enriched uranium], is excluded, because MOX is only a secondary contributor for fission rate. 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. In addition, we do not consider unconventional fuels such as thorium based or chemical form such as nitrides or carbides.

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.

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 than 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 [7]. 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, 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.

Five years after UOX SNF discharge, fissile Pu-239 is the main isotopic component in the UOX-SNF with its value decreasing with discharge burnup. 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; (b) cooling years of the original UOX-SF, and (c) storage years of Pu after reprocessing. Lead and lag time 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.

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. On the other hand, a significant decrease of Pu-239 and an increase of Pu-240 is seen in the MOX-SNF. The isotopic content of the plutonium is degraded on the subsequent recycle because of the presence of increasing proportion of the higher mass isotopes. As time out of reactor increases, the pressure in the fuel rod will increase due to the fission product decay to produce helium.

3. EFFECTS OF HIGH BURNUP UOX AND MOX FUEL ON THE BACK-END OF THE FUEL CYCLE

The components of SFM 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. Economics and non-proliferation were also considered, but not analyzed in detail.

TABLE 1. EFFECTS OF HIGH BURNUP UOX AND MOX FUEL\ON 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 behaviour 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 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 analysing 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 behaviour 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 behaviour especially fuel-cladding breach, hydrogen generation and materials compatibility, need to be evaluated under both normal, and accident conditions.

Properties and behaviour 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.

An overriding consideration in many phases of the back end of the fuel cycle is criticality. 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. The higher residual enrichment might be offset by burnup credit.

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 SFM components revealed:

3.1. 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.

3.2. 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, 3) potentially a lower maximum fuel cladding temperature, and 4) an increased decay time in the pool prior to placement in dry storage.

3.3. 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.

3.4. Reprocessing

Reprocessing plants are designed and licensed for maximum conditions of burnup and enrichment. Presently, the burnup limits are 40–55 GWd/MTU. Extensions are planned to meet even higher burnups. Increased decay heat places additional duty on plant cooling

systems. 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.

Increased neutron activity requires radiometric instruments (used to control criticality) to be recalibrated. 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.

3.5. 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.

3.6. 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.

3.7. 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.

3.8. 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 favourable fuel isotopics for proliferation (less Pu-239 and more Pu-240 and Pu-238). Despite the increased burn-out of the fissile U-235 isotope, high burnup fuels tend to also have higher residual

enrichment levels due to the higher initial U-235 enrichment required to achieve the high burnup levels.

4. CONCLUSIONS

Although experience has shown the safe feasibility of using high burnup and MOX fuels in the reactor, it appears that longer cycles, higher radiation and heat load, and changes in 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. Increasing spent UOX burnup requires either longer decay times before dry storage, transportation, or disposal, or more storage/transportation/disposal container capacity. 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.

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STORAGE AND MANAGEMENT OF SPENT FUEL OF FAST REACTORS IN INDIA

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Abstract

Deploying fast breeder reactors on a commefcial scale is vital for India in order to utilise the vast thoirium reserves and to meet the long term energy needs of the country. As a first step, a 40 MWt, 13.5 MWe Fast Breder Test Reactor (FBTR) was constructed and has been in operation since 1985. FBTR utilises mixed carbide fuel. The spent fuel subaasembly of FBTR is stored in an argon inerted container and cooled by air. India took the construction of indigenously designed 500 MWe Prototype Fast Breeder Reactor (PFBR) in September 2003. The reactor is in advanced stage of construction. It is planned to construct six more FBRs of 500 MWe each by the year 2023. All the 500 MWe fast reactors utilise oxide fuel, the most proven fuel in the case of the fast reactors. The spent fuel subassembly after internal storage for one campaign of 8 calendar months is washed to remove sodium and stored in water in a spent fuel storage lined with SS 304L liner and with a provision of leak monitoring of the liner welds. The failed subassembly is stored in a leak tight container before putting in the spent fuel stoage. The road to getting faster growth in Indian indigenous nuclear power programme is through use of metallic fuel in fast breeder reactors. The metallic fuel design gives high breeding ratio and thus shorter doubling time. Both sodium and mechanical bonded metallic fuel design are under development. The growth of fast reactors is sensitive to the time period the spent fuel is out of the reactor i.e. from the time spent fuel is out of the reactor, cooled, reprocessed, refabricated and put back into the reactor. Storage of spent fuel subassemblies in sodium and co-located pyro-reprocessing facility is planned for the metallic fuel reactors. The paper describes in brief the system for storage and management of spent fuel of FBTR, PFBR and metallic fuel fast reactors.

1. INTRODUCTION

The FBR programme in the country began with the construction of a 40 MWt, 13.2 MWe, Fast Breeder Reactor (FBTR), which has been in operation since 1985. FBTR serves as a test bed for irradiation of fuels and structural materials. Operation of FBTR has given valuable operating experience with systems of FBRs including sodium and fuel handling systems [1] and has led to the design of 500 MWe Prototype Fast Breeder Reactor (PFBR). Indigenously developed mixed carbides of PuC-UC is used as the fuel in FBTR. As part of the second stage of the Indian nuclear power programme, PFBR which is a sodium cooled pool type FBR is under construction at Kalpakkam and is expected to go critical by Sept 2011. PFBR design is based on mixed oxides of PuO_2-UO_2. It is planned to construct six more oxide fuelled reactors of 500 MWe capacity by 2023 and metal fuelled FBRs of 1000 MWe thereafter.

The SFM programme in India is based on closing the nuclear fuel cycle with reprocessing of the spent fuel and re-fabrication & re-utilization as new fuel. The type of fuel used / proposed to be used also varies ranging from carbide to oxide/metal. The fuel cycle programme is also being developed to cater to the requirements of various types of fuel. The technology of reprocessing of the carbide fuel has been established with the successful reprocessing of FBTR fuel pins of 155 GWd/t burnup at the Compact Reprocessing facility for Advanced fuels in Lead shielded cells (CORAL) facility [2,3]. Fast reactor fuel cycle facility is planned to be co-located with PFBR to reprocess the spent fuel of PFBR along with its refabrication. Research & development has been initiated on reprocessing of metallic fuel.

2. SPENT FUEL HANDLING AND STORAGE IN FAST REACTORS — AN OVERVIEW

Fast reactors are characterized by high fissile enrichment (20-25% of heavy metal for power reactors) with resultant peak burnup of 100 GWd/t and above. The high burnup of the fuel leads to higher decay heat of the subassemblies immediately after shutdown of the reactor and which decays exponentially with time. To maintain clad integrity, the temperature of the clad is limited during handling and storage of the subassemblies (700^{o}C for oxide fuel). The temperature limit during storage is generally lower than the clad temperature limit during normal operation. Sodium used as the coolant, being a liquid metal has high thermal conductivity and high heat removal capability. The maximum decay heat rating is in general, not more than 40 kW in refuelling sequences that employ the direct transfer method without internal storage [4]. In most of the reactors, the subassemblies are transferred to peripheral core locations (internal storage) and are stored for one or more campaigns in order to reduce the decay heat suitable for handling in gaseous environment (to meet the clad temperature limits). Rotatable plugs and in-vessel handling machine(s)/ handling flask (s) are used to transfer the subassemblies to internal storage locations. From the internal storage locations, the subassemblies are further transferred out of the main vessel in a sodium filled pot/leak tight inerted flask /leak tight inerted cell to the sodium washing facility after the decay heat is reduced to a level suitable for handling (5 kW). The subassemblies after washing are sent to the spent fuel storage for long term storage / storage to reduce the decay heat suitable for reprocessing. A comprehensive review of the fuel handling options including spent fuel storage in various fast reactors is given in [5].

