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Very High Burn-ups in Light Water Reactors

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FOREWORD

At the 2003 meeting of the NEA Nuclear Science Committee, it was decided to establish an expert group that would carry out a technical assessment of very high burn-up fuel cycles in current light water reactors (LWRs). Specifically, the average discharge burn-ups considered would span the range between 60 GWd/t (the limit of current experience) and ~100 GWd/t. The scope would include assessments of the implications for enrichment, fuel (UO_2) fabrication, in-core fuel management, thermal-hydraulics, fuel performance, spent fuel source terms and fuel cycle economics.

The first meeting of the expert group was held in January 2004. It was decided to produce a state-of-the-art report which would provide fuel vendors and utilities with a guide as to the technical feasibility of extending LWR discharge burn-ups well beyond current levels and identify the research required to support very high burn-ups.

Specific chapters of the report cover:

- the history of LWR fuel burn-up evolution, fuel design and utility demands;
- issues needing to be addressed to attain very high burn-ups;
- the implications for fuel fabrication of high initial fuel enrichments;
- the impact on in-core fuel management;
- economic implications for different parts of the fuel cycle;
- research and development needs.

It should be noted that this study seeks to identify the outstanding scientific and technical issues associated with high burn-up, but does not attempt to propose solutions to the difficulties identified. The report will be of interest to both nuclear utilities and fuel cycle specialists.

Acknowledgments

The expert group would especially like to thank Gyorgy Hegyi and Czaba Maraczy from the KFKI Atomic Energy Research Institute in Hungary for their contributions to core design and economics calculations for VVER-440.

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LIST OF ABBREVIATIONS

ANL	Argonne National Laboratory
ATWS	Anticipated transient without scram
BOC	Beginning of cycle
BUCCX	Burn-up credit critical experiment facility
BWR	Boiling water reactor
CAs	Control assemblies (moveable assemblies in context of VVER-440)
CILC	Crud-induced localised corrosion
CPR	Critical power ratio (BWR)
DNBR	Departure from nucleate boiling ratio
E110	Low-corrosion zirconium alloy
ECCS	Emergency core cooling system
EOC	End of cycle
EPMA	Electron probe micro analysis
EPRI	Electric Power Research Institute
FISPIN	Nuclear inventory code used in the UK
GWd	Giga-Watt day
GWd/t	Giga-Watt day per tonne (initial heavy metal)
H/U	Hydrogen/uranium volume ratio
HBS	High burn-up structure
HEU	High-enriched uranium
ICRP	International Commission on Radiological Protection
IFBA	Integral fuel burnable poison (ZrB_2 coating)
ILW	Intermediate-level waste
IRI	Incomplete (control) rod insertion
JAERI	Japan Atomic Energy Research Institute
LEU	Low-enriched uranium
LHGR	Linear heat generation rate (units kW/m)
LLLP	Low-leakage loading pattern
LOCA	Loss of coolant accident
LTA	Lead test assembly
LWR	Light water reactor
M5	Low-corrosion zirconium alloy
MIMAS	Belgian/French process for MOX fuel fabrication
MLHGR	Maximum linear heat generation rate (units kW/m)
MOX	Mixed-oxide $(U,Pu)O_2$
MOX/EUS	Mixed-oxide/enriched uranium support
MTC	Moderator temperature coefficient
NERI	Nuclear Energy Research Initiative
ORIGEN	Nuclear inventory code used in US and other countries

PCA-2b	Low-corrosion zirconium alloy
PCI	Pellet-clad interaction
pcm	Per cent milli – unit of reactivity corresponding to 10^{-5} $\Delta k/k$
PCIOMR	Pre-conditioning interim operating management recommendations
PIRT	Phenomenon identification and ranking tables
PWR	Pressurised water reactor
RCCA	Reactivity control cluster assembly
RIA	Reactivity insertion accident
SCC	Stress corrosion cracking
SDM	Shutdown margin
SIMS	Secondary ion mass spectrometry
SNL	Latest series Russian designed pressurised water reactor
SWU	Separative work unit
USDOE	United States Department of Energy
USNRC	United States Nuclear Regulatory Commission
VHLW	Vitrified high-level waste
VVER-1000	Late series Russian-designed pressurised water reactor
VVER-440	Early series Russian-designed pressurised water reactor
WA	Working assemblies (non-moveable assemblies in the context of VVER-440 reactor)
XRF	X-ray fluorescence
ZIRLO	Low-corrosion zirconium alloy marketed by Westinghouse
PUREX	Plutonium uranium reduction extraction
UREX	Uranium reduction extraction
β_{eff}	Effective delayed neutron fraction

GLOSSARY

Assembly

In the context of a LWR, a fuel assembly is a mechanical structure containing a collection of fuel rods joined together into a single unit to facilitate loading and unloading. An LWR fuel assembly variously comprises the fuel rods, the structural support grids, top and bottom nozzles, control rod guide tubes, instrument tubes, channel box, burnable poison rods, water channels, mixing grids, etc. Other commonly employed terms (the use of which differs between different reactor types and countries) include fuel bundle and fuel element. Fuel assembly is the preferred nomenclature in this report, and will be used for BWR, PWR and VVER.

Availability factor

The availability factor for a power plant over a period of time, usually expressed as a percentage, is the fraction of the time for which the plant is available to generate power. Availability factors are normally quoted annually.

Average assembly burn-up

The burn-up averaged over a single fuel assembly at discharge is the average assembly burn-up. This is the most useful characterisation of discharge burn-up as the average assembly burn-up effectively defines the demand implied by a specific fuel management scheme. In this report the average discharge burn-up is the principal burn-up unit which will be used; thus, it is the average discharge burn-up which is referred to when it is stated that the burn-up range examined will be from 60 to 100 GWd/t.

Batch fraction

The batch fraction is the fraction of fuel discharged during a refuelling outage. LWRs are batch-loaded reactors which are shutdown periodically for refuelling and a number of fuel assemblies are discharged and replaced by fresh fuel. The batch fraction is defined as the number of fresh fuel assemblies m divided by the number of fuel assemblies in the core M:

$$\text{Batch fraction} = m/M$$

If the batch fraction corresponds to a simple fraction, such as 1/2, 1/3, 1/4 or 1/5, all the fuel loaded fresh in a particular refuelling is discharged at the same time after 2, 3, 4 or 5 fuel cycles respectively. The batch fraction need not necessarily correspond to a rational fraction, in which case a *split batch* scheme applies and a proportion of fuel assemblies are discharged after an integral number $n - 1$ of cycles, whereas the remainder are irradiated for n cycles [where $n = int(M/m)$].

Capacity factor (or load factor)

The capacity factor, usually expressed as a percentage, is the integrated electrical output of the plant over a period of time divided by the output that the plant would provide in the same time if it were able to provide 100% of its nominal power output for the entire time period. In practice power plants are sometimes unavailable for power production due to a combination of refuelling and maintenance outages, i.e. unscheduled breakdowns. When a plant is available, there may not be a demand for 100%

of its power output or there may be power restrictions in place that prevent the full power output being achieved. A combination of these two factors leads to typical capacity factors for LWRs of approximately 85%, with the best-performing units able to achieve over 90%. Capacity or load factor is normally quoted annually.

Discharge burn-up

The discharge burn-up of nuclear fuel is usually defined as the thermal energy output during the lifetime of the fuel divided by the initial mass of heavy metal (denoted HMi). The unit of burn-up (or thermal energy production) is Giga-Watt days per tonne of initial heavy metal. One GWd corresponds to an energy of 1E12 × 24 × 60 × 60 Joules = 8.64E13 Joules, and is a more convenient unit than Joules. The energy output is related to the initial heavy metal mass as a fixed reference point; the actual heavy metal mass varies with burn-up as the fissile elements are converted to fission products. Other units commonly used include MWd/tHM (1 GWd/tHM = 1000 MWd/tHM) and MWd/kgHM (1 GWd/tHM = 1 MWd/kgHM). A burn-up of 1 GWd/tHM corresponds roughly to a fission fraction of 0.1%; therefore fuel at 50 GWd/tHM will have fissioned approximately 5% of the heavy atoms.

In this report GWd/tHM is abbreviated to GWd/t, it being understood that it is burn-up relative to initial heavy metal.

Lead rod burn-up

This is the average burn-up of the lead rod within an assembly at discharge. The lead rod burn-up will always exceed the average assembly burn-up.

High burn-up LWR cladding

Standard Zircaloy-2 and Zircaloy-4 cladding tends to be susceptible to water-side corrosion at local burn-ups in excess of ~40 GWd/t. LWR fuel vendors have developed advanced Zircaloy cladding materials (e.g. M5, E110, PCA-2b and ZIRLO), which are corrosion resistant up to significantly higher local burn-ups.

Pellet or local burn-up

This is the highest burn-up *locally* in an individual fuel rod in an assembly, corresponding to the single pellet with the highest burn-up at discharge. The peak pellet burn-up will always exceed the lead rod and average assembly burn-ups. The peak pellet burn-up would usually be expected to occur on the lead burn-up rod in the assembly.

Levelised fuel cycle cost

The levelised fuel cycle cost is calculated by summing all the fuel cycle costs incurred over the operating lifetime of a reactor and dividing by the electricity output. The most convenient units are $/MWh. The levelised cost may be undiscounted, in which no monetary discount factor is applied to either the costs or the electrical output to allow for the time value of money. Alternatively, time discounted costs and electrical outputs are used to determine the discounted levelised cost.

Reload fraction

In a batch-loaded reactor, a certain fraction of the fuel assemblies are discharged during refuelling operations and replaced with fresh assemblies. The number of fresh assemblies loaded divided by the total number of assemblies in the core is called the reload fraction. This is often assumed to be a constant once the reactor reaches equilibrium operation, even though in practice there may often be variations even for a nominal equilibrium reload.

Split batch loading scheme

See *Batch fraction*.

Zircaloy-2

Zirconium-based alloy containing 1.2-1.7% Sn, 0.07-0.20% Fe, 0.05-0.15% Cr and 0.03-0.04% Ni. Zircaloy-2 is used as the cladding for BWR and CANDU fuel rods, as well as for other reactor applications such as the pressure tubes in CANDU reactors.

Zircaloy-4

Zirconium-based alloy used for PWR fuel assembly components such as fuel cladding tubes, fuel assembly grids, control rod guide tubes and fuel assembly shrouds.

EXECUTIVE SUMMARY

Historically, average discharge burn-ups of light water reactors (LWRs) have steadily increased with time as experience has accumulated and as technological developments have progressed. One of the main benefits has been to decrease the fuel cycle cost. Another benefit has been the increased operational flexibility that high burn-ups allow. Average discharges are currently in the region of 50 GWd/t and for burn-ups up to and at least a little beyond this figure, there is a clear economic incentive to continue the trend. But the question for which there is no definitive answer at present is whether this historic trend will continue indefinitely or whether there will be a technological limit to PWR burn-ups. The question is very important for utilities and fuel fabricators, as it is probably the single most important technical unknown affecting the future LWR fuel cycle, and is thus the subject of this expert group study.

This study resulted from a discussion at the June 2002 meeting of the OECD/NEA Nuclear Science Committee, at which it was first suggested to set up a project to address very high burn-ups (specifically average discharge burn-ups in the range 60-100 GWd/t). The outcome was the setting-up in 2003 of the Expert Group on Very High Burn-ups in LWRs, which was charged with the single task of delivering a state-of-the-art report on high burn-ups in LWRs. It was felt that the report should concentrate on LWRs because that is where the bulk of experience and knowledge resides, but much of the analysis will also be applicable to other reactor types, even if not all the details are transferable.

It was decided to limit the scope of the study to encompass only conventional LWR fuel assembly designs, which conform to the assembly geometries used in present LWRs and that use conventional oxide fuels. It is recognised that present-day assembly designs will need to evolve to some extent to accommodate the very high burn-ups considered here (through the use of different cladding materials, for example, or by adjusting plenum lengths), but substantial changes to fuel designs (such as change of fuel material away from oxide, or major changes in geometry) are not considered.

It is unclear whether the almost constant rate of increase of historical region average discharge burn-ups will continue in the future. It is conceivable that there may be a lack of incentive for increasing burn-ups much beyond 60 GWd/t (for instance, due to the need for enrichments higher than the current 5.0 w/o criticality limit) or that technological barriers may intervene. The incentives may differ between countries and individual utilities such that even if there are no technological limitations not all utilities may choose to adopt very high burn-ups. The purpose of the report is to systematically explore the incentives and technological issues as far as current knowledge allows. It is by no means considered inevitable that utilities will adopt very high burn-ups and there is the possibility that the historic burn-up evolution may eventually cease. This is ultimately a decision for individual utilities and governments and it is hoped that this report will assist them in making informed judgements.

The export group report systematically reviews the implications of very high burn-ups on all technological aspects of the LWR fuel cycle. Within the constraints of the study, the analysis is necessarily restricted and is mostly limited to qualitative assessments with some quantitative analysis where practical. A number of technological issues are identified as being key to achieving very high burn-ups and various recommendations for facilitating progress are made. An assessment of the fuel cycle economics is undertaken, with some specific examples analysed quantitatively. The economics assessment identifies circumstances under which very high burn-ups continue to provide an economic incentive.

Chapter 1

INTRODUCTION

The history of LWRs has seen a gradual progression toward higher and higher fuel discharge burn-ups. For PWRs, VVERs and BWRs, current discharge burn-ups are approaching double those of 30 years ago. PWR burn-ups are now typically close to 50 GWd/t, and VVERs and BWRs are not far behind. The reasons for this gradual evolution of burn-ups are first that the levelised fuel cycle cost unambiguously decreases with increasing discharge burn-up for burn-ups up to 50 GWd/t. Second, utilities benefit operationally from high burn-ups because they increase operational flexibility. Third, at high burn-ups the volume of fuel discharged is reduced, which may have a monetary and/or an operational advantage (such as a reduction or even the elimination of additional spent fuel storage capacity). The attainment of higher burn-ups has been possible because of a combination of accumulated operational experience, improved fuel designs and fuel materials developed by fuel vendors and assisted by improved understanding of fuel behaviour. Utilities currently benefit from the considerable past investments the new fuel designs and materials represent.

For a given energy output, the mass of fresh fuel loaded and the mass of spent fuel discharged (both normally quoted in terms of initial heavy metal mass) are inversely proportional to the average discharge burn-up; doubling the average discharge burn-up halves the masses of fresh and spent fuel for a given energy output. This provides a direct economic benefit for those fuel cycle cost components that scale with fuel throughput, specifically fuel fabrication costs, fuel assembly component costs, fresh fuel transport costs and spent fuel storage, transport and processing costs. However, for uranium fuel the overall fuel cycle cost is affected by the higher initial enrichment and there is a fine balance between these various opposing effects that can result in different behaviour with increasing burn-up. In some circumstances that can be envisaged, overall fuel cycle costs could continue to decrease even at burn-ups very much higher than current values, whereas in other circumstances there may be an optimum economic burn-up.

High licensed burn-ups increase operational flexibility by allowing the utility more scope to optimise the fuel management scheme with less likelihood of being constrained by burn-up limitations. Higher burn-ups might allow a utility to lower the batch fraction, reducing the number of fresh fuel assemblies needed at each reload, thereby benefiting from the decreased uranium ore and enrichment requirements that multi-batch core loadings allow. In addition, there are potential resource and economic savings if the operator is able to specify an integral inverse batch fraction (i.e. exactly one-third, one-quarter or one-fifth of the core replaced in every cycle), so that all fuel assemblies in a batch are irradiated for the same number of cycles (split batch schemes wherein some assemblies are irradiated for n cycles, with others only irradiated for $n - 1$ cycles are less resource efficient because those assemblies discharged after $n - 1$ cycles are under-utilised). Another example might be increased scope to reuse previously irradiated fuel assemblies to reduce the fresh fuel requirement. For some utilities, higher burn-ups might allow cycle lengths to be extended and economic gains to be attained through the higher capacity factors thereby achievable (by extending the time between refuelling outages, capacity loss due to refuelling operations is reduced).

In addition to the direct economic benefit from reducing spent fuel arisings, there may be indirect benefits such as effective augmentation of spent fuel storage capacity. Thus a utility with a spent fuel store sufficient for a given number of years of operation might be able to operate for longer before having to invest in new storage capacity. This is potentially a very powerful incentive even if, as is theoretically possible, a utility is charged for spent fuel management according to cumulative energy output rather than spent fuel mass/volume. A utility might benefit indirectly from having a high burn-up capability, even if it has no plans to extend burn-ups itself, in that fuel capable of operating reliably at extended burn-ups might be expected to offer increased reliability at lower burn-ups. Since LWR utilities put fuel reliability at the top of their list of fuel requirements, this spin-off benefit should not be overlooked.

A natural question to ask is whether the historic trend of increasing burn-ups will continue. The issue is complex, and this report cannot assume to provide a definitive answer. Rather, it seeks to impart partial answers and guidance where possible. More importantly, it attempts to compile a comprehensive list of all the specific technical issues that will need to be addressed and to map out the limits of our understanding. LWRs are expected to be the mainstay of the world's commercial nuclear power supply for at least 20 or 30 years, and the question of how far discharge burn-ups will progress is important for understanding how the industry will need to evolve.

This report resulted from a discussion at the June 2002 meeting of the OECD/NEA Nuclear Science Committee, at which it was first suggested to set up a project specifically to address high burn-ups. The outcome was the setting-up in 2003 of the Expert Group on Very High Burn-ups in LWRs, which was charged with the single task of delivering a state-of-the-art report on high burn-ups in LWRs. It was felt that the report should concentrate on LWRs because that is where the bulk of experience and knowledge resides, but much of the analysis will also be applicable to other reactor types, even if not all the details are transferable.

For the purposes of this report, very high discharge burn-ups are defined as region average burn-ups from 60-100 GWd/t. The region average burn-up is the discharge burn-up averaged over an entire batch of fuel assemblies discharged at the same time. The 60-100 GWd/t range takes the analysis largely beyond the range of current LWR experience (although there is some experience of experimental test rods having been taken into this burn-up range) and looks very much to the future of LWRs. It is not intended that this range of average burn-ups reflect any specific target. The lower limit of 60 GWd/t was chosen as a convenient round figure that is just outside the boundary of current operational experience (LWR region average burn-ups are currently close to 50 GWd/t, although a small number of individual assemblies have been irradiated up to 60 GWd/t and more). The upper limit was chosen as the highest value for which present knowledge can reasonably be extrapolated.

It is not at all clear whether the almost constant rate of increase of historical region average discharge burn-ups will continue in the future. It is conceivable that there may be a lack of incentive for increasing burn-ups much beyond 60 GWd/t or that technological barriers may intervene. The incentives may differ between countries and individual utilities such that even no technological limitations exist, not all utilities may choose to adopt very high burn-ups. The purpose of the report is to systematically explore the incentives and technological issues as far as current knowledge allows. It is by no means considered inevitable that utilities will adopt very high burn-ups and there is the possibility that the historic burn-up evolution will eventually cease. This is ultimately a decision for individual utilities and it is hoped that this report will assist them to make an informed judgement.

It was decided to limit the scope of the study to encompass only conventional LWR fuel assembly designs, which conform to the assembly geometries used in present LWRs and that use conventional oxide fuels. It is recognised that present-day assembly designs will need to evolve to some extent to

accommodate the very high burn-ups considered here (through the use of different cladding materials, for example, or by adjusting plenum lengths), but substantial changes to fuel designs (such as change of fuel material away from oxide, or major changes in geometry) are not considered. However, the potential impact of very high burn-ups for mixed-oxide (MOX) fuel is addressed.

The report is structured as follows: Chapter 2 summarises the history of LWR burn-up evolution. Chapter 3 lists the technical issues that arise with very high burn-ups in LWRs. The technical aspects are then developed further in Chapters 4 (fuel cycle issues), Chapter 5 (fuel management, core design and core safety/dynamics) and Chapter 6 (operational, safety and licensing issues). Chapter 7 analyses the economics of very high burn-ups. Finally, Chapter 8 discusses research and development needs.

Chapter 2

HISTORY OF LWR BURN-UP EVOLUTION, FUEL DESIGN AND UTILITY DEMANDS

Light water reactors (LWRs) are currently the most prevalent reactor type used for power production. Many LWRs operating today will be operational well into the 21st century and there is every prospect that LWRs will form the mainstream of the widely anticipated resurgence of nuclear power. LWR technology has reached a very high degree of maturity, and continues to evolve. This chapter reviews the historical evolution of LWR fuel designs and region average discharge burn-ups.

2.1 Introduction

The LWRs that began commercial operation in the United States during the late 1950s and early 1960s became the prototypes of the current generation of LWRs used throughout the world today. These nuclear plants are not only the most widely used, but also the most technologically advanced from an engineering standpoint, with advanced versions of both the boiling water reactor (BWR) and pressurised water reactor (PWR) having come online within the past decade. The Russian-designed PWRs (VVER-440 and VVER-1000), though different in detail, have much in common with Western PWRs and have developed in parallel with them. All three reactor types have been developed to a very high degree of maturity and have demonstrated a high level of safety and reliability.

As the plant designs have evolved, so has LWR fuel technology. During the early stages of LWR development, fuel was initially of limited reliability because of various failure mechanisms that were experienced. As the cause and effect of these failure mechanisms came to be understood, changes were made in the fuel design, fabrication processes and in-reactor operating procedures that led to improved fuel reliability, higher availability, longer operating cycles with shorter and less frequent refuelling outages, steadily increasing burn-up levels and lower fuel cycle costs. With deregulation of the power generation market during the 1990s, there was continued pressure for fuel designs to meet ever more demanding specifications and operating conditions, including power plant up-rate, changing water chemistry environments (to help extend plant lifetimes) and even higher burn-ups to achieve better fuel cycle economy. New cladding and structural materials with high corrosion resistance combined with evolutionary fuel assembly design enhancements to attain higher thermal margins, have been key to maintaining good fuel reliability under very challenging circumstances. This chapter briefly reviews the history of fuel design improvements for BWR, PWR and VVER in the context of the utility's needs and demands.

2.2 BWR fuel design evolution

The BWR fuel assembly provides a flexible platform to allow changes when the need arises. Advances in fuel manufacturing capabilities and technologies have enabled fuel vendors to steadily increase the number of fuel rods and reduce fuel rod diameter within the same channel box dimensions in order to meet the utility's needs, for example, operation at higher power densities, higher fuel burn-up, more flexible operating strategies, etc.

The fuel assembly initially adopted in the 1960s was composed of a 6×6 array of fuel rods. From the very beginning, zirconium alloys (e.g. Zircaloy-2) were chosen as the cladding material, although stainless steel was used for a period of time because of early fuel failures experienced with zirconium alloy cladding. The fuel failure mechanism was later proved to be a result of fluorine contamination (originating from the UF_6 used in the enrichment process) causing a harmful interaction with the Zircaloy cladding and improved control of fluorine eliminated this mechanism as a cause of failures. Since that time, Zircaloy-2 became the universal choice of cladding for BWRs and continues to be used today.

Evolutionary design development and improvements in vendor manufacturing capability led to the introduction of a 7×7 fuel assembly to reduce the design maximum linear heat generation rate (MLHGR). The adoption of urania/gadolinia (UO_2/Gd_2O_3) pellets for reactivity control and a re-crystallised annealing process for the cladding became standard design features of the 7×7 fuel. In the 1970s an effort was made to further reduce the design MLHGR, leading to the introduction of an 8×8 fuel assembly. The MLHGR limit existing at that time, 56.4 kW/m, was reduced to the current limit of 40.8 kW/m. The centre temperature of a fuel rod is largely determined by the MLHGR, and spreading the heat generation load over a larger number of fuel rods reduces the peak centre temperature, thereby increasing operating margins.

With regard to the fuel failure mechanisms during this period, the predominant ones were localised internal hydriding of the Zircaloy cladding and pellet-clad interaction (PCI). Excessive moisture in the ceramic fuel pellets resulted in hydriding of the cladding, eventually leading to a loss of cladding integrity. Improved fabrication techniques, careful quality control and the introduction of a “getter” in the fuel rod plenum to absorb moisture have essentially eliminated clad hydriding as a failure mechanism. The reduction of average linear heat generation rate (LHGR) by the introduction of the 8×8 design proved to be an effective countermeasure against PCI, particularly when combined with the imposition of the pre-conditioning interim operating management recommendations (PCIOMRs). PCIOMRs imposed a limit on the rate at which the LHGR of exposed fuel was allowed to increase and restricted the LHGR within a specified upper limit. These pre-conditioning limits significantly reduced the PCI failure rate in the 1970s, but at a substantial fuel cycle cost penalty.

An innovative solution, involving the co-extrusion of a high-purity zirconium layer on the inner surface of the Zircaloy-2 cladding, was introduced during the first half of the 1980s to mitigate susceptibility to stress corrosion cracking (SCC). This new “barrier” feature, so named due to its accepted PCI resistance, allowed plants to regain some flexibility in power manoeuvring and to significantly improve their capacity factors, since the restrictive PCIOMRs could be relaxed or eliminated entirely. In the early 1980s the average fuel burn-up was under 30 GWd/t, but increased rapidly after that, achieving a value in excess of 45 GWd/t today. Utility needs for further fuel cycle cost reductions are likely to push the average fuel burn-up to at least 50 GWd/t in the near term. Of course, an increase in fuel enrichment was needed to achieve these higher burn-up values. This adversely affected the neutron spectrum, leading to a slight deterioration of neutron economy and transient characteristics. Therefore an increase in the water-to-fuel ratio was needed in the channel box region to offset these negative effects. This lead to the introduction of a 9×9 fuel design, which maintained the necessary fuel inventory and incorporated a large central water rod to achieve an appropriate moderator inventory. This also had the benefit of further reducing the average LHGR. Also, as higher fuel burn-up with increased enrichment leads to a degradation in the thermal-hydraulic capabilities of the fuel assembly, further design enhancements such as a high performance spacer grid for improved critical power ratio (CPR), “part-length” fuel rods, and axial zoning of gadolinia and enrichment to improve the shutdown margin (SDM) were adopted. The next BWR fuel design innovation was a 10×10 fuel assembly, first developed and introduced in Scandinavian reactors. Most

major fuel vendors today offer their own unique 10×10 design which enables utilities to up-rate reactor power levels by as much as 20%, although the typical range is 5-10%. Since the current average fuel enrichment in BWRs of around 4 w/o is sufficient to give average burn-up levels of 50 GWd/t, the potential exists for BWR fuels to theoretically achieve even higher burn-ups while staying within the current 5 w/o enrichment limit.

In the latter half of the 1970s and early 1980s a number of high-power density US BWR-4 plants experienced crud-induced localised corrosion (CILC), where Zircaloy-2 cladding susceptible to nodular corrosion provided a mechanism for the deposition of metallic impurities, mainly copper and zinc, onto the cladding surfaces. The copper and zinc originated from the corrosion and erosion of the brass condenser tubes. Most of the affected plants replaced their brass condenser tubes with tubes made of stainless steel or titanium, or installed pre-filters to more effectively remove the soluble species from the feedwater. In addition, vendors made adjustments to the manufacturing process of the incoming Zircaloy material to improve its nodular corrosion resistance. Such BWR fuel failure experiences clearly indicate the need to maintain optimum water chemistry environments to avoid excessive amounts of impurities that may be harmful to the fuel. A more recent fuel failure mechanism observed in BWRs is that of debris fretting. This is being addressed by improved maintenance procedures during refuelling and maintenance outages and the introduction of “debris-resistant” lower tie plate designs.

2.3 PWR fuel design evolution

PWR fuel assemblies consist of a square array of fuel rods supported by a skeleton comprising the top and bottom nozzles, reactivity cluster control assembly (RCCA) guide tubes and spacer grids. PWR fuel assembly designs are more closely integrated with the reactor design since the RCCAs insert into the guide tubes, thereby tightly constraining the assembly design (changing the lattice spacing in a PWR fuel assembly would be incompatible with the pressure vessel internal components, including the RCCAs and with the control rod housings in the pressure vessel head unit). These structural constraints have limited the degree to which vendors have been able to modify the original design concept for PWR fuel. Various fuel rod arrays, ranging from 14×14 to a proposed 19×19 array and various assembly lengths, have been adopted according to plant type.

The evolution of the PWR fuel design has focused on improvements in material properties, thermal-hydraulic performance and fabrication processes. With regard to past fuel failure mechanisms, internal cladding corrosion by hydriding and cladding creep collapse were prevalent during the 1960s and 70s. As with BWRs, hydriding was caused by excessive moisture in the fuel rod, which was mitigated by improved moisture control in fabrication. Cladding creep collapse was caused by excessive fuel densification leading to opening of gaps between pellets. Measures to improve fuel density control and the ability of the cladding to resist creep deformation have successfully eliminated cladding creep collapse. Another early failure mechanism was attributed to a phenomenon called “baffle jetting” in which jets of water impinged on fuel rods located near the core periphery, resulting in rod vibration and eventually failure. Baffle jetting was limited to plants in which a high pressure differential existed across the core baffles in the inlet coolant stream. Modification of the vessel internal structures were made which significantly decreased the cross-flow at the baffle locations, effectively eliminating this failure mechanism.

Debris fretting became a major cause of fuel failures in the 1980s and into the 1990s. Countermeasures similar to those adopted for BWRs were implemented. Improved foreign material exclusion procedures were followed by plant personnel and fuel assembly designs were fitted with debris filters in the bottom nozzle. Consequently, this fuel failure mechanism has become less prevalent.

During the 1980s, increasing emphasis on fuel cycle cost reduction led to higher burn-up fuel designs with correspondingly higher uranium enrichments. Utilities decided to operate with extended cycles (three-batch, 18-month cycles or four-batch annual cycles) that necessitated enrichments in excess of 4 w/o ^{235}U. At these higher burn-ups, fission gas release and the internal pressure criteria became limiting and fuel design changes, such as reduced fill pressure and internal volume increase, were introduced as a response. Cladding materials with increased corrosion resistance were demanded in the 1990s to achieve even higher fuel burn-up while maintaining fuel reliability. Nb-containing zirconium alloys introduced by some PWR fuel vendors have shown satisfactory corrosion performance to average assembly burn-ups of more than 60 GWd/t. These new cladding materials have also exhibited better dimensional stability and higher creep resistance. New cladding materials, such as low tin alloys, M5, PCA-2b and ZIRLO have been introduced to improve operating margins.

In the past 10 years, in the face of growing electricity demand, some PWRs have been up-rated by as much as 10%, with initial enrichments now as high as 4.6 w/o. Core reload management has been further optimised and peaking factor criteria relaxed in order to meet the demands imposed by up-rating. Other significant developments have included the introduction of mixer vanes and intermediate mixer grids designed to increase thermal-hydraulic margins. The use of thinner RCCA guide tubes has in some instances led to a loss of rigidity of fuel assemblies during irradiation and the consequent increased risk from incomplete rod insertion (IRI). Very recently, grid-to-rod fretting has become the dominant fuel failure mechanism in some plants, possibly caused by irradiation-induced relaxation of grid-retaining forces at the higher burn-ups that are now routine. Through these developments, licensed maximum average assembly discharge burn-ups gradually evolved from approximately 37 GWd/t in the 1970s to typically around 55 GWd/t today, and as high as 60 GWd/t in Germany and Switzerland.

Both “in-reactor” irradiation experience and data from test reactors have underlined the potential importance of phenomena such as the rim effect and thermal conductivity degradation that may become more prevalent at higher burn-ups. While some of these new effects are taken into account in the fuel behaviour calculations, others still require further investigation.

2.4 VVER fuel design evolution

The older generation of Russian-designed PWRs are designated VVER-440. Fuel assembly designs for VVER-440 plants differ from western PWRs in several respects: Firstly, the assembly is hexagonal in shape and the fuel rods are positioned on a triangular lattice. Secondly, the fuel assemblies are shrouded, similar to BWRs, and there are no control rod guide thimbles. In place of the RCCAs used in western PWRs, VVER-440 reactors have movable fuel assemblies known as control assemblies (CAs). These are long fuel assemblies with a fuelled axial region and an absorber axial region. In the most reactive position the fuelled region is inserted in the core. Emergency shutdown and other control rod functions are performed by sliding the assemblies such that the absorber region is in the core; the contrast between fuelled and absorber regions enhances the reactivity worth compared with conventional control rods.

Fuel assembly designs for VVER-440 have evolved in a similar way to those of western PWRs. Initial enrichments and discharge burn-ups started off quite low and have gradually increased as operational experience has been accumulated. In response to similar operational pressures facing PWR utilities, experience has been gained with high enrichment (4.4 w/o) fuel assemblies intended for five annual irradiation cycles and average assembly discharge burn-ups of approximately 50 GWd/t. Appendix A gives details of the history of fuel management for the KOLA-3 plant used as a lead plant for high enrichment/high burn-up fuel assemblies. Other advances with VVER-440 fuel designs include

the first use in 1998 of urania/gadolinia fuel in the KOLA-4 plant in the working (i.e. non-moveable) fuel assemblies (WA) fuel assemblies intended for five-year irradiation cycles. Details are given in Appendix A.

Since 2003, second-generation fuel assemblies have been irradiated in KOLA-3. These differ from the assemblies of the first generation in several respects: the fuel pin pitch is increased from 12.2 to 12.3 mm (which increases the moderator/fuel ratio); the fuel column length is increased; the diameter of the central hole in the fuel pellets is decreased from 1.6 to 1.35 mm; the outer diameter of the fuel rods is reduced from 9.1 to 9.07 mm (which also increases the moderator/fuel ratio); and the use of urania/gadolinia burnable poison and the addition of hafnium plates to the docking unit of the CAs.

The hafnium plates in the CAs were introduced in response to the more stringent demands placed on the core design with high burn-up fuel assemblies. In particular restrictions related to the permissible values of the linear heat rate jumps of fuel rods, as well a requirement to operate the reactor in manoeuvring mode led to the need to upgrade the VVER-440 CAs. Hafnium plates were introduced into the design of the docking unit of these assemblies in order to decrease the peaks of energy deposition in the fuel rods of the peripheral rows of the neighbouring WAs. The first batch of such CAs (12 assemblies) with an average enrichment of 3.82 w/o was loaded into the core of Novovoronezh NPP 4 in 2000. These CAs are currently being operated in the reactor cores at Novovoronezh NPP 4 and KOLA-2 and -3.

The operational experience and data from post-irradiation investigations and code predictions suggest that VVER-440 fuel is capable of operating up to an average assembly burn-up of 55 to 60 GWd/t with a high degree of confidence. The data from post-irradiation investigations indicate that following burn-up extension:

- The mechanical properties of the cladding do not change significantly and remain at a sufficiently high level to ensure fuel rod integrity.

- Cladding deformation (both in the radial and axial directions), including the fuel-cladding interaction, should not result in exhaustion of operability of fuel assemblies.

- Crack development is not observed in the cladding, such that accumulation of damage in the cladding should not constrain the operability of fuel assemblies.

- Corrosion and hydriding of cladding should not result in significant deterioration of operability of fuel assemblies.

- Gas release from the fuel and reduction of the volume of free space under the fuel cladding should not lead to a situation wherein the pressure under the fuel cladding exceeds the coolant pressure.

The second generation of Russian-designed PWRs are designated VVER-1000. The fuel assembly designs for the VVER-1000 plants retain the triangular lattice and hexagonal overall outline, but have evolved to more closely resemble western PWR assemblies. The VVER-1000 fuel assembly does not have the shroud that characterised its predecessor in the VVER-440. The VVER-1000 fuel assembly uses control rod clusters similar to those of western PWRs that insert into guide thimbles, thus dispensing with the need for a moveable assembly. Although the moderator/fuel ratio is slightly different to that of western PWRs and the triangular lattice slightly modifies the neutronic behaviour, the VVER-1000 assembly behaves very similarly. The initial enrichment versus discharge burn-up relation is very little different and many of the statements in this report relating to western PWRs are

transferable to VVER-1000. Of course, the specific design details, such as fuel pin design, materials, thermal-hydraulic correlations, mechanical design, etc., have been developed independently and would need to be validated separately at very high burn-ups.