Spent fuel storage in fast reactors is basically of two types:

(a) Wet storage and
(b) Dry storage.

2.1. Wet storage

Wet storage utilizes a bulk liquid medium directly in contact with the subassemblies to remove the decay heat and also to provide the required shielding. It can be an open type water pool or a closed vessel filled with liquid metal like sodium/NaK. The geometric configuration is such that the assemblies remain subcritical during storage (Keff < 0.95). The chemistry of the medium in contact with the subassemblies is controlled such that the integrity of the clad is maintained without adverse corrosion effects. The temperature of the medium is also maintained such that the decay heat of the subassemblies are removed effectively under both normal/loss of cooling conditions in order to maintain the temperature limits of the clad/coolant medium/structural members of the storage system.

Sodium storage has also been used in fast reactors primarily as an intermediate storage immediately after direct discharge from the reactor vessel. It uses a vessel filled with sodium housing storage locations for the subassemblies. A redundant vessel is provided around the vessel to take care of any leaks in the primary vessel. The primary vessel is closed at the top by top shield and the sodium free level is covered by an inert cover gas. The vessel has its own cooling system to remove the decay heat and heating system to maintain the sodium temperature. Adequate redundancy is provided to take care of loss of cooling/heating systems. The design of sodium storage is covered by ANSI standard ANSI/ANS 54.2 [6]. The

advantages of sodium storage are (i) decay heat can be effectively removed due to use of sodium, (ii) fuel handling time is less since only exchange of spent subassemblies between the reactor vessel and the sodium storage is done during fuel handling while the other operations like subassembly washing, loading of fresh fuel and spent SA discharge can be done later during reactor operation thus providing higher plant capacity factor and (iii) full core unloading of subassemblies with larger decay heat is possible within the shortest possible time in case of any need to examine the reactor internals due to fuel handling incidents or in-service inspection requirements. The disadvantage is its being a costly option since the sodium storage is like a mini reactor with safety features required similar to a reactor system. The leakage of the ex-vessel storage drum in Superphenix-1 led to its subsequent abandonment of storage and conversion to a transit cell filled with inert gas. However, sodium storage still remains one of the attractive options being considered for future fast reactors in some countries.

Water pool storage has been used in many fast reactors and is the most popular method of storage. The feedback of similar storage in thermal reactors has led to maturity of this concept such that today there are many water pools in the world operating successfully for many decades. The design of water pool storage is covered by ANSI standards ANSI/ANS 57.2 [7], ANSI 57.7 [8] and several IAEA documents [9–12]. Water being relatively inexpensive as compared to sodium also makes this option highly economical. Early water pools had concrete walls coated with epoxy paint, but later designs with austenitic stainless steel liners with in-built leakage detection & collection arrangements has reduced the possibility of leakages from the system significantly. Typically, in a once through cycle, spent fuel is stored in water pool storage for a long period till the decay heat and radioactivity of the fuel comes to a level such that these can be considered for other means of storage. This has led to the concept of dry storage of fuel for long term storage/disposal.

2.2. Dry storage

Dry storage involves the storing the fuel in steel or concrete vaults with active or passive cooling using air or gas. The dry storage concept is covered in [9, 13]. The subassemblies are contained within leak tight containers to avoid direct spread of radioactivity. The containers are filled with a liquid like sodium, lead, lead-antimony, etc. to enable dissipation of decay heat or an inert gas. In certain cases, the subassemblies are kept within helium filled casks and the casks are stored & cooled by natural/forced convection. Dry storage has the advantage that it does not result in direct spread of radioactivity to the environment since the fuel is encapsulated in containers/casks and hence is considered for extended long term storage. It is suitable when the decay heat of subassemblies has been reduced to a level to permit heat removal by natural/forced cooling using air since loss of cooling will lead to rapid rise of clad/structural temperatures.

3. SPENT FUEL STORAGE IN FAST REACTORS IN INDIA

3.1. Fast breeder test reactor (FBTR)

Figure 1 shows the details of fuel subassembly of FBTR. The subassembly is hexagonal in shape (width across flats = 49.8 mm, overall height of SA = 1661.5 mm) and has a head suitable for gripping externally by the gripper of handling flasks.

Fig. 1 also shows the scheme of spent fuel discharge followed in FBTR. The spent fuel is discharged and stored in the peripheral locations of the core for one campaign (45 days) in order to reduce its decay heat to less than 400 W. At this decay power level, no active cooling is required during its handling by various flasks. Two rotatable plugs along with a fixed fuel handling canal provided in small rotatable plug are used for in-vessel transfer. In-vessel transfer operations are carried out using a transfer flask. The spent SA is discharged out of the reactor through the fuel handling canal using discharging flask. The flask transfers the spent SA to a leak tight argon filled discharge pot located in discharge pit. Twelve locations are provided in discharge pit. From the discharge pit, the pot along with the SA is handled by secondary flask to the Irradiated subassembly storage. The secondary flask is also used to transfer the SA from the storage pit to the Radio metallurgical lab (RML) pit from where the spent SA is transferred into the hot cell of RML using a special flask. Inside the hot cell, the SA wrapper is cut, the pins are cleaned with alcohol and are then sent to the reprocessing plant in a shipping cask.

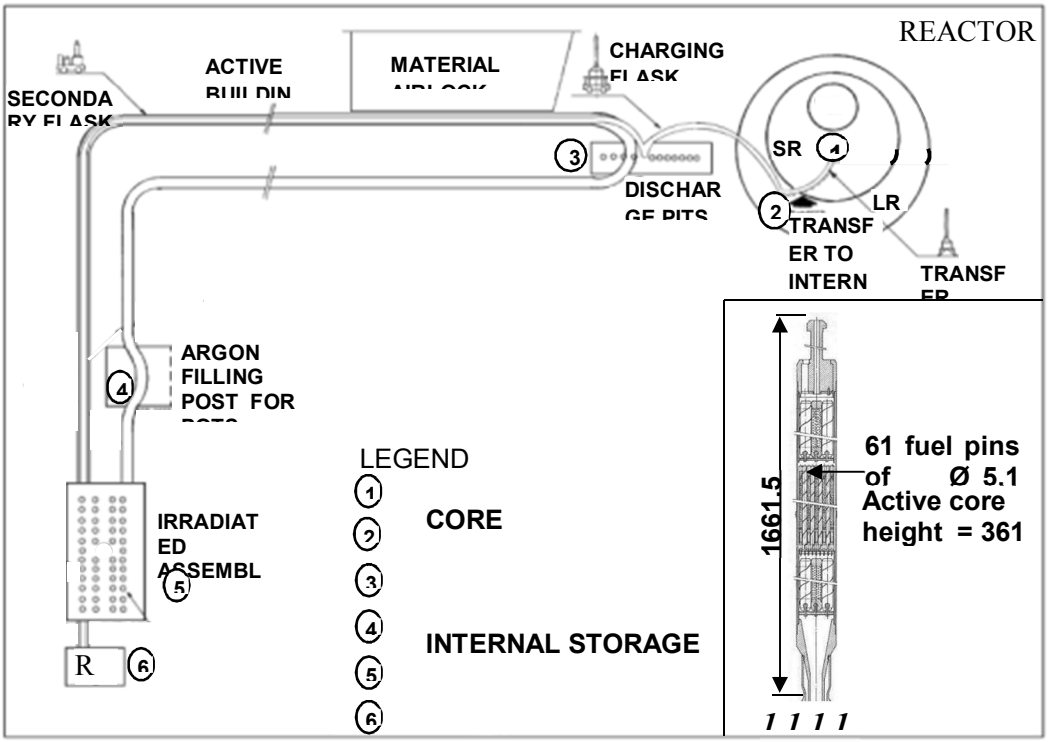


FIG. 1. Spent subassembly handling scheme in FBTR.