2.5 MOX

UO_2/PuO_2 mixed-oxide (MOX) fuel has long been used in LWRs as a partial substitute for low-enriched UO_2 fuel. The UO_2 component of MOX normally comprises depleted uranium and the PuO_2 component constitutes the main source of fissile material. MOX use in LWRs is a means by which plutonium separated during reprocessing can be reused and uranium resources conserved. MOX usage in PWRs dates from experience in Belgium in the 1960s and is now deployed on a commercial scale in PWRs in Belgium, France, Germany, India and Switzerland. Japan, Russia and the USA all plan to start commercial-scale MOX usage in LWRs in the near future. MOX has also been used in BWRs in Germany, and Japan is planning to use MOX in its BWRs in the near future. Current practice is to use fractional core loadings of MOX (with typically between 30 and 50% MOX core fraction), with the rest of the core being made up of conventional UO_2 assemblies. This strategy avoids some technological limitations that would affect 100% MOX-loaded cores in current LWRs, none of which have been designed for this specific purpose. Although the technological base for MOX irradiation is less extensive than for UO_2, it is nevertheless very substantial. Commercial experience of the irradiation of MOX assemblies in PWRs and (to a lesser extent) BWRs has been gained over many years, with cumulative MOX fuel manufacture amounting to approximately 1 500 tonnes. The current balance between PWR and BWR MOX manufacture can be gauged from the MOX commitments for 2005, which are approximately 150 tonnes for PWR and 20 tonnes for BWR.

Plans to establish MOX use in VVER-1000 reactors are under way as part of the Russian Federation/US collaboration related to weapons plutonium disposition. In PWRs and BWRs, MOX fuel irradiation experience has generally been positive, being mostly comparable to that of UO_2 assemblies. MOX assembly burn-ups have gradually increased as operational experience has accumulated, though in some countries (e.g. France) they have lagged somewhat behind those of UO_2 due to burn-up limitations in licensing that have applied to MOX. This was largely due to the smaller operational database for MOX assemblies, which can imply larger margins for uncertainties in some safety case arguments. Where larger uncertainty margins are needed, it is not usually because the intrinsic properties of MOX are inferior to UO_2, but rather because the statistical base for establishing the uncertainty margins is smaller, which sometimes makes it difficult to justify applying the same uncertainty allowances as for UO_2. This is the situation that has applied in France for some years, but now that cumulative MOX experience is much increased, the prospect of achieving parity between MOX and UO_2 in France is in view.

In February 2003, the US utility Duke Energy filed an application to allow use of MOX lead test assemblies (LTAs) at the Catawba Nuclear Power Station in South Carolina. These assemblies use surplus plutonium as part of the ongoing US-Russian plutonium disposition programme mentioned in the NCS report. In March 2005, following a lengthy public hearing before the Atomic Safety and Licensing Board, the NRC approved use of the MOX lead test assemblies at Catawba. This is the first time MOX fuel has been used in a US power reactor in support of the disposition programme.

During the MOX hearing [1], the NRC staff provided testimony regarding similarities in the behaviour of urania- and plutonia-based fuels. NRC is aware that regulatory authorities in some other countries [2] may consider some aspects of the behaviour of UO_2 and MOX to differ fundamentally.

Chapter 3

ISSUES NEEDING TO BE ADDRESSED TO ATTAIN VERY HIGH BURN-UPS

The intention of this chapter is to highlight the main technical issues resulting from very high burn-ups that will be discussed in more detail later.

The past history of burn-up evolution has been characterised by a long process of incremental improvements based on a gradually increasing operational and experimental database. It is expected that this incremental process will continue for some time yet, as it is widely considered that LWR burn-ups have not yet reached their ultimate optimum. Considerable effort will be needed over many years in order to demonstrate safe LWR operation with very high discharge burn-ups (60 to 100 GWd/t average assembly burn-up). Significant investment would be needed from LWR fuel vendors, research institutions and to some extent utilities and regulatory authorities. The research and development needed to demonstrate safe and reliable operation at very high burn-ups would necessarily take a long time to complete, and progress towards very high burn-ups would take place incrementally. Even if burn-ups approaching 100 GWd/t prove desirable, it is not envisaged that they will be routinely achieved within much less than 10 to 20 years.

Although there may be some countries in which back-end concepts and strategies are already fixed, and where flexibility for increasing burn-ups may be limited, for the majority of LWR utilities the motivation for adopting very high burn-up cycles is potentially very strong. In some circumstances, very high burn-ups may reduce fuel cycle costs and this is very important for utilities; fuel cycle economics is an area where a utility can directly influence costs and many utilities, particularly those operating in a competitive market, are under very strong competitive pressures to minimise costs. For many utilities, direct fuel cycle cost reductions may play a secondary role to reducing spent fuel arisings; many utilities operate with rigid operating constraints such as limited spent fuel storage capacity that need careful management to maximise their plant's operational lifetime. For utilities in this position, potential reductions in spent fuel arisings with very high burn-ups may be the key to maximising generating revenue over their plant's lifetime and may therefore equate to a very large economic benefit. Very high burn-ups also allow a utility increased flexibility in choosing an optimal combination of cycle length and refuelling fraction, potentially yielding significant economic and operational benefits.

Against this background, a large number of issues can be identified that would need to be addressed and ultimately resolved before very high burn-ups can be implemented in routine LWR operations:

- Beyond an average discharge burn-up of approximately 65 GWd/t, the initial enrichment for LWRs will be in excess of the 5.0 w/o licensing limit of current fuel fabrication plants. The 5.0 w/o threshold is also important for fresh fuel transport criticality safety. It is thought that average assembly burn-ups of 100 GWd/t could be attained in LWRs with average assembly enrichments of approximately 8 w/o. However, the enrichment zoning used in BWR assemblies to counter local flux peaking effects implies that the peak rod enrichment will need to be

higher (up to ~10 w/o). The dependence of initial enrichment with average discharge burn-up is discussed further in Section 4.1.

- Fresh assemblies with higher enrichments will require adaptation of fuel enrichment facilities, transport casks, pool and dry storage facilities at nuclear power plants and repositories. These questions are addressed in Chapter 4.

- The higher decay heat output and neutron emissions of very high burn-up fuel, along with the possibility of the spent fuel having a higher reactivity, has possible implications for spent fuel storage pond operations and criticality assessments. At discharge, the decay heat loading and neutron emission from very high burn-up fuel will diminish more slowly, with possible repercussions for subsequent transport of the fuel to other storage locations or reprocessing. There are also possible implications for a utility's requirement for spent fuel storage capacity, which may be adversely affected if very high burn-up fuel has to be stored for longer periods. However, an indisputable benefit of high burn-ups is that there is a proportional reduction in the mass of spent fuel discharged that may in some cases, depending on individual utilities' particular circumstances, be sufficient to offset these negative trends. A need to store spent fuel for longer periods also has implications beyond just reactor operation in that it may delay final emptying of the fuel ponds after reactor closure and therefore delay the start of pond decommissioning operations. These issues are discussed in Section 4.6.

- The higher minor actinide inventory of very high burn-up fuel, which causes enhanced decay heat and neutron emissions, has possible repercussions for reprocessing plants and may require changes of design and/or operating procedures. An additional factor is the incorporation of high-level waste in glass, which in current plants is limited by neutron emissions; at very high burn-ups the increased inventory of ^{244}Cm may reduce incorporation rates and lead to increased volumes of vitrified waste. The isotopic make-up of plutonium recovered from very high burn-up fuel will be of poor quality, which has particular implications for plutonium recycling in thermal reactors. These issues are addressed in Section 4.8.2.

- There is a lack of knowledge as to whether the direct disposal of very high burn-up fuels in a geological repository may have an adverse impact on the subsequent long-term integrity and leach rates from the waste packages. The implications for any conditioning process to which spent fuel may be subjected prior to disposal are also unknown. These issues are addressed in Section 4.10.

- Very high average discharge burn-ups require very high average ^{235}U enrichments in the core and this in turn reduces the reactivity worths of absorbing materials such as ^{10}B, gadolinia and of neutron absorbing materials used in control rods. For PWRs this implies higher critical boron concentrations and/or higher burnable poison loadings, while for BWRs higher burnable poison loadings, possibly supplemented by and increased control rod insertion will be needed. In LWRs the core boron injection system may need to be re-evaluated to test if it meets more onerous requirements due to the higher reactivity of very high burn-up fuel. At the same time, the differential in reactivity between the fresh fuel and the partly burnt fuel in the core increases, leading to a tendency for higher radial power peaking factors. In BWRs, there may need to be an increase in re-circulation flow (to compensate for the increased power mismatch in a high burn-up core. The core support structure may require adaptation with regard to fuel element lift-off. These issues are discussed in Chapters 5 and 6.

- Additional validation of the nuclear codes and nuclear data (for in-reactor and post-discharge calculations) may be needed to cover very high burn-ups. This is addressed in Section 5.1.5.

- Cores containing very high burn-up fuels will have different safety and transient characteristics that will need to be assessed against the various safety criteria. Such changes may require the safety analyses to be re-assessed. These aspects are addressed in Section 5.2.

- Very high burn-ups imply longer fuel dwell times and consequently more severe fuel mechanical performance duties. All the fuel thermo-mechanical behavioural aspects are affected, notably fission pellet restructuring, gas release, cladding corrosion and dimensional stability. At very high burn-ups current fuel behaviour experience will no longer be valid, necessitating lengthy and costly irradiation trials to demonstrate satisfactory performance. Fuel behaviour codes will need to be extensively validated for very high burn-ups, possibly with new theoretical methods needed to meet the new demands. The thermal-hydraulic and thermo-mechanical issues are considered in Chapter 6.

These issues are discussed in more detail later in this review. Attempts are made to quantify the likely extent to which parameters such as decay heat, neutron emissions and other parameters amenable to calculation are affected by going to very high burn-ups.

Chapter 4

FUEL CYCLE ISSUES

This chapter discusses the impact of very high burn-ups on the broader aspects of both the front-end and back-end of the fuel cycle. There are also important implications for fuel in-core management and neutronics behaviour, fuel design, fuel thermal-hydraulic and fuel thermo-mechanical behaviour, but these aspects are deferred until Chapters 5 and 6 where there are considered in appropriate detail. The starting point for this chapter is to establish a relationship between initial enrichment and average discharge burn-up that is later used to allow a more quantitative discussion of irradiated fuel properties.

4.1 Dependence of initial enrichment on burn-up

This section establishes a relation between initial ^{235}U enrichment and region average discharge burn-up. This enrichment/burn-up relation is used in this chapter to illustrate how various irradiated fuel parameters vary with average discharge burn-up. It is also used in Chapter 7 to illustrate how fuel cycle economics calculations vary with burn-up. The precise relation between initial enrichment and average discharge burn-up will vary depending on the reactor type, the specific fuel design, the fuel management scheme and the kind of burnable poison used. Since there is little in the literature pertinent to very high burn-ups, any enrichment/burn-up relation used for very high burn-ups must be considered subject to uncertainty. For the irradiated fuel parameters considered in Chapter 4, the impact of uncertainties in the enrichment/burn-up relation used is relatively inconsequential. However, there are important implications for fuel cycle cost calculations, as highlighted in Chapter 7.

4.1.1 Initial enrichment versus burn-up relation for very high burn-ups

For average discharge burn-ups up to those already achieved routinely (~50 GWd/t), the relation between initial enrichment and discharge burn-up is well known. An underpinning requirement for this chapter is to have some indication of the initial enrichments required to achieve the very high burn-ups of interest in this study. Initial enrichments as a function of average discharge burn-up were established for PWRs using a combination of two published studies [3,4] and calculations carried out specifically for this report. Specific calculations were also carried out for VVER-440 reactors. Unfortunately, no such calculations were available for BWR; the illustrative results presented in Chapters 4 and 7 are thus restricted to PWR and VVER-440.

Recently published results using the CASMO-SIMULATE 3-D nodal calculations [3] provide PWR initial enrichments for average discharge burn-ups from 45-94 GWd/t. These enrichments are listed in Table 1 and were calculated for a standard 17 × 17 PWR with a thermal output of 3 500 MW and an initial fuel loading of 90 tHM. The calculations assume a fuel management process with a fixed quarter-batch refuelling scheme and the cycle burn-up is varied to achieve the required average assembly discharge burn-up. The *modus operandi* entailed choosing the average initial enrichment and then relying on the 3-D nodal calculations to yield the corresponding average discharge burn-up, which explains why the initial enrichments have rational values, whereas the burn-ups are irrational.

The calculations have been optimised to minimise the radial power peaking factor. Gadolinia (Gd_2O_3) integral burnable poison is used to suppress excess reactivity and radial power peaking and the total gadolinia loading increases with discharge burn-up. Residual neutron captures in the even gadolinium isotopes contribute to a burnable poison reactivity penalty and the initial enrichments in Table 1 account for this effect. The gadolinia loadings tend to increase with burn-up, with both the average number of gadolinia rods per assembly and the gadolinia concentration increasing. The average number of gadolinia rods per assembly is typically 16 at an average discharge burn-up of 60 GWd/t, increasing to as high as 28 at the highest average burn-up. Various different gadolinia concentrations are used, ranging from 4.0 w/o to 9.0 w/o, though there is no simple relation with burn-up. In every case, the ^{235}U enrichment of the gadolinia rods was assumed to be 1.0 w/o lower than that of unpoisoned fuel rods, which is necessary in order to ensure that the gadolinia rods are never limiting for thermo-mechanical behaviour. This is standard practice, although the precise offset between the enrichment of unpoisoned and poisoned fuel rods varies between fuel vendors.

Table 1. Reference initial ^{235}U enrichment versus average assembly burn-up for a 17 × 17 PWR operating with quarter-core refuelling (according to [3])

Cycle length input (GWd/t)	Average assembly discharge burn-up	Average initial ^{235}U enrichment
11.2	44.9	3.8
13.6	54.3	4.5
16.0	64.1	5.4
18.5	73.8	6.5
21.0	84.0	7.5
23.4	93.7	8.5

Though the burn-up/enrichment dependence in Table 1 is considered to be a realistic representation, the fuel loading patterns were not fully optimised and it is quite likely that the calculated enrichments are somewhat pessimistic. Moreover, these calculations represent only one particular scenario. Other scenarios based on different refuelling schemes will have a slightly different set of initial enrichments. The need to develop detailed fuel assembly designs for very high burn-ups and to refine the loading patterns and burnable poison loadings, etc., will all influence the initial enrichments. More fundamental modifications to the fuel assembly design (for example changing the H/U ratio) would also affect the initial enrichments. Overall, it is judged that the initial enrichment estimates are reliable to within ±0.2 w/o at the low end of the burn-up range and ±0.5 w/o at the high end. For the purpose of providing illustrative estimates for this chapter, these initial enrichments are considered adequate.

The initial enrichment relation of Table 1 indicates that the maximum average discharge burn-up achievable within the 5.0 w/o fabrication limit is a little over 60 GWd/t. This figure may be slightly pessimistic, because of the gadolinia residual absorption penalty and because Table 1 applies to a four-batch fuel management scheme (whereas a five- or six-batch scheme or higher are known to yield slightly higher average discharge burn-ups for the same initial enrichment).

For average discharge burn-ups up to ~62 GWd/t, the overall dependence of initial enrichment with average discharge burn-up is supported by a recent study using the Westinghouse nuclear design code suite PHOENIX-P/ANC [4]. The Westinghouse study provides the initial enrichment versus burn-up relation for the Westinghouse reference 17 × 17 plant for average discharge burn-ups up to a maximum of 61.9 GWd/t. The Westinghouse analysis assumes that the plant operates on a fixed 12-month refuelling cycle and generates realistic core loading patterns as the number of feed assemblies

is varied between 64 and 48 (corresponding to a batch fraction varying from ~0.33 to ~0.25 in a 193-assembly core). Zirconia diboride (IFBA) burnable poisons are used to control excess reactivity and radial power peaking. Table 2 shows the enrichment/average discharge burn-up relation according to the Westinghouse study.

Table 2. Initial ^{235}U enrichment versus average assembly burn-up for a Westinghouse reference 17 × 17 PWR operating on a 12-month fuel cycle (according to [4])

Number of feed assemblies	Average assembly discharge burn-up	Average initial ^{235}U enrichment
64	46.441	3.894
60	49.537	4.077
56	53.075	4.283
52	57.154	4.537
48	61.917	4.834

Extrapolation of the trend from Table 2 points to a maximum average discharge burn-up achievable within the 5.0 w/o fabrication limit of approximately 65 GWd/t, assuming that the cycle length is maintained at 12 months and that integral fuel burnable absorber (IFBA) burnable poison is used (and which is described later in Section 4.1.2). It is known that IFBA burnable poison has a zero residual absorption penalty and with any other burnable poison the achievable burn-up will be somewhat lower. This is consistent with the inference made from the data in Table 1.

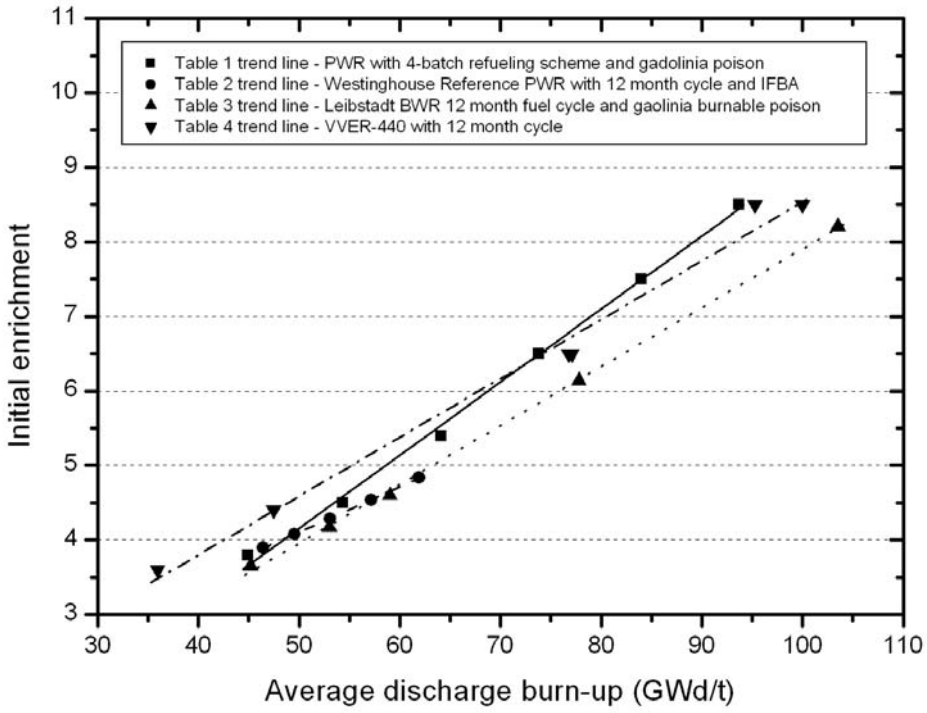
The trend of the Westinghouse has recently been confirmed by unpublished calculations carried out by NOK/KNP for the Leibstadt BWR using the HELIOS/PRESTO-2 code package [5,6]. The NOK/KNP calculations assumed a fixed 12-month cycle length, as in the Westinghouse study, but extend to burn-ups in excess of 100 GWd/t. The results are summarised in Table 3. The three low burn-up cases have zero burnable poison loading and the NOK/KNP data are very close to that of the Westinghouse data (though with a slightly steeper gradient). However, the two high burn-up cases use gadolinia burnable poison to reduce the beginning of cycle (BOC) soluble boron requirement. The gadolinia poison loading is less than that assumed in Table 1 and the trend line for the NOK/NNP data represents an intermediate case between the Westinghouse trend, which has zero burnable poison penalty and the high gadolinia loading assumed in Table 1.

Table 3. Initial ^{235}U enrichment versus average assembly burn-up for Leibstadt BWR

Number of feed assemblies	Average assembly discharge burn-up	Average initial ^{235}U enrichment
152	45.2	3.65
128.5	53.0	4.17
116	59.0	4.60
88	77.8	6.14
66	103.5	8.21

The trend lines for the data in Tables 1, 2 and 3 are shown in Figure 1 (which also includes a trend line for VVER-440 considered later). There is quite a large spread of enrichments, reflecting differences in fuel management schemes, burnable poison types and other factors. For the purposes of illustrating the dependence of irradiated fuel properties on average discharge burn-up in this chapter,

Figure 1. Initial enrichment versus average discharge burn-up trend lines corresponding to Tables 1, 2, 3 and 4



the Table 1 trend line was adopted as the reference enrichment/burn-up relation. It should be noted that the Table 1 trend line is particularly steep partly because these calculations assume a fixed quarter core refuelling fraction. This is neutronically less efficient than the alternative strategy of lowering the refuelling fraction to achieve high burn-ups, which is the assumption made in the Table 2 and Table 3 data. Also, the Table 1 data include a significant burnable poison absorption penalty and, as was noted earlier, the calculations are not fully optimised and may therefore contain an element of pessimism. Nevertheless, the Table 1 enrichments are perfectly adequate to use as the basis for illustrations presented later in this chapter.

For BWRs, the initial enrichment versus burn-up dependence is expected to be similar and the illustrations are considered to be qualitatively valid for BWRs as well. A recent publication [7] has highlighted the limitations on BWR utilities caused by the 5.0 w/o fabrication limit. BWR fuel designers are being forced to use increased numbers of fuel rods of 4.95 w/o enrichment in order to achieve the burn-ups required with extended fuel cycles and power uprates. This has adversely affected radial power peaking within the fuel assemblies, which in turn has forced the core designers to increase the number of fresh assemblies in order to maintain CPR and LHGR margins. Average discharge burn-ups can be limited to as low as 44 GWd/t with the current 5.0 w/o fabrication limit; the 5.0 w/o limit therefore represents a significant obstacle to a further increase of average discharge burn-ups in BWRs and Ref. [7] appeals for it to be relaxed.

Table 4 shows an initial enrichment versus average discharge burn-up relation for VVER-440, which is also included in Figure 1. This is based on unpublished calculations carried out by KFKI specifically for this report; the calculations are described in more detail in Chapter 5, as they are used there to illustrate the dependence of core design and safety parameters on discharge burn-up. The calculations were made for a VVER-440 of 1 375 MW thermal output operating with a fixed cycle

Table 4. Initial ^{235}U enrichment versus average assembly burn-up for VVER-440

Number of feed assemblies	Average assembly discharge burn-up	Average initial ^{235}U enrichment	Burnable poison type
102	35.9	3.6	None
77	47.5	4.4	None
48	76.8	6.5	None
37	100.0	8.5	None
48	77.1	6.5	Gadolinia
37	95.3	8.5	Gadolinia

length of 325 effective fuel power days, using the KARATE code system. The core contains a total of 349 fuel assemblies, each with 126 fuel rods. The calculations in the first four data sets did not include burnable poisons, whereas the last data points include gadolinia. Overall, the initial enrichments are somewhat higher than those of the PWR, which may be due to the different H/U ratio of VVER-440 fuel assemblies.

Finally, to conclude this section it may be helpful to emphasise the distinction between the average discharge burn-up of the core as a whole and the discharge burn-ups of individual batches of assemblies, which can be higher than the average. This is potentially a source of confusion. Consider for example a split-batch scheme in which some assemblies reside in the core for n cycles, whereas the remainder reside for $(n - 1)$ cycles. This occurs when the refuelling fraction does not correspond to a rational fraction such as 1/3, 1/4 or 1/5, which results in the requirement to "split" a fuel batch into two sub-batches, one of which is discharged earlier than the other. With a split-batch scheme the assemblies discharged after n cycles will have an average burn-up in excess of the average values shown in Tables 1 and 2. However, the burn-up of the remaining assemblies discharged after $(n - 1)$ cycles will be lower. It is incorrect to take the burn-up of assemblies discharged after n cycles in a split-batch scheme as being representative of the average discharge burn-up for the entire core. Though it is possible that individual assemblies of 5.0 w/o initial enrichment could achieve burn-ups in excess of 70 GWd/t in a split-batch scheme, the average discharge burn-up for the entire core would necessarily be less than 65 GWd/t and therefore consistent with Table 1.

4.1.2 Choice of burnable absorber

The choice of burnable absorber is an important consideration for very high burn-ups and it has already been seen how this can affect the initial enrichments when gadolinia is used. The residual neutron absorption associated with the even isotopes of gadolinia (which remain after the highly absorbing 155 and 157 isotopes have been removed by neutron capture) is responsible for a significant increase in the initial enrichments of Table 1 (estimated from calculations of residual absorption in the even gadolinia isotopes to be approximately 0.3 w/o at the upper end of the burn-up range). Gadolinia was chosen in Ref. [3] because it is very effective at controlling radial power peaking, for which the ability to separately control the initial neutron capture rate (which is to a first approximation proportional to the total number of gadolinia rods and insensitive to the gadolinia concentration) and the burn-out rate (which results from the strong self-shielding effect and depends on the initial gadolinia concentration) is very useful. It is very important to carefully optimise the fuel and burnable poison loading patterns to minimise the initial enrichment; it has already been noted that the data in Table 1 are not fully optimised and that the residual absorption penalty in this data may be pessimistically high.

In PWRs, alternatives to gadolinia are IFBA and erbia, both of which are used by Westinghouse. IFBA consists of a thin coating of zirconium diboride (ZrB_2) on the surface of UO_2 fuel pellets in which ^{10}B is the burnable isotope. IFBA has the advantage that residual absorption in ^{11}B is effectively zero, so there is no residual absorption penalty. Westinghouse has demonstrated [4] that IFBA is effective at controlling reactivity and radial power peaking for average burn-ups as high as 62 GWd/t and it is possible that it may also be effective at higher burn-ups. This has not yet been demonstrated, however, and it is possible that the total absence of self-shielding with IFBA (which implies lack of control of the burn-out rate) may make it difficult to achieve the necessary degree of control of the radial power peaking. The use of boron enriched in ^{10}B may be helpful in this respect, but it does have adverse implications for helium build-up, which is a feature unique to IFBA. Neutron captures on ^{10}B nuclei generate helium which builds up in the fuel rod free volume. Careful design is needed to ensure that the helium build-up does not become too large, and this will be more difficult to demonstrate at very high burn-ups.

Erbia (Er_2O_3) burnable poisons are in use in a small number of PWRs and share many of the characteristics of gadolinia. Erbia is very effective at controlling radial power peaking and may be advantageous over gadolinia. Erbia suffers from higher residual absorption than gadolinia, due to the relatively low capture cross-sections of the absorbing isotopes, but it is claimed that erbia can be advantageous with very long fuel cycles. It is therefore possible that erbia might be naturally suited to very high burn-ups, but this has not been demonstrated.

Gadolinia is the only burnable poison used in BWRs. The very demanding requirements in BWRs regarding reactivity control can probably only be met by gadolinia.

4.2 Enrichment

Enriched uranium hexafluoride UF_6 is available commercially from various suppliers who operate either gaseous diffusion or gas centrifuge plants. Supplies of enriched UF_6 are also available from the Russian Federation, consisting of surplus military highly-enriched uranium (HEU) that has been blended down to low-enriched uranium (LEU).

The IAEA defines HEU as any material with a ^{235}U enrichment of 20.0 w/o or greater, which requires a higher degree of control than LEU. As indicated in Table 1, an initial enrichment of 8 to 9 w/o would be sufficient for an LWR to attain average discharge burn-ups of 100 GWd/t or more. Even allowing for the possibility of the need for multiple enrichments at these high burn-ups, there should remain a considerable margin to the 20.0 w/o HEU threshold.

Commercial enrichment suppliers have not yet been called upon to supply enrichments in excess of 5.0 w/o for commercial LWRs, the reason being the current 5.0 w/o fuel fabrication limit (see Section 4.3). For those enrichment suppliers who depend on gaseous diffusion plants, there may be criticality implications if they are asked to supply enrichments in excess of this threshold. Gaseous diffusion plants can be designed to operate at high enrichments if required and indeed have in the past been used to supply HEU. Such plants tend to use large gaseous diffusion units for the lower enrichment stages, because these provide economies of scale. However, above a threshold enrichment, criticality safety requirements dictate that the separation unit sizes decrease. For existing gaseous diffusion plants supplying LEU to LWRs there is the possibility that criticality safety limits would restrict the enrichment which could be supplied. The design details of commercial enrichment plants are not openly available and the suppliers themselves would need to provide specific guidance as to any limitations that might apply. Gas centrifuge plants are not subject to criticality limitations in the separation units because of the low operating pressure; the inventories of ^{235}U in each separation unit are so small at the operating pressures that it is not possible to accumulate a critical mass.

Both gaseous diffusion and gas centrifuge plants would need to consider criticality safety at the enriched UF_6 filling stations. Supply of UF_6 enriched in excess of 5.0 w/o entails the availability of suitable transport cylinders and on being able to satisfactorily demonstrate criticality safety in the filling process. The UF_6 transport cylinders currently in use are licensed only to a maximum enrichment of 5.0 w/o.

For both diffusion and centrifuge suppliers, the decision as to whether to offer enrichment services for > 5.0 w/o will be made on commercial grounds. These are likely to depend on the likely cost of any plant modifications that may be required, the cost of obtaining the necessary criticality safety clearances and an assessment of the actual prospects for a market actually materialising. For gaseous diffusion plants in particular, an important factor may be the limited economic lifetime of the plants, considering that centrifuge plants are gradually displacing them.

Although no commercial enrichment plants have yet been licensed for > 5 w/o, because of the lack of any commercial demand, there are indications that the situation may soon change. Indeed, the Georges Besse 2 gas centrifuge plant in France, scheduled to begin operating in 2007, is designed for up to 6.0 w/o, with the possibility of extension to 8.0 w/o without requiring re-licensing. It is also believed that the URENCO cascades could be re-licensed to similar enrichments if a business requirement to do so was identified.

For blended-down HEU, the only consideration would be the availability of suitable transport cylinders, and there is no reason why this route should not be used for enrichments in excess of 5.0 w/o. However at the current rate of consumption, the designated reserves of HEU will be completely used up by 2014, so the potential scope for the use of HEU for this purpose is restricted, unless further HEU is released to the commercial market.

4.3 Fuel and fuel assembly fabrication

4.3.1 Uranium oxide fuels

All the operations involved in fabricating low-enriched uranium fuel do so under strict criticality control regimes. These involve a combination of fissile mass control, geometric and moderation control to ensure that a critical mass cannot be assembled inadvertently. Neutron detector systems are used to monitor and warn of the occurrence of a criticality.

All the fuel fabrication plants currently in operation function with a 5.0 w/o enrichment restriction; material with an enrichment greater than 5.0 w/o is not allowed in the plants. In practice, individual UF_6 cylinders provided by uranium enrichment suppliers are subject to an uncertainty of ±0.05 w/o. When several such cylinders are combined to form a single manufacturing batch, the uncertainty on the average enrichment falls to ±0.02 w/o. Therefore in order to guarantee that the 5.0 w/o limit is not exceeded, fuel fabrication plants need to work with UF_6 with a nominal enrichment no higher than 4.98 w/o. There is already evidence that the 5.0 w/o limit is beginning to become restrictive, especially for BWR utilities. With average assembly discharge burn-ups in both PWRs and BWRs approaching 50-60 GWd/t, average assembly enrichments in the region of 4.5 w/o are already needed. Since BWR fuel assemblies use radial enrichment grading to counteract within-assembly flux peaking, this implies that the highest enrichment rods need to be close to 5.0 w/o, and it is now not unusual for BWR fuel designers to specify the highest enrichment region at the limit. Clearly, average assembly burn-ups much in excess of 60 GWd/t in both BWRs and PWRs will require relaxation of the 5.0 w/o limit.

As discussed in Section 4.1, enrichments in the range 5.0-8.0 w/o would be necessary to achieve discharge burn-ups up to 100 GWd/t in LWRs, with maximum enrichments in BWR assemblies perhaps as high as 9.0 or 10.0 w/o. For such enrichments it would be necessary to review the criticality control procedures for all areas of a fuel fabrication plant.

In the order which they occur during the fabrication process, these are:

1. UF_6 cylinder receipt.
2. Re-conversion plant used to convert UF_6 to pelletable UO_2 powder/granules.
3. Blending equipment used to blend powders from different batches.
4. UO_2 pellet press.
5. Pellet sintering furnaces.
6. Pellet grinding area.
7. Pellet inspection area.
8. Intermediate product store.
9. Fuel rod manufacturing and inspection areas.
10. Fuel assembly manufacturing and inspection areas.
11. Finished fuel assembly storage.
12. Transportation of finished fuel assemblies to power plant.
13. Handling and storage of finished assemblies at power plant.
14. UO_2 residues recovery operations at fuel fabrication plant.

For any of today's fuel fabrication plants, reviewing the criticality control systems in all these areas will be a large undertaking, with no guarantee that the existing facilities can be criticality cleared for higher enrichments. It is possible that for many existing fuel fabrication plants the only option would be to replace some or all of the existing facilities. In order to licence a higher enrichment limit, it may be necessary to impose restrictions that effectively reduce the throughput of the plant. In the event that there is a need for capital investment and/or throughput restrictions, there will be substantial implications for the cost of fuel fabrication; the fabricators will need to recover their investment and will charge an additional premium for fuel fabrication. The decision to invest in relaxing the enrichment limit will depend on the fuel fabricators being able to satisfy themselves of the market need and that there is a low risk of them not recovering their investment.

While many of the items listed above are not expected to be problematic, particular areas for concern are those of the finished fuel storage and finished fuel transport (Items 11 and 12 above). In finished fuel storage the existing criticality safety cases are only just able to satisfy the criticality licensing requirements and increasing the enrichment is likely to make the current approaches to criticality control unworkable. Whereas high-enriched fuel assemblies are likely to incorporate increased

burnable poison loadings, current criticality safety rules do not always permit credit to be taken for the presence of poisons (although in some countries, such as Hungary, burnable poison credit is allowable). Addressing criticality of finished fuel storage will therefore require a concerted technical effort. The implications, particularly methods and validation, for criticality analysis of finished fuel storage and transport are discussed further in Section 4.4.

Detailed modifications to the fuel fabrication plant may be required to deal with any changes in the mechanical design of the fuel assembly. The fuel rod design, fuel cladding material, fuel assembly skeleton and structural materials are all aspects that may evolve with increasing burn-up. However, it is not envisaged that major modifications of the fabrication plant will be required for the evolutionary design changes expected for LWR fuel assemblies. The need for improved cladding materials is discussed further in Section 6.4.

If very high burn-ups are to be achieved in practice, criticality control in fuel fabrication and fresh fuel transport is one of the key hurdles to be overcome. This will necessitate a substantial investment that will require an assured market. On the other hand, the market will not exist until the first fuel fabrication company commits itself. Predicting whether and when this will occur is one of the greatest uncertainties.

4.3.2 MOX fuels

MOX fuel fabrication plants also operate with criticality limits, usually a limit on the maximum fissile content of MOX fuel (^{239}Pu + ^{241}Pu). The strong thermal absorption cross-section of MOX fuels allows high fissile loadings to be used, such that existing MOX fuel fabrication plants may be able to supply fuel assemblies capable of supporting very high burn-ups without significant modification. A previous OECD/NEA study [8] highlighted a limitation of approximately 12.0 w/o on the *total* plutonium content to ensure the moderator void coefficient is negative, and it is likely that this limit will be more restrictive in practice than the criticality limit. The highest average assembly burn-up achievable within the 12.0 w/o limit depends on the isotopic composition of the plutonium being used.

Table 5 gives an indication of the how the initial plutonium content of PWR MOX fuel increases with average assembly discharge burn-up. In this specific case, the plutonium source is assumed to be from LWR fuel with a discharge burn-up of ~45 GWd/t (isotopic make-up 2% ^{238}Pu, 53% ^{239}Pu, 24% ^{240}Pu, 14% ^{241}Pu, 6% ^{242}Pu and 1% ^{241}Am). The initial plutonium loading values in Table 5 were obtained from single assembly lattice code calculations with a constant 600 ppm soluble boron concentration and assuming a three-batch fuel management scheme.

Table 5. Initial plutonium loading versus discharge burn-up for a PWR MOX assembly using LWR plutonium discharged at 45 GWd/t

Average assembly discharge burn-up (GWd/t)	35	45	55	65	75
Initial total plutonium content (w/o)	6.2	8.0	9.6	11.2	13.0

For this specific plutonium isotopic make-up, a burn-up of approximately 70 GWd/t is achievable within the 12.0 w/o void coefficient limit. Higher burn-ups could be reached with a better plutonium isotopic quality or using a more efficient (e.g. four- or five-batch) refuelling scheme. An alternative approach is the use of enriched rather than depleted uranium as the diluent component of the MOX matrix. This has been investigated in France and is considered later in Section 4.9.2.

4.4 Criticality safety

As discussed in Section 4.1, enrichments in excess 5 w/o will be required to achieve very high burn-ups. The use of fuel enrichments greater than the 5 w/o criticality clearance may require significant changes to fuel fabrication facilities, transportation casks and fuel storage at nuclear power plants [9,10]. Modifications of the fuel fabrication facilities will be needed to address criticality concerns associated with these higher enrichments. Modifications of fuel shipping containers may be needed to change the separation between fuel assemblies, incorporate additional neutron absorbers and limit the amount of water that could be added during a shipping accident. The fresh fuel and spent fuel storage racks, or dry storage containers may need to be re-evaluated and modified as well. If higher enrichment fuel is discharged from the core at or near its nominal burn-up, it should be no more reactive than current fuel with < 5 w/o enrichment; however, there must still be a section of the spent fuel storage or fresh fuel racks available to handle lower burn-up fuel during outages and unplanned full-core offloads.

Since the criticality analysis will be revised for all facilities, new experimental data supporting that analysis will be important, especially for enrichments greater than 5 w/o. One recent criticality study at the Sandia National Laboratories (SNL) was supported by the US DOE Nuclear Energy Research Initiative (NERI) programme [11,12]. This NERI project has performed reactor physics calculations to address fuel processing, handling, fabrication, storage, in-core and transportation issues and is performing critical experiments using 7 w/o Pathfinder fuel (i.e. a BWR super-critical water prototype reactor built in the USA in the 1960s) in the SNL Burn-up Credit Critical Experiment (BUCCX) facility [13]. New burn-up credit experiments using higher enrichments may also be needed using the same BUCCX and types of experiments performed in the 1999 NERI project.

Monte Carlo methods are typically used for the criticality safety investigations of the storage and transport facilities. One of the difficult problems encountered when using these Monte Carlo methods, e.g. MCNP, is the source convergence problem, which arises when the system consists of “loosely coupled” regions with small fission sources. For example, the storage pool contains highly burnt-up fuel, which represents a weak neutron source in the middle of a very large neutron absorber, water and poison interspersed in the fuel racks. Larger fission rates may be produced in the ends of the fuel, but the centre of the fuel is highly depleted, so the fuel end regions are loosely coupled. The Monte Carlo source calculation may result incorrectly in a nearly zero fission rate. Benchmark calculations would be important exercises to ascertain the fidelity and reliability of Monte Carlo methods for use in criticality safety analyses.

Burn-up credit in criticality assessments, in which account is taken of the reduction in fissile material and build-up of neutron absorbers with burn-up, is potentially very beneficial with respect to spent fuel storage requirements. The potential gains are larger at very high burn-ups, though the benefit may not scale exactly proportional to burn-up because the issue of neutronic de-coupling also affects burn-up credit. This occurs because, due to the inhomogeneous axial and radial burn-up distributions of irradiated fuel, the average burn-up may not be a reliable indicator of the true multiplication factor (the axial heterogeneity is the dominant effect, which accounts for the terminology “end effect” which is used). The end effect tends to become more pronounced the higher the burn-up and also increases for longer cooling times. An additional factor is that the end effect tends to increase in importance with increasing initial enrichment. All these adverse factors will need to be taken into account in the burn-up credit analyses for very high burn-ups. It is important to highlight that the analysis will be complicated, as there are many interacting trends that must be taken into account.

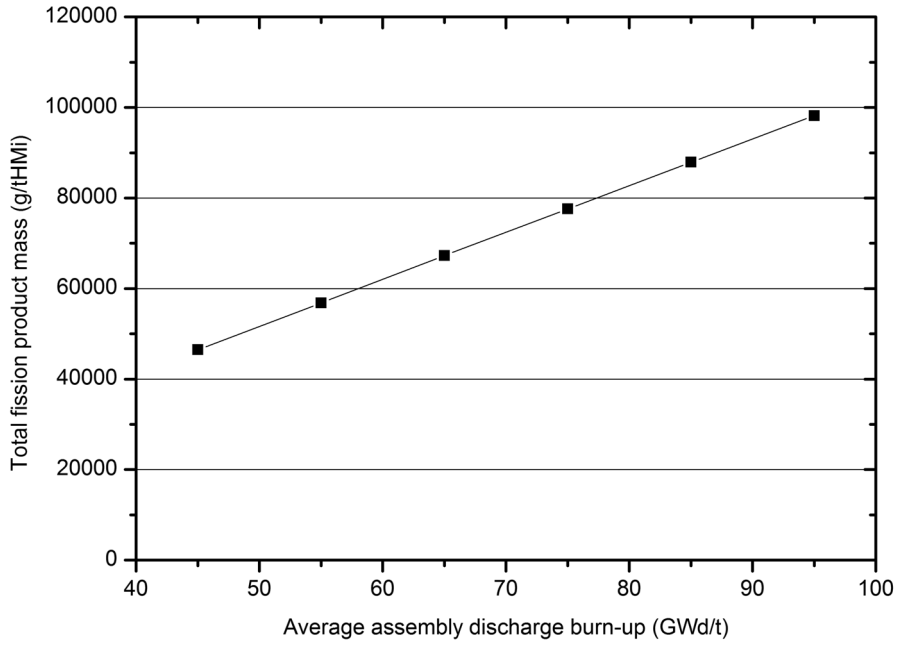
4.5 Spent fuel isotopic inventories, decay heat output and neutron emissions

This section considers the general behaviour of the build-up fission products, transuranics and activation products with burn-up and of the related issues of decay heat output and neutron emissions. The burn-up dependence of spent fuel isotopic inventories is a very important factor that affects the back-end of the fuel cycle. More specific aspects of spent fuel inventories are discussed as necessary in Sections 4.6 and 4.7. Decay heat output and neutron emissions are important considerations affecting many different stages of the back-end of the fuel cycle. This section presents sample calculations that were performed using the FISPIN inventory code [14]. A recent publication [15] describes very similar calculations for a PWR that generally confirm and support the data presented here.

4.5.1 Fission products

The integrated yield of fission products to a good approximation increases linearly with discharge burn-up. This is illustrated in Figure 2, which shows the total mass of fission products for PWR UO_2 assemblies five years after discharge. The initial enrichment varies with discharge burn-up and is consistent with the reference enrichment/burn-up relation from Table 1. The mass of fission products is shown in grams per tonne of initial heavy metal.

Figure 2. Total mass of fission products (g/tHM) as a function of discharge burn-up



There is in principle a small deviation from strict linearity due to the fact that as burn-up proceeds the dominant fissioning nuclides change (e.g. from ^{235}U in fresh UO_2 fuel to ^{239}Pu and ^{241}Pu late in life) and some fission products yields differ depending on the fissioning nuclide. However, this is not noticeable in the plot. Stable and long-lived fission products therefore accumulate roughly linearly with increasing discharge burn-up. Shorter-lived fission products reach saturation when their decay rate matches their production rate and their inventory at discharge is roughly independent of discharge burn-up. The accumulation of stable and long-lived gaseous fission products in the fuel matrix poses difficult challenges for the fuel design and increases the potential release of fission gases into the free internal volume of the fuel rods.

4.5.2 Transuranics

The build-up of the transuranic nuclides proceeds via multiple neutron captures; for each unit increase in atomic mass above that of the parent nuclide, a single neutron capture is required. Those nuclides which are created after only one neutron capture build up first, followed by nuclides which require successively larger numbers of neutron captures. The underlying process is one in which the build-up of the more massive nuclides is exponential with burn-up. Whereas those nuclides with large fission cross-sections (e.g. ^{239}Pu and ^{241}Pu) can eventually reach an equilibrium when their rate of removal by fissions and captures combined equals the rate at which they are created, the non-fissioning heavy nuclides continue to accumulate with burn-up.

Section 4.8.2 considers the accumulation of neptunium, americium and curium as a function of discharge burn-up. For the specific example considered, the concentration of ^{244}Cm at discharge increases by a factor of around 6 for a doubling of the discharge burn-up. This sharply rising behaviour is typical of the heavier transuranics.

4.5.3 Activation products

Long-lived activation products tend to accumulate proportional to burn-up, as the rate at which they are generated is governed by the neutron fluence (i.e. flux multiplied by time). For a given fuel design, the accumulation of activation products in the fuel cladding or the coolant, for example, would increase linearly with burn-up. However, for very high burn-ups new cladding materials may well be required and/or possibly modified coolant chemistries. Such changes could imply a change in the concentration of precursor elements which might result in activation products accumulating at a different rate with burn-up.

Activation of the fuel rod cladding and the fuel assembly components is important, because in the case of a reprocessing it contributes to the total radioactivity of intermediate-level waste (ILW). Doubling the burn-up of fuel, for example, will double the activity of the fuel assembly structural materials, but the mass of fuel discharged will be halved so there is no overall sensitivity. There will be small effects because of end-component activation caused by the axial flux shape differences, but these are not thought to be significant in this context.

4.5.4 Decay heat output

Figure 3 shows the decay heat outputs of PWR UO_2 assemblies with discharge burn-ups ranging from 45 to 95 GWd/t, for cooling times from 0.1 day to 10 000 days (27 years). The decay heat outputs were calculated using the FISPIN inventory code and the initial enrichment varies with discharge burn-up consistent with the reference enrichment/burn-up relation of Table 1. At short cooling times the decay heat output shows very little burn-up dependence (the short-term decay heat output increases only by a few per cent from 45 GWd/t to 95 GWd/t and is not very evident on the logarithmic plot shown). However, the heat output at high burn-ups is significantly elevated at long cooling times. For a given threshold heat output for transport or transfer to interim storage, longer cooling times are required at high burn-ups.

Figure 4 shows the corresponding plot for MOX fuel, though for a narrower range of burn-ups: 45, 55 and 65 GWd/t. The MOX fuel assemblies are assumed to use plutonium recycled from PWR assemblies and the initial plutonium concentration varies with discharge burn-up to maintain energy equivalence with UO_2 assemblies discharged at the same burn-up; the average initial plutonium concentrations are 7.7, 9.3 and 11.0 w/o respectively.

Figure 3. Decay heat output (kW/tHM) versus discharge burn-up – UO_2

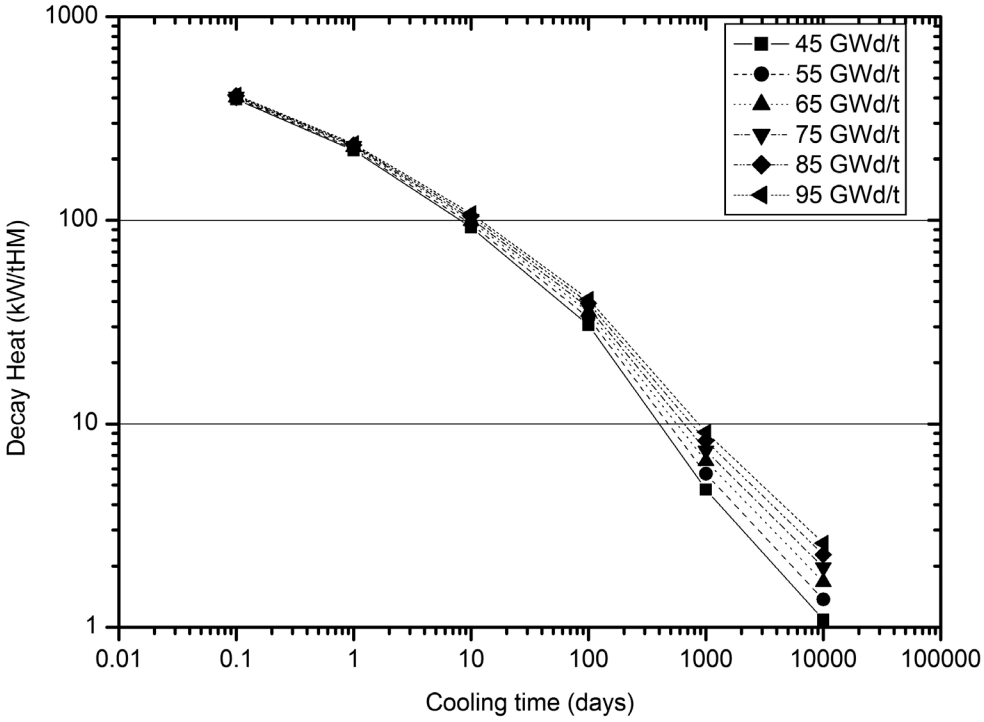
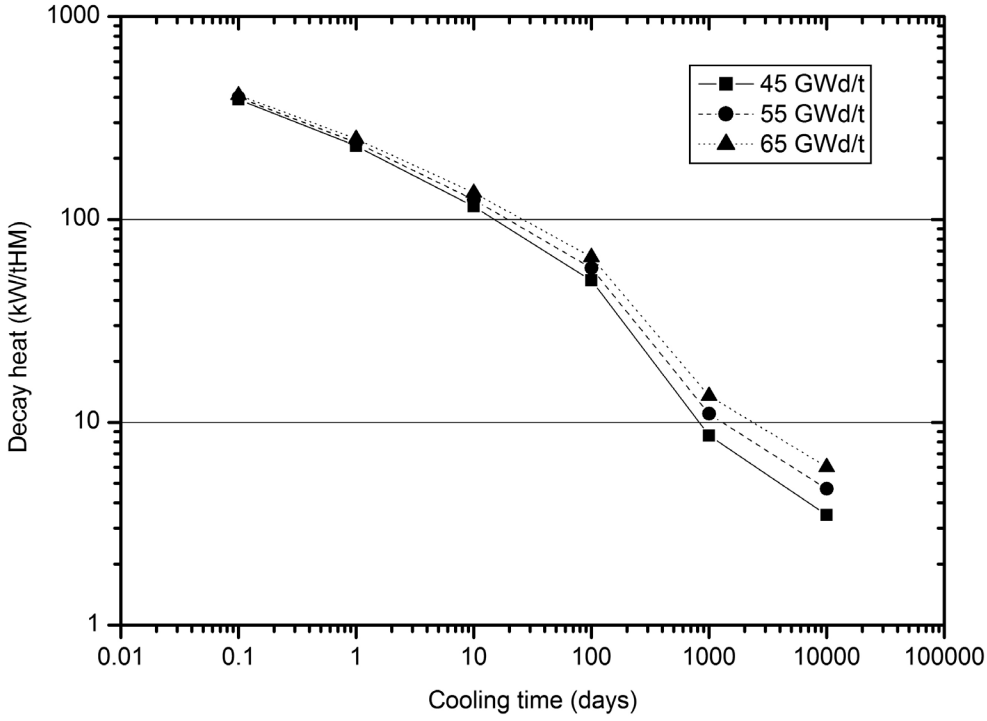


Figure 4. Decay heat output (kW/tHM) versus discharge burn-up – MOX



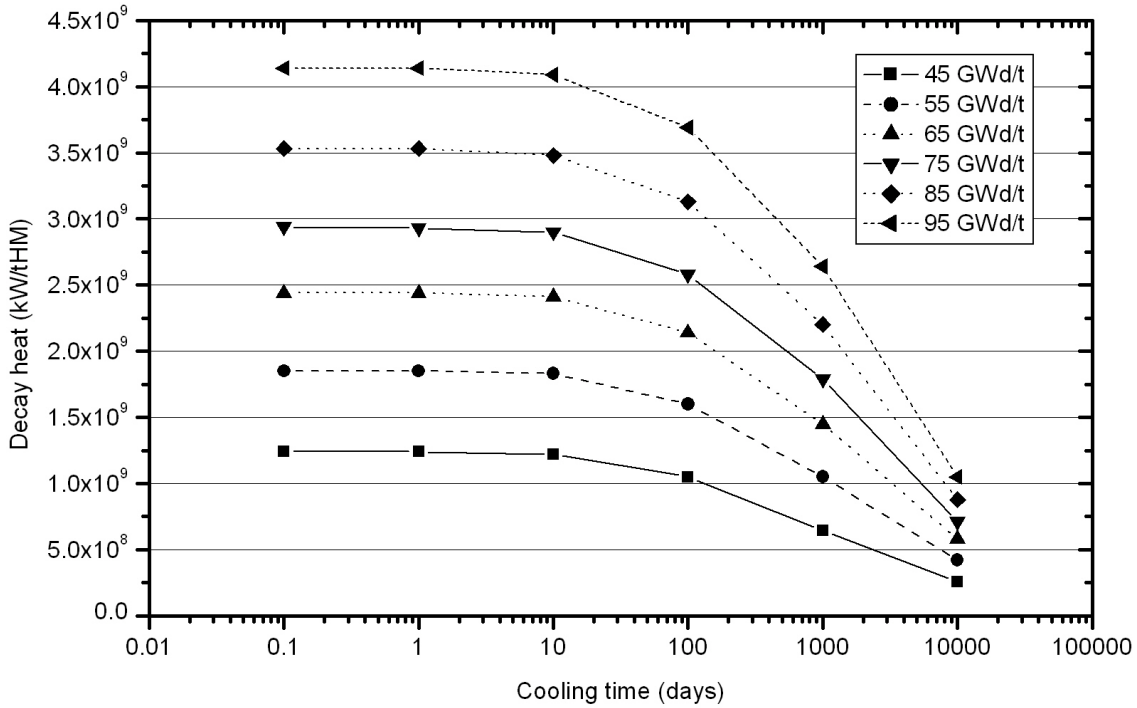
At short cooling times the decay heat output from MOX is very close to that of UO_2 fuel, but declines more slowly with time. It is noticeable from Figures 3 and 4 that the heat output of MOX at long cooling times is more sensitive to burn-up than UO_2.

4.5.5 Neutron output

Irradiated fuel emits neutrons from spontaneous fissions and (α,n) reactions on the oxygen atoms of the UO_2 matrix. Neutron emission is not a limiting factor while the fuel is in wet storage, but it is an important factor that needs to be accounted for in the shielding analyses for irradiated fuel transport flasks and interim fuel storage modules, as well as other aspects of the back-end of the fuel cycle.

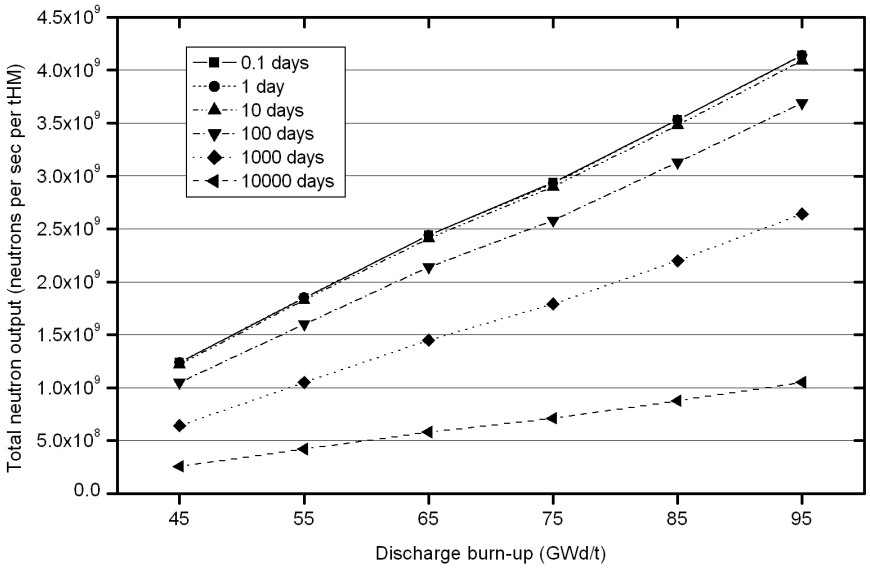
Neutron outputs for high burn-up assemblies are more intense immediately after reactor shutdown and remain higher for longer, as illustrated below. Figure 5 shows neutron outputs calculated by the inventory code FISPIN for UO_2 as a function of cooling time. These are the same FISPIN runs that were used to generate Figure 3. The neutron outputs include the neutron sources from both spontaneous fissions and (α,n) reactions. The units are neutrons per second per tonne initial heavy metal. Figure 6 shows the same data plotted against discharge burn-up on the x-axis. Increasing the discharge burn-up from 45 GWd/t to 95 GWd/t increases the neutron source by a factor ~4 at all cooling times. There is a slight non-linearity noticeable in Figure 6. This is thought to be a genuine effect that results from the effect of burnable poisons; neutron output is sensitive to the initial enrichment and at burn-ups between 65 and 75 GWd/t there is an increase in the burnable poison loading that distorts the initial enrichment versus burn-up relation of Table 1 that was used to generate the data for Figure 6.

Figure 5. Neutron output (neutron/s/tHM) versus cooling time – UO_2



At first sight the almost linear dependence of neutron output with discharge burn-up evident from Figure 6 seems a surprising result, and it might reasonably be expected that the build-up of the main contributing nuclides (^{242}Cm and ^{244}Cm) with burn-up would be closer to an exponential form. This is

Figure 6. Neutron output (neutron/s/tHM) versus discharge burn-up – UO_2



indeed observed if the build-up of these isotopes is followed during the depletion of a particular fuel assembly with a lattice code such as CASMO-4. However, the CASMO-4/FISPIN calculations used to generate the curves in Figure 6 make use of the reference initial enrichment versus burn-up relation from Table 1, such that the higher burn-up data points start off with a correspondingly higher initial enrichment. This causes a reduction in thermal flux and therefore the history of the neutron fluence (i.e. flux × time) is different for the high burn-up data points. The almost linear dependence of neutron output with burn-up represents the net outcome of these opposing effects.

Figures 7 and 8 show the corresponding neutron output plots for MOX fuel, though for a narrower range of burn-ups of 45, 55 and 65 GWd/t (again with initial plutonium concentrations of 7.7, 9.3 and 11.0 w/o respectively). The MOX neutron output curves are very similar to those of UO_2, but it should be noted that quantitatively the MOX values are almost an order of magnitude larger.

Figure 7. Neutron output (neutron/s/tHM) versus cooling time – MOX

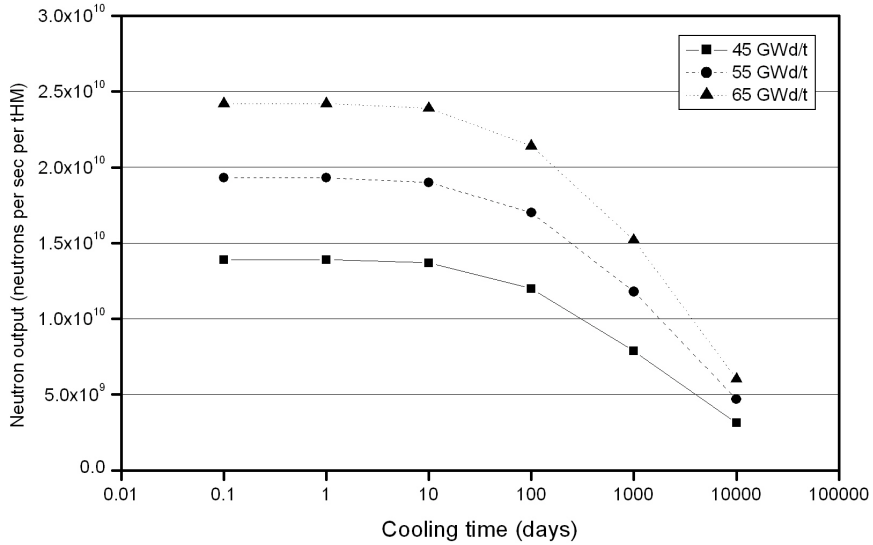
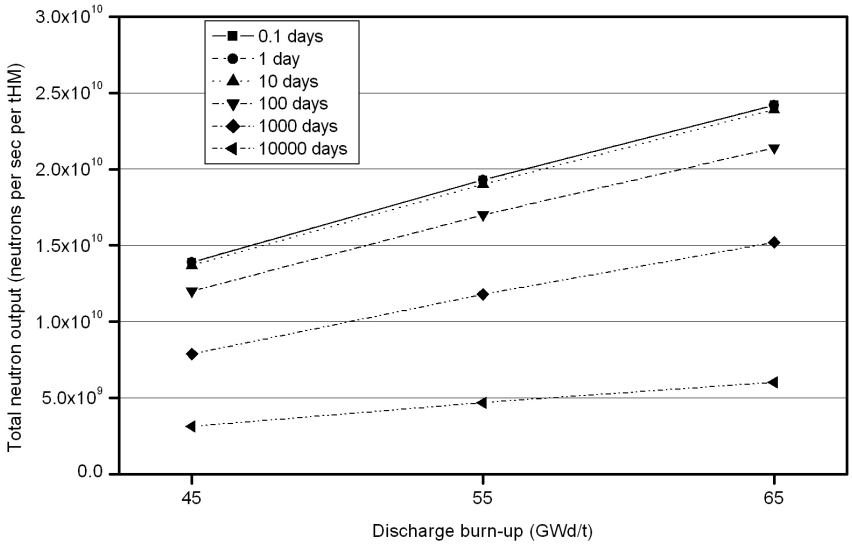


Figure 8. Neutron output (neutron/s/tHM) versus discharge burn-up – MOX



4.6 Spent fuel storage

Interim storage of spent fuel is potentially an area where very high burn-up fuel could be greatly advantageous for a utility. A doubling of discharge burn-up would halve the volume of spent fuel accumulated over the lifetime of an LWR plant. A reduction in spent fuel storage capacity could potentially represent a major strategic benefit to a utility if this means that the need for extra storage capacity is avoided; very high burn-ups could in some circumstances allow a utility to avoid having to apply for political authorisation to extend its spent fuel storage capacity. Indeed, it is conceivable that some utilities might face closure of their plants once their spent fuel capacity is exhausted and therefore very high burn-ups might be the most pragmatic means of achieving a lifetime extension beyond this point.

However, this simplistic scenario is complicated in practice by the higher heat loading and neutron output of very high burn-up fuel discussed in the preceding section. For spent fuel storage, the implications of high burn-ups are that the decay heat output and neutron emissions will not have to be reduced to below the threshold for transfer to interim storage or transport off-site until a much longer cooling time has elapsed. A doubling of the burn-up may therefore not achieve a doubling of effective storage capacity, as the fewer number of assemblies discharged per year is offset by the increased cooling time.

Dry spent fuel storage casks are being used increasingly by utilities whose storage ponds are full, and there are currently 18 dry storage systems which have been approved by the US NRC [16] (see Table 6). The increased heat loading and neutron output may potentially reduce the number of assemblies that are permitted to be loaded in a single storage unit. The implication is again that a doubling of the burn-up will not automatically halve the number of interim storage units needed. It is difficult to make any general statements, as each of the commercially available interim storage systems will have different licensing requirements and, indeed, each utility's position is likely to be unique. The possible need to re-license existing interim storage systems for very high burn-ups represents an additional economic cost that would eventually be passed on to the utilities. Each utility would need to conduct its own cost/benefit analysis to decide on the optimum discharge burn-up for its own particular circumstances.

Table 6. Dry spent fuel storage designs currently approved by US NRC

Supplier	System designator
BNFL Fuel Solutions Corp	VSC-24
BNFL Fuel Solutions Corp.	Fuel Solutions
General Nuclear Systems, Inc.	CASTOR V/21
Holtec International	HI-STAR 100
Holtec International	HI-STORM 100
NAC International, Inc.	NAC-UMS
NAC International, Inc.	NAC-MPC
NAC International, Inc.	NAC S/T
NAC International, Inc.	NAC-C28 S/T
Transnuclear, Inc.	NUHOMS-24P
Transnuclear, Inc.	NUHOMS-52B
Transnuclear, Inc.	NUHOMS-61BT
Transnuclear, Inc.	NUHOMS-32PT
Transnuclear, Inc.	NUHOMS-24PHB
Transnuclear, Inc.	TN-24
Transnuclear, Inc.	TN-68
Transnuclear, Inc.	Advanced NUHOMS-24PT1
Transnuclear, Inc.	TN-32

4.7 Spent fuel transport

Spent fuel transport may be required if a utility needs to transfer its spent fuel into interim storage containers to a storage pond at another location on- or off-site or to a reprocessing plant. The licensing conditions for transport flasks impose limits on total heat loading and neutron output from the fuel that must not be exceeded. For a given cooling period between discharge and transport, very high burn-up LWR fuel will have higher heat output and neutron emissions. The implications are either that the fuel will need to be cooled for a longer time before it is transported, or the number of assemblies loaded in each transport flask will be reduced. A combination of these two approaches is, of course, also possible.

A further consideration is that of criticality control restrictions, which may engender an economic penalty unless the licensing authority allows burn-up credit; without burn-up credit, increasing the boron content of the baskets may prove expensive.

Retaining the spent fuel in the cooling ponds for a longer cooling time may potentially affect the storage capacity available, with consequent economic and operational ramifications. It is possible that the lower mass throughput of very high burn-up fuel may offset the longer cooling time. Each utility would need to examine its own specific situation. Reducing the capacity of transport flasks will be seen as a direct economic cost by the utility, with the number of transport movements and hence the cost scaling inversely with the reduction in capacity.

4.8 Spent fuel conditioning/reprocessing

After a period of cooling spent fuel may be processed in some way, either conditioned for direct disposal or reprocessed. This section considers the impact of very high burn-ups on spent fuel conditioning for direct disposal and reprocessing, both of which may be affected by the higher decay heat and neutron outputs of very high burn-up fuel.

4.8.1 Conditioning

Spent fuel conditioning is a generic term for any process that spent fuel may be subject to prior to its being committed for geological disposal. This may amount to no more than monitoring the fuel for leakage of fission products, followed by its being placed in a disposal container. At the other extreme, conditioning might entail mechanical dismantling the fuel rods followed by some form of reprocessing operation that eventually led to the encapsulation of the radioactive inventory in some immobilisation matrix such as glass or synthetic rock. Vagueness is unavoidable because few countries have any advanced plans for geological disposal of spent fuel and most have no expectation of disposal within the next 50 years or so. Those countries which have more advanced plans for geological disposal have generally favoured minimum intervention with the fuel, with the fuel rods being transferred intact to the disposal container.

Enhanced heat emissions and neutron emissions from very high burn-up fuel will have direct economic and operational implications for whatever conditioning process applies. There will be a requirement to adequately shield all parts of the conditioning plant, which could result in higher shielding requirements and therefore higher construction costs. These would ultimately be passed on to the utilities, possibly in the form of a higher conditioning cost per tonne of spent fuel.

In cases where spent fuel conditioning involves a high level of processing of the fission products and actinides, the incorporation rate of radioactive elements in the various encapsulation matrices may be very sensitive to the discharge burn-up. The volume of encapsulation matrix produced per tonne of spent fuel will increase with discharge burn-up. This is the same consideration that affects reprocessing, and is discussed further in Section 4.8.2.

It is conceivable that a utility might nevertheless see a reduction in spent fuel conditioning costs for very high burn-up fuel if the penalty on the unit cost is outweighed by the reduction in spent fuel discharge volume. However, this is not guaranteed. In the case where a country builds a spent fuel conditioning plant specifically to deal with its own spent fuel arisings, it is possible that the total cost of constructing the plant and its operation will be relatively insensitive to the total throughput. Under such circumstances, the operator of the conditioning plant may be forced to scale its unit costs inversely with spent fuel throughput (i.e. proportional to discharge burn-up) in order to recover the same overall cost. In this case, the utility would see a more or less constant spent fuel conditioning cost irrespective of discharge burn-up.

In terms of total cost for spent fuel conditioning a range of possibilities is therefore envisaged, with some utilities seeing a reduction in fuel costs in inverse proportion to discharge burn-up, while other utilities might see a cost-neutral effect or even an increase in total conditioning costs with burn-up. At the present time it would be premature to attempt to make any quantitative estimates, since there are too many unknowns.

4.8.2 Reprocessing

The higher heat loading and neutron emissions of very high burn-up fuel have important implications for the design and operation of a reprocessing plant. For existing reprocessing plants, neutron shielding may be a factor that limits the maximum burn-up of fuel that is accepted into the process. Neutron output may also have implications for the degradation rate of the solvent used in the chemical extraction process (which also depends on α, β and γ radiolysis), possibly decreasing its lifetime and increasing the corresponding waste stream volume. Heat output is an important factor that affects the behaviour of the plant in design basis accident conditions. Again, there is the potential for

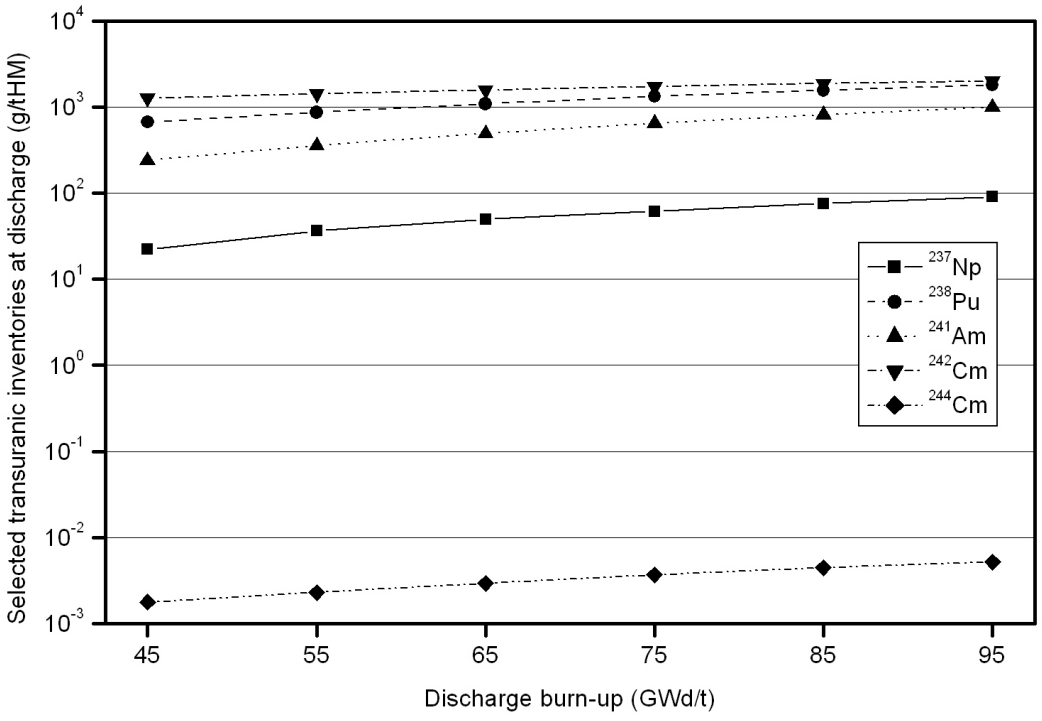
the higher heat output to limit the maximum burn-up of fuel that can be processed and/or restrict the throughput of fuel that can be safely accepted. Depending on the specific situation, restricted throughput may lead to higher unit prices being borne by utilities.