The carbide fuel used in FBTR is by nature pyrophoric. Hence, the spent fuel is kept in contact with an inert gas medium (argon). Fig. 2 shows the details of the spent fuel storage in FBTR. A dry storage concept with subassemblies stored by forced convection flow of air is adopted. The arrangement consists of a rectangular concrete pit, 4.8 m wide × 21.9 m length × 2.78 m height in which the SA are stored in triangular pitch. Air is circulated across the active portion of the SA to provide the required cooling. The whole storage area is divided into fissile and non-fissile storage zones. A total of 203/619 storage positions are provided in the fissile/fertile zones including provision for emergency core unloading. The fissile zone can be used to store fertile subassemblies, but not vice versa. The pit is closed at the top by cast iron blocks for shielding. Shielding plugs made of cast iron with its upper part similar to discharge pot is used as dummy shielding for the vacant locations to avoid radiation streaming from the

stored spent subassemblies. 3 × 50% blowers are provided along with provision of emergency power supply to ensure availability of air cooling. The low decay power of subassemblies enables use of dry storage for spent subassemblies.

Failed subassemblies are stored in the same argon filled pots as intact subassemblies with provision of sealing by solidified lead in addition to the normal seals provided. Storage and handling of the pots with failed subassemblies is similar to intact subassemblies.

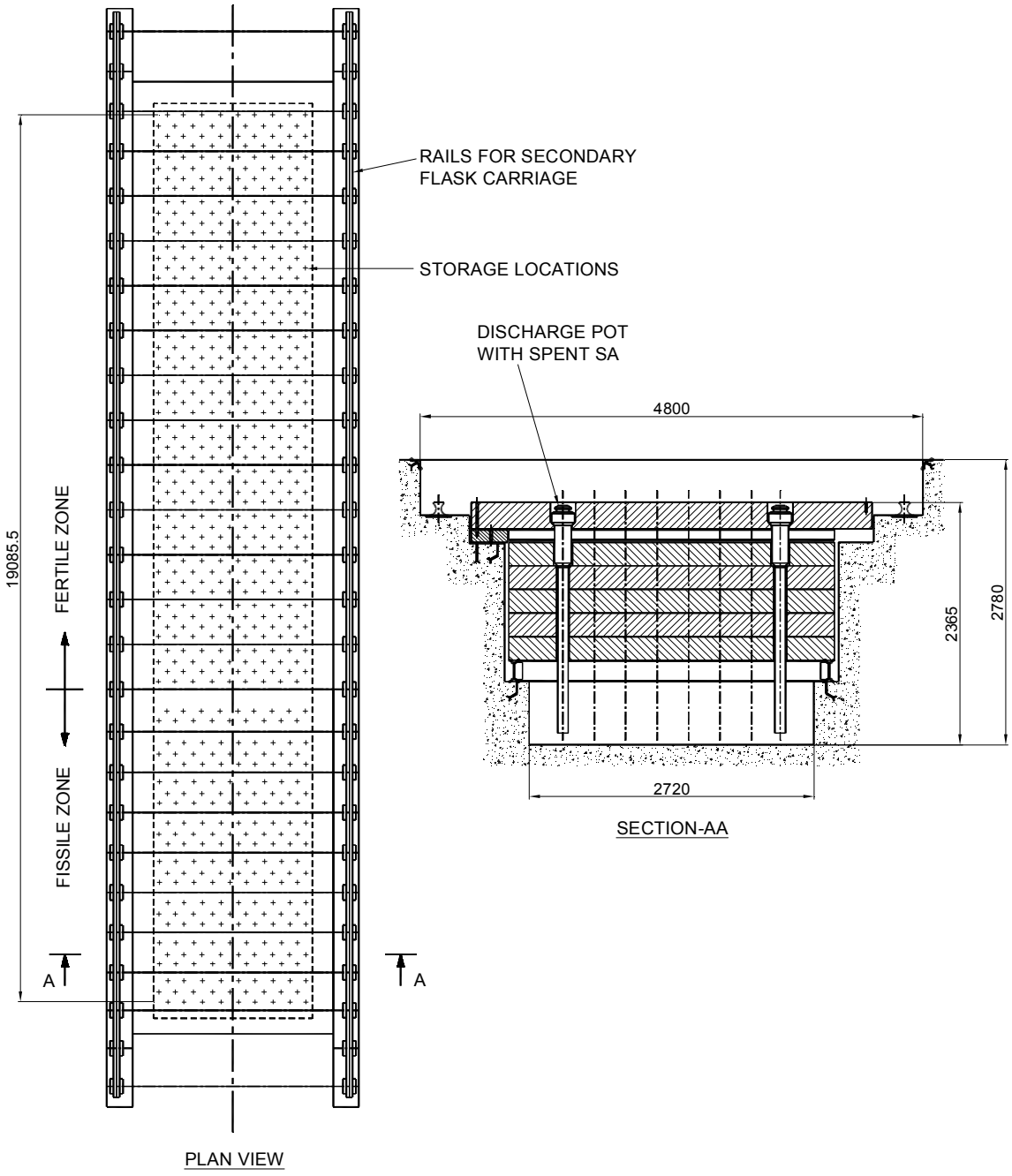


FIG. 2. Details of the spent fuel storage in FBTR.

3.2. Prototype fast breeder reactor (PFBR)

Fig. 3 shows the details of PFBR fuel subassembly. The subassembly is 4.5 m in height, hexagonal in shape (width across flats = 135 mm), weighs 250 kg and has a head suitable for gripping internally by the handling machines.

Fig. 4 shows the spent SA handling scheme. A combination of two rotatable plugs and one offset arm type transfer arm is used for in-vessel handling. The spent SA is transferred to internal storage locations in the periphery of the core. The average residence time of fuel SA is 2 y and that of blanket SA is 3 y. 62 fuel, 33 blanket and 4 absorber SA are replaced in each fuel handling campaign. On the periphery of the core, 156 storage locations are provided for storing spent SA. They are then loaded into a sodium filled transfer pot. The inclined fuel transfer machine transfers the pot along with SA to the ex-vessel transfer port (EVTP) located within the nitrogen filled fuel transfer cell. The cell transfer machine, which is a straight pull type machine, transfers the spent SA from the EVTP to the washing facility where the sodium sticking to the SA is cleaned by steam-nitrogen process. Four washing vessels are provided to provide redundancy and to ensure maximum availability of the system since the sodium washing is done as part of the fuel handling operations. The SA after cleaning is stored in the spent subassembly storage bay (SSSB), which is a demineralized water pool in order to reduce the decay heat further to a level suitable for reprocessing. The SA after required storage are then sent to the reprocessing plant in the shipping cask. The crane handling the shipping cask is designed single failure proof to preclude load fall avoiding damage to the SSSB. The bay is also not designed for aircraft crash, since PFBR site meets the Atomic energy regulatory board stipulation of minimum screening distance of 8 km from major airports (PFBR site is 47 km from the nearest Chennai airport).

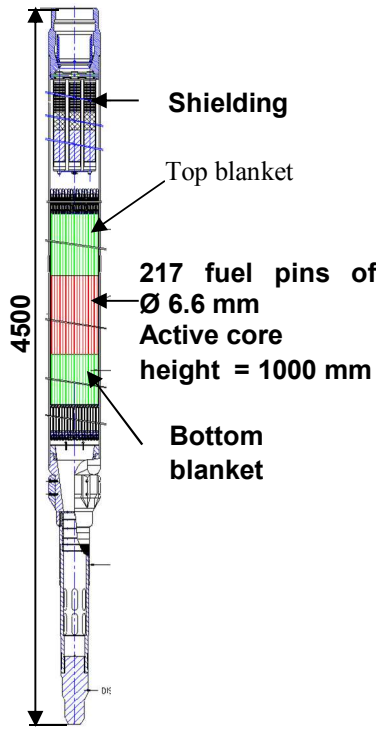
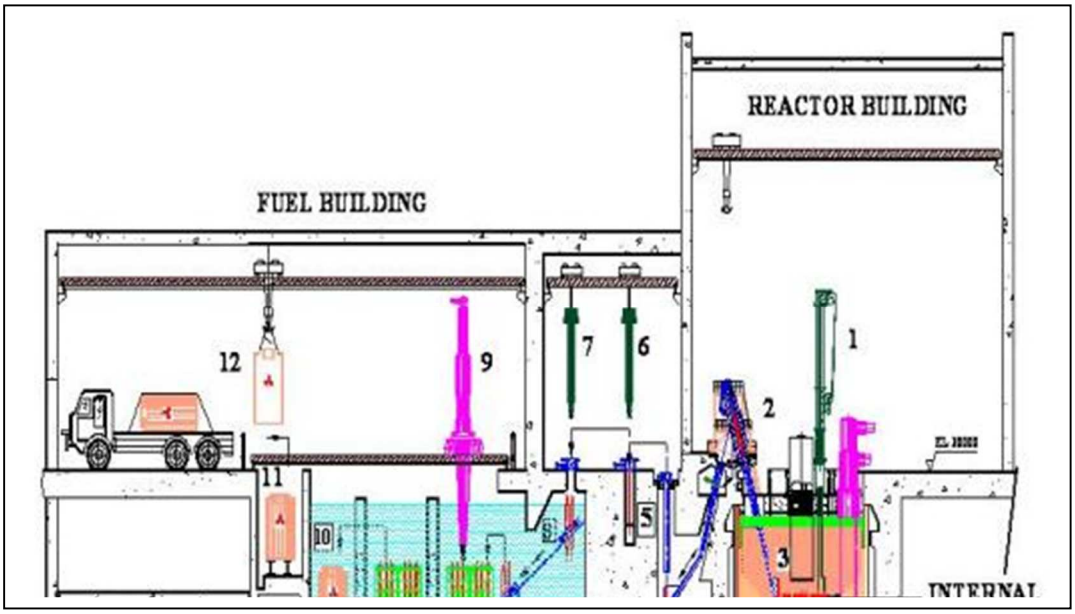


FIG. 3. PFBR fuel subassembly.



1. TRANSFER ARM
2. INCLINED FUEL TRANSFER MACHINE
3. IN VESSEL TRANSFER POSITION
4. EX VESSEL TRANSFER POSITION
5. SPENT SA WASHING FACILITY
6. CELL TRANSFER MACHINE (Fresh SA)
7. CELL TRANSFER MACHINE (Spent SA)
8. UNDER WATER TROLLEY
9. SPENT SA TRANSFER MACHINE
10. SHIPPING CASK LOADING BAY
11. SHIPPING CASK WASHING FACILITY
12. CASK TRANSFER TO REPROCESSING FACILITY

FIG. 4. Spent SA handling scheme in PFBR.

The design objective is to discharge SA at a decay power < 5 kW from the main vessel to the spent SA storage bay and at < 2.5 kW to the reprocessing plant. For the maximum rated subassembly, the decay power is < 3.5 kW after internal storage for one campaign. Detailed thermal hydraulic calculations have been carried out to estimate the temperature of the clad at various stages of handling for both normal and abnormal (stuck conditions).

Fig. 5 shows the details of the spent SA storage bay (SSSB). The bay is an open water pool made of ordinary concrete. Both single and double tank constructions were studied and a single tank lined with stainless steel liner with in-built provisions of leak detection and collection was selected from economical considerations. The bay is divided into four compartments, namely SSSB-1, SSSB-2, cask loading and cask washing. Water pool level is maintained in SSSB-1, SSSB-2 and cask loading which are interconnected by a passage wall.

Provision is made for insertion of a leak tight partition door in between the water filled compartments, whenever the need arises for maintenance intervention. SSSB-1 and SSSB-2 are used to store the spent SA / container with failed SA and cask loading compartment is used to park the spent SA transport cask for loading of SA inside the cask. The inner concrete surface of the bay is lined with SS 304 L liner. The walls / floor are lined with 3 /6 mm thick liner respectively. The spent SA storage racks are bolted on embedded plates anchored to the floor. The total storage capacity of the bay is 711 adequate to meet the requirements of two normal fuel handling campaigns plus one full core emergency core unloading. The water

column of 4.5 m above spent SA head and the concrete thickness provide the required radiological safety for operating personnel. Due to the restricted entry, the design dose rate at accessible areas in the storage bay area is less than 10 μSv/h.

Water in the SSSB is potentially radioactive and to detect any leak through the liner welds, an in-built leak collection and detection arrangement is provided just below each liner weld on both floor & walls (Fig. 6). The leakage collection ducts are made up of carbon steel channel sections, which are covered with a SS 304 L plate at the top to form a box structure. Two grooves are machined on the top surface of the channel and leakage collection holes are drilled at regular intervals just prior to liner plate welding. The leakage collection channels are interconnected to form a number of modules. The different modules are connected to the leakage collection header with redundant leak detection arrangement for each module and finally the leaked water is collected in leakage collection sump. In case of any leak in the bay liner, the water gets collected in the sump, which is collected and finally disposed through the liquid effluent system.

The bay is designed as a seismic category 1 structure (SSE loading). The storage racks store the SA vertical with the geometric pitch between adjacent SA enough to maintain the whole arrangement sub-critical (Calculated K_{eff} is < 0.8 as compared to allowable value of 0.95). Ventilation ducts are provided just above the water level to exhaust the evaporating water to maintain the required temperature and humidity near the bay. During normal operation the maximum decay heat generated from the stored spent SA is 480 kW and under full core unloading condition, the maximum heat generated from the stored spent SA is 1690 kW. A dedicated cooling system is provided to remove the decay heat from the stored SA and to maintain the water temperature within specified limits.