There is a tendency for the inventories of many transuranics to increase very significantly with burn-up. As an illustration, Table 7 and Figure 9 show FISPIN discharge inventories for ^{237}Np, ^{238}Pu, ^{241}Am, ^{242}Cm and ^{244}Cm in UO_2 PWR fuel as a function of discharge burn-up between 45-95 GWd/t. These calculations correspond to a cooling time of 1 000 days after discharge and the reference enrichment/burn-up relation of Section 4.1 has been assumed once again. ^{238}Pu and ^{244}Cm are important for heat loading, while ^{242}Cm and ^{244}Cm are important contributors to neutron output. This shows a very steeply rising trend over this limited range that will continue at higher burn-ups. Even allowing for the reduced spent fuel mass, the ^{237}Np, ^{238}Pu, ^{241}Am, ^{242}Cm and ^{244}Cm inventories per unit energy output will clearly increase with discharge burn-up.

Table 7. Dependence of ^{237}Np, ^{238}Pu, ^{241}Am, ^{242}Cm and ^{244}Cm concentrations (after cooling for 1 000 days) on discharge burn-up

Discharge burn-up (GWd/t)	^{237}Np concentration (g/tHM)	^{238}Pu concentration (g/tHM)	^{241}Am concentration (g/tHM)	^{242}Cm concentration (g/tHM)	^{244}Cm concentration (g/tHM)
45	6.67E+02	2.40E+02	1.26E+03	1.77E-03	2.21E+01
55	8.66E+02	3.57E+02	1.42E+03	2.30E-03	3.63E+01
65	1.09E+03	4.94E+02	1.57E+03	2.92E-03	5.01E+01
75	1.33E+03	6.45E+02	1.73E+03	3.68E-03	6.15E+01
85	1.56E+03	8.13E+02	1.87E+03	4.43E-03	7.57E+01
95	1.80E+03	9.96E+02	2.00E+03	5.20E-03	9.06E+01

Figure 9. Dependence of ^{237}Np, ^{238}Pu, ^{241}Am, ^{242}Cm and ^{244}Cm concentrations on discharge burn-up



Present reprocessing plants use glass to encapsulate and immobilise the high-level waste. The incorporation rate of waste can be limited by the neutron output from higher actinides such as ^{244}Cm. A low incorporation rate due to a high concentration of ^{244}Cm in the high active liquor implies an increase in the number of HLW glass canisters needed for a given mass of fuel processed. This would represent an increased operational cost that would be passed on the utilities. The cost would need to include the costs associated with any additional high-level waste storage capacity.

An additional consideration for reprocessing plants will be that of criticality. The reprocessing plant design needs to guarantee that an inadvertent criticality is not possible in the worst credible scenario. The possibility of having to process higher-enriched fuel assemblies that have only been partly irradiated will need to be allowed for in the design, and this may have a potential adverse economic impact if, for example, the sizes of the dissolver and chemical separation systems are affected.

Any reprocessing plants built in the future would benefit from being able to take into account from the outset of the design process the increased neutron emissions and heat output. However, there will be obvious implications for construction costs, operating costs and maximum throughput. Advanced PUREX, UREX and non-aqueous processes may not be so acutely affected, but there will remain the potential for higher unit costs.

4.9 Uranium/plutonium recycle

Reprocessing of spent nuclear fuel recovers the uranium and plutonium which can then be recycled. This section considers the impact of very high burn-ups on recycled uranium and plutonium.

4.9.1 Uranium recycle

It is possible to recycle the uranium recovered from spent LWR fuel, either in LWRs or in other reactor types. Some French, German, Japanese and Swiss LWRs are already using recycled uranium fuel and it is expected that utilisation of reprocessed uranium will gradually increase in the future. During irradiation the initial ^{235}U inventory is depleted by fission and neutron captures, while other minor uranium isotopes ^{232}U, ^{234}U and ^{236}U build up. The minor uranium isotopes are each important for various reasons.

^{232}U is important because it is short-lived (68.9 year half-life) and its decay chain includes ^{208}Tl, which emits a very intense gamma ray at 2.6 MeV. Once the ^{232}U decay chain reaches equilibrium after about 2 years (corresponding to the 1.9 year half-life of ^{228}Th), the ^{208}Tl gamma represents a very important source of personnel dose in fuel fabrication operations of typically tens of micro-Sieverts per hour. Although ^{232}U is typically present at 1 to 10 ppb (10^{-9}) levels, it nevertheless has a potentially large impact on uranium recycle.

^{234}U is a naturally occurring isotope that is a decay product of ^{238}U. It tends to accumulate gradually in irradiated fuel due to (n,2n) reactions on ^{235}U. It has a small but significant role as an alpha emitter which contributes to internal dose uptake in fuel fabrication operations. ^{234}U also has a small neutron absorbing effect in-reactor. However, since neutron captures on ^{234}U lead to ^{235}U, the absorption effects tend to become less significant at higher burn-ups, since the ^{235}U so generated contributes to fissions. ^{236}U is built up principally by neutron captures on ^{235}U and is important as a neutron absorber. It builds up to a concentration which can approach 1 w/o at very high burn-ups and its neutron capture effect needs to be compensated for by over-enriching the ^{235}U, typically by ~0.2 w/o.

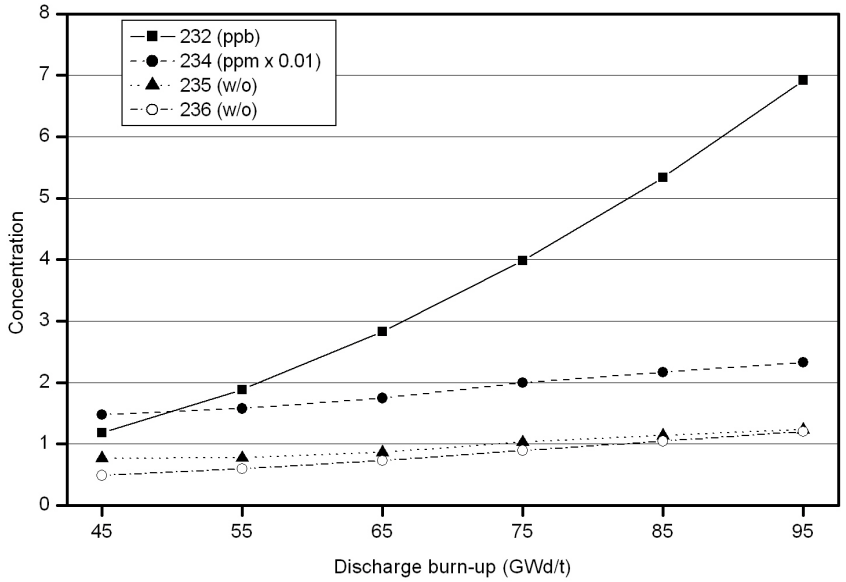
Reprocessed uranium can be recycled either by blending with HEU or by re-enrichment. With the blending route, the ^{235}U concentration is made up to that required in fresh fuel by dilution of HEU. The concentrations of the minor uranium isotopes are slightly reduced in the blending process and their impact is kept to a minimum. Re-enriching involves the conversion of the reprocessed uranium to UF_6 and feeding it to an enrichment plant. The minor uranium isotopes are unavoidably concentrated in the product, which amplifies their impact on fuel fabrication operations and reactor performance. Due to its small mass, ^{232}U is concentrated almost entirely in the product stream, which in the most limiting case at high burn-ups could result in concentrations in re-enriched uranium as high as 50 to 100 ppb.

Table 8 and Figure 10 show FISPIN discharge concentrations of ^{232}U, ^{234}U, ^{235}U and ^{236}U as a function of discharge burn-up for the reference enrichment/burn-up relation of Section 4.1. The burn-ups range from 45 GWd/t to 95 GWd/t. For burn-ups up to 65 GWd/t, for which there is irradiation experience, the calculations can be regarded as definitive. For higher burn-ups they are less certain, but are still judged reasonably reliable. The relatively high residual ^{235}U contents at very high burn-ups are a result of the high gadolinia poison loadings and incomplete loading pattern optimisation; the residual absorption of the gadolinia means that the ^{235}U is not as fully depleted as it would otherwise be. This effect is specific to the burnable poison type and loading.

Table 8. Uranium isotopic concentrations of irradiated PWR UO_2 fuel as a function of discharge burn-up

Discharge burn-up (GWd/t)	^{232}U (ppb)	^{234}U (ppm)	^{235}U (w/o)	^{236}U (w/o)
45	1.18	148	0.7682	0.4891
55	1.89	158	0.7753	0.5974
65	2.83	175	0.8638	0.7306
75	3.98	200	1.0311	0.8937
85	5.34	217	1.1393	1.0477
95	6.92	233	1.2358	1.2039

Figure 10. Uranium isotopic concentrations of irradiated PWR UO_2 fuel as a function of discharge burn-up



There is a clear trend for the ^{232}U, ^{234}U and ^{236}U concentrations to increase with discharge burn-up. The impact on fuel fabrication plant dose uptake and in-reactor neutron absorption will clearly be more significant for very high burn-ups. There is a slight trend for the residual ^{235}U to increase with burn-up, thought to be an effect of the residual absorption of the gadolinia burnable poisons. This slightly increases the recycle value of the material at high burn-ups.

4.9.2 Plutonium recycle

Recycle of plutonium is a strategic priority in some countries that operate LWRs, and this section considers the impact of very high burn-ups on that aspect of the fuel cycle. The fissile quality of plutonium recovered from spent UO_2 fuel tends to deteriorate with increasing discharge burn-up. The fissile quality or fissile fraction is expressed as the ratio of ^{239}Pu and ^{241}Pu to total plutonium

$$\text{fissile fraction} = [^{239}Pu+^{241}Pu]/\text{Pu-total}$$

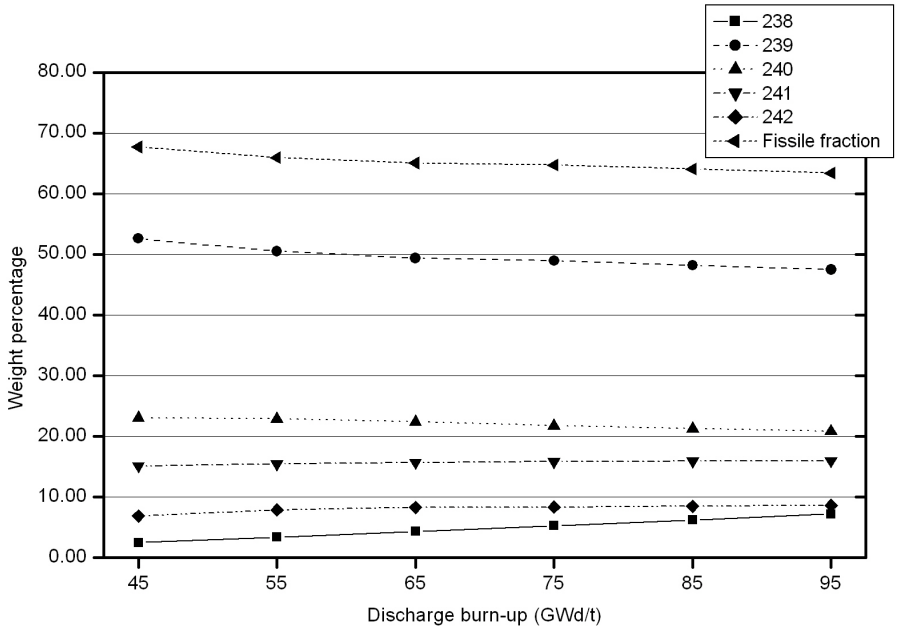
since the even plutonium isotopes do not contribute significantly to fissions in the thermal neutron spectrum of an LWR. It is usually desirable that the fissile fraction should be as high as possible, as the presence of even plutonium isotopes only increases neutron absorption in the fuel. Table 9 and Figure 11 show the trend in plutonium isotopic fractions for ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and fissile fraction for a PWR UO_2 fuel assembly for discharge burn-ups ranging from 45 GWd/t to 95 GWd/t, assuming the reference initial enrichment/burn-up relation of Table 1. The calculations were made using the FISPIN inventory code. ^{239}Pu tends to decrease with burn-up, so that the fissile fraction gradually deteriorates with discharge burn-up. While ^{240}Pu and ^{241}Pu remain roughly constant, ^{238}Pu and ^{242}Pu gradually accumulate.

The implication of deteriorating plutonium fissile quality is that MOX fuel using poor quality plutonium will need a higher initial plutonium concentration. Increasing the initial plutonium concentration to maintain the initial fissile loading is insufficient to maintain reactivity; it is necessary to increase the initial fissile loading to a yet higher level to compensate for increased absorption in the even isotopes. These two effects combined cause the initial plutonium concentration to be quite sensitive to the discharge burn-up of the UO_2 assemblies from which the plutonium is extracted. Bearing in mind that the 12 w/o limit on initial plutonium needs to be respected to ensure that the void coefficient of the MOX assemblies does not become positive, there is the potential for difficulties to arise with some combination of poor fissile quality and high discharge burn-ups for the MOX assemblies.

Table 9. Plutonium isotopic compositions from PWR UO_2 assemblies as a function of discharge burn-up

Discharge burn-up (GWd/t)	^{238}Pu (%)	^{239}Pu (%)	^{240}Pu (%)	^{241}Pu (%)	^{242}Pu (%)	Fissile fraction (%)
45	2.61	54.07	23.98	12.07	7.26	66.14
55	3.56	52.26	23.73	12.36	8.09	64.62
65	4.60	49.36	23.92	12.49	9.63	61.85
75	5.65	47.53	23.73	12.50	10.59	60.03
85	6.72	45.66	23.59	12.40	11.64	58.06
95	7.76	44.04	23.39	12.25	12.55	56.29

Figure 11. Plutonium isotopic make-up of PWR UO_2 assemblies as a function of discharge burn-up



The estimated initial plutonium concentration for a PWR MOX assembly to attain 95 GWd/t in a five-batch fuel management scheme is in excess of 12 w/o even for good quality plutonium with a fissile fraction of 70%. For a fissile fraction of 60%, the highest discharge burn-up attainable within the 12 w/o void coefficient limit in a five-batch fuel management scheme is estimated to be about 75 GWd/t. Thus clearly, for the very high burn-ups of interest in this report, the 12 w/o limit may become restrictive. A technical solution extensively researched in France might be able to circumvent this restriction, while remaining compatible with current LWR core geometry. This is the MOX/EUS (mixed-oxide/enriched uranium support) plutonium assembly, which CEA developed as a potential means of establishing multiple recycle of plutonium in PWRs [17].

MOX/EUS is a mixed-oxide assembly for PWRs in which the PuO_2 is a secondary contributor to fissions, the bulk of the fissions being from enriched UO_2. Because the PuO_2 only plays a secondary role, the nuclear performance of the assembly is close to that of a normal enriched UO_2 assembly. This contrasts with a conventional MOX assembly, where depleted or natural UO_2 is used as the carrier and PuO_2 accounts for the majority of fissions. The MOX/EUS assembly mechanical design is identical to that of a conventional UO_2 assembly, so that there are no mechanical or thermal-hydraulic compatibility issues. Since the nuclear characteristics of MOX/EUS are mostly determined by the uranium, there is relatively little sensitivity to the isotopic quality of the plutonium and the reactivity coefficients are more favourable.

In additions to MOX/EUS, various other technical options for plutonium recycle have recently been reviewed by the OECD/NEA [17]. Among these options is inert matrix fuel (IMF), which is principally intended for one-time irradiation of plutonium as a means of disposing of surplus. IMF fuel contains an inert diluent to replace the uranium oxide matrix of MOX fuel. This eliminates the generation of fresh ^{239}Pu from ^{238}U neutron captures. The potential capability for IMF to generate very high energy outputs (i.e. equivalent to up to 100 GWd/t in conventional assemblies) has not yet been addressed in the literature.

4.10 Waste/radiotoxicity/environmental impact

This section considers the impact of very high burn-ups on waste volume/mass, radiotoxicity and environmental impact.

4.10.1 Waste arisings

The waste arisings in the fuel cycle principally comprise the irradiated fuel itself and the depleted uranium tails. As discussed in Section 4.8, the volumes of high-, medium- and low-level radioactive waste arising from the spent fuel conditioning or reprocessing operations depend on the specific process, and there is the potential for there to be a significant burn-up sensitivity. The volume of depleted uranium tails increases with burn-up because of the higher initial enrichment requirement.

4.10.2 Radiotoxicity

The International Commission on Radiological Protection (ICRP) publishes recommended dose coefficients for use in evaluating the ingestion and inhalation radiotoxicity of radioactive nuclides [18]. For evaluating the radiotoxicity of spent nuclear fuel that is disposed of in a geological repository, ingestion dose coefficients are most relevant, as the pathway to the environment is via groundwater leaching. The ingestion dose coefficients take account of biological and physical factors, such as how much a nuclide is retained in the body and the energy deposition and biological damage per decay. The radiotoxicity is most simply expressed in Sieverts (Sv), which are the preferred unit here. The activity of spent fuel can be expressed in Sv per tonne of initial heavy metal (Sv/tHM), but a more meaningful measure is the radiotoxicity per unit of energy production, as this automatically accounts for the reduced fuel throughput at high burn-ups.

Figure 12 shows the radiotoxicity of spent PWR fuel as a function of cooling time and discharge burn-up. The units are Sv/tHM. The radiotoxicity calculation was carried out with the FISPIN inventory code and used the ICRP-72 recommended effective dose coefficients for ingestion. The reference initial enrichment/burn-up relation of Section 4.1 was again assumed, so that the initial enrichments assumed in the FISPIN calculations increase with discharge burn-up to match the increasing reactivity requirements. The prominent "knee" in the curves at 10^4 years corresponds with the half-life of ^{239}Pu, the decay chain of which controls the total radiotoxicity up to this time.

There is a clear burn-up dependence, with the highest burn-up having the highest radiotoxicity per tonne. This is understandable, as the specific activity of the spent fuel is naturally expected to increase with discharge burn-up. However, this graph does not account for the fact that a higher energy output is generated by each tonne of fuel at high burn-ups, or equivalently, that fewer tonnes of fuel are needed to generate a given amount of energy. Figure 13 below the same data on the basis of Sv per TJ (electrical) energy output.

This alternative plot levelises the burn-up dependence, but the relatively small remaining burn-up dependence is complicated. Up to 100 years' cooling time, there is a slight trend for high burn-ups to have higher radiotoxicity per unit energy. However, at approximately 100 years there is a pinch point where all the different burn-ups almost coincide. Between 100 years and 100 000 years the burn-up trend is reversed, with the low burn-up cases giving the highest radiotoxicity measure. At 100 000 years there is a second pinch point beyond which there is a slight trend for radiotoxicity to decrease with increasing burn-up.

Figure 12. Radiotoxicity versus cooling time and discharge burn-up in Sv/tHM – UO_2

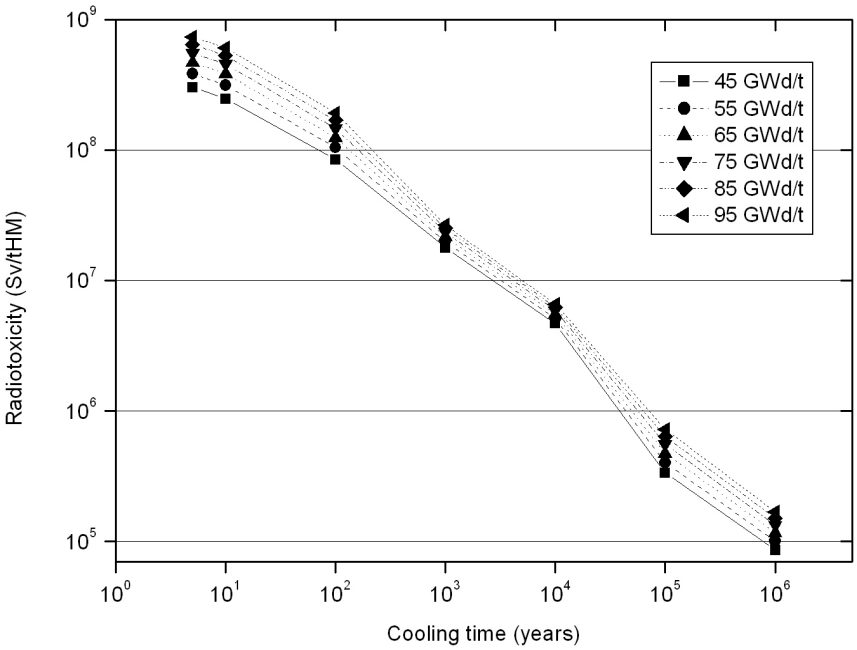
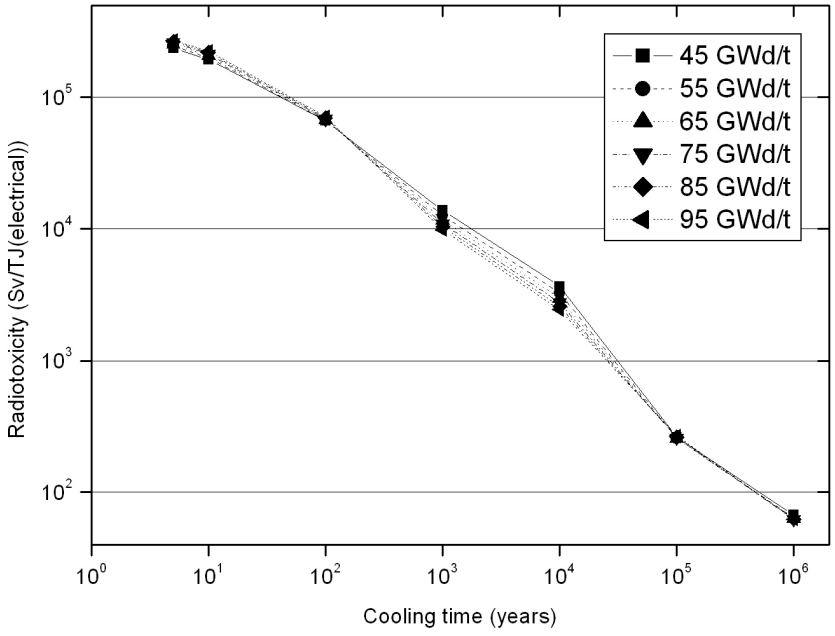


Figure 13. Radiotoxicity versus cooling time and discharge burn-up in Sv/TJ(electrical) – UO_2



In terms of ingestion toxic potential, there is very little sensitivity to discharge burn-up when measured in units that account for the variation of fuel throughput with discharge burn-up. What little sensitivity remains is dependent on cooling time, with mostly much less than a factor of 2 bounding radiotoxicities over the entire range from 45 GWd/t to 95 GWd/t. Compared with the uncertainties in the effective dose coefficients, it is arguable whether the differences are really significant at all when integrated over time. The conclusion is that very high burn-ups have very little impact on the radiotoxicity of spent UO_2 fuel.

For MOX fuel the position is very similar. Figures 14 and 15 show the Sv/tHM and Sv/TJ(electrical) plots for MOX for a narrower burn-up range of 45 to 65 GWd/t. These are based on the same FISPIN calculations referred to in Section 4.5. Though the radiotoxicity per unit mass Sv/tHM increases with discharge burn-up, the radiotoxicity per unit energy output is much less sensitive. There is no reason why this trend should not continue for very high burn-ups, and the conclusion is that there is relatively little sensitivity of MOX radiotoxicity with burn-up.

Figure 14. Radiotoxicity versus cooling time and discharge burn-up in Sv/tHM – MOX

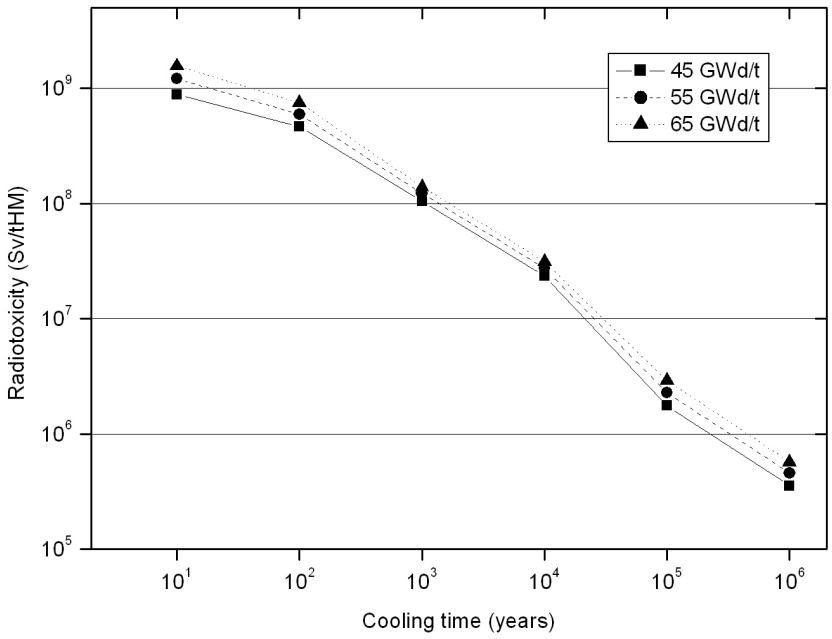
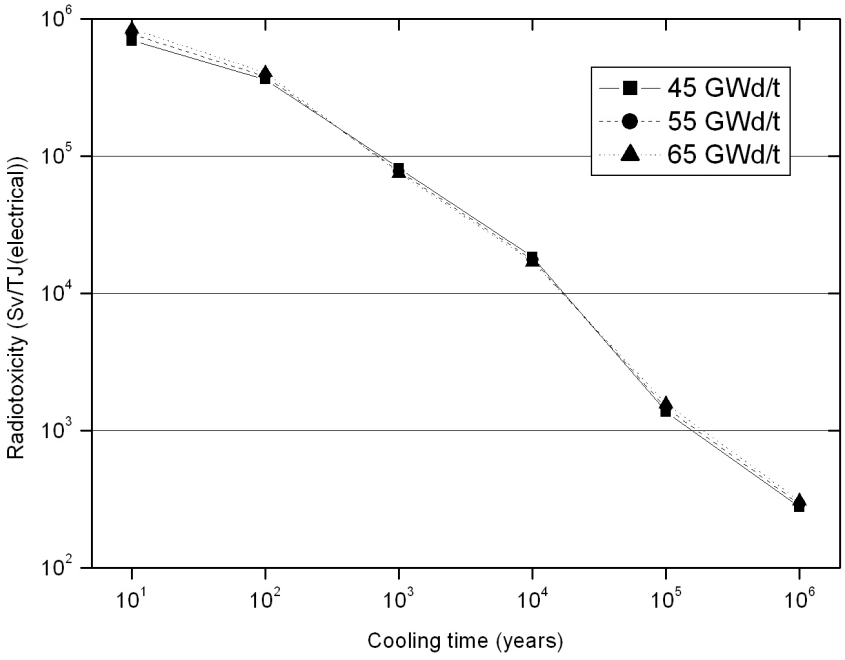


Figure 15. Radiotoxicity versus cooling time and discharge burn-up in Sv/TJ(electrical) – MOX



4.10.3 Environmental impact

The environmental impact directly attributable to fuel operations at a PWR or BWR plant are very small and are most probably very plant specific, depending on the number of fuel failures in operation, the cooling pond chemistry, etc. It is likely that such local factors will have a more important effect on environmental impact than discharge burn-ups. For spent fuel conditioning or reprocessing operations, it is again likely that the specific process details will be a more significant determinant than discharge burn-up. Within a deep geological repository the specific details of leaching rates and groundwater flow are likely to be more important factors than discharge burn-up.

Chapter 5

FUEL MANAGEMENT, CORE DESIGN AND CORE SAFETY/DYNAMICS

This chapter considers the implications of very high burn-ups for in-core fuel management, core design and the safety/dynamic characteristics of the core. The particular details vary depending on whether the reactor type is a PWR, VVER or BWR, and there is also some variation between different plants belonging to a specific class. For illustration, much of the discussion here uses VVER-440 as an explicit example and although the details are specific to that type, many of the principles apply more generally. The aim of this chapter is to consider the impact of very high burn-ups on the *safety-related* core neutronics characteristics and to identify those (if any) which are most likely to be limiting. Generally, the nuclear design bounding or frame parameters are evaluated over the course of the core design, taking into account the uncertainties of the calculations and the changes originating from the differences of the expected future cycles. It cannot be ruled out that these bounding parameters may be modified at very high burn-ups and the related initiating events of the safety analyses are recalculated. Only reactivity-type parameters are investigated here.

5.1 Fuel management, in-core design

There are two approaches by which the discharge burn-up of a fuel cycle can be increased. One involves decreasing the reload fraction, leaving the cycle length and reactor power unchanged. For example, the reload fraction might be decreased from one-third to one-quarter, so that the fuel residence time increases from three to four cycles. For a fixed cycle length, the discharge burn-up increases in inverse proportion to the reload fraction. The other approach can potentially provide a larger economic benefit, and involves increasing the cycle length while the reload fraction remains constant. Assuming refuelling outage times are unaffected, longer cycles imply higher capacity factors and therefore higher income from generation. With this approach the discharge burn-up increases in direct proportion to cycle length. A combination of smaller reload fraction and increased cycle length is also possible. Irrespective of the approach chosen, increasing the discharge burn-up requires higher initial enrichments if cycle length and power are not decreased. However, decreasing the reload fraction benefits from a slight increase in efficiency as the reload fraction tends towards zero (corresponding to infinite-batch refuelling), so that for any given discharge burn-up a lower initial enrichment is needed. Due to the higher initial enrichment, both approaches significantly affect the in-core fuel management and care is needed to ensure that the in-core parameters, particularly power peaking factors, reactivity feedback coefficients and shutdown margins, remain within acceptable ranges. Higher discharge burn-ups can also be attained by up-rating reactor power. If the reload fraction and the time elapsed during a cycle is kept the same, the burn-up increases in proportion to the up-rating.

The implications of burn-up increase with a constant cycle length can be understood from the linear reactivity model (see Appendix B). There are several key points that emerge from a linear reactivity approach that reinforce the comments in the introductory paragraph:

- As noted, high discharge burn-up can be attained only by increasing the neutron production capability (k-infinity) of the fresh fuel assemblies (see relationship B4 in Appendix B) and this normally implies a higher initial enrichment. A small increase in burn-up can be achieved by modifying the H/U ratio, though for the current generation of LWRs this has very limited potential because the H/U ratio is already very close to the optimum.

- The difference in reactivity between the fresh fuel and the partly burnt fuel in the core is increased, leading toward a tendency for higher radial power peaking factors, especially at beginning of cycle (BOC). This effect may be counteracted to some extent by using an increased loading of burnable poison.

- As the initial enrichment increases, the fast/thermal flux ratio also increases, corresponding to a "harder" neutron spectrum. The harder neutron spectrum influences the reactivity coefficients and control rod worth as detailed in Section 5.2.

- The reload fraction must decrease in inverse relation to the burn-up increase (see relationship B1 or B3 in Appendix B).

- For a fixed cycle length, the BOC reactivity reserve (or excess reactivity) needed to counteract the decrease of reactivity throughout a cycle is only slightly affected by very high burn-up (see relationship B6 in Appendix B); shutdown margins, BOC critical boron concentrations and consequently moderator temperature coefficients are not too severely impacted. In PWR and VVER, there is a tendency, however, for the boron reactivity coefficient to reduce in magnitude due to the higher enrichments needed to achieve high burn-ups. This implies a higher critical boron concentration at the start of a cycle, which in turn affects reactivity coefficients and shutdown margins.

Similarly, the implications of the alternative approach of increasing burn-up by increasing the cycle length are:

- The initial enrichment needs to be increased more when the cycle length is extended at fixed reload fraction (see relationship B4 in Appendix B) and the consequences of the higher enrichment are more evident. For example, the maximum discharge burn-up attainable with the current 5.0 w/o criticality limit will be smaller if the burn-up increase is achieved by increasing the cycle length.

- If a large increase in cycle length is made and if the discharge burn-up is restricted for reasons of fuel performance or licensing, it may be necessary to increase the reload fraction, which will adversely affect the initial enrichment (see relationships B1 and B2 in Appendix B). This is a trade-off that many utilities have made in the past 20 years or where they accept a penalty on fuel cycle cost in order to be able to make a larger gain from the higher capacity factor that extended cycles allow.

- In the absence of burnable poison, the BOC reactivity reserve increases (see relationship B6 in Appendix B). This implies higher BOC critical boron concentrations and consequently less negative moderator temperature coefficients. Shutdown margins can also be adversely affected. The use of burnable poisons is an effective means of compensating these trends and is also advantageous for decreasing the radial peaking factors at BOC.

- Longer cycles are favoured by utilities who are not in a position to achieve reliably short outages, e.g. as a consequence of delays to re-start caused by licensing requirements. The

implication is that, by reducing the average outage time, thus improving the plant load factor, a reduction of the overall power generating cost can be achieved with longer cycles. However, this is dependent on the level of replacement power cost.

The simplest strategy for increasing the average discharge burn-up is by simply reducing the cycle length. In accordance with the linear reactivity model (discussed in Appendix B), decreasing the cycle length (and therefore reducing the fractional core loading) allows a higher discharge burn-up to be achieved for any given initial enrichment, because of the improved neutron economy. Therefore the shorter the cycle, the higher the average discharge burn-up achievable with a given ^{235}U enrichment of the fuel assemblies. This in turn, means better fuel utilisation and lower specific fuel cycle costs. However, this strategy is not often adopted by utilities, as it implies more frequent reload outages and reduced availability and capacity factors (and consequent loss of generating revenue) that would outweigh the fuel cycle cost savings. An alternative means of increasing average discharge burn-ups is to extend cycle lengths by means of coast-downs, although this again involves a trade-off between replacement power cost and reduced fuel cycle cost.

Table 10 illustrates one strategy for the particular example of a VVER-440 for which core design models were generated specifically for the purpose of this report. The KARATE code system [19] was used to model equilibrium fuel management schemes with average discharge burn-ups ranging from 36 to 100 GWd/t. Table 10 gives the initial enrichments, average fuel core residence time and average discharge burn-up for the six cases studied as calculated by KARATE. The KARATE models were also used to calculate reactivity coefficients, shutdown characteristics and other safety-related core parameters. Although the results are specific to VVER-440, many of the qualitative trends will also apply to PWRs and BWRs.

Table 10. Main characteristics of the VVER-440 equilibrium cycles

Cycle	No. of fresh fuel assemblies/cycle*	Enrichment of fresh assemblies [%]	Average lifetime of fuel in core [cycle]	Average discharge burn-up [GWd/t]
Case 1, without burnable poison	102	3.6	3.42	35.94
Case 2, without burnable poison	77	4.4	4.47	47.46
Case 3, without burnable poison	48	6.5	7.27	76.76
Case 4, without burnable poison	37	8.5	9.69	100.05
Case 5, same reloading as in Case 3 but 6 pins/assembly (from the 126)* with burnable poison	48**	6.5	7.27	77.06**
Case 6, same reloading as in Case 4 but 12 pins/assembly with burnable poison	37**	8.5	9.69	95.33**

* The VVER-440 core consists of 349 fuel assemblies, 126 pins/assembly.

** One additional partially burnt fuel had to be loaded and reloaded in each cycle to compensate for the residual reactivity of the burnable poison, therefore the calculated discharge burn-up is perturbed.

Cases 1 to 4 show the effect of increasing average discharge burn-up from 36 to 100 GWd/t without the use of burnable poisons. The average discharge burn-up is increased by reducing the number of fresh assemblies loaded in each refuelling outage and the initial enrichment increases from 3.6 to 8.5 w/o. The absence of burnable poisons has important implications for reactivity coefficients, as discussed later in this chapter. Cases 5 and 6 are the same as Cases 3 and 4 respectively, but with gadolinia burnable poison rods to lower the beginning of cycle (BOC) critical boron concentrations and improve the reactivity coefficients and shutdown parameters. In Case 5 the fresh fuel loading was adjusted slightly to maintain the average discharge burn-up almost constant. For Case 6, however, no such adjustment was made and the reduction in average discharge burn-up from 100 to 95 GWd/t is a direct result of the residual absorption penalty associated with gadolinia poison rods. In each case the reactor is assumed to have a thermal power output of 1 375 MW and the cycle length is constant at 325 effective full power days (efpd).

5.1.1 Burnable poison requirement

Whichever approach is used to increase the burn-up, the burnable poison requirement may be affected. Significant burn-up extension will require a higher initial enrichment and will increase the initial multiplication factor (k-infinity) of fresh fuel. At the same time, the multiplication factor of the highest burn-up fuel in the core will be reduced, increasing radial flux/power peaking. Cycle length extension will in addition increase the BOC reactivity reserve. Both effects can be counteracted with burnable poisons, which can be used to reduce the initial reactivity of fresh assemblies, maintain the specified shutdown margin and satisfy power peaking and thermal limits.

During their first cycle of operation, virtually all the PWRs currently operational used discrete burnable absorber rods inside the assembly, or within the “spider” assembly replacing the control rod fingers. At the low burn-ups of the early reload cores, few PWRs needed to use burnable poisons and those that did need to use them had only relatively modest burnable poison requirements for which discrete burnable poison rods were adequate. The use of control rod guide tubes for discrete burnable absorber rods constrains the reactor fuel management design, since those fresh assemblies cannot be placed in the control rod assembly positions (which are in fixed locations). Furthermore, with fewer fresh assemblies containing these burnable absorbers, an even distribution is more difficult to achieve. The higher burn-up levels may also significantly change the control rod worths if the very high burn-up (three or four times burnt assemblies) must be placed in control rod positions. A further disadvantage of discrete burnable poisons is that they add to the volume of intermediate-level waste arisings and require separate handling and storage in the fuel ponds. For these reasons, discrete burnable poisons have largely been superseded by integral burnable poisons, but they remain as a technical option that might be useful in supplementing integral burnable poisons with very high burn-up fuel management schemes.

In integral burnable poisons, the poison material is incorporated as part of the fuel pellet or fuel rod. Examples are urania/gadolinia, urania/erbia or zirconium diboride (ZrB_2) coatings, all of which locate the poison inside or on the surface of the fuel pellets and have been used in PWRs. Theoretically, the poison material could be located in the fuel cladding, or in the fuel assembly spacer grids, though this has not yet been adopted in practice.

Integral burnable poisons have the advantage that separate handing of poison rods is eliminated and that residual neutron captures in the burnable poison rod mechanical structure is eliminated. Urania/gadolinia and urania/erbia have an appreciable residual neutron capture penalty from the even mass isotopes remaining after the high cross-section odd isotopes have undergone neutron capture. However, the residual capture penalty with the zirconium diboride coating is almost zero. Although it

is possible to place integral poison assemblies under a control rod, this can reduce the reactivity effect of the control rod (as on insertion the control rod will enter an assembly with a lower thermal neutron flux) and there may in practice be constraints on the number of fresh assemblies in control rod locations.

In PWRs, the constraints imposed by control rod locations, burnable absorber penalty issues, rod worth, and fuel loading pattern requirements such as core and local power distributions, may become more difficult to attain with very high burn-up concepts. Core symmetry issues (where the designer wants to be able to load like assemblies in locations at 90 or 180 degrees during the next cycle to reduce flux tilts) may become more difficult to achieve with fewer assemblies per batch or cycle, as may be necessary with the use of very high burn-up loading patterns.

In BWRs, urania/gadolinia poisons are universally used. The impossibility of using soluble boron reactivity control in BWRs means that burnable poisons have to play a much more significant role than they do in PWRs, and the initial reactivity hold-down and poison depletion rate are more critical in a BWR. For a BWR core designer, the ability to independently control the initial reactivity hold-down (which is proportional to the number of urania/gadolinia rods) and the burn-out rate (which depends on the gadolinium concentration) is major factor in favour of urania/gadolinia. While it is desirable in BWRs to avoid excessive reactivity hold-down with control rods (to avoid the effects of PCI), the demands of very high burn-up fuel cycles may demand deeper control rod insertion to be used. In turn, this may require further improvements with regard to PCI resistance.

The application of particular types of burnable poisons for very high burn-up cores will have to be evaluated carefully to ensure that the core-wide fuel rod and assembly duties remain acceptable, and there are several complicating factors to be addressed: There may be licensing "penalties" arising from the increased uncertainty on within-assembly peaking associated with poison depletion for example. Boron placed inside or on the surface of the fuel pellets (as with zirconium diboride coating) may be limited because of helium gas pressurisation produced by the (n,α) capture reaction.

Very high burn-up concepts may need to develop innovative burnable poison concepts to avoid such constraints and minimise the residual neutron capture penalty. One approach may be the selective enrichment of specific poison isotopes, rather than just using the natural concentration levels [20]. For example, enriched ^{10}B is being considered for both fixed and soluble poisons in MOX cores, while it would be theoretically beneficial to use gadolinium enriched in ^{157}Gd.

As an illustrative example, Table 11 gives BOC critical boron concentrations for the six VVER-440 fuel management cases presented earlier in Table 10. This also shows the burn-up reactivity reserve. Critical boron concentrations are given at beginning of cycle hot full power (BOC HFP) and beginning of cycle hot zero power (BOC HZP) core conditions. The units are grams of molecular H_3BO_3 per kg of coolant. In western PWRs the units normally used are part per million (ppm) by weight of elemental boron in the coolant and conversion from g/kg molecular H_3BO_3 to ppm elemental boron is achieved by dividing by 0.00572.

The reactivity reserve is the reactivity worth of soluble boron expressed in terms of percentage Δk. Even though the cycle length is constant, the magnitude of the reactivity reserve decreases with increasing average discharge burn-up. The reason for this is that the absolute value of the thermal flux is smaller for the high burn-up cases (this is a consequence of the increased fissile density needed for high burn-ups) and this lowers the inherent depletion time constant of the fissile material. If the reactivity reserve is expressed in terms of the critical boron concentration (which involves dividing the burn-up reactivity reserve by boron density coefficient), then the burn-up reactivity reserve is practically independent of discharge burn-up with exception of the two burnable poison cases, where the burn-up reserve is lower by about 1 g/kg (or 1% Δk).

Table 11. Reactivity characteristics and boron density reactivity coefficients for VVER-440 equilibrium cycles

Case	Average discharge burn-up [GWd/t]	Critical boron concentration at BOC HFP [g/kg]	Burn-up reactivity reserve [%Δk]	Critical boron concentration at BOC HZP [g/kg]
1	35.94	7.265	8.803	10.493
2	47.46	7.572	8.384	11.318
3	76.76	7.446	6.014	12.187
4	100.05	7.951	5.667	**13.698**
5	77.06	6.356	5.095	10.977
6	95.33	6.152	4.543	11.375

* Safety-related parameters exceeding the present limit values are denoted by bold figures.

The boron coefficient is affected in the high burn-up cases by the hardening of the neutron spectrum being significantly reduced in magnitude at high burn-ups. The decrease in the boron reactivity coefficient has a large impact on the BOC HZP critical boron, such that at the highest burn-up and with no burnable poison (i.e. Case 4), the required boron concentration exceeds the maximum allowable limit. However, the burnable poisons are effective in reducing the critical boron concentrations, as can be seen by comparing Cases 5 and 6 with corresponding un-poisoned cases (Case 3 and Case 4 respectively).

5.1.2 Low-leakage loading patterns (LLLP)

Improved fuel utilisation can be achieved by enhancing core loading using low-leakage loading patterns (LLLP). The traditional means of doing this is to load the oldest and most highly burnt fuel (which has the lowest k-infinity) at the core periphery, thereby improving neutron economy by reducing neutron leakage [21-23]. This also serves to reduce reactor pressure vessel neutron embrittlement. However, there are other constraints that apply which might become more difficult to satisfy with very high burn-ups. For example, the highest burn-up assemblies may need to be placed near fresh assemblies in order to have a better neutron source-sink relationship. Such constraints may make LLLP more difficult to achieve, but nevertheless it should still be possible to achieve better performance than the older, less economical, out-in loading patterns used in the 1970-1980s.

A consequence of LLLP is that the fuel will need to be able to sustain a higher power level. The more highly-enriched fuel needed for higher burn-up is a further reason for this requirement. In BWRs there is margin available in the critical power ratio (CPR), so further power improvement (i.e. up-rating) is feasible. The improvements in CPR achieved by the newer BWR designs over the past 10 years have shown that a 10% improvement can be translated into about 2% less fuel consumption or a corresponding reduction in enrichment. LLLP using very highly irradiated assemblies may also impact other safety margins, such as those imposed by LOCA criteria, critical heat flux or DNBR margins, shutdown margins, etc.

5.1.3 Other considerations

It is also important to note that the use of soluble boron in PWRs for beginning of life reactivity suppression must satisfy tight constraints such as solubility limits (e.g. typically < 2 200 ppm in PWRs), moderator temperature coefficient limits (zero or negative MTC) and operational chemistry limitations.

For example, some PWRs are constrained by the reactor water storage tank temperature and boron injection system limits (i.e. active heating of borated water storage tanks is required to keep the boron in solution). It is important that very high burn-up fuel loading strategies should be assessed very carefully from the perspective of excess reactivity control options, based on penalties, chemistry, flexibility, etc. Again, the use of enriched soluble boron is an option. As noted earlier, in BWRs the importance of burnable poisons for reactivity suppression is considerably accentuated because of the impossibility of using soluble poison for reactivity hold-down.

In very high burn-up fuel ^{239}Pu contributes a larger fraction of total fissions and causes the delayed neutron fraction (β_{eff}) to decrease. This will affect the progression of some transients, particularly those that are limiting at end-of-life conditions, and these will have to be re-analysed. Design basis accidents that have rapid cool-downs need to be considered since the stronger negative temperature coefficients combined with the reduced delayed neutron faction can make some transients (i.e. main steam line and feed line breaks in a PWR, especially at end-of-life, at hot zero power conditions) more severe.

Since current MOX core designs have typically been limited to 30-40% MOX fraction, reductions in the delayed neutron fraction should not be a problem with very high burn-up fuel assemblies, as long as the delayed neutron fraction is kept within the licensing basis for the plant for end-of-life conditions. However, the current Combustion Engineering System 80+ PWR is designed for full-core MOX, and there are initial design options of the EPR and AP1000 to accommodate full-core MOX, where the delayed neutron fraction may be affected to a greater extent.

Many utilities would like to achieve longer cycles (18-24 months), with fuel loading flexibility, core/assembly symmetry and the ability to produce more power with a core up-rating. For the fuel an up-rating means that the core must produce more power in the fresh and once-burnt assemblies, but with the constraints met for LLLP requirements, safety margins, etc. The use of very high burn-up fuel may reduce the scope for up-rating in combination with longer cycles.

EOC coast-down core conditions may cause unwanted conditions in the balance of plant, steam generators, etc. The EOC coast-down to “stretch” the fuel cycle will also have to be carefully evaluated on a case-by-case basis. An EOC coast-down operational period impacts not only the current cycle but also subsequent cycles with the fuel that is reloaded into the core. The history effects (i.e. spectral, burn-up, coolant chemistry) can impact the future core patterns and the location of fresh assemblies. For example, if the current cycle length, as well as spectral and thermal conditions change, core design may become difficult because the reload for the upcoming refuelling may contain too few or too many fresh fuel assemblies. Shutdown margins, safety parameters (e.g. moderator temperature coefficients), and other key core-wide parameters can be harder to maintain.

The constraints on core-wide and local power conditions (hot spots, peak-to-average flux levels) need to be carefully analysed using well-established, i.e. benchmarked and licensed, reactor physics, thermal-hydraulic analysis and fuel performance evaluation tools and further validation may be required to ensure the codes are still accurate at very high burn-ups.

5.1.4 High burn-up core design studies

The Electric Power Research Institute (EPRI) and DOE have studied the impact of very high burn-up strategies on optimum cycle length for both greater and less than 5 w/o enrichment [24,25]. These studies looked at different fuel cycle lengths and feed assembly options for both BWR and PWR core designs. It was found that increasing discharge burn-up tended to increase steady-state core peaking factors, initial boron concentrations (PWRs) and fuel duty, axial offset anomaly in PWRs, and

corrosion potential in BWRs. Although the economics of using greater than 5 w/o enrichment may be challenging, it was found that very high burn up levels (> 80 GWd/t) could be achieved using higher enrichments and various feed assembly and loading pattern options. Furthermore, specific in-core analyses showed that safety margins (e.g. moderator temperature coefficient, flux peaking factors, Fq) could be adequately maintained. A recent study of very high burn-up fuel management schemes in a PWR [3] has also arrived at the same conclusions and shown that flux/power peaking can successfully be controlled using urania/gadolinia burnable poison.

5.1.5 Core design methods for very high burn-ups

High burn-up core analysis will require improved depletion routines in the current static reactor physics tools to accommodate the increased concentrations of fission product isotopes and higher actinides caused by very high burn-up. Several computer codes based on the ORIGEN depletion model have been developed to address this problem and have been benchmarked [26]. The accurate determination of depletion isotope chains becomes more important for very high burn-up conditions, especially for transuranic isotopes (Pu, Np, Am, Cm, etc.) and fission products.

The evaluation of any very high burn-up concept will involve the use of detailed reactor physics calculations for both steady-state and core depletion conditions, and reactor kinetics analyses for specific transient and accident scenarios. The use of current nodal reactor physics tools (e.g. CASMO, SIMULATE [27,28]) has been assessed for large burn-ups, MOX and UO_2 cores, and extreme conditions (i.e. branch calculations for fuel temperature, burn-up, moderator temperature and void, spectral history, etc.) needed for in-core fuel management, neutronics and safety assessments. The current reactor physics tools have been deemed adequate for these cases, provided that sufficient cross-section data and energy groups have been used for the analysis. However, the relative ratio of transuranic isotopes and fission capture chains for very high burn-ups may require more thorough investigation of energy-dependent cross-section data uncertainty.

Finally, the use of nodal diffusion theory codes vs. higher-level neutron transport codes for very high burn-up core analysis will have to be evaluated, especially for instances where very highly irradiated assemblies are placed next to fresh assemblies with highly poisoned rods, or integral burnable absorbers are present. Although stochastic transport theory codes using Monte Carlo methods can be used for reactor physics benchmarks and some analyses on a limited basis, such Monte Carlo tools can be very computationally intensive, and may not be fast enough to be used for performing the large number of computations needed for in-core fuel management assessments. The OECD/NEA has proposed and completed various code-to-code and code-to-experiment benchmarks to ascertain the adequacy of current nodal reactor physics tools to be applied to extremely difficult problems [29-34].

5.2 Core safety and dynamics parameters

This section considers the impact of very high burn-ups on the *safety-related* core neutronics parameters (i.e. reactivity feedback coefficients, kinetics parameters which influence the dynamical behaviour of the reactor) and attempts to identify those (if any) which are most likely to be limiting.

PWRs and VVERs both use the “nuclear design bounding parameters” approach to simplify safety assessments. This involves the designation of bounding values for the various safety-related parameters such as reactivity coefficients, power peaking factors, control rod reactivity worths, etc. The bounding values are chosen such that the actual calculated values are always less limiting. Therefore if the safety analysis uses the bounding values the results will be pessimistic. The bounding parameters approach is

very advantageous in that it largely decouples the safety assessments from the detailed within-cycle and cycle-by-cycle variations of the actual parameter. If the bounding values are chosen correctly, the safety assessment carried out for the first operating cycle will remain valid for the subsequent reload cycles, even though the relevant parameters may vary from cycle to cycle and during a single cycle. Care is needed to ensure that the bounding parameters approach does not introduce excessive pessimism and in some circumstances additional safety assessment calculations based more closely on the actual parameter values may be required. Nevertheless, the bounding parameters approach is very practical and successful.

The nuclear design bounding parameters are evaluated over the course of the core design, taking into account the uncertainties of the calculations and any changes expected in future cycles. It is quite possible that some of these bounding parameters may be modified at very high burn-ups, and in this section the bounding parameters for PWR and VVER are discussed individually and the implications of very high burn-ups discussed case by case, using VVER as an illustrative example.

BWRs use a different strategy from PWRs and VVERs in that they rely more on flux mapping of the core during operation to calibrate and re-adjust the nuclear design models. Using the calibrated models, checks and adjustments are made to ensure that the operating limitations will remain satisfied. Although the approach is different, many of the burn-up trends for PWR and VVER will also apply to BWRs, though the details will be different.

5.2.1 Limitations of the power distribution: maximum linear heat rate, maximum pin power, maximum assembly power

In PWRs, hot channel and hot pin calculations are carried out to evaluate the minimum departure from nucleate boiling ratio (DNBR). In BWRs, the corresponding calculations feed into the critical power ratio (CPR) evaluations. The hot channel and hot pin calculations also have a direct impact on maximum fuel temperature, maximum clad temperature and maximum radial average enthalpy. Moreover, important fuel behavioural characteristics (internal pressure, fatigue, elongation, etc.) during the long-term normal operation are also impacted.

As a consequence of the higher initial enrichment needed for very high burn-ups, the difference in reactivity between the fresh fuel and the partly burnt fuel in the core is increased, leading to a tendency for higher radial power peaking factors. This tendency may need to be controlled with higher burnable poison loadings. At high burn-ups, the axial power distribution can be modified significantly, and this effect may need to be offset with modifications to the fuel assembly design (such as axial heterogeneity of burnable poison and/or the enrichment distribution).

Recent 3-D nodal calculations for high burn-up PWR cycles [3] illustrate the unavoidable trend for the radial peaking factor to increase with discharge burn-up. This work shows that the radial power peaking factor $F_{\Delta H}$ increases at a rate of approximately 0.14% per 1 GWd/t increase in discharge burn-up. This trend occurs in spite of efforts to control power peaking with increased burnable poison loading. The observation in such studies is that although burnable poisons can be very effective at controlling radial power peaking early in a cycle, there is a tendency for the radial power peaking to rebound later in the cycle as the poison becomes depleted. This burn-out effect is unavoidable if the burnable poison design is also to successfully minimise residual absorption at the end of a cycle. It may be possible to suppress this effect if some residual absorption can be tolerated at EOC conditions, but if so this is likely to have a significant economic penalty because of the adverse implications for initial enrichment.

5.2.2 Maximum (least negative) moderator temperature reactivity coefficient

The maximum (i.e. least negative) moderator temperature coefficient bounding limit is important for several initiating events in a VVER and PWR: uncontrolled withdrawal of control rod(s), start-up of an inactive coolant loop with lower boron concentration (VVER only), control rod ejection, decrease of the reactor coolant inventory (for example LOCA) and some cases of decreased heat removal by the secondary system. For all these initiating events moderator feedback plays an important role in limiting the resulting power excursion and the bounding limit should be more positive than the calculated moderator temperature coefficients.

In PWRs and VVERs, very high burn-ups may involve extending the intervals between refuelling outages, and with long fuel cycles higher critical boron concentrations and consequently less negative moderator temperature coefficients at BOC can be expected.

Table 12 gives the maximum moderator temperature coefficients for the six VVER-440 equilibrium fuel cycles of Table 10. The maximum (i.e. least negative values) occur at beginning of cycle hot full power (BOC HFP) core conditions. The units are per cent mill (pcm) per degree Kelvin; 1 pcm is equivalent to 10^{-5} Δk.

Table 12. Moderator temperature reactivity coefficients for VVER-440 equilibrium cycles

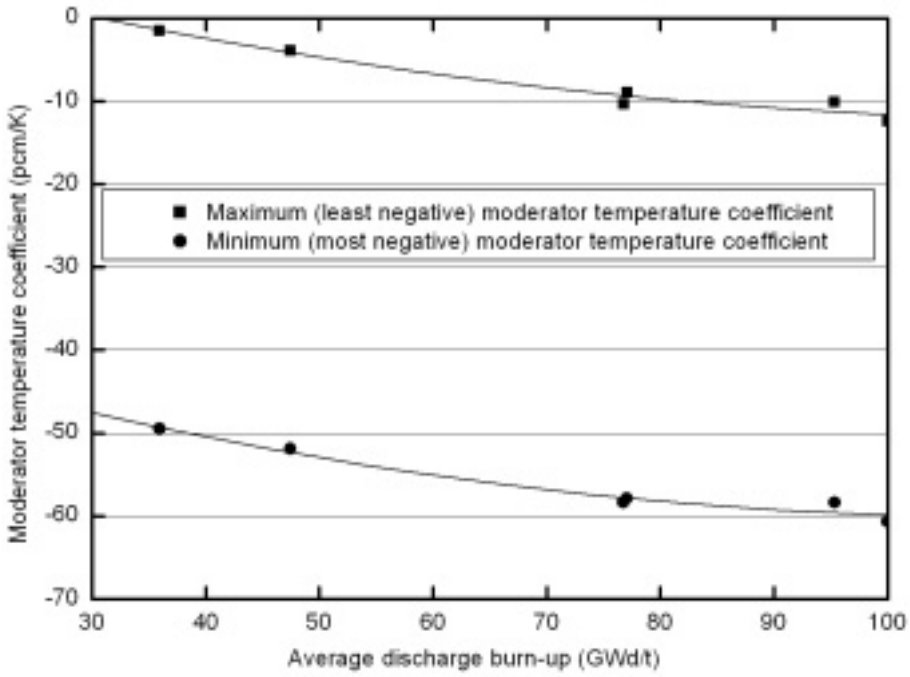
Case	Average discharge burn-up [GWd/t]	Maximum isothermal reactivity coefficient at BOC HFP [pcm/K]	Minimum moderator temperature reactivity coefficient at EOC HFP [pcm/K]
1	35.94	-1.5	-49.5
2	47.46	-3.9	-51.9
3	76.76	-10.3	-58.4
4	100.05	-12.4	-60.7
5	77.06	-8.9	-57.9
6	95.33	-10.1	-58.4

Figure 16 shows the same data plotted against average discharge burn-up; the upper curve shows the maximum moderator temperature coefficient (BOC, HFP). A logarithmic trend line has been fitted to the individual data points. The points corresponding to Cases 5 and 6, with gadolinia burnable poison, do not fit the same trend as the un-poisoned cases; they lie slightly above the un-poisoned trend. Nevertheless, the general trend for the maximum moderator temperature coefficient with increasing discharge burn-up is favourable, as the margin between the calculated values and the safety limit widens.

Similar behaviour would be expected for PWRs, though the precise functional dependence will be different because VVERs have a slightly lower moderator/fuel ratio than PWRs. The burnable poison loading, in combination with the soluble boron concentration also influences the moderator temperature coefficient, especially at BOC conditions; the burnable poison loading is actually determined in part by the need to ensure that the moderator temperature coefficient is negative at BOC.

The moderator temperature coefficient in MOX fuel tends to be slightly more negative than in UO_2 cores, such that this bounding limit should be satisfied in MOX-loaded cores.

Figure 16. Moderator temperature reactivity coefficient in VVER-440 equilibrium cycles as a function of average discharge burn-up



5.2.3 Minimum moderator (most negative) temperature reactivity coefficient

The relevant initiating events include: start-up of an inactive coolant loop with lower temperature (VVER only) and increased heat removal by the secondary system (for example main steam line break accident). These events result in a reactivity excursion due to the entry of coolant into the core at lower than the nominal temperature. The bounding limit should be more negative than the actual calculated values.

Table 12 lists the minimum moderator temperature coefficients for the six VVER-440 equilibrium cycles. The lower trend line in Figure 16 plots the same data as a function of average discharge burn-up. Again, the two cases with burnable poisons fit a slightly different trend to the unpoisoned cases. The hardening of the neutron spectrum and the differences between ^{235}U and ^{239}Pu fission cross-sections lead to more negative moderator temperature coefficients at high burn-ups irrespective of whether or not there is cycle length extension [31]. This is an adverse trend, as it will increase the reactivity insertion resulting from the ingress of cold water into the core. For MOX cores the slightly more negative moderator temperature coefficient may reduce the margin to this bounding limit.

5.2.4 Maximum (least negative) fuel temperature reactivity coefficient

The relevant initiating events include: uncontrolled withdrawal of control rod(s), start-up of an inactive coolant loop with lower temperature or/and boron concentration, control rod ejection.

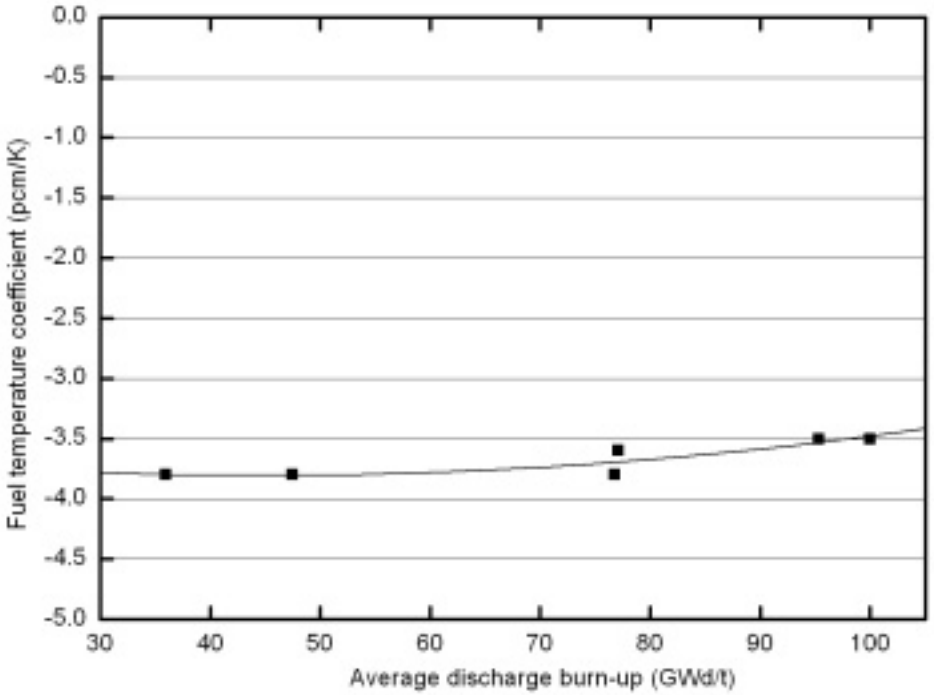
The fuel temperature reactivity coefficient determines the peak power excursion and is especially important because it is a fast-acting feedback mechanism and is guaranteed to always be negative. The bounding limit should be less negative than the calculated values.

The magnitude of the fuel temperature coefficient tends to increase with burn-up during a cycle and also increases with decreasing power. The limiting condition for this limit is therefore at BOC HFP core conditions. Table 13 lists the calculated fuel temperature coefficients at BOC HFP for the six VVER-440 equilibrium cycles, and Figure 17 shows the same data plotted against average discharge burn-up, together with the associated trend line. There is little variation with increasing discharge burn-up, so that satisfactory fulfilment of this limit is not likely to be affected [31]. There is only a slight impact from burnable poison loading, such that the same trend line applies quite well to un-poisoned and poisoned cores. Similar behaviour would be expected in a PWR.

Table 13. Temperature reactivity coefficients for VVER-440 equilibrium cycles

Case	Average discharge burn-up [GWd/t]	Fuel temperature reactivity coefficient at BOC HFP [pcm/K]
1	35.94	-3.8
2	47.46	-3.8
3	76.76	-3.8
4	100.05	-3.5
5	77.06	-3.6
6	95.33	-3.5

Figure 17. Fuel temperature reactivity coefficient in a VVER as a function of assembly burn-up for two initial enrichments



5.2.5 Minimum (most negative) fuel temperature reactivity coefficient

The relevant initiating events include: increased heat removal by the secondary system and decrease of the reactor coolant inventory.

Increased heat removal from the primary circuit, i.e. due to increased heat transfer to the secondary circuit or vaporisation of the primary coolant, causes the fuel to cool and increases reactivity. This can contribute towards reducing the margin to re-criticality and for conservatism the bounding limit should be more negative than the calculated values. Due to the weak burn-up dependence noted in Figure 17, satisfaction of this limit is not likely to be affected at very high burn-ups [31].

5.2.6 Maximum (least negative) boron concentration reactivity coefficient

The relevant initiating events include: increased heat removal by the secondary system and decrease of the reactor coolant inventory. In both of these initiating events, at a late phase of the transient the reactivity is controlled by boron injection from the emergency core cooling system (ECCS), which decreases core reactivity. For conservatism the bounding limit should be more positive than the calculated boron reactivity coefficients.

Table 14 gives calculated boron density reactivity coefficients at BOC HFP and BOC HZP core conditions for the six VVER-440 equilibrium cycles. The boron coefficient is maximum (i.e. least negative) at BOC core conditions, when the boron concentration is highest. Figure 18 plots the same data versus average discharge burn-up, along with the trend lines.

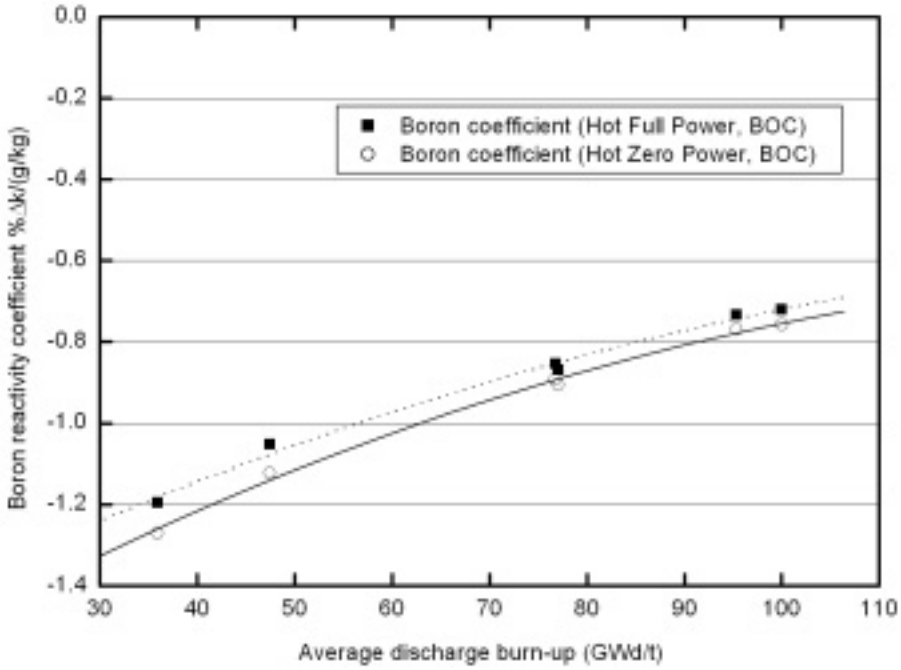
Figure 18 shows that for VVER-440, increasing burn-up reduces the magnitude of the boron reactivity coefficient. The trend line is only slightly sensitive to whether the cores contain burnable poisons. The same general trend is also expected in PWRs. This is a spectral effect associated with the decrease in thermal/fast flux ratio as the enrichment increases and may become a significant adverse factor at very high burn-ups. The same effect is also very characteristic of MOX cores and in current PWRs can influence the highest MOX core fraction that can be allowed. The boron reactivity coefficient is not relevant to BWRs.

Table 14. Reactivity characteristics and boron density reactivity coefficients for VVER-440 equilibrium cycles

Case	Average discharge burn-up [GWd/t]	Boron density reactivity coefficient at BOC HFP [%Δk per g/kg]	Boron density reactivity coefficient at BOC HZP [%Δk per g/kg]
1	35.94	-1.196	-1.271
2	47.46	-1.052	-1.123
3	76.76	**-0.853**	**-0.892**
4	100.05	**-0.721**	**-0.757**
5	77.06	**-0.868**	**-0.906**
6	95.33	**-0.734**	**-0.769**

* Safety-related parameters exceeding current limit values are denoted by bold figures.

Figure 18. Boron concentration reactivity coefficient in a VVER as a function of assembly burn-up for two initial enrichments



5.2.7 Minimum (most negative) boron concentration reactivity coefficient

The relevant initiating events include: malfunction of the volume and boron control system and start-up of an inactive coolant loop with lower boron concentration. In these initiating events, the injection of coolant with a lower boron concentration than nominal increases reactivity. For conservatism the bounding limit should be more negative than the calculated values.

As Figure 18 illustrates, the boron reactivity coefficient tends to become less negative as the initial enrichment increases, so that satisfactory fulfilment of this limit is not likely to be adversely affected at very high burn-ups. Since the magnitude of the boron coefficient is particularly small in MOX cores, this bounding limit is not likely to be transgressed with MOX fuel.

5.2.8 Shutdown margin and minimum trip reactivity

For all initiating events, assurance of subcriticality following reactor trip is a basic design requirement. Specifically, the design requirement is to demonstrate a sufficiently high reactivity worth of control rods such that the reactor can at all times be shut down safely. The shutdown margin is essentially the reactivity worth of control rods minus the reactivity effect of moderator and fuel temperature feedback. At high burn-ups, spectral hardening causes the reactivity worth of control rods to decrease in magnitude. This can be seen for the VVER-440 equilibrium cycles in the third column of Table 15, which gives the EOC HZP trip reactivity worth. The lower trend line in Figure 19 shows the same data plotted versus average discharge burn-up.

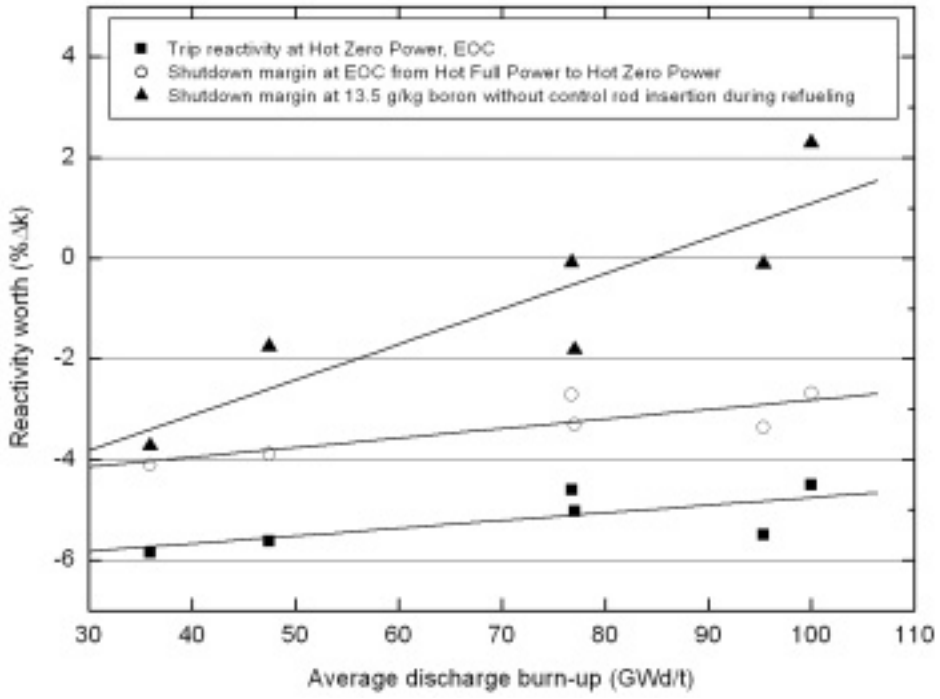
The corresponding shutdown margin from hot full power to hot zero power at EOC (fourth column in Table 15 and middle trend line in Figure 19) decreases with burn-up (the same trend is also seen in PWRs [35]), though for VVER-440 it remains within acceptable bounds. However, the

Table 15. Shutdown characteristics for VVER-440 equilibrium cycles

Case	Average discharge burn-up [GWd/t]	Trip reactivity at EOC HZP* [%]	Shutdown margin from EOC HFP to HZP * [%]	Shutdown by 13.5 g/kg boron without control rods during refuelling [%]	Re-criticality temperature after scram from EOC HFP* [°C]
1	35.94	-5.84	-4.10	-3.72	150.1
2	47.46	-5.61	-3.88	-1.75	158.2
3	76.76	-4.60	-2.71	**-0.08**	**205.0**
4	100.05	-4.49	-2.68	**2.30**	**207.1**
5	77.06	-5.02	-3.30	-1.81	188.7
6	95.33	-5.49	-3.36	**-0.11**	187.1

* The most effective control rod is stuck at the upper position. Safety-related parameters exceeding the present limit values are denoted by bold figures.