Following are the temperatures of the storage bay water under different conditions.

- Normal operation: 318 K (45°C);
- Full core unloading: 328 K (55°C);
- Loss of cooling (max): 338 K (65°C).

The cooling circuit is provided with centrifugal pumps (3 × 50%) and heat exchangers (3 × 100%). During normal operation, two pumps and two heat exchangers are in service. During full core unloading condition, all pumps and heat exchangers are put in service. The water suction is taken at 0.5 m below the surface of the water level in the bay in each compartment. A make-up water connection is provided to the bay to maintain the level of water in the bay compensating for all possible water losses.

A dedicated purification system is provided to remove suspended, dissolved and solid contamination in the pool, to keep the bay water clear for better visibility required for under water operations and to maintain the purity of water from corrosion considerations. A siphon break arrangement is provided to avoid drainage water from the bay. Detailed thermal hydraulic analysis carried out indicates that the water temperature rises to 65^{o}C in 44 / 12 hours time under normal operation/full core unloading conditions respectively. Hence, no emergency power supply is provided to the equipment of cooling system.Intact SA are stored in direct contact with the bay water while failed SA are stored in leaktight water filled containers to avoid direct contamination of the bay. The subassemblies are canned in containers at special canning facilities provided within the fuel transfer cell. Prior to discharge to the reprocessing plant, the failed subassemblies are once again recanned in nitrogen filled containers.

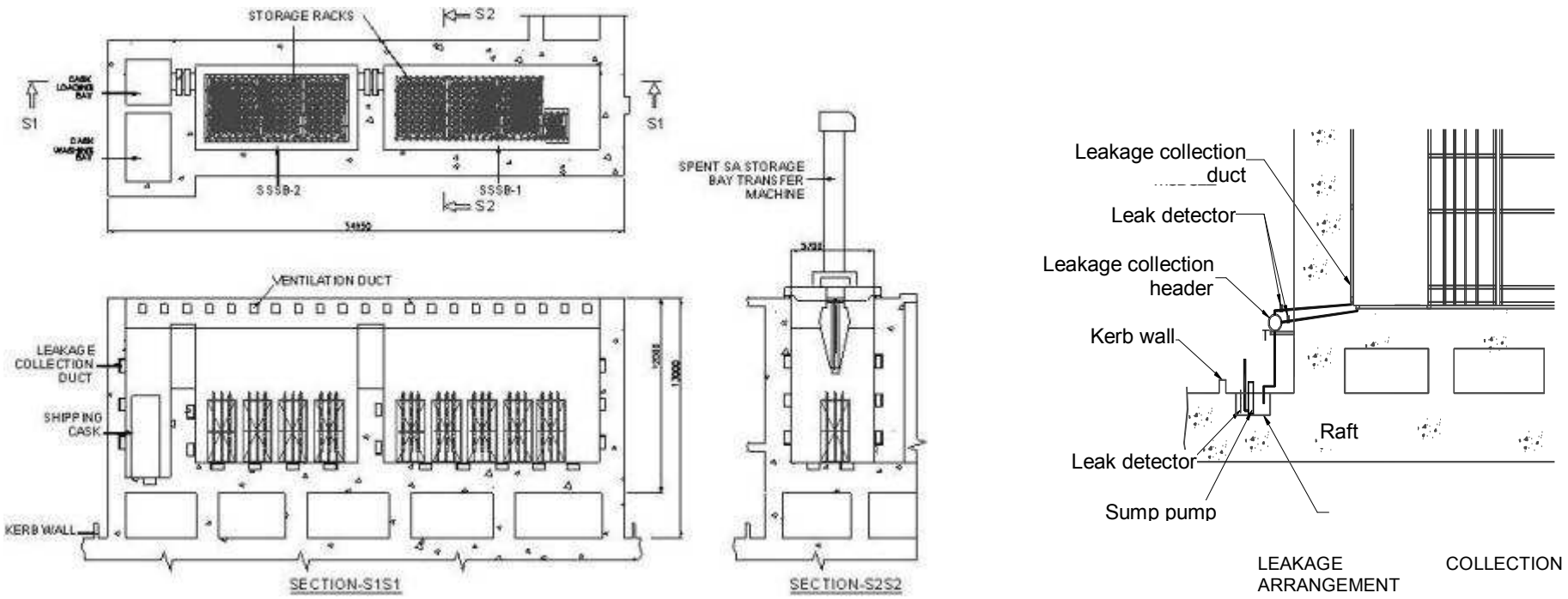


FIG. 5. Spent subassembly storage bay In PFBR.

3.3. Future oxide fuelled FBRs

The next six reactors planned after PFBR are oxide fuelled reactors of twin unit design. The spent fuel washing and spent fuel storage in water as in PFBR will be used except that the spent fuel storage bay will be common for the twin units. The storage capacity will be sized to meet the normal requirements of both the units and provision for emergency core unloading for one unit.

3.4. 1000 MWe metal fuel FBRs

Indian nuclear power programme attaches great importance to the development of metal fuel fast reactors to provide faster growth as metal fuel has potential to give high breeding ratio and shorter doubling time. Both the sodium bonded and helium bonded metal fuel designs are under development. The growth of FBRs is sensitive to the time period fuel is out of the core, i.e., from the time spent fuel is out of the reactor, cooled, reprocessed, refabricated and put back into the reactor. Spent fuel storage in sodium coupled with co-located fast reactor fuel facility based on pyro-reprocessing are the design objectives for future 1000 MWe reactors. The first metal fuel based fast breeder is planned to go operational by the year 2027.

4. FUEL REPROCESSING AND REFABRICATION

In the Indian context, closing the fuel cycle with as low as possible out of pile fissile inventory is an important requirement for increasing the share of nuclear power at the earliest. The development of this complex technology is carried out in four phases [11,12]. The first phase is the developmental phase in which major R&D issues have been addressed [13,14]. The second phase is the construction and operation of a pilot plant, called CORAL (COmpact Reprocessing facility for Advanced fuels in Lead shielded cells). The third phase is construction and operation of Demonstration of Fast reactor Fuel Reprocessing Plant (DFRP) to gain experience in fast reactors fuel reprocessing with high availability factors and plant throughput. Construction and operation of the commercial plant (FRP) for reprocessing of PFBR fuel is the fourth phase.

The CORAL facility, has reprocessed spent mixed carbide fuel from FBTR of different burnups and cooling periods Modified PUREX process is deployed for reprocessing these fuels. Progressive reprocessing exercise of 25, 50, 100 and 155 GWd/t burnup fuel pins beginning with a campaign of very low burnup fuel pins, has been successfully carried out in this facility. The successful reprocessing of the spent fuel with a burnup of more than 155 GWd/t and short cooling period has aided in shaping the fast reactor fuel reprocessing technology of India.

Demonstration Fast reactor fuel Reprocessing Plant (DFRP), is currently under construction and is scheduled for commissioning in 2011. In this plant, regular reprocessing of FBTR spent mixed carbide and oxide fuels will be carried out. After commissioning of DFRP, the reprocessing of FBTR fuel reprocessing will be routinely done in DFRP while CORAL will continue to operate as a research facility for validating alternate process flowsheets. Apart from the closing the fuel cycle of FBTR, DFRP will enable demonstration of the reprocessing of mixed oxide fuel subassemblies of PFBR spent fuel. This will enable optimizing the plant

operating conditions for the regular PFBR fuel reprocessing plant, which is current in design stage.

It is planned to set up Fast Reactor Fuel Cycle Facility (FRFCF) at Kalpakkam, near PFBR site, to close the fuel cycle of PFBR and cater to the reload fuel requirement of PFBR. FRFCF comprises of a spent fuel reprocessing plant, fuel fabrication plant, reprocessed uranium plant, core subassembly plant and waste management plant. FRFCF is designed to meet the annual requirement of 100 nos. of fuel subassemblies and 50 nos. of blanket subassemblies of PFBR. FRFCF is engineered to be expandable to meet the requirements of two more 500 MWe oxide fuel fast reactors proposed to be located at Kalpakkam. FRFCF is targeted for start of construction in 2011 and completion by 2015.

5. CONCLUSION

Fast breeder reactors are expected to play a major role in future towards fulfilling the energy demands thereby providing energy security to the country. The fast reactor programme in India is based on closing the fuel cycle with reprocessing the spent fuel and redeployment as fresh fuel back into the reactor. Spent fuel storage in FBTR is based on dry storage in air cooled argon filled containers while PFBR utilizes a wet water pool storage. For the future FBRs with metallic fuel, sodium storage is envisaged for enhancing capability to handle short cooled fuel. Reprocessing of the spent fuel of FBTR has been established by successful operation of the CORAL facility. A demonstration plant for regular reprocessing of FBTR fuel and demonstration of PFBR fuel is under construction. A co-located fast reactor fuel cycle facility to close the fuel cycle of PFBR is under design and will be completed by 2015. The same facility will be expanded to meet the requirements of two more oxide fuelled reactors.

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LWR SPENT FUEL MANAGEMENT FOR THE SMOOTH DEPLOYMENT OF FBR

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Abstract

Fast breeder reactors (FBR) and FBR fuel cycle are indispensable to prevent the global warming and to secure the long-term energy supply. Commercial FBR expects to be deployed from around 2050 until around 2110 in Japan by the replacement of light water reactors (LWR) after their 60 years life. The FBR deployment needs Pu (MOX) from the LWR-spent fuel (SF) reprocessing. As Japan can posses little excess Pu, its balance control is necessary between LWR-SF management (reprocessing) and FBR deployment. The fuel cycle systems were investigated for the smooth FBR deployment and the effectiveness of proposed flexible system was clarified in this work.

1. INTRODUCTION

Atomic Energy Commission of Japan published the Framework for Nuclear Energy Policy in 2005, which indicates the long-term nuclear energy plan for research, development and utilization in Japan [1]. Before the transition period from LWR to FBR until around 2050, Rokkasho reprocessing plant (RRP) will treat 800 t/y LWR-SF and recovered Pu will be supplied to LWR as mixed oxide (MOX) fuel. As generation rate of LWR-SF in the near future will be about 1200 t/y, interim storage LWR-SF will be about 400 t/y. The 2nd reprocessing plant that will start operation at around 2050 will treat about 1200 t/y LWR-SF to recover Pu (MOX) for the deployment of FBR and to reduce LWR-SF storage amount as shown in Figure 1. Harmonization of LWR and FBR cycles is quite important through the 2nd reprocessing plant for the LWR/FBR (L/F) transition (FBR deployment).

2. FUEL CYCLE SYSTEMS FOR THE L/F TRANSITION

In order to manage the LWR-SF and to deploy FBR smoothly, the flexible fuel cycle initiative (FFCI) system was proposed and compared with reference (ordinarily considered) system for the various FBR deployment scenarios. Figure 1 shows both systems. The reference system consists of full LWR reprocessing to supply FBR-fresh fuel (FF) from LWR-SF and full FBR reprocessing to recycle FBR fuels in FBR cycle. The reference system recovers Pu (+U) from LWR-SF in LWR reprocessing plant and Pu (+U) from FBR-SF in FBR reprocessing plant. On the other hand, the FFCI system consists of uranium removal as the LWR reprocessing to separate most U from LWR-SF, and full FBR reprocessing to recover Pu(+U) from U removal residue (recycle material) and recycle FBR fuels [2]. The U removal can be carried out by the solvent extraction, crystallization, fluorination or precipitation. Temporary storage

of the recycle material (RM) is adopted to control the Pu balance between LWR reprocessing and FBR deployment. If the FBR deployment is smooth, the RM goes directly to FBR reprocessing plant without temporary storage.

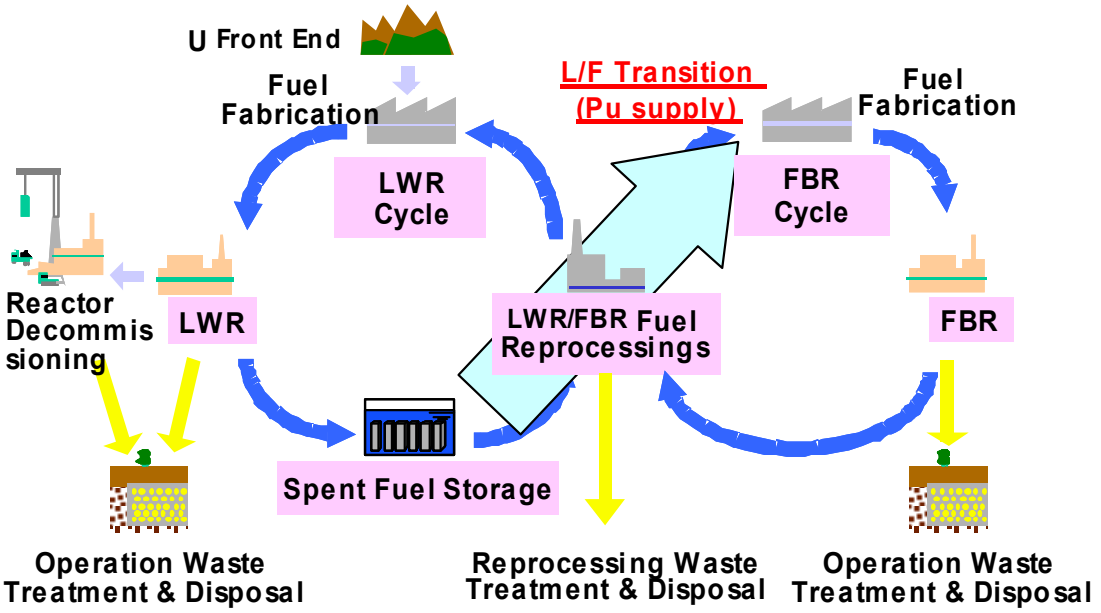

FIG. 1. Transition from LWR cycle to FBR cycle (L/F transition).