Figure 19. Trip reactivity worth and shutdown margins in VVER-440 equilibrium cycles as a function of average discharge burn-up



shutdown margin during refuelling with 13.5 g/kg soluble boron (fifth column in Table 15 and upper trend line in Figure 19) shows a very adverse trend (due to the combined effect of reduced control rod reactivity worth and smaller soluble boron reactivity worth) and eventually goes outside acceptable bounds in the absence of burnable poisons. Similar results would be expected for PWR cores.

Table 15 also shows the re-criticality temperature for EOC HFP core conditions. This is the temperature at which the shutdown margin is eroded in a cool-down fault and the core becomes critical again. There is a limiting temperature for re-criticality which is exceeded in Cases 3 and 4 (neither of which have burnable poisons).

All the shutdown parameters are sensitive to the presence of burnable poisons and different trend lines apply to the poisoned and un-poisoned cases; Figure 19 only shows one trend line for each parameter for clarity. Similar results would be expected for PWRs, though the details are very specific to the particular plant.

The smaller control rod reactivity worths that are characteristic of MOX cores, combined with more negative moderator temperature coefficients, further reduce the shutdown margins. Indeed, the minimum shutdown margin requirement can limit the maximum MOX fraction in the core.

5.2.9 Excess reactivity at BOC

The relevant initiating event is malfunction of the volume and boron control system.

With extended cycles, higher critical boron concentrations and consequently higher excess reactivity at BOC can be expected. MOX cores tend to have lower BOC excess reactivities (due to the combination of low thermal flux and the effect of fertile conversion in ^{240}Pu), which may be beneficial at very high burn-ups.

5.2.10 Maximum reactivity of one control assembly

The relevant initiating event is control rod ejection or withdrawal. The hard neutron spectrum at high burn-ups leads to a smaller control rod worth; consequently, compliance with this limit is not likely to be affected. The same is also true for MOX cores.

5.2.11 Maximum reactivity of the working control rod group

The relevant initiating event is uncontrolled control rod group withdrawal. The comments in Section 5.2.10 are also applicable here.

5.2.12 Isothermal re-criticality temperature

The relevant initiating event is increased heat removal by the secondary system. This bounding limit is affected in the same way as the shutdown margin.

5.2.13 Effective delayed neutron fraction

All transients controlled by non-zero excess reactivity are influenced by the effective delayed neutron fraction β_{eff}. β_{eff} decreases with increasing burn-up, due to the increasing contribution to fissions from ^{239}Pu (which has a smaller delayed neutron fraction than ^{235}U). The decrease in β_{eff} is not very large, and its impact is not expected to be very significant. This can be seen from Table 16, which gives calculated BOC and EOC effective delayed neutron fractions for the six VVER-440 equilibrium cycles.

MOX cores tend to have lower delayed neutron fractions, though at low burn-ups this is only a limited effect. It is only slightly dependent on cycle burn-up and is not expected to be a major concern at very high burn-ups.

Table 16. Effective delayed neutron factors for VVER-440 equilibrium cycles

Cycle	Average discharge burn-up [GWd/t]	Burn-up state	β_{eff}
1	35.94	BOC	0.0059337
		EOC	0.0051941
2	47.46	BOC	0.0059075
		EOC	0.0052580
3	76.76	BOC	0.0058890
		EOC	0.0054028
4	100.05	BOC	0.0059578
		EOC	0.0055013
5	77.06	BOC	0.0057697
		EOC	0.0053861
6	95.33	BOC	0.0057626
		EOC	0.0054908

5.3 Pressure vessel irradiation embrittlement

Very high burn-up fuel cycles will require a reassessment of pressure vessel neutron fluence. Neutron irradiation of the pressure vessel can induce changes in material properties, especially near weld locations, that can result in embrittlement. Weld embrittlement was a very major concern in some early LWRs and though more modern plants are less vulnerable, it is still an area that needs to be examined very carefully.

Very high burn-up fuel will have a small impact on the high-energy neutron fluence (> 1 MeV), largely due to the change in fission contribution between ^{235}U and Pu. The average number of neutrons from ^{235}U fissions is lower that for ^{239}Pu and ^{241}Pu fissions and the energy release per fission is also slightly smaller. As fuel is irradiated, plutonium fissions gradually contribute more and more to the fast neutron fluence, and this is reflected in a change in the fast neutron production per unit energy output. For very high burn-up fuel cycles, the relative contributions of ^{235}U and Pu fissions over the lifetime of the fuel will be different and the impact of this change on vessel fluence will need to be assessed. It is not thought to have a very large effect on UO_2 cores, but it is potentially more significant for MOX cores.

A possibly larger impact might be seen from changes in the loading pattern. As discussed in Section 5.1.2, there may be constraints in applying LLLP in very high burn-up cycles. One of the main benefits of LLLP has been to reduce pressure vessel neutron fluences. If it is necessary to use less aggressive LLLP in very high burn-up cycles, there may be an unavoidable adverse impact on pressure vessel fluences. It is recommended that the dependence of vessel fluence on burn-up should be systematically investigated. A review of the adequacy of neutron transport codes for calculating vessel fluence at high burn-ups is also considered necessary. If the pressure vessel lifetime is adversely affected, this will be an important input to the overall operational and economic justification for very high burn-ups. This question is one which will need to be addressed on an individual basis for each plant and it is not possible to attempt to address it in this review.

5.4 Research and development requirements for very high burn-ups

Nuclear design codes for LWRs (typically comprised of an evaluated nuclear data library, a lattice code and a whole-core nodal code) are currently not validated for application to very high average discharge burn-ups. Moreover, the nuclear design codes typically only model a limited number of fission products and actinides that may not be adequate to maintain accuracy at very high burn-ups and may need to be extended. Another possible limitation is that the few-group constants for the nodal calculations of the entire core (and those for the pin-wise calculations) are parameterised according to the burn-up and other technological parameters (e.g. density, temperatures, and concentrations of selected isotopes) and the parameterisation will need to be extended at high burn-ups. It is therefore considered that further refinement of the depletion chain modelling and few-group parameterisations will be needed to maintain accuracy at very high burn-ups. In addition, further extension of the validation databases will be necessary if very high burn-ups are adopted. Further validation could be obtained from lead assembly loadings in commercial LWRs and by continual update of the databases as average discharge burn-ups gradually increase.

Additional validation of the nuclear codes and nuclear data may be needed to cover very high burn-ups. Experimental data may be needed to demonstrate adequate validation and some suitable experimental facilities, such as PROTEUS [36], are potentially available. Current and completed tests [37 at the PROTEUS facility using UO_2 and MOX assemblies with high burn-up provide the opportunity for validation of nuclear reactor physics codes. It is recommended that these experimental results be used for future OECD/NEA benchmark exercises.

Chapter 6

OPERATIONAL, SAFETY AND LICENSING ISSUES

This chapter considers operational, safety and licensing questions relating to the thermal-hydraulic performance of LWR fuel assemblies and fuel thermo-mechanical behaviour. In keeping with the limited scope of the expert group, thermal-hydraulics and fuel behaviour are considered for current fuel designs only.

6.1 Thermal-hydraulics

Thermal-hydraulics issues may potentially arise at very high burn-ups. Areas identified include critical heat flux, internal heat transfer within the fuel pin and indirect effects due to changes in assembly design. The following sub-sections consider these issues in turn.

6.1.1 Critical heat flux

In both PWRs and BWRs, the critical heat flux ultimately limits the power density in the core and is the key parameter that needs to be considered in the thermal-hydraulics analysis. Heat fluxes in excess of the critical value take the fuel cladding into a different heat removal regime where the clad temperature rises suddenly, with increased risk of cladding failure.

6.1.1.1 Oxidation and crud layer

During the long-term irradiation of fuel in the reactor, the build-up of corrosion and crud production can influence the critical heat flux. These processes lead to an increased wetted cladding surface area, and the general view is that the critical heat flux increases with increasing crud thickness [38,39]. However, some measurements seem to contradict this conclusion [40], and it cannot be excluded that the effect and its sign may be dependent on parameters such as the cladding material, water chemistry and temperature during the irradiation. Additional critical heat flux measurements for cladding materials to be used at high burn-up are advisable. These measurements should be made on highly oxidised fuel pins with a well-established crud layer.

6.1.1.2 Core axial and radial power distribution

In PWRs and BWRs the power distributions can change significantly at high burn-ups. Lower powers in the upper part of a PWR or VVER core can lead to relatively low burn-up values in this region due to the partially inserted control rods and the higher temperature. At EOC, when the control rods are withdrawn, the power distribution can become more top-peaked and the axial position of the minimum DNBR can re-position higher up the core because of the more onerous coolant conditions there. The axial power gradient is large in this upper region, while the critical heat flux correlations are

usually measured at a constant axial power. This effect can be taken into account by using a “form factor” or an additional safety reserve. This phenomenon is more likely to be pronounced and probably needs to be studied for very high burn-ups, but is not thought likely to be a significant problem.

6.1.2 Heat transfer inside the fuel pin

It is well known that in LWRs the fissionable ^{239}Pu mostly originates from the resonance capture of ^{238}U and accumulates next to the pellet surface because of the strong self-shielding (this is a temperature-dependent effect). As a consequence, at high burn-ups the power and the burn-up distributions are concentrated near the pellet surface. Heat transfer parameters (gap conductance, fuel conductivity, thermal diffusivity, fuel heat capacity and stored energy) in the fuel depend on burn-up and temperature. In steady-state and slow transients, the heat transferred to the coolant is equal to that produced inside the pellet and the outer cladding surface temperature is dependent only on the coolant conditions, therefore the above effects do not play any role. However, for fast transient events such as control rod ejection or fast boron dilution, the time dependence of heat transfer to the coolant and consequently the DNB ratio are significantly influenced by the heat transfer parameters and especially the gap conductance [41]. It is important that realistic thermal models should be applied, especially at very high burn-ups. For example, it has been shown that the use of a flat radial power distribution instead of the realistic one is non-conservative. Similarly, the use of gap conductance assuming zero burn-up leads to non-conservative estimates of the DNBR (at zero burn-up the fuel/clad gap is open, effectively insulating the cladding from the fuel and delaying the clad temperature rise, which is non-conservative).

6.1.3 Indirect effects on thermal-hydraulics due to the potential fuel assembly design modifications

At very high burn-ups, fuel assembly design modifications may be beneficial for various reasons, and several areas for possible design changes suggest themselves.

The adverse impact of the harder neutron spectrum and the higher burn-up or/and higher boron concentration may need to be counter-acted by increasing the H/U ratio. The lattice pitch and the number of fuel pins per assembly may also be subject to modification for similar reasons. To counter the expected increase in the assembly-wise power peaking factor at BOC, increased coolant mixing inside the assembly and increased flow area in the most highly loaded sub-channels may be advantageous; these would ensure a lower sub-channel temperature rise.

Depending on the extent of assembly design modifications, several aspects of the thermal-hydraulic design may need to be adapted, including coolant mixing, sub-channel heat-up, coolant flow rate in different parts of the assembly, hydraulic resistance, critical heat flux modification due to the change of the assembly geometry and the avoidance of assembly lift-off. In BWRs, there may need to be an increase in re-circulation flow (to compensate for the increased power mismatch in a high burn-up core, which has implications for assembly lift-off).

6.2 Thermal-mechanical fuel performance

Assuring satisfactory thermal-mechanical performance of the fuel is a very important requirement for very high burn-ups. Historically, LWR fuel designs have evolved as discharge burn-ups increased, and as knowledge of physical mechanisms and materials properties has improved to the extent that modern LWR fuel assemblies routinely achieve very high reliability. It will be very important to ensure that these high standards of reliability not be compromised at very high burn-ups.

Fuel performance is dependent on a variety of phenomena which need to be correctly quantified to ensure an adequate fuel behaviour predictive capability. Burn-up extension exposes the fuel to increasing challenges and therefore implies a need for qualified models and codes at high burn-ups for safety assessments of different operational conditions. While the database for burn-ups > 70 GWd/t[1] is limited at present, some trends regarding performance developments can be reliably inferred; it is considered that models based on fundamental principles can be constructed that will allow extrapolation to 100 GWd/t.

Property changes are in general a continuous function of exposure, which in many countries has made it possible to allow higher and higher burn-ups in small increments, backed up by lead irradiation assemblies and test reactor experience. As the permissible burn-ups are further increased beyond current limits, it is not expected that any fundamentally new phenomena will be seen. In fact, in the past 20 years only one fundamentally new phenomenon has emerged that had not previously been recognised when burn-ups were much lower (< 35 GWd/t), namely the so-called "rim effect", discussed in the next section.

6.2.1 Rim structure formation

Since the relatively recent realisation of the importance of rim structure formation, usually referred to as the "rim effect", considerable efforts have been made to arrive at an empirical as well as phenomenological and mechanistic understanding. The present understanding cannot be considered complete, and further work will be needed at the phenomenological level, especially for very high burn-ups. Rim structure formation has important influences on thermal, fission gas release and PCI behaviour under normal operation and transient conditions.

The formation of high burn-up structure (HBS, also known as "rim structure formation" or "rim effect") occurs in nuclear fuel materials as a consequence of excessive fission damage and in-growth of fission products. HBS is also referred to as rim structure, as it is first formed near the outer surface (rim) of UO_2 fuel pellets where the burn-up is highest because of resonance neutron capture by ^{238}U forming fissile ^{239}Pu. It can also be observed in Pu-rich agglomerates of MOX MIMAS fuel. The HBS is fully developed at about 70 GWd/t local burn-up and is characterised by sub-micrometer-size grains and large micrometer-sized pores.

Rim structure formation starts at the pellet periphery where the number of fissions is highest due to the build-up of plutonium. The same structure has also been generated in highly-rated UO_2 fuel disks kept at sufficiently low temperature and in plutonium-rich agglomerates of MOX MIMAS fuels, showing that the number of fissions is the key parameter for this phenomenon to develop. Fully developed HBS is characterised by a loss of optically definable grains, i.e. the development of sub-micrometer-size grains. Electron probe micro analysis (EPMA) shows that the matrix is depleted of fission product xenon (Xe) with a concurrent formation of high concentration of pores with a volume fraction of 10-15%, depending on burn-up. X-ray fluorescence (XRF), micro-coring and secondary ion mass spectroscopy (SIMS) show that Xe is retained in the pores.

Apart from burn-up, several parameters play a role in the HBS formation and may influence the threshold value at which HBS starts to develop: HBS formation occurs only at sufficiently low temperatures (< 900-1 000°C), as demonstrated by restructuring of Pu-rich agglomerates of MOX fuels and with disk fuel irradiation at different temperatures. There are some indications that the threshold burn-up might be increased by increasing grain size and higher hydrostatic stresses. A similar effect is

[1] When not specified otherwise, the burn-up refers to the mass of heavy metal and assembly average.

likely to be induced by higher temperature and lower fission density, as this phenomenon is related to a competition between creation and recovery of point defects as well as to fission gas super-saturation in the matrix followed by fission gas bubble formation at low temperature. This is clearly a field where further research and development will be necessary in order to precisely determine the influence of the different parameters.

For burn-ups approaching 100 GWd/t, a much larger volume will be affected by HBS formation. Late in life, high burn-up fuel will be operated at low power and thus low temperature (< 1 000°C), in which case the entire pellet cross-section may, in principle, eventually be affected by HBS formation. While fuel behaviour during normal operation (swelling, fission gas release, failed fuel) may be affected, the overriding impact of HBS formation is expected to be on off-normal behaviour (e.g. fuel relocation, fission gas release and dispersal during LOCA/RIA).

6.2.2 Thermal behaviour

As most material properties and phenomena depend on temperature, a thorough knowledge of fuel thermal behaviour is essential for modelling and safety analyses. The thermal behaviour of nuclear fuel is a fundamental aspect of the safe and reliable operation of nuclear power plants. Ref. [42] is a useful source of information on the behaviour of high burn-up LWR fuel.

The thermal behaviour of ceramic fuel (UO_2, MOX) is dominated by fuel conductivity and gap conductance. The latter shows a complex evolution due to the counteracting phenomena of pellet-clad gap closure leading to improved gap conductance and fission gas release which degrades the gap conductance. However, at high burn-up gap closure is complete and changes in gas composition due to continuing fission gas release have only a small influence on fuel temperatures as evidenced by experimental data. No significant change of this situation is expected to occur in the range 70-100 GWd/t.

The bonding layers of zirconia and fission product compounds between the fuel and cladding surfaces grow during irradiation and act as a thermal barrier. The exact composition of these layers is not known, but their conductivity is very low, similar to those of UO_2 and ZrO_2. Fuel behaviour modelling codes should be able to adequately account for the change of heat flow resistance that increases as some of the cladding metal is transformed to oxide, but also decreases as the contact between fuel and cladding is improved. However, as these layers grow in thickness with increasing burn-up it will be necessary to obtain experimental data regarding the properties of these layers for the highest burn-ups.

The degradation of the thermal conductivity of UO_2 and MOX fuel has been investigated for many years, and a considerable amount of in-pile and out-of-pile data has been accumulated. In-pile temperature data indicate a linear trend of increasing fuel temperatures at constant power, and this observation can be modelled with a modification of the 1/T phonon term which most conductivity correlations contain explicitly. Some theoretical considerations suggest a saturation effect as the pellet burn-up approaches 100 GWd/t, but so far the existing data base does not support this. The linear trend can be assumed to continue into the 70-100 GWd/t range.

Fuel porosity has an effect on fuel temperatures and the porosity changes with burn-up, most notably in the rim area affected by HBS formation. The pores in the rim area are spherical and their effect on the temperature distribution can be evaluated analytically and easily incorporated in fuel modelling codes. The correlation between local burn-up and pore formation is known and is extendable to even higher burn-ups.

One purpose of temperature calculations is to estimate the thermal energy stored in the fuel and the specific heat capacity C_p needs to be known. Burn-up has only a minor influence on C_p as shown by measurements on simulated and real burn-up fuel. The change is compatible with the slight increase of number of atoms due to fissioning and can be predicted from first principles. Current correlations can therefore be applied to even higher burn-ups.

6.2.3 Fission gas release

The fission gas inventory in a fuel rod increases proportionally with burn-up, while the decreasing fuel thermal conductivity and changes of fuel microstructure reduce the ability of the fuel to retain the fission gas. This, together with the rod void volume reduction caused by cladding creep-down and fuel swelling, increases the potential for high rod internal pressures with increasing burn-up. Moreover, even when the gas is retained in the fuel – especially at grain boundaries – its sudden release may potentially impair fuel integrity in, for example, reactivity transients which cause rod overheating.

Fission gas release studies have been conducted to explore short- and long-term release kinetics for different types of fuel as well as limiting and mitigating factors. Investigations of the effect of large grain fuel on release behaviour have confirmed the predicted delay of fission gas release caused by the longer diffusion path to the grain surface. Large grains are related to a smaller surface area of grain boundaries and hence probably to a lower amount of retained gas at these grain boundaries; therefore it is expected that the burst release during transients can be reduced in large grain fuels, but of course only as long as HBS formation has not taken place.

Rod overpressure studies at the Halden Reactor Project [43] have shown that a large overpressure (> 100 bar) under PWR operating conditions is needed to cause clad lift-off and increasing fuel temperatures. Where lift-off is the limiting criterion, this may provide a margin for even higher burn-ups than currently licensed.

The possibility of gas release from the high burn-up structure could aggravate fission gas release and rod pressure development. Current experience indicates that the fission gas is to a large extent retained in highly pressurised pores in the HBS affected zone. However, the possibility cannot be excluded that HBS that develops at higher temperatures and is subsequently subjected to higher temperatures during irradiation (as the HBS grows deeper into the pellet) may show a less favourable behaviour. Such a trend has already been observed, but further research and development is needed to address this question.

As burn-up increases, more and more gas is trapped in bubbles at grain boundaries (both in the HBS areas and in regions that have retained the original microstructure). The consequences are enhanced fission gas release due to the accumulation of open grain boundary bubbles, and grain boundary segregation leading to fuel embrittlement and inter-granular fracture, which can have an impact on performance especially during abnormal transients.

Ref. [44] is a useful source of data on fission gas release in water reactor fuels.

6.2.4 Pellet-clad interaction

Pellet-clad interaction is less prevalent as a failure cause. This is due to the combination of improved fuel designs and restrictions enforced by regulation or fuel vendor specification. While there should be no immediate concern that this situation should change drastically as burn-up is increased further, continued control of cladding oxidation and ensuing hydrogen pick-up is essential for keeping

PCI-induced failures at the present low level. It seems that both the outer rim zone and the pellet-clad bonding layer contribute favourably to good PCI behaviour of high burn-up fuel as far as the available experience indicates.

Doped fuels with large grains (chromia, niobia, alumino-silicate, etc.) are being considered because of their possible benefits for improved gas retention, but they also exhibit an enhanced visco-plastic behaviour and favourable crack patterns with numerous radial cracks on the pellet surface mitigating local stress concentrations in the cladding. Such fuels may also mitigate the effect of axial ratcheting which can be observed in conjunction with power cycling since stresses induced in the cladding are relaxed faster by fuel creep.

Ref. [45] provides an up-to-date perspective on PCI in water reactor fuels.

6.2.5 Fuel and cladding dimensional changes

After fuel densification is complete, which occurs relatively early during irradiation, fission product swelling is the major cause of dimensional changes of the fuel. An extensive database exists regarding swelling data, and experimental results are compatible with theoretical considerations which account for the extra volume required by the fission products. Matrix swelling and swelling due to the rim effect can therefore be expected to continue inexorably with increasing burn-up at the same rate as for current fuels.

At high burn-ups fission gas swelling may become accentuated due to the ever growing number of atoms not being released from high burn-up fuel under normal operational conditions at relatively low temperatures. Furthermore, fission gas swelling may be much more important for high-temperature accident situations than for normal operation conditions. Burn-up extension should therefore be supported by experimental and theoretical work relevant to fission gas swelling.

There is evidence that fuel bonded to the cladding transmits its inexorable matrix swelling to the cladding, inducing permanent elongation. This must be taken into account by providing sufficient head room for the free end of the fuel rods in an assembly.

6.2.6 Failed fuel behaviour

If failure occurs, the tightly bound column of high burn-up fuel impedes steam transport and is therefore protective against the development of secondary fuel failure degradation. In the event of extended failure, the presence of large amounts of fuel with tiny grains at the fuel periphery (i.e. the region affected by HBS) might induce an increased risk of release of fissile material to the coolant. Current methods for estimating release may not be adequate and may need to be developed further; this is another area where research and development is required.

6.3 MOX fuel behaviour

While most of the issues discussed in the previous section are generic, MOX fuel has some specific features that are briefly outlined in this section.

MOX fuel is currently irradiated to a lower burn-up than UO_2 fuel because the operational database experience is not yet as extensive and MOX fuel has some special features which may imply a higher research and development requirement in order to achieve the same burn-up as UO_2.

For core physics reasons, the linear heat rating at high burn-up may be higher in MOX rods than in UO_2 rods. Secondly, fuel thermal conductivity is slightly lower in MOX than in UO_2 and the centreline temperature is higher in MOX pellets even at the same linear power. As a consequence, fission gas release is generally higher in MOX rods than in UO_2 rods, and the requirements concerning inner rod internal pressure may be more difficult to fulfil.

As the fission of plutonium atoms induces more metallic fission products than uranium fission, there is a higher risk of departure from stoichiometric composition at high burn-up, which might have a deleterious effect on the thermal conductivity of highly burnt MOX fuel. Although experimental results show that molybdenum acts as a buffer, this point has to be kept in mind.

Another special feature of MOX fuel results when the fissile atoms are not homogeneously distributed in the pellets on a microscopic scale, especially in MOX fabricated by the MIMAS process (blending of a master-mix of UO_2 and PuO_2 with depleted UO_2 powder). Inside these Pu-rich agglomerates, the local burn-up can be very high: HBS formation occurs and most of the fission gas inside these agglomerates is stored in bubbles, increasing the potential for higher gas release during transients.

The creation of helium during irradiation is much higher in MOX fuel than in UO_2 fuel, mainly because of the decay of ^{242}Cm, generated by neutron captures in Pu and Am. In BWR fuel rods with low initial helium pressure, the helium so generated is released and contributes to the internal pressure of the rods; in PWR rods with higher initial helium pressure, the same effect could occur at very high burn-ups.

Research and development is already necessary and in progress to develop a MOX fuel able to reach an assembly burn-up of 70 GWd/t. As the burn-up qualification of MOX fuel does not yet extend even to 70 GWd/t, it is too early to comment on the possibilities of achieving even higher burn-ups. But taking into account the isotopic concentration of plutonium recovered from UO_2 fuel irradiated at 70 GWd/t or more, the plutonium content of a MOX fuel aiming at a burn-up higher than 70 GWd/t would exceed the void coefficient barrier of 12 w/o content (see Section 4.3.2). Therefore a burn-up around 70 or 75 GWd/t is likely to be considered as an ultimate burn-up for MOX fuel in a standard LWR recycling scheme.

6.4 Fuel design

The discussion in the previous section has highlighted various phenomena potentially affected by high and very high burn-ups. Some of these can be influenced by fuel design modifications in order to conserve or restore the margins to limits imposed by regulatory and operational constraints.

6.4.1 Fuel

The development of doped fuels is actively being pursued in the nuclear industry. Dopants such as chromium and niobium are used to control the microstructure in such a way that improved performance is obtained. Fuel dopants are used to:

- increase the average grain size from about 10 μm to > 50 μm and thus the fission gas retention capability, both during normal operation and during transients;

- enhance creep properties for better PCMI performance;

- delay the high burn-up structure formation by increasing the grain size;

- improve thermal conductivity (for example by incorporating the fuel grains within a metallic precipitate matrix).

Additives are also required to compensate for excess reactivity of fresh fuel. Gadolinium is already widely used as neutron poison in BWRs and also offers advantages for other reactor types. Erbium is considered for the same purpose. These additives affect not only the nuclear, but also the thermo-mechanical properties of the fuel. In particular, gadolinia fuel has a lower thermal conductivity and is generally used with slightly lower uranium enrichment. It is therefore essential that sufficient data on the performance at very high burn-up be accumulated and that compatibility with other dopants be demonstrated.

6.4.2 Fuel rod

The increase of internal pressure due to fission gas release and volume reduction is one of the most limiting factors regarding burn-up extension. In addition to the development of fuels with better fission gas retention capability, several adjustments of the fuel rod design are possible to keep the inner pressure within permissible limits. Among these are:

- Increase of internal volume by increasing the plenum length. Though easily implemented, it can only be achieved at the expense of the amount of fuel in the core and therefore reduced total power production.

- Decrease of the as-fabricated fill gas pressure. Although this countermeasure is easy to implement, any reduction must be carefully balanced with the function of initial rod pressure to provide a restraint against uncontrolled clad creep-down and collapse, and to minimise the reduction of gap conductance due to contamination with released fission gas.

- Introduction of plenum subdivision (to adjust the effective plenum volume) and getter materials in the plenum (to absorb gaseous fission products and reduce internal pressurisation).

The limiting phenomenon of clad lift-off is the result of complex interactions between fuel temperatures, fission gas release, fuel swelling and cladding creep. Improved fuel rod designs as well as the development of new and improved existing cladding alloys with respect to mechanical and creep properties will be important for burn-ups beyond current limits.

6.4.3 Rod cladding

External clad corrosion was the first serious limitation that was encountered in PWRs during the drive towards higher burn-up. For rod burn-ups higher than 60 GWd/t, Zircaloy-4 cladding typically develops a thick layer (100 μm) of zirconia generated by oxidation of the zirconium alloy by the coolant. This oxidation is associated with the pick-up of hydrogen which in turn induces embrittlement of the material. Furthermore, for a corrosion depth exceeding 100 μm, spalling of the zirconia layer may occur and lead to accelerated hydriding and embrittlement.

A large research and development effort has been performed all over the world to develop and qualify new zirconium alloys with better corrosion resistance with the addition of niobium (M5, ZIRLO, E110, MDA [46], NDA [47]). Table 17 gives the elemental compositions of some zirconium alloys which are at the stage of industrial development.

Table 17. Elemental compositions of new zirconium alloys

Zirconium alloy	Sn	Fe	Cr	Ni	Nb	O
Zircaloy-2	1.2-1.7	0.07-0.2	0.05-0.15	0.03-0.08	No data	1 100 to 1 400 ppm
Zircaloy-4	1.2-1.7	0.18-0.24	0.07-0.13	No data	No data	
M5	No data	No data	No data	No data	1.0	
ZIRLO	1.0	0.1	No data	No data	1.0	
E110	No data	No data	No data	No data	1.0	
MDA	0.7-0.9	0.18-0.24	0.07-0.13	No data	0.45-0.55	
NDA	1.0	0.27	0.16	0.01	0.1	

The available results provide evidence that the new alloys can reach rod burn-ups of 70 to 80 GWd/t with very low clad corrosion (see examples in Figures 20, 21 and 22) and low hydrogen pick-up (of the order of 100 ppm instead of more than 500 ppm), and there is some hope that much higher burn-ups, e.g. 100 GWd/t, might be achievable.

However, it should be noted that qualification of these alloys for such high burn-up needs further research and development in order to obtain a complete picture of their mechanical properties at very high burn-ups and under all relevant conditions such as RIA and LOCA events.

Figure 20. Maximum corrosion depth in M5 alloy compared with Zircaloy-4 [48]

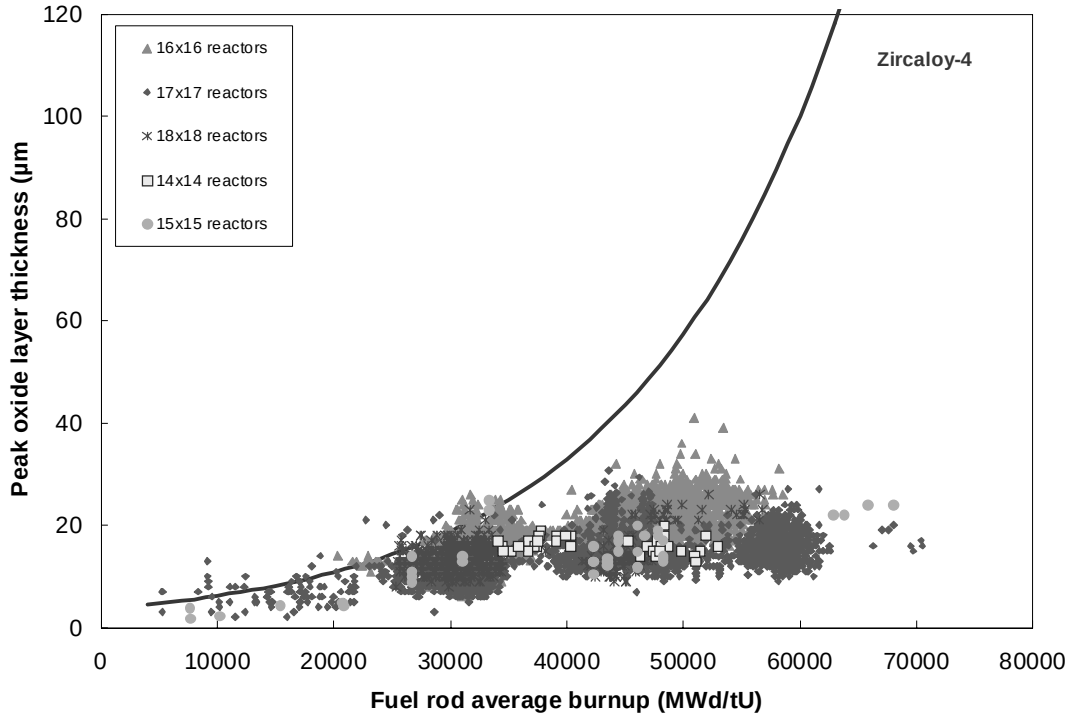


Figure 21. Corrosion depths in low tin ZIRLO compared with standard ZIRLO [49]

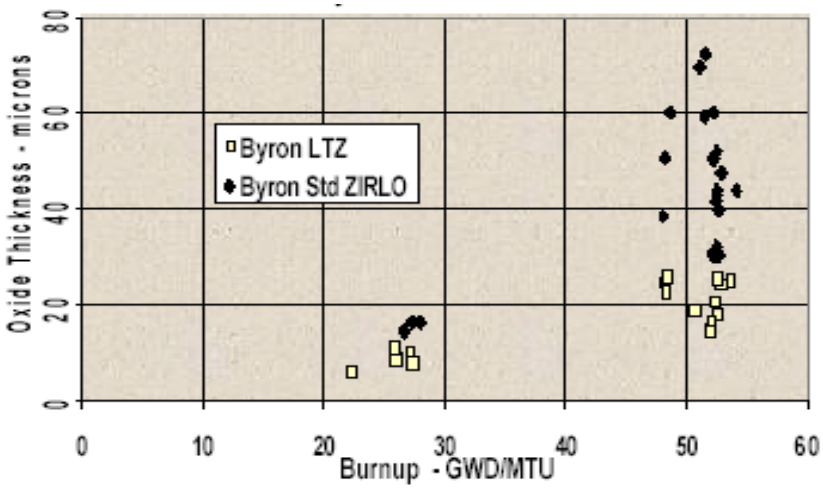
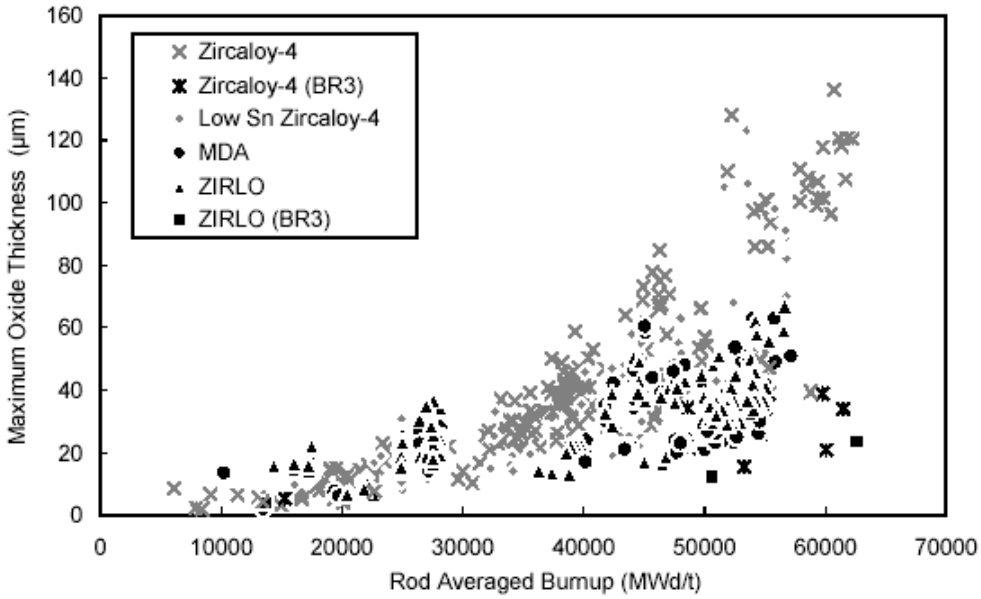


Figure 22. In-reactor corrosion behaviour of MDA cladding compared with Zircaloy-4 and ZIRLO [50]



6.4.4 Fuel assembly mechanical structures

Extended burn-up implies extended in-core residence time. The fuel assembly structure and components must therefore be able to maintain their function for a longer time. Ref. [51] provides a recent review of fuel assembly structural behaviour. Fuel assembly designs, materials and components must be optimised with respect to:

- Minimal and uniform dimensional changes, especially growth, in order to avoid warping.
- Susceptibility to hydrogen pick-up and embrittlement.

- Improved rod-to-grid fretting resistance, as the average time available for fretting is prolonged before unloading. New designs of grids have already been implemented for that purpose, but their behaviour still needs to be qualified at very high burn-up.