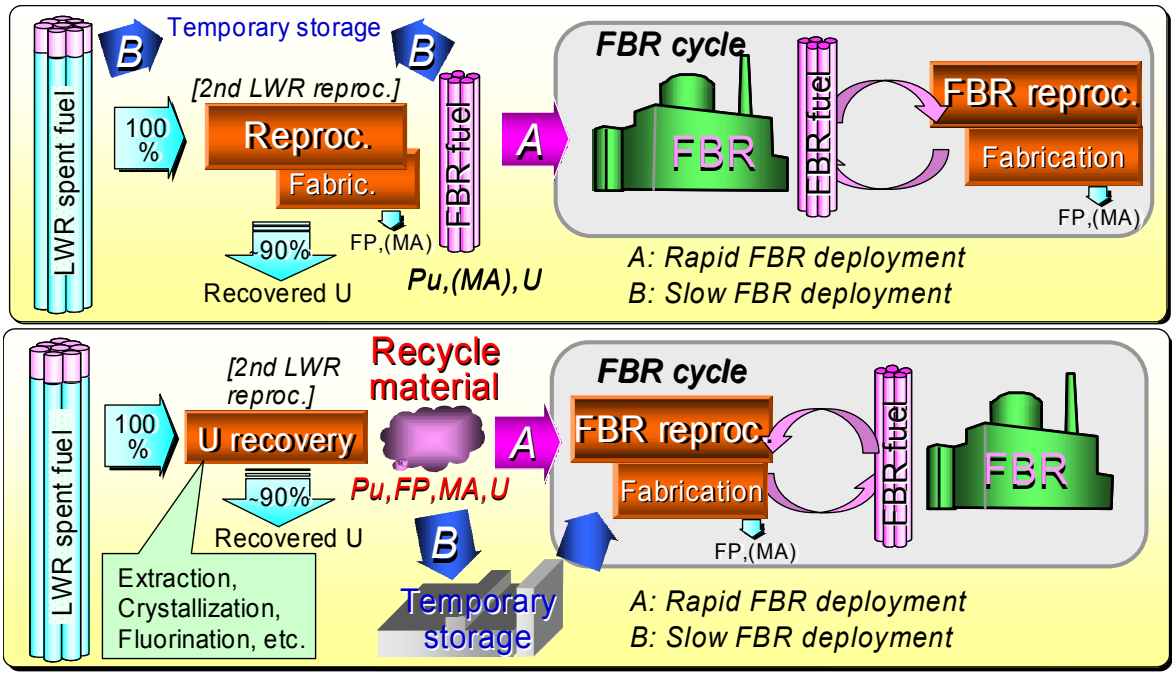

FIG. 2. Reference (above) and FFCI (below) systems for L/F transition.

The capacity of 2nd reprocessing plant would be adjusted for the scheduled FBR deployment rate (rapid FBR deployment), but some material must be temporarily stored in case of

unscheduled stagnation (slow FBR deployment). The material is LWR-SF before LWR reprocessing or FBR-FF (LWR reprocessing product) for the reference system and recycle material for FFCI system. Recovered uranium from LWR reprocessing would be utilized in LWR again after re-enrichment for both systems, which means that the purification process is needed for 2nd LWR reprocessing in both systems.

Partitioning and transmutation technologies of minor actinides (Np, Am, Cm) have been developed to reduce disposal site volume and radio-toxicity of high-level waste in the future fuel cycle system. High-level waste would be generated from LWR and FBR reprocessing plants for the reference system, and only from FBR reprocessing plant for the FFCI system. This means partitioning and transmutation technology will be applied only for FBR reprocessing for the FFCI system.

3. CHARACTERISTICS OF THE SYSTEM

The characteristics of the FFCI system were basically investigated and compared with the reference system for flexibility, economy, proliferation resistance and compatibility of the RM storage system. At first Pu balance analysis was carried out to evaluate the amounts of LWR-SF, LWR reprocessing, FBR reprocessing and Pu storage. The calculation conditions were mainly based on the Nuclear Energy Policy Framework. Reprocessed Pu storage amount (temporary storage material for the reference system) is limited up to 30 t (20 t for fissile Pu), the same value for RRP. People will consider the non-proliferation more strictly in the future. Other temporary storage materials, LWR-SF and recycle material (RM) have high dose rate and high proliferation resistance.

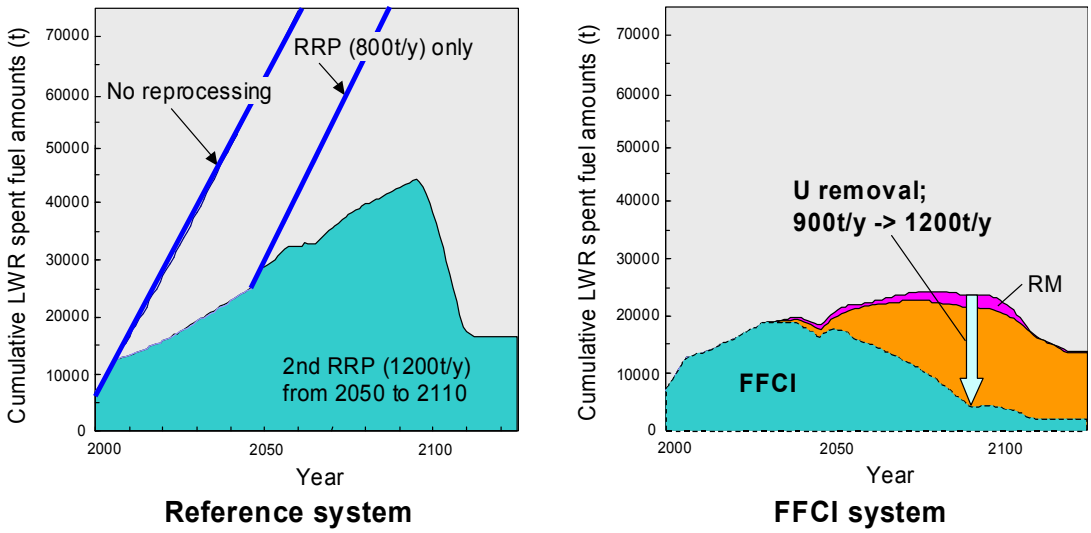

FIG. 3. Amounts of LWR-SF for reference and FFCI systems.

The amounts of LWR-SF are shown in Fig. 3 for the reference and FFCI systems. The left figure also shows the amounts without reprocessing and only RRP. Reprocessing is effective for the reduction of LWR-SF amounts. The RRP will be operated for 40y and the rapid increase of LWR-SF amount after its life, which necessitates the 2nd reprocessing plant with higher capacity for LWR-SF. The deployment rates of FBR indicated in the Framework are rapid (~2 GWe/y) at the beginning and the end of the transition period and slow (~0.5 GWe/y)

between. The operation period of 2nd reprocessing plant and U removal plant is from 2050–2110 in this calculation. Reference system temporarily stores LWR-SF at the slow FBR deployment period and shows large amount of LWR-SF to be stored even after 1200 t/y reprocessing plant operation after 2050. The result for the reference system indicates the needs of much higher reprocessing capacity or 3rd reprocessing plant. FFCI with 1200 t/y U removal can continue the reprocessing even at the slow period and reduce LWR-SF amount steadily after 2050. The result for FFCI system indicates the needs of slightly higher U removal capacity or next U removal plant.

Other important factors were also quantitatively evaluated such as LWR reprocessing, FBR reprocessing and Pu storage and shown in Figs 4, 5 and 6, respectively. The operation period of 2nd reprocessing plant and U removal plants is 2048-2087, and 3rd reprocessing and 2nd U removal plants is 2088-2020 in these calculations.

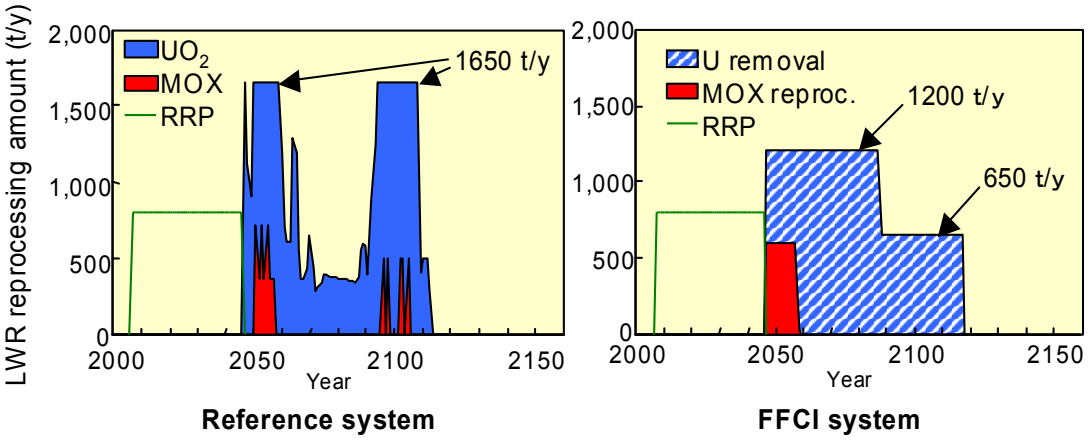

FIG. 4. Amounts of LWR reprocessing for reference and FFCI systems.

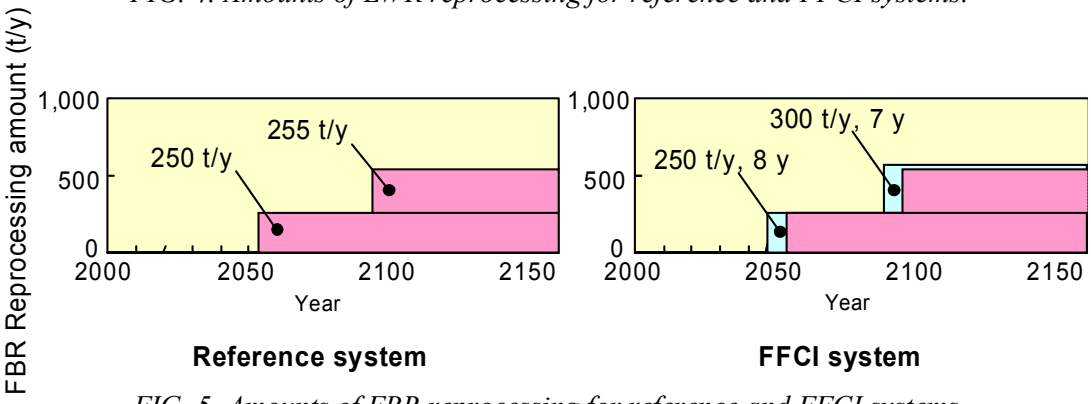

FIG. 5. Amounts of FBR reprocessing for reference and FFCI systems.

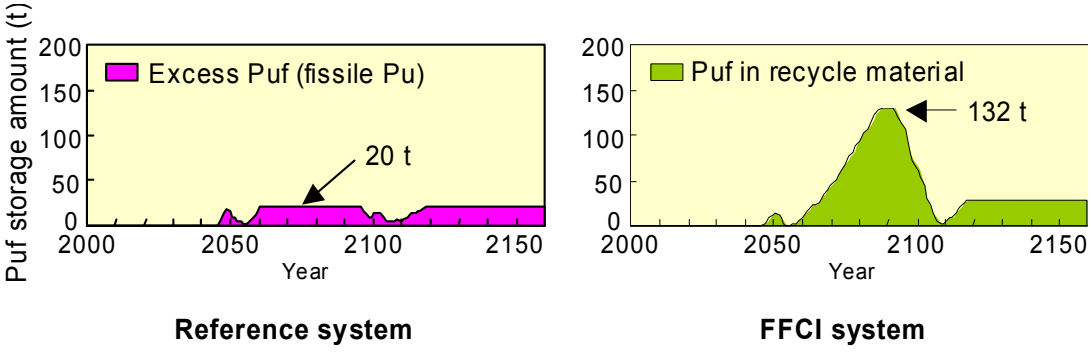

FIG. 6. Amounts of temporary Puf storage for reference and FFCI systems.

Fig. 4 shows the LWR reprocessing amounts for the reference system and U removal amounts for the FFCI system. Red part shows the reprocessing of LWR-MOX-SF that has higher Pu content than LWR-UO_2-SF and suitable for the Pu (MOX) supply to FBR during rapid FBR deployment period. Reference system needs 2nd and 3rd reprocessing plants with 1650 t/y capacity each. FFCI system needs 1200 t/y and 650 t/y capacities for 1st and 2nd U removal plants, respectively.

Fig. 5 shows the FBR reprocessing amounts for both systems, which suggests the 2 step capacity increase at around 2050 and 2090. Reference system needs 250 t/y capacity at around 2055 and 255 t/y around 2095. FFCI system needs earlier construction of FBR reprocessing plants to supply initial Pu (MOX) for FBR deployment (Fig. 2). The capacities are 250 t/y and 300 t/y at around 2047 and 2088, 8 y and 7 y earlier than the reference system, respectively.

Figure 6 shows the Pu storage amounts. Reference system is controlled by the limit of 20 t of fissile Pu, Puf (30 t Pu). FFCI system stores 132t Puf at maximum in the recycle material with other nuclides and high proliferation resistance.

The above results clarify the effectiveness of FFCI system. Although FFCI system needs early construction (and slightly higher capacity) of FBR reprocessing plant, it can reduce LWR-SF efficiently, decrease the capacity and function (no Pu/U recovery) of LWR reprocessing plants, and increase the availability of LWR reprocessing.

The analyses including cost estimations are also conducted for other FBR deployment scenarios, different FBR deployment start time, different FBR deployment rate, and so on [3]. These investigations revealed that the FFCI system could flexibly respond to the various FBR deployment scenarios, reduce the fuel cycle cost about 30% lower than the reference system and have high proliferation resistance for RM compared with Pu product.

The vault storage facility of vitrified high-level waste is applied for RM storage, which needs the heat removal and criticality safety analyses. Simulated RM was prepared using U and stable elements to get the heat conductivity values. The values were found to depend on the powder particle size and the heat conduction analyses clarified that the temperatures at important points of the RM storage facility could be kept under the limited ones [3].

The criticality safety was analysed by the same method for re-criticality for severe accident of FBR under the conditions of coolant loss, core melting and heavy elements (actinides) sedimentation at the bottom. Preliminary analyses clarified that the RM storage facility was also safe for criticality even in the severe accident case [3]. Thus the FFCI system was proved to have good abilities for the flexible FBR deployment.

4. CONCLUSIONS

The FFCI system was proposed for the transition period from LWR to FBR and quantitatively compared with the reference system. The investigations clarified that FFCI system can reduce the LWR-SF amount effectively, must construct the FBR reprocessing plants several years earlier, can decrease the capacity of future LWR reprocessing (U removal) plants, and reduce the fuel cycle cost.

ACKNOWLEDGEMENT

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