- Improved hold-down spring design to accommodate fuel assembly growth and to prevent fuel assembly bowing that can induce incomplete control rod insertion.

The experimental data indicate that cladding elongation follows fuel swelling when bonding between fuel and cladding is established. More axial space must be provided to accommodate the increasing length unless some margin can be gained by more growth-resistant cladding materials.

6.5 Transient/accident behaviour

There is a general consensus that in most transients there are no fundamental mechanisms affecting fuel behaviour which are specifically impacted at high burn-ups. As noted earlier in Section 6.2, most of the relevant fuel behaviour mechanisms vary in a continuous manner with burn-up. While the precise history of a transient will be different at high burn-ups, such that a different outcome is possible regarding fuel survival or failure, the behaviour of the fuel will not be affected by any new physical phenomena. The exceptions however, specifically reactivity-initiated accidents (RIA) and loss of coolant accidents (LOCA), are considered in this section.

6.5.1 Fuel failure in reactivity-initiated accident

In terms of the occurrence of fuel rod failure (cladding failure) during a reactivity-initiated accident (RIA), a high burn-up effect would manifest itself as a failure at a lower fuel enthalpy level due to PCMI (pellet-cladding mechanical interaction) during an early phase of the transient. Possible PCMI failure at the lower enthalpy level would occur primarily due to ductility reduction of the cladding in higher burn-up fuels. Hydride cluster formation, resulting in a peripheral dense hydride layer (hydride-rim) in stress-relieved zirconium alloy cladding and radially-oriented hydride clusters in re-crystallised zirconium alloy cladding, is the cause of ductility reduction in combination with irradiation hardening. Accordingly, the improved performance of recently developed alloys (M5, ZIRLO, MDA, NDA, etc.) in terms of corrosion resistance could improve the survivability of high burn-up fuels in an RIA. However, qualification of higher burn-up fuels with these cladding materials for RIAs is needed with additional and extensive experiments in order to demonstrate improved survivability.

In order to evaluate the PCMI loading and to determine the threshold for PCMI failure, the pellet expansion mechanism in the early phase of the RIA transient as well as cladding ductility reduction must be clarified. The role of fission gas accumulation in fuel pellets in the PCMI loading remains an outstanding question. Researchers in Japan [52,53] and in the US [54] have separately argued that the cladding deformation is caused primarily due to thermal expansion of the UO_2 matrix and the fission-gas-induced pellet expansion is negligible in the early phase of RIA transients, based on existing data and analysis. On the other hand, a theoretical approach proposed by French researchers [55] predicts that thermal expansion of fission gas accumulated in grain boundaries provides a strong PCMI load to the cladding even during the early stage. It is assumed in the theory that the thermal expansion of fission gas accumulated in bubbles at grain boundaries induces a rapid mechanical de-cohesion of these grain boundaries, thus providing a strong PCMI load to the cladding; however, this mechanism has not yet been confirmed. Therefore, a key question concerning the pellet behaviour has to do with the fission-gas-induced expansion caused by pressure loading of heated fission gas confined in grain boundaries. The mechanical properties and strength of the grain boundaries, as well as how early and how fast the pressure elevation and subsequent reduction occur, should be determined.

At higher burn-ups a stronger PCMI load can be expected due to the HBS formed at the pellet periphery and in particular the large amount of fission gas accumulated in the structure, but the role played by the HBS bubbles remains unclear at this stage.

6.5.2 Fission gas release in reactivity-initiated accident

It is assumed that the fission gas accumulated in grain boundaries is released from the fuel pellets to the rod plenum during an RIA. A larger inventory of the inter-granular gas in higher burn-up fuels accordingly results in a larger fission gas release. The effect of the HBS formation on fission gas release remains unclear.

6.5.3 Post-failure event in reactivity-initiated accident

When cladding fails during an RIA, contact between dispersed fuel pellets and coolant water produces mechanical energy as a post-failure event. Grain boundary separation accelerates fuel fragmentation resulting in a larger surface area of finely fragmented fuel particles. Although the fuel particles are in solid form, thermal interaction between fuel fragments and coolant water can produce mechanical energy. One can assume that HBS formation will result in more violent post-failure events, enhanced fragmentation and resulting stronger mechanical energy generation. However, the effect of the HBS on the events also remains unclear. There is an observation in the NSRR experiment, which shows a bonded rim layer among finely fragmented post-test fuel particles.

6.5.4 Rod failure in loss-of-coolant accident

During a loss-of-coolant accident (LOCA) fuel rods are exposed to steam at high temperature and subsequently quenched by emergency core cooling water. The fuel cladding becomes brittle due to severe oxidation during the high-temperature period, and thermal shock resistance is reduced. Since cladding embrittlement is dependent primarily on the amount of oxidation, it is important to evaluate the oxidation kinetics. Corrosion layer growth, the higher hydrogen concentration and the larger neutron dose in higher burn-up fuels are all factors which can affect the oxidation kinetics. It has been established that the corrosion layer has a protective effect on steam oxidation at high temperatures and the effect would be maintained at higher burn-ups. The effect of hydrogen on oxidation kinetics has also been investigated. The increase in hydrogen concentration generally increases the oxidation rate above 1 223 K. However, the increase is limited to 5% if the hydrogen concentration is below 800 ppm. If the concentration becomes higher than that at very high burn-ups, the effect of hydrogen could become significant in the oxidation kinetics. Radiation damage should anneal out with temperature increase during the LOCA transient and is believed to have no direct effect on the oxidation kinetics. Neutron irradiation at high burn-ups may have an influence on the oxidation kinetics if it causes pronounced changes in microstructure, such as the accumulation of precipitates.

The influence of hydrogen absorbed before the transient on the thermal shock resistance of oxidised cladding has been examined. The non-fracture/fracture boundary is reduced with an increase in the hydrogen concentration, though the extent is dependent on the axial constraint condition on quenching. The influence of the reactor irradiation, including corrosion and hydrogen absorption, on the thermal shock resistance are currently being investigated in the US and Japan. Evaluation of thermal shock resistance should be based on results from those experiments.

Cladding materials will need to be improved or newly developed for further burn-up extension. (This would be of benefit for corrosion and hydrogen absorption under reactor operation conditions.) On the other hand, careful studies are required because changes in alloy components may cause different behaviours under LOCA conditions.

The effects of a decrease in gap size and chemical bonding between pellet and cladding should be taken into account for cladding failure in specifying the fuel rod geometry. These factors may affect rupture behaviour, oxidation at the inner surface and the secondary hydriding from the inner surface. The NRC/ANL high burn-up fuel test programme and the LOCA tests at the Halden reactor will give some information on these effects.

Ballooning and rupture behaviour are also important from the viewpoint of coolant channel blockage and fission gas release. It is well known that the circumferential increase caused by ballooning is strongly connected with the phase structure of Zircaloy at failure. Recent studies have indicated that the phase transformation temperature changes following an increase in hydrogen concentration and the circumferential increase by ballooning is affected by it. Therefore, the circumferential increase of high burn-up fuel cladding can be expected if the phase transformation temperature changes with hydrogen concentration increase are taken into account. A possible reduction of the failure temperature has also been suggested with an increase in a hydrogen concentration. It is probably attributable to a decrease of cladding strength with hydrogen absorption. More investigation may be necessary into cladding strength at failure temperatures.

6.5.5 Fuel relocation in loss-of-coolant accident

Fuel pellets crack during irradiation in a reactor. It has been pointed out that pellet fragments can relocate and accumulate in the ballooned region caused by mechanical shock at rupture. This phenomenon increases the peak clad temperature and oxidation at the rupture position compared with the situation assuming no relocation, though the increase depends on the filling ratio of fragments. Information on the possibility and timing of the fuel fragment relocation and the filling ratio, etc., will be obtained by the in-pile LOCA tests at the Halden reactor. At very high burn-ups, significant bonding between pellets and cladding may reduce the possibility of the fuel fragment relocation.

Chapter 7

ECONOMICS

This chapter considers the economics of very high burn-up fuel cycles and attempts to illustrate how the fuel cycle cost varies with average discharge burn-up. The fresh fuel requirement, which can be measured either by the number of assemblies or the mass of fuel loaded, varies inversely with average discharge burn-up. This represents a direct economic saving for some fuel cycle cost elements such as fuel assembly component procurement, new fuel fabrication cost and some back-end costs that scale proportional to the number of assemblies. On the other hand, the investment in each fuel assembly increases because of the higher initial enrichment requirement and if there are any technological improvements to the fuel assembly design required for very high burn-ups. There is also the possibility that some back-end unit costs (i.e. the cost per kg), may increase because of the increased technological demands of handling very high burn-up fuel. The burn-up dependence of the overall fuel cycle cost depends on the balance between these competing effects. In this report the possible beneficial effect of increased burn-up on planned unavailability and power upgradings has not been studied.

It is clear that for average discharge burn-ups up to at least 60 GWd/t, fuel cycle costs decrease with burn-up. This explains the historic trend of LWR burn-ups having steadily increased with time. Whether this trend still applies to very high burn-ups > 60 GWd/t is not clear. In fact, the fuel cycle economics is so dependent on the specific situation that might apply to a particular country or utility that there is probably no single conclusion that is generally applicable. This chapter discusses the issues and presents some examples that illustrate how fuel cycle costs can vary with burn-up under different circumstances.

7.1 Total generation costs

Nuclear power plant costs can be categorised under four main headings: capital and investment costs, operation and maintenance (O&M) costs, fuel cycle costs and decommissioning costs. The capital cost encompasses all the costs associated with civil engineering, procurement and installation of components and all the associated mechanical, electrical systems, etc. This is sometimes referred to as the “substantive cost”, being the actual cost of the concrete, metal, engineering, etc. Because the construction period of nuclear power plants typically lasts for several years, this represents a large investment prior to any income flow from generation. Historically most of the reactors currently operational were funded by governments. However, in the increasingly prevalent private market, such investment is obtained by borrowing from private investors. Irrespective of the source of capital finance, borrowing large sums of money for substantial periods of time entails a cost and a risk. The risk is usually framed in terms of the monetary discount rate, the rate of which varies between countries and is usually significantly higher for private investors. The discount rate is a measure for the amount in which future cash flows are weighted with respect to current cash flows. The more uncertain future cash flows, the higher the discount rate that is applied by the decision maker. Given the lifetime of nuclear reactors, the monetary discount rate is therefore one of the most significant factors that influences the capital cost component.

Future cash flows are dependent on inflation (for those cash flows that follow “normal” yearly cost increases) or escalation (for those cash flows with a yearly increase higher than inflation). While for a thorough investment decision using a net present value calculation each cash flow should be inflated on a yearly basis and discounted back to the present time, often only a discount rate (above inflation) is used.

With respect to capital investment costs there is a tendency for the capital costs of newly built plants to be higher than in the past, mainly because of increasingly stringent safety requirements. For new plants in deregulated markets an even more important factor affecting investment costs will be the high monetary discount rates demanded by private investors; these reflect the perceived high financial risk of investment in nuclear power plants. On the other hand, the emergence of evolutionary LWR designs with simplified safety systems is a positive development that would counteract rising investment costs. A specific factor which varies from country to country is that the investment costs are dependent on the number of replica plants deployed, being lower for a country such as France where a large number of plants have typically been built to the same design.

For a reactor which is already operational, the capital cost component is usually regarded as a fixed, historic and sunk cost over which the utility has no influence. In the context of this report, the use of very high burn-up fuel cycles can have no influence on these historic capital costs. However, there may be an indirect influence if very high burn-ups allow the overall capacity factor to be increased; increased operating capacity implies higher generating revenues which would reduce the levelised production costs, most conveniently expressed in $/MWh. Realistically, high burn-up fuel cycles might yield an improvement of only a few per cent in capacity factor. Even so, the reduction in levelised costs could potentially be very worthwhile from a utility’s perspective.

Operating and maintenance encompasses all the costs associated with operating a nuclear power plant (salaries, rent, insurance, materials, plant maintenance costs, waste treatment/disposal and the costs associated with refuelling), though it is usual practice to exclude the cost of the fuel itself. The levelised operating and maintenance costs usually follow inflation, but are insensitive to the monetary discount rate as they are spread more or less uniformly out in time, with roughly the same time profile as the generating revenue and the two discount factors cancel out. The opportunities for savings on O&M costs are limited because of minimum regulatory standards that affect the number of staff required and the required level of maintenance. The absolute operating and maintenance cost would not normally be affected to any significant extent by the adoption of very high burn-up fuel cycles, but there may be a modest (though still worthwhile) benefit for the levelised operating and maintenance cost if higher capacity factors can be achieved. Although the potential savings usually only amount to a few per cent, or even fractions of a per cent, they nevertheless equate to large absolute savings that are very worthwhile to a utility. Operating and maintenance costs are subject to systematic trends, such as the need to comply with ever more stringent legal and licensing requirements, balanced by the implementation of more efficient operating methods and in some countries (such as the USA), by the pooling of experience and knowledge through the merging of small utilities into large ones.

The fuel cost is normally treated separately from operating and maintenance. It includes, at the front-end, the cost of procuring uranium ore, conversion and enrichment services, fuel fabrication and transport. Significant front-end costs arise before production and a high discount rate especially penalises the costs of the first reactor core. At the back-end the component costs are those of spent fuel storage, transport and either conditioning/encapsulation for direct disposal or reprocessing, depending on whether a once-through or recycle strategy is preferred by the utility or mandated by political constraints. Included within the back-end costs are the cost of disposal of spent fuel in the eventual repository either as intact fuel or vitrified high-level waste. Both reprocessing and direct disposal would also produce varying quantities of intermediate- or low-level waste whose conditioning and

disposal is also included in the fuel cycle cost. A high discount rate decreases the costs of back-end cost items which are incurred after a long-term delay, such as the disposal of spent fuel. Very high burn-up fuel cycles can have a direct influence on both the absolute and levelised fuel cycle costs, as will be seen later. Fuel cycle costs are subject to systematic external factors such as rising and falling uranium ore prices and also to very specific factors, such as the costs of radioactive waste storage. An important factor for fuel cycle costs is the long intervals needed for fuel to pass through the various stages of the fuel cycle.

Decommissioning costs for a modern LWR can be expected to amount to several hundred million dollars. Nevertheless, since these costs are not incurred until many years after the plant is shutdown, discounting effects typically reduce the levelised decommissioning cost to 1% or less of the total levelised generating cost. Different countries have various approaches to providing for decommissioning. In some countries a provisioning discount rate is applied to the decommissioning component using a lower discount rate than applies to the other cost components. This is designed to be pessimistic and the levelised decommissioning cost is increased substantially. Nevertheless, even with such pessimism, the levelised decommissioning cost remains a very small fraction of the overall cost. As such, the effect of very high burn-up fuel cycles on the decommissioning levelised cost is negligible, certainly smaller than the inherent uncertainties on what the eventual decommissioning costs will actually be.

To summarise, the capital costs usually represent the largest single component, typically representing somewhat more than half of the total generation cost (typically 50-60%), while operating and maintenance and fuel cycle costs each account for typically 20-25%. Decommissioning costs are very small, typically less than 1% of total generation costs. The precise breakdown of costs varies from country to country and from utility to utility. There is no single economic model that represents all situations and care is required when interpreting statements on the effect on fuel cycle economics of very high burn-ups.

7.2 Fuel cycle cost structure

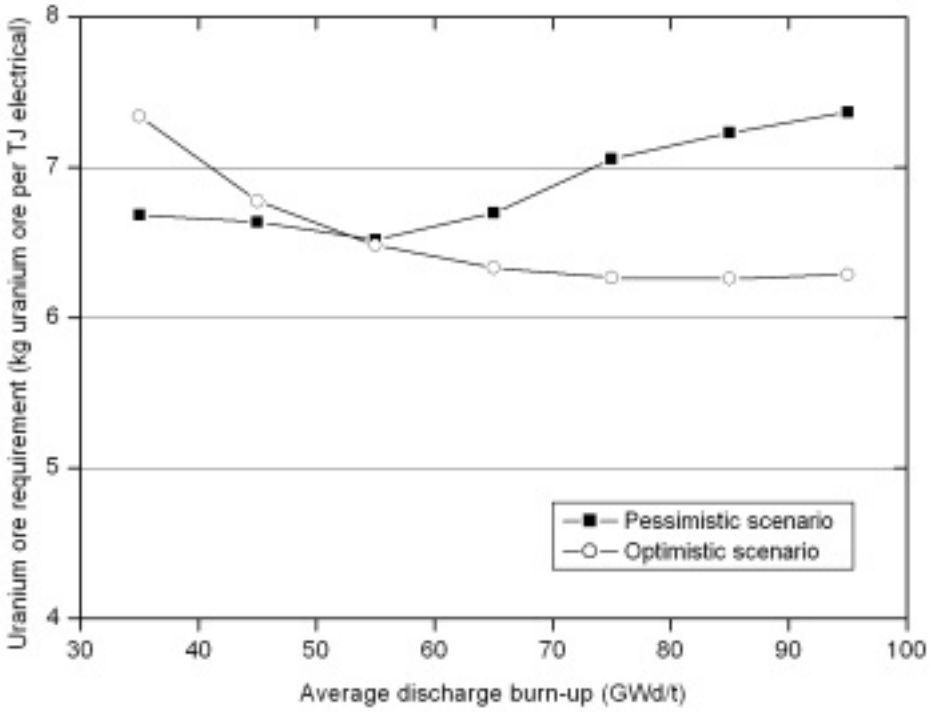
This section considers the fuel cycle cost in more detail, breaking it down into its component parts.

7.2.1 Uranium procurement and conversion

Uranium procurement and conversion can logically be considered together. For LWRs the uranium needs to be converted to uranium hexafluoride UF_6 as a precursor to enrichment. The natural uranium mass needed to provide 1 kg of enriched fuel is given by the enrichment feed factor $f = (e_{prod} - e_{tails})/(e_{feed} - e_{tails})$, where e_{prod} is the enrichment of the enriched uranium product stream, e_{tails} is the enrichment of the depleted uranium tails stream and e_{feed} the enrichment of the feed stream to the enrichment plant. For given feed and tails enrichments, the feed factor increases linearly with product enrichment.

The higher initial enrichments needed for high average discharge burn-ups implies increased uranium ore procurement and uranium conversion costs per fuel assembly. On the other hand, the number of fresh fuel assemblies is inversely proportional to burn-up. The net result of these competing effects can best be visualised by considering the uranium utilisation, defined as the mass of uranium ore needed to supply 1 TJ of electrical energy. The uranium utilisation is illustrated in Figure 23 for two PWR scenarios, one which assumes the initial enrichment versus burn-up relation of Table 1 (in which burn-up extension is achieved by increasing the cycle length with a fixed four-batch refuelling scheme) and a more optimistic enrichment/burn-up relation in which the cycle length is constant and the refuelling fraction decreases.

Figure 23. Variation of uranium ore utilisation versus average discharge burn-up for two enrichment/burn-up relations



While the pessimistic curve in Figure 23 incorporates the effect of residual neutron absorption in gadolinia, the optimistic curve assumes zero burnable poison penalty. This latter curve uses an extrapolation of the Westinghouse initial enrichment/burn-up relation from Table 2 and should be considered as the most optimistic case conceivable. A quadratic fit was made to the Westinghouse data an extrapolated to very high burn-ups. The initial enrichment ε as a function of average discharge burn-up B_d (in units of GWd/t) is:

$$\varepsilon = 0.0002B_d^2 + 0.0395B_d + 1.6378$$

While extrapolation so far outside the range of data is difficult, the resulting trend is consistent with the Leibstadt data of Table 3 (see Chapter 4).

The pessimistic case suggests a shallow minimum at approximately 55 GWd/t average burn-up, while the more optimistic curve does not reach a minimum until 85 GWd/t. The difference in behaviour of the two curves is due to a combination of factors: the burnable poison residual absorption effect; incomplete optimisation of the loading patterns, and also because of the different strategies adopted for increasing burn-up in the two scenarios.

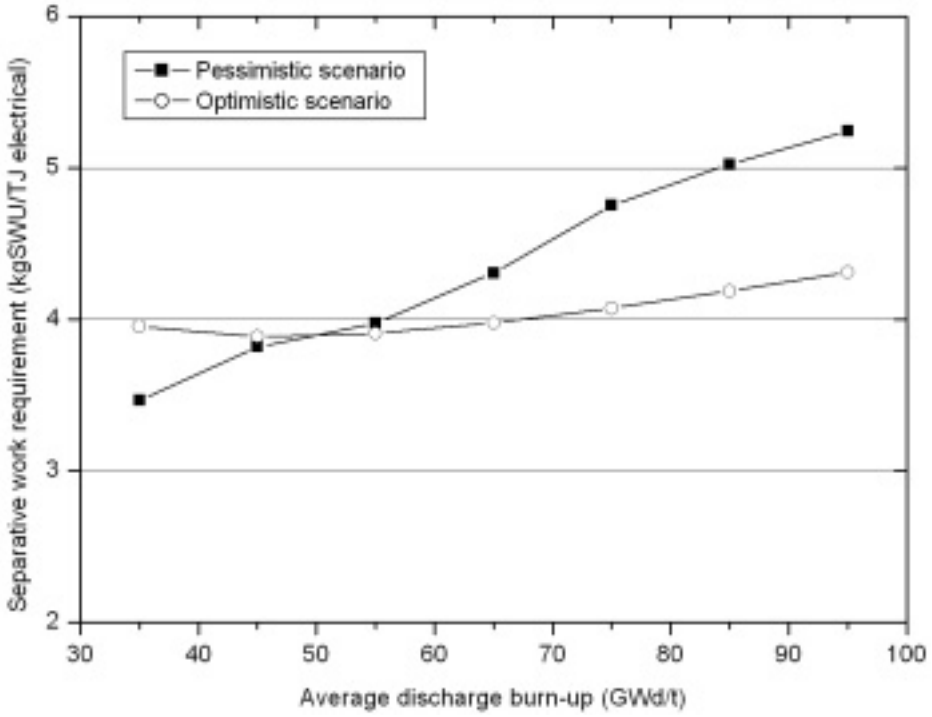
The two curves in Figure 23 are best regarded as defining best- and worse-case scenarios. It is not certain how realistic the optimistic curve is and whether it is actually achievable in practice. Furthermore, different utilities will choose different fuel management strategies to suit their particular requirements. By adopting these two scenarios as best and worst scenarios, it will be possible here to indicate quantitatively the likely range of variation of fuel cycle cost versus burn-up, within which any specific utility's specific analysis might be expected to lie.

The levelised cost of uranium ore and conversion as a function of average burn-up exactly matches Figure 23. The absolute value of the levelised cost depends on the market price for uranium ore and conversion services. Uranium ore prices have historically been very low but have recently risen sharply to approximately 70 $/kg, which increases the relative weight of ore and conversion costs in relation to overall fuel cycle costs. It is thought [56] that uranium prices are unlikely to return to the very low values ~25 $/kg that have prevailed for the past 20 years until the recent price rises. Therefore the sample case calculations specified in Appendix C have been carried out using a value of 50 $/kg.

7.2.2 Separative work requirement

Enrichment costs are charged according to the number of separative work units (SWUs). The unit SWU is that of mass and the cost is normally quoted in $/kgSWU. The separative work requirement depends on initial enrichment in a complex manner. As discharge burn-up increases, the separative work requirement per kg of finished fuel increases, but the fuel requirement decreases. The specific separative work requirement per unit energy output varies with discharge burn-up as shown in Figure 24 for the same pessimistic and optimistic scenarios used previously in Figure 23. While the separative work requirement for the pessimistic scenario increases with burn-up, that for the optimistic scenario shows a shallow minimum at approximately 50 GWd/t before rising slightly for higher burn-ups.

Figure 24. Specific SWU requirement versus average discharge burn-up



The increase in specific separative work requirement with burn-up implies that the total demand for enrichment plant services will increase for very high burn-up scenarios. This has the potential to affect SWU prices. Even if it does not, the influence of enrichment unit cost on the overall fuel cycle cost will grow. There is a trend for enrichment unit costs to decrease with time as enrichment plants become ever more efficient to operate and if it continues, it will be favourable for very high burn-ups.

7.2.3 Fuel fabrication

Since the fuel requirement scales inversely with discharge burn-up, the fuel fabrication requirement will decrease with increasing burn-ups. If the unit cost of fuel fabrication in kg/tHM were to remain constant, the levelised cost of fuel fabrication would scale inversely with average discharge burn-up as well. However, this simplistic argument is unlikely to apply in practice because very high burn-ups will impose increasingly difficult demands on the fuel that will necessitate improvements to the fuel assembly design and to the basic materials. Apart from the increased cost of more complex fuel assembly components and fabrication, fuel vendors will need to invest substantial amounts of money in R&D to support very high burn-ups and they will need to recover this by charging a premium for high burn-up fuel designs. Such a premium will to some extent offset the inverse relationship of fuel fabrication cost with burn-up, though to what extent it is impossible to predict.

A very significant consideration is the imperative to relax the 5.0 w/o enrichment limit and the possibility that fuel vendors may need to invest heavily in upgrading or even replacing their facilities to accommodate it. This can only be expected to have a major influence on unit prices for fabrication. The fact that demand for fuel fabrication will decrease with very high burn-up fuel cycles may be sufficient in itself to distort fuel vendors' pricing, as they may need to carry fixed overheads on a diminishing supply base or may lose some of the cost advantages associated with higher throughputs. There is therefore the possibility that a trend to very high burn-ups will distort the market situation for fuel fabrication.

7.2.4 Spent fuel management

Since the spent fuel mass decreases inversely with burn-up, very high burn-up fuel cycles would benefit from a reduction in the spent fuel management levelised cost as long as unit costs remain constant. Whereas this may apply to a once-through fuel cycle with direct disposal of intact fuel assemblies, the discussion of Sections 4.5 through 4.8 highlight the complex technical issues that will affect spent fuel transport, storage and any conditioning/reprocessing operations. In light of these complexities, it is difficult to envisage a situation wherein spent fuel management unit costs will not increase for very high burn-up fuel cycles. This will offset the benefit of reduced spent fuel mass to an extent which is impossible to predict at this stage. In any case, the relationship between spent fuel management unit costs and discharge burn-up is likely to be affected by local considerations that are specific to individual countries and utilities.

The situation is complicated further in that some countries (USA, Spain) impose a levy on generation to cover spent fuel management. In the specific case of the USA, the levy is charged proportional to electricity generation and does not take into account the smaller spent fuel volume that will arise with high burn-ups. In this case therefore, there is in principle no incentive for burn-up increase. Nevertheless in both countries utilities are increasing burn-up; in the USA because of the lack of storage capacity for spent fuel; in Spain because utilities are concerned about a possible a change of legislation towards volume dependent costs.

7.3 Investments and infrastructure

Up to the present time LWRs have been able to progressively increase discharge burn-ups without requiring any fundamental changes to the underlying investment base. Of course, substantial investment has been needed for R&D to justify increasing burn-ups and to continually upgrade facilities to meet changing environmental and regulatory requirements. But the R&D investments have

supported small incremental changes that have not demanded a fundamentally new infrastructure. This report has highlighted that only limited increases in discharge burn-up are feasible within the limitation of the existing LWR infrastructure and there will come a point where key investment decisions will need to be taken.

The most pressing of these decision points is that of extending the 5.0 w/o fabrication limit. For BWRs it is already clear that utilities would benefit from relaxing this limit. For PWRs the enrichment limit is not quite so imminent, but it will be within 5 or 10 years. Considerable new investment in fuel fabrication plants will be needed if this limit is to be relaxed. It is probable that significant investment will be needed to upgrade existing fuel fabrication plants to relax the limit and it is possible that some existing fabrication plants may need to be partially or completely replaced. Such a decision is a difficult one for fuel fabricators, and they would need to be certain that the investment is justified by a demand from the utilities.

The 5.0 w/o limit is particularly onerous because it affects not just the fuel fabrication plant, but also other aspects of the infrastructure such as transport. There may be technological uncertainties associated with, for example, transport criticality clearance that will increase the investment uncertainty and risk for a fuel fabricator; transport is crucial because a criticality clearance for a transport container would need to be sanctioned internationally for it to be universally applicable. Since time scales for licensing approval of fuel fabrication plants and transport containers for fuels > 5.0 w/o can reasonably be expected to be long, criticality clearance for > 5.0 w/o may already be a critical path item; if LWR discharge burn-ups are to continue their upward trend this issue probably needs to be addressed in the very near future.

There are potentially many in-reactor implications, including possible upgrades to the boronation systems in a PWR and the re-circulation flow in BWR (as noted in Chapter 4). The arrangements for fresh fuel handling and storage will need to be reviewed. There are also possible implications for plant lifetime, discussed earlier. This is relevant since there is a well-established trend to increase plant life beyond 30 years, possibly up to 60 years. A utility must factor all these investment requirements into its decision as to whether to pursue very high discharge burn-ups. It is possible that such considerations would be sufficient to dissuade some utilities from very high discharge burn-ups, particularly for plants with only a limited remaining lifetime.

The back-end aspects of very high burn-ups, such as higher decay heat outputs, higher neutron outputs, higher fission product and minor actinide inventories are different in that there is no single threshold point at which current facilities can no longer be used. Existing reprocessing plants, for example, have limitations on heat output and shielding that might be obstacles in the longer term if their entire spent fuel throughput was from very high burn-up stock. However, if there is a mix of moderate and high burn-up fuel assemblies these plants can to some extent counteract the effects of high burn-up by blending. There may also be the possibility of blending with historic low burn-up stocks.

Eventually, however, it may be necessary to upgrade or replace the existing reprocessing facilities if very high burn-ups become prevalent. This will be a major investment decision for reprocessing companies that will have an impact on reprocessing unit prices. Similar arguments will apply to the direct disposal conditioning plants that will be necessary at some point in the future. The capital and operating costs of whatever operations are carried out in such plants will be affected by the source terms for decay heat, neutron output, etc., and will affect unit prices in the same way.

The investment decisions noted here represent the key uncertainties that prevent a definitive economic analysis of the fuel cycle cost for very high discharge burn-ups. The nature of these investment decisions is very specific to different countries, individual utilities and fuel cycle service

providers. There cannot conceivably be a single answer that would apply to all situations. It may well be necessary for fuel cycle service providers to cater to varying individual requirements. All that can be achieved in this study is to sketch out guidelines to assist decision makers with the imperfect knowledge that currently exists. This is attempted in Section 7.4.

7.4 Examples of economic approaches

7.4.1 Results from earlier studies

The OECD/NEA has previously carried out studies of PWR fuel cycle costs, the most recent of which was published in 1994 [57]. However, the scope of these studies did not include evaluating the effect of increasing discharge burn-up; the 1994 study only considers a single discharge burn-up of 42.5 GWd/t, so is not directly relevant to this review except in that it provides a fixed reference point for comparison. The only systematic effort to investigate the effects of discharge burn-up on PWR fuel cycle costs was the IAEA WREBUS study [58]. Though now very dated, it was intended to address the question of whether there was a fuel cycle cost incentive to increase discharge burn-ups from the then-prevalent 30 GWd/t to 50-60 GWd/t. The WREBUS study concluded that fuel cycle costs would decrease at least up to discharge burn-ups of 50-60 GWd/t, but was unable to determine if the trend would continue to higher burn-ups. The principal reason for this is that the WREBUS study did not have a reliable initial enrichment versus burn-up relation for burn-ups of 60 GWd/t or more. At that time there was no experience of average discharge burn-ups much in excess of 30 GWd/t, and the extrapolation to 60 GWd/t was a considerable extension of the then-available knowledge. The WREBUS study noted that it was possible to identify an optimum discharge burn-up in some scenarios (such as the US back-end model with a fixed levy on generation). This suggested that there may be a trend for fuel cycle costs to increase for burn-ups beyond ~60 GWd/t, but the study was unable to confirm this suspicion.

The present report aims to continue the work of the WREBUS, but with the advantage of a broader remit that allows all the technical aspects of very high burn-ups to be reviewed as well as the economics. The initial enrichment versus discharge burn-up relations of Section 4.1 are considered to improve on the information available to the WREBUS working group and should allow a more reliable analysis of fuel cycle costs up to ~100 GWd/t. However, this analysis still cannot make any pretensions to be definitive, as many of the economic inputs such as fuel fabrication prices, spent fuel management unit prices, etc., cannot be specified with any confidence for very high burn-ups. Even so, the analyses presented in Section 7.4.2 provide some useful indications within the limits of current knowledge.

7.4.2 PWR economics sample case

An economics analysis was carried out for a sample PWR scenario. This is fully specified in Appendix C. The objective was to carry out an analysis of the fuel cycle levelised cost as a function of discharge burn-up assuming the pessimistic and optimistic initial enrichment versus burn-up relations of Section 7.2.1, while benchmarking with other computer programs and calculation methods for the same input variables. As discussed earlier, the two scenarios can be regarded as defining upper and lower bounds within which any more detailed and realistic analyses might be expected to fit.

While the OECD/NEA study of 1994 [57] is now dated, the basic economics inputs used largely remain valid, and it was decided to use very similar inputs in the sample case. This is helpful as it provides a common reference point for comparison. The sample case represents a modern four-loop PWR of 1 200 MWe.

For consistency with Ref. [57], the base case assumes encapsulation/direct disposal as the back-end route, with encapsulation and disposal costs deferred 40 years after fuel discharge. As in [57], it is assumed that back-end costs are discounted at the same rate as other costs.

Two sets of economics inputs were specified for the sample case. In the first all the economics parameters were simplistically assumed to be independent of discharge burn-up. This applies to fuel fabrication and spent fuel storage transport and eventual disposal. While it is accepted that this is an unrealistic model, it will allow a lower limit to be set and highlights the economic dependencies due to the fundamental physical factors, such as the initial enrichment versus discharge burn-up relation. The second set of economic inputs made some arbitrary assumptions as to how fuel fabrication and back-end costs might vary with discharge burn-up. This serves to illustrate how burn-up-dependent economic inputs might affect fuel cycle costs. Combined with the pessimistic and optimistic enrichment versus burn-up relations, the two sets of economic inputs lead to four cases:

- optimistic enrichment, burn-up-independent unit costs;
- pessimistic enrichment, burn-up-independent unit costs;
- optimistic enrichment, burn-up-dependent unit costs;
- pessimistic enrichment, burn-up-dependent unit costs.

Sensitivity to the monetary discount rate was addressed, with values from 5% to 10% covered. Recognising that financial discounting is not used in all countries, a separate sensitivity case with undiscounted costs was included. Economics analyses can be quite sensitive to the discount rate, especially for fuel for which there is a substantial lead time prior to any revenue being produced. Variation of the fuel fabrication costs and spent fuel management costs as a function of discharge burn-up is already covered by the burn-up-dependent model.

Sections 7.4.2.1 and 7.4.2.2 describe economics calculations carried out by Nexia Solutions and NRG using their respective fuel cycle economics codes FCE [59] and DANESS. These illustrate different trends for fuel cycle costs as a function of average discharge burn-up. Section 7.4.2.3 describes calculations performed for VVER-440, which were carried out using a different set of assumptions.

7.4.2.1 PWR economics sample case – Nexia Solutions results

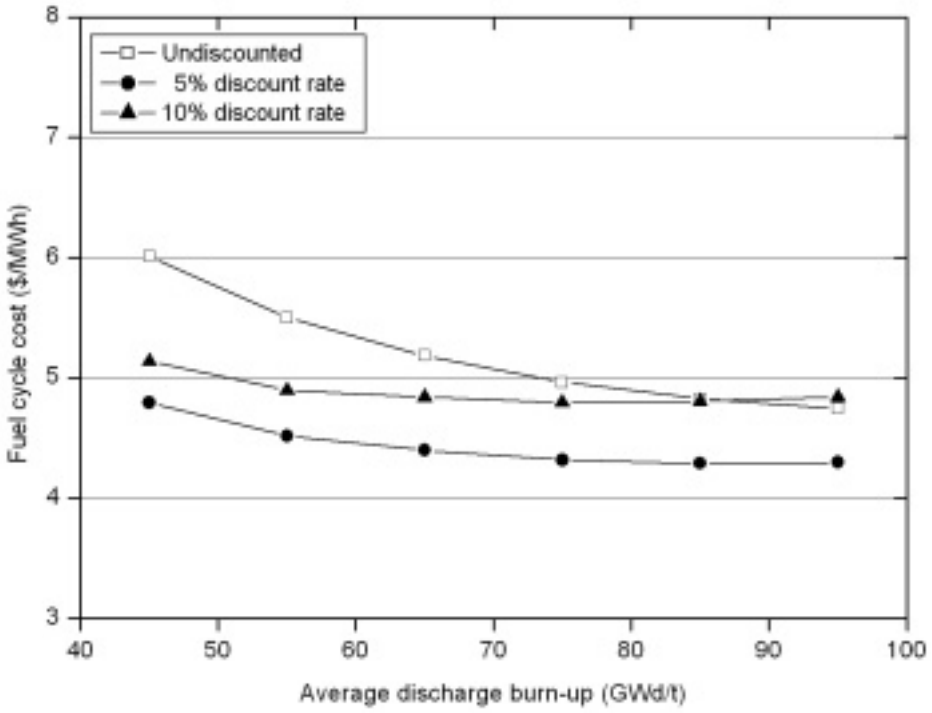
Nexia Solutions carried out an analysis of the PWR economics sample case using its in-house fuel cycle modelling code FCE. This calculates the levelised fuel cycle cost over the entire lifetime of a reactor. The FCE code explicitly analyses the first core and all the transition cores leading to the equilibrium cores, calculating the initial enrichment for all these equilibrium cores needed to approach equilibrium smoothly. The FCE code therefore models a more realistic scenario than the sample case specified, taking full account of the different costs incurred in early non-equilibrium cycles and the last cycle, this being a departure from the sample case, as specified. Although the detailed results are affected to some extent by the extra complexity of the FCE model, the general trends are not, and the results are still useful for illustration here.

Tables 18 to 21 and Figures 25 to 28 show FCE levelised fuel cycle costs as a function of average discharge burn-up and discount rate for the economics sample case for the two scenarios. The optimistic scenario (Table 18 and Figure 25) with burn-up-independent back-end unit costs shows a decreasing trend of fuel cycle cost, levelling off at the higher burn-ups.

Table 18. Fuel cycle levelised cost ($/MWh) versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-independent unit costs (evaluated by Nexia Solutions using FCE)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	6.011	5.500	5.184	4.965	4.829	4.746
5% DR	4.794	4.517	4.399	4.318	4.291	4.299
10% DR	5.138	4.897	4.842	4.797	4.804	4.839

Figure 25. FCE fuel cycle levelised cost versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-independent unit costs



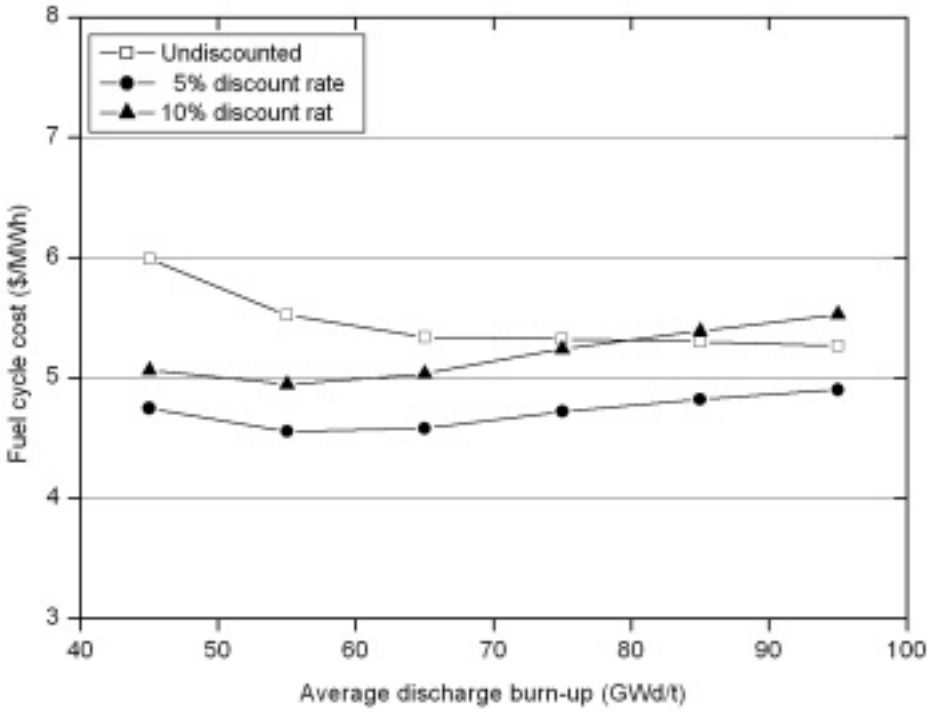
The undiscounted case has the highest fuel cycle cost because in this case the deferred back-end costs do not benefit from discounting and therefore represent a much larger fraction of the overall fuel cycle cost. The 10% discount case has a slightly higher fuel cycle cost than the 5% case. This is because the income from electricity generation is also discounted and the relative timings of costs (some of which are incurred before starting production) and incomes in this scenario slightly favour the 5% case. The same behaviour was noted in [57].

The pessimistic scenario (Table 19 and Figure 26) shows decreasing costs for the undiscounted case, while the two discounted cases show a minimum at approximately 55 GWd/t. The difference in behaviour compared with the optimistic scenario is due to the combined effects of burnable poison residual absorption, non-optimised loading pattern and the assumption of a fixed quarter core loading scheme. The larger weighting of the deferred back-end costs in the undiscounted case causes the monotonic behaviour for that case.

Table 19. Fuel cycle levelised cost ($/MWh) versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-independent unit costs (evaluated by Nexia Solutions using FCE)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	5.986	5.525	5.339	5.324	5.301	5.267
5% DR	4.747	4.557	4.579	4.722	4.821	4.901
10% DR	5.065	4.946	5.035	5.242	5.391	5.528

Figure 26. FCE fuel cycle levelised cost versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-independent unit costs



With burn-up-dependent fuel fabrication and back-end costs, the optimistic scenario (Table 20 and Figure 27) the fuel cycle cost shows a shallow minimum at 55 GWd/t, with an increasing trend between 65 and 95 GWd/t. The pessimistic scenario with burn-up-dependent fuel fabrication and back-end costs (Table 21 and Figure 28), shows an increasing trend with burn-up and there is no minimum (although the gradient between 45 and 55 GWd/t is quite flat).

Table 20. Fuel cycle levelised cost ($/MWh) versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-dependent unit costs (evaluated by Nexia Solutions using FCE)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	6.011	6.044	6.148	6.268	6.421	6.609
5% DR	4.794	4.746	4.804	4.868	4.966	5.088
10% DR	5.138	5.081	5.174	5.255	5.371	5.509

Figure 27. FCE fuel cycle levelised cost versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-dependent unit costs

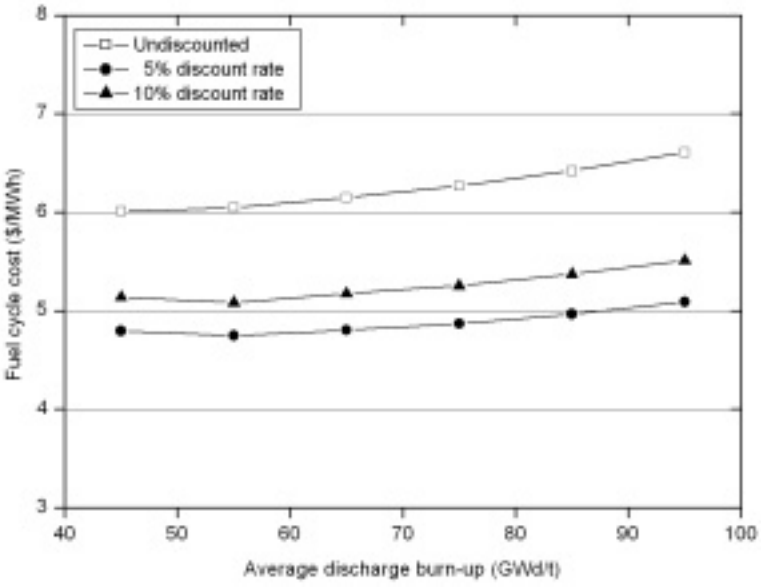
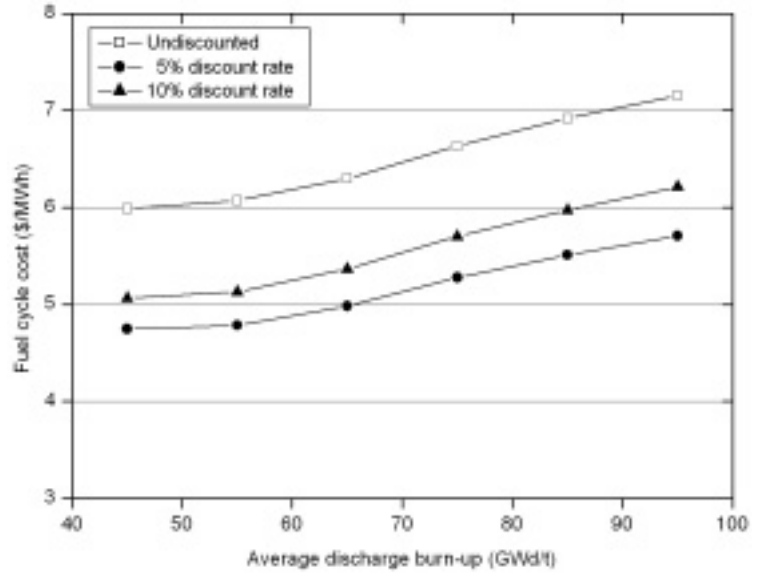


Table 21. Fuel cycle levelised cost ($/MWh) versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-dependent unit costs (evaluated by Nexia Solutions using FCE)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	5.986	6.070	6.299	6.634	6.920	7.154
5% DR	4.747	4.785	4.983	5.277	5.510	5.707
10% DR	5.065	5.130	5.367	5.703	5.971	6.213

Figure 28. FCE fuel cycle levelised cost versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-dependent unit costs



For countries where monetary discounting does not apply, it is possible to conclude that there is a clear fuel cycle cost benefit from increasing burn-ups irrespective of which enrichment/burn-up relation applies, provided that the unit costs for fuel fabrication and back-end services are burn-up-independent (or at least not too steeply rising with burn-up). In these circumstances, there is a clear economic incentive to increase average discharge burn-ups as high as possible.

However, if a non-zero monetary discount rate applies, then the qualitative behaviour of fuel cycle costs can change depending on the enrichment/burn-up relation and only in the most optimistic case (with zero burnable poison residual absorption and burn-up increased by decreasing the reload fraction) does the fuel cycle cost show a benefit from very high burn-ups.

It is difficult to draw any definite conclusions from this analysis that are applicable in all circumstances, as there are different qualitative behaviours depending on the economic model that applies. The economic incentive for very high burn-ups is clearest in the case where an undiscounted economic model applies, as the increasing fuel cycle costs apply irrespective of the initial enrichment/ burn-up relation. However, even in this case the fuel cycle cost benefit of very high burn-ups can be negated if the unit costs increase sufficiently with burn-up. Where economic discounting applies, there are situations where there is an economic benefit from very high burn-ups, but this only applies with the optimistic initial enrichment/burn-up relation. For a more pessimistic relation a minimum fuel cycle cost is obtained at approximately 55 GWd/t.

7.4.2.2 PWR economics sample case – NRG results with DANESS

NRG also carried out a fuel cycle cost analysis of the PWR economics sample case using the DANESS fuel cycle modelling tool. Tables 22 to 25 and Figures 29 to 32 show DANESS levelised fuel cycle costs as a function of average discharge burn-up and discount rate for the economics sample case for the two scenarios and with burn-up-independent and burn-up-dependent fuel fabrication and back-end costs. The DANESS model takes full account of the different costs incurred in early non-equilibrium cycles and the last cycle, similar to the FCE model (see Section 7.4.2.1).

The optimistic scenario (Table 22 and Figure 29), with burn-up-independent back-end unit costs shows a decreasing trend of fuel cycle cost for the undiscounted case only, while the discounted cases show a minimum between 55 and 65 GWd/t. Though the undiscounted case agrees quite well with the corresponding FCE results (Table 18 and Figure 25), the discounted behaviour is qualitatively different. This must reflect differences in the discount approach used in the two codes, at least in part caused by the departure of FCE model from the sample case specification, that are outside the scope of this study.

The DANESS fuel cycle costs for the pessimistic scenario (Table 23 and Figure 30) show the same trends as the corresponding FCE calculations (Table 19 and Figure 26), though the discounted behaviour is slightly different.

Table 22. Fuel cycle levelised cost ($/MWh) versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-independent unit costs (evaluated by NRG using DANESS)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	6.40	5.73	5.42	5.26	5.17	5.14
5% DR	4.83	4.55	4.55	4.66	4.80	4.98
10% DR	5.40	5.13	5.30	5.56	5.87	6.21

Figure 29. DANESS fuel cycle levelised cost versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-independent unit costs

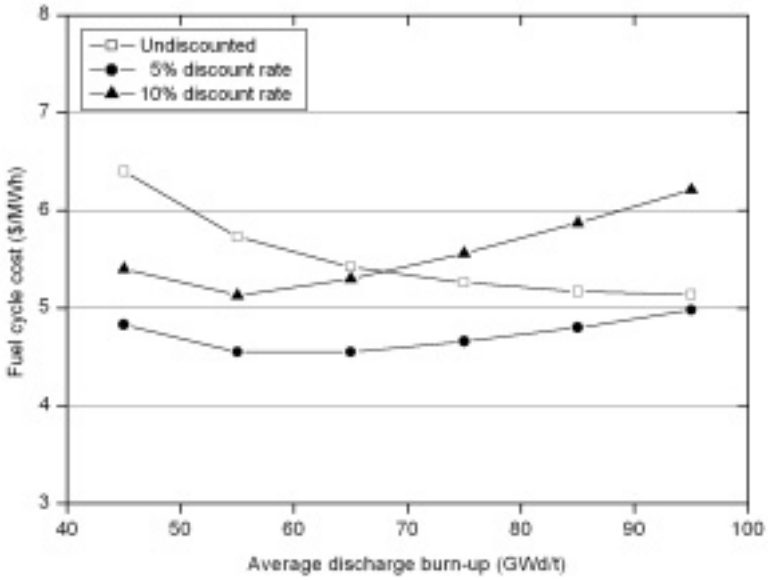
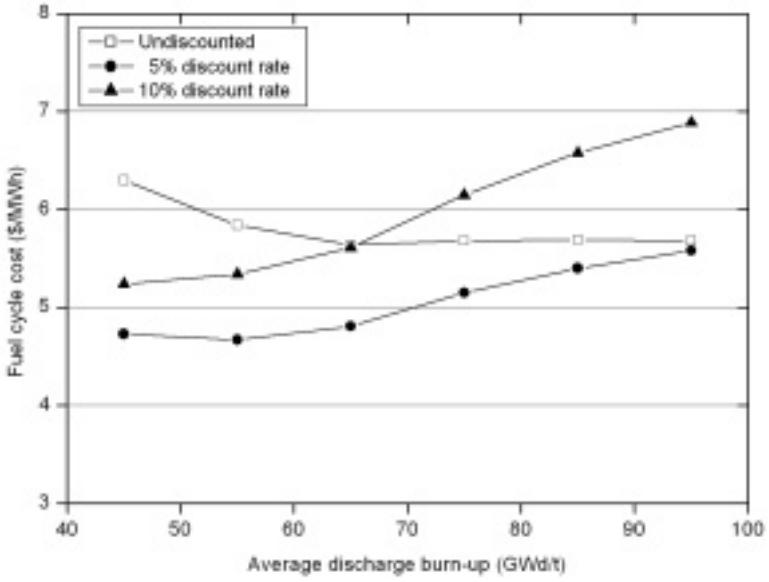


Table 23. Fuel cycle levelised cost ($/MWh)versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-independent unit costs (evaluated by NRG using DANESS)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	6.30	5.84	5.64	5.68	5.69	5.68
5% DR	4.73	4.67	4.81	5.15	5.40	5.58
10% DR	5.24	5.34	5.61	6.15	6.58	6.89

Figure 30. DANESS fuel cycle levelised cost versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-independent unit costs



For both optimistic and pessimistic scenarios, DANESS fuel cycle costs with burn-up-dependent fuel fabrication and back-end costs (Tables 24 and 25; Figures 31, and 32) are much closer to the FCE costs and show the same general trends. Irrespective of the discount rate there is an increasing trend with burn-up for both the optimistic and pessimistic scenarios.

Table 24. Fuel cycle levelised cost ($/MWh) versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-dependent unit costs (evaluated by NRG using DANESS)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	6.40	6.27	6.36	6.52	6.69	6.89
5% DR	4.83	4.69	4.81	5.00	5.22	5.47
10% DR	5.40	5.24	5.48	5.83	6.19	6.60

Figure 31. DANESS fuel cycle levelised cost versus average discharge burn-up and discount rate – optimistic initial enrichment/burn-up relation and burn-up-dependent unit costs

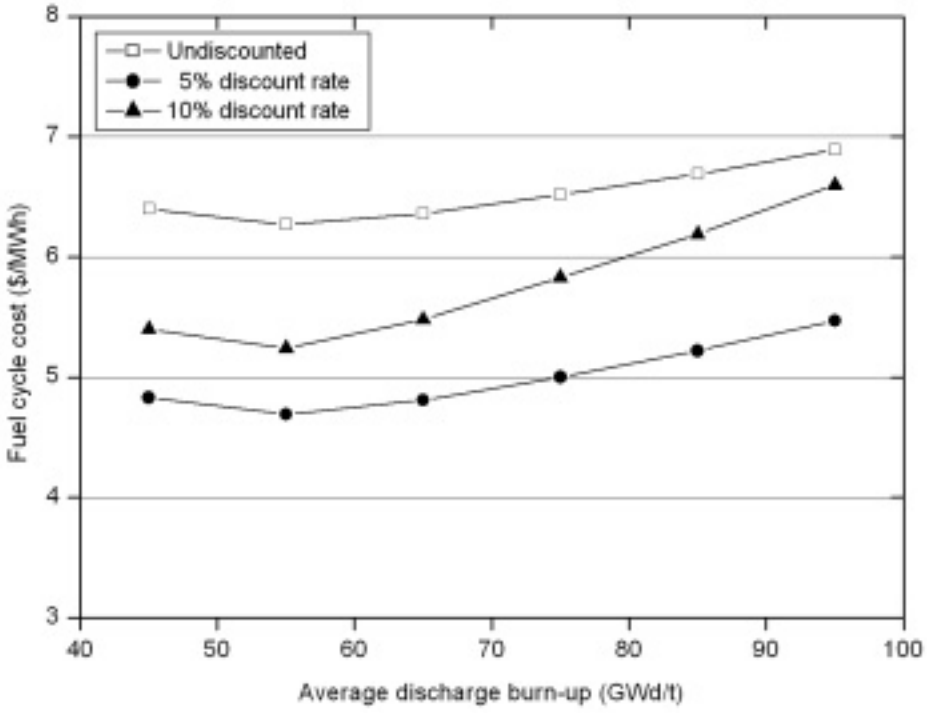
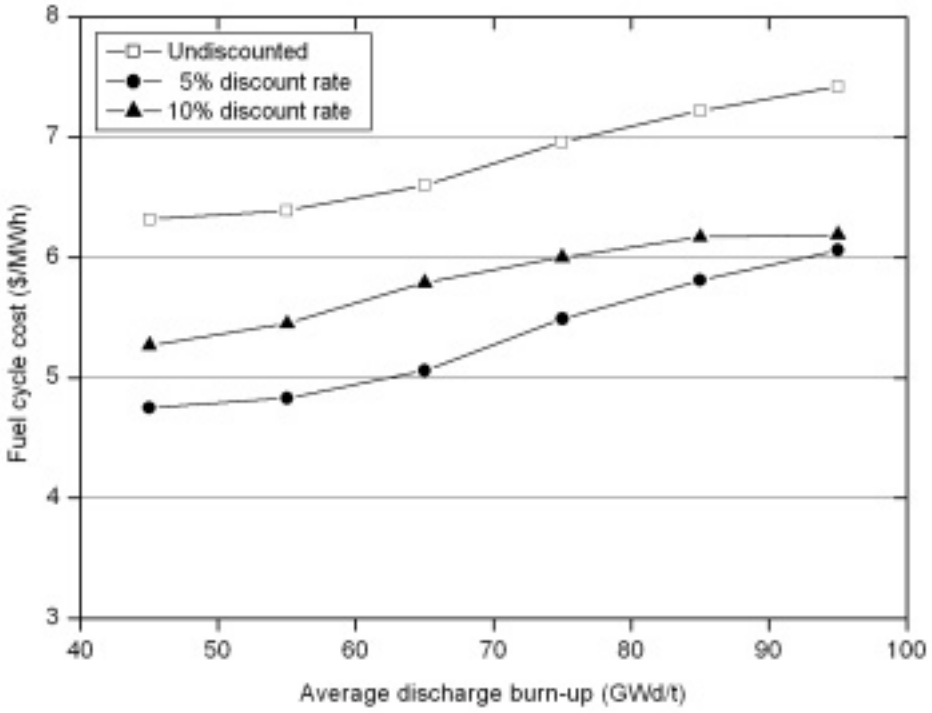


Table 25. Fuel cycle levelised cost ($/MWh) versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-dependent unit costs (evaluated by NRG using DANESS)

Burn-up (GWd/t)	45	55	65	75	85	95
Undiscounted	6.32	6.39	6.60	6.96	7.22	7.42
5% DR	4.75	4.83	5.06	5.49	5.81	6.06
10% DR	5.27	5.45	5.79	6.00	6.17	6.19

Figure 32. DANESS fuel cycle levelised cost versus average discharge burn-up and discount rate – pessimistic initial enrichment/burn-up relation and burn-up-dependent unit costs



7.4.3 VVER-440 economics sample case

An economics sample case analysis was carried out by KFKI in Hungary for a VVER-440 to complement the PWR sample cases discussed in the preceding section. Table 26 summarises the characteristics of seven fuel cycles considered in the sample case. These use the same initial enrichments as the fuel cycles presented earlier in Table 10, though there is one additional case and in some cases small adjustments have been made to the number of feed assemblies and the average discharge burn-ups. The fuel cycles can be classified into two types: in the first, the large reactivity reserve of high burn-up cycles is compensated by increased boron concentration, while in the second one gadolinia burnable poison rods are used to counteract the reactivity reserve, which is also helpful in assuring sub-criticality of the fresh fuel during the transport and storage. Instead of a fixed batch fraction, a fixed cycle length was assumed which is advantageous for maximising the discharge burn-up and for minimising the extra cost of burnable poisons, but disadvantageous with respect to the availability factor, which is constant.

In Hungary the normal approach used for power plant economics is to use undiscounted cost calculations. Table 27 defines the two component cost scenarios that were used in the analysis. Although these are designated “optimistic” and “pessimistic”, as for the PWR sample cases, these terms are used in a different sense. The optimistic scenario is designed to favour very high burn-ups; low uranium and enrichment prices are more favourable for the high initial enrichments needed at high burn-ups, as is the high fabrication cost. A fabrication cost penalty of $700 per gadolinia pin was assumed in the analysis. The economics calculations assume that the reactor is operating in equilibrium and does not account for the different costs incurred in early non-equilibrium cycles or the last cycle before reactor closure.

Table 26. VVER-440 cycle characteristics

No. of feed assemblies	Average assembly discharge burn-up (MWd/t)	Average initial ^{235}U enrichment (w/o)	No. of fuel rods per assembly containing gadolinia burnable poison
102	35.9	3.6	0
77	48.3	4.4	0
48	77.5	6.5	0
37	100.5	8.5	0
78	47.7	4.4	3
49	75.9	6.5	12
40	92.9	8.5	24

Table 27. VVER-440 sample economic scenarios

	Optimistic scenario	Pessimistic scenario
Uranium ore price ($/kg)	25	40
Enrichment cost ($/kgSWU)	80	95
Fabrication cost/assembly ($)	50 000	40 000
Back-end cost/assembly ($)	60 000	60 000

Regarding fresh fuel transport and storage, it was assumed that the use of burnable poison addresses criticality concerns caused by the increased enrichment. By invoking burn-up credit, sub-criticality during spent fuel storage and transport can also be assured. A large margin in the biological shielding requirement is assumed, such that no additional shielding costs are incurred at high burn-ups.

The enrichment cost is a linear function of the separative work units (SWUs). The specific SWU requirement (Figure 33) depends only on the enrichment/discharge burn-up relationship, therefore these values are the same for both scenarios, but differ for the two fuel cycle types (with and without burnable poison). The comparison of the total fuel costs for the two economics scenarios and the two fuel cycle types is shown in Figure 34. Without burnable poison there is a monotonic decrease with burn-up; with burnable poison a minimum is observed at 80 GWd/t due to the residual absorption penalty.

In Figure 35, the incremental costs due to the use of burnable poisons is evident. It can be seen that at the highest discharge burn-up about the half of the burnable poison penalty originates from the residual reactivity decrease and the remaining part from the extra gadolinia pin fabrication cost.

Regarding the safety-related reactor physics parameters, the main conclusion is that high burn-ups can be reached only by a considerable increase of the initial enrichment, which leads to a lower moderator temperature feedback, boron efficiency and consequently, at fixed boron concentration, to a lower shut margin. The use of burnable poison is effective in resolving this problem, but it penalises the fuel cycle cost. If burnable poison is not used, a monotonic decrease of the fuel cycle cost with burn-up is obtained up to 100 GWd/t, otherwise a minimum is observed at 80 GWd/t for VVER-440 cores. It should be noted that the VVER H/U ratio is optimised for the lower burn-up values. By increasing the H/U ratio, less negative moderator temperature coefficients could be reached even at high burn-up and the economic penalty originating from the burnable poison application could be significantly reduced, but this is outside the scope of this report.

Figure 33. Specific SWU requirement for VVER-440 for un-poisoned and poisoned fuel cycles

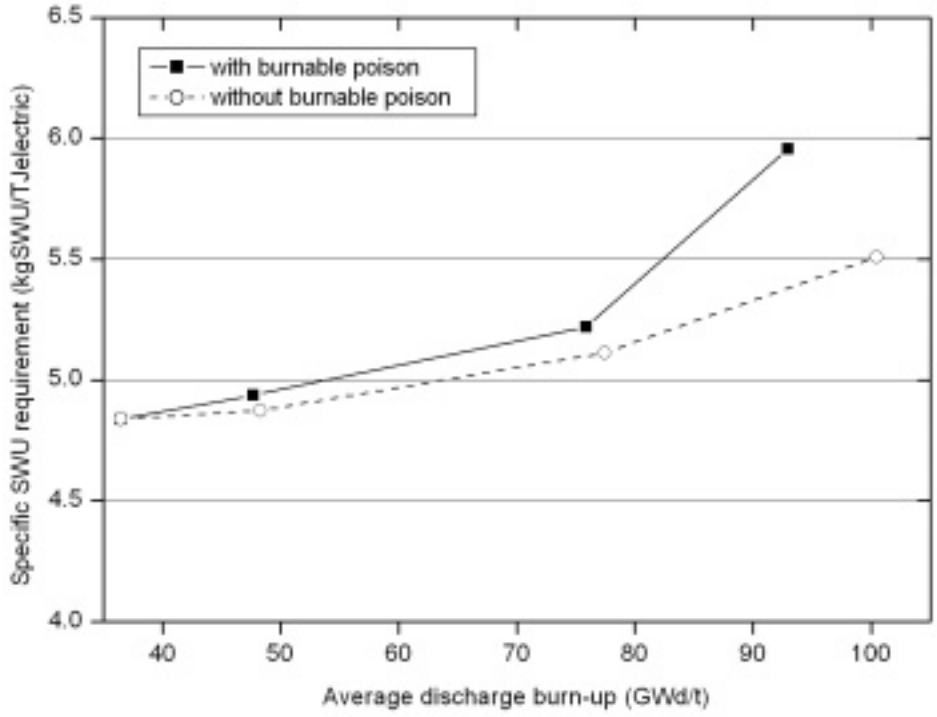


Figure 34. Comparison of the total fuel costs for the two economic scenarios for poisoned and un-poisoned fuel cycles

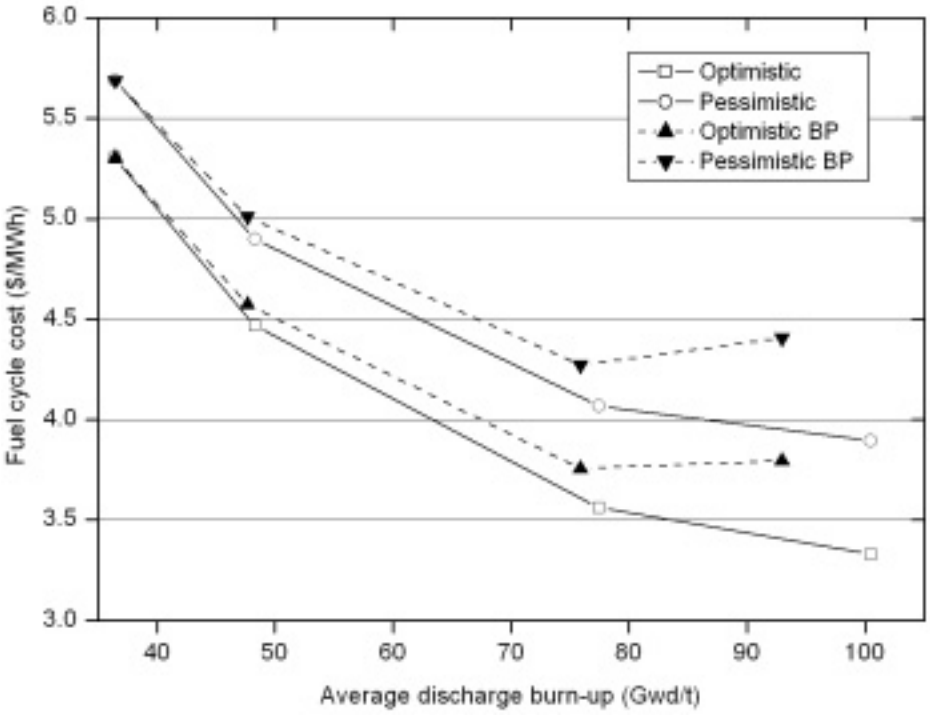
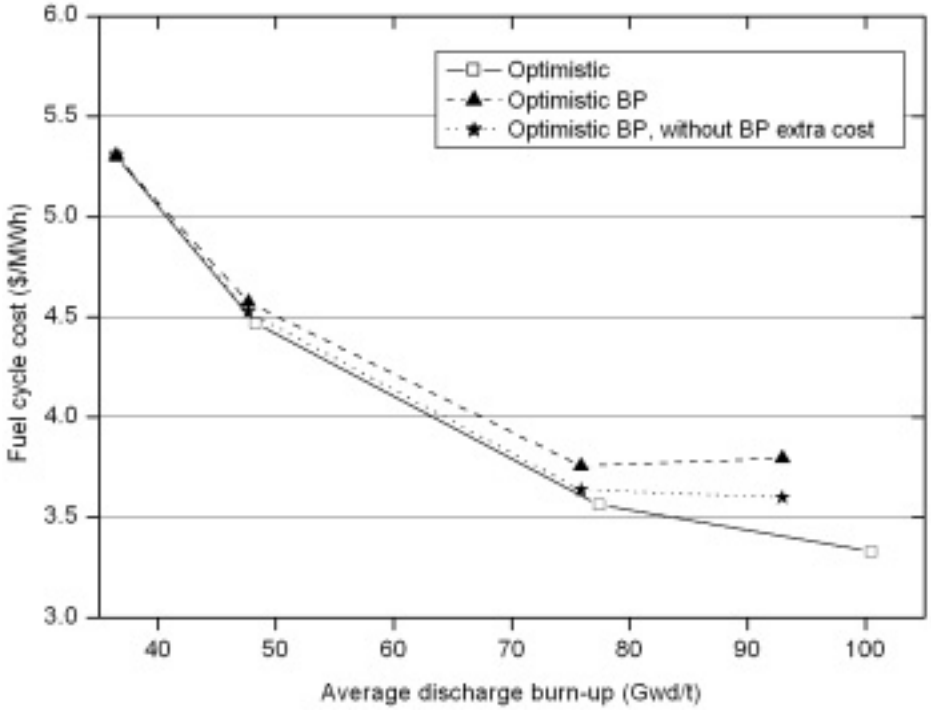


Figure 35. Comparison of the total fuel cycle costs with and without the extra gadolinia cost (optimistic scenario)



Chapter 8

RESEARCH AND DEVELOPMENT NEEDS

This Chapter reviews the research and development needs for very high burn-up LWR fuels. Many of these have been noted in earlier chapters and they are brought together here for clarity.

8.1 High enrichment experiments

In order to achieve very high burn-ups, initial enrichments in excess of 5 w/o will be required which may require significant changes to fuel fabrication facilities, transportation casks and fuel storage at nuclear power plants [24,25]. Modifications of the fuel fabrication facilities will be needed to address criticality concerns associated with these higher enrichments. Fuel shipping containers will also require re-licensing and possibly redesign. Modifications of the shipping containers may be needed to change the separation between fuel assemblies, incorporate additional neutron absorbers and limit the amount of water that could be added during a shipping accident.

Fresh fuel and spent fuel storage racks or dry storage containers may need to be re-evaluated and modified as well. Experiments may be required to ensure that current spent fuel racks can accommodate spent fuel with initial higher enrichments (> 5 w/o) that have achieved high enough burn-up, based on burn-up credit analysis [25]. Since higher enrichment fuel would normally be discharged from the core after a substantial burn-up, it would be no more reactive than current fuel with < 5 w/o enrichment; however, a portion of these racks must still allow for the storage of fresh fuel or lower burn-up fuel during outages and unplanned full core offloads.

Since criticality analysis will be revised for all facilities, new experimental data supporting that analysis will be important. New criticality experiments and further research and development may be required to have benchmark information to support licensing of higher enrichments. One such experimental research effort is presently underway at the Sandia National Laboratories (SNL), supported by the US DOE Nuclear Energy Research Initiative (NERI) programme [11]. This NERI project has performed reactor physics calculations to address fuel processing, handling, fabrication, storage, in-core and transportation issues. The SNL Burn-up Credit Critical Experiment (BUCCX) facility [13] will be used to perform criticality measurements with 7 w/o uranium Pathfinder fuel using several fuel rod configurations, and will provide criticality benchmark experimental data for 7 w/o conditions by the end of 2006. New burn-up credit experiments using higher enrichments may be needed using the same BUCCX and types of experiments performed in the 1999 NERI project [4]. The BUCCX can be used for criticality experiments for even higher enrichment fuel specimens when they become available.

Further experiments using the CEA facilities (e.g. MINERVE and OSMOSE programmes at Cadarache and the Valduc IPSN “Appareillage B” facility) may be needed to extend the range of burn-up credit and reactivity effect measurements for higher enrichment fuel and very high burn-up conditions [60]. Experiments will also need to be performed for burn-up credit with actinides and fission products combined, i.e. “full” credit, and not just “actinide only” credit for very high burn-ups.

"Full" burn-up credit experiments, including radiochemical assays, would be useful for spent fuel safety and licensing issues. In addition, reactivity worth experiments would be valuable for individual fission products, rather than just integral fission product data.

8.2 Fuel design and fuel performance testing

Very high burn-up fuel may rather behave differently than fuel discharged at the existing burn-up limits and will require advanced fuel cladding that can not only withstand higher neutron irradiation levels, but also endure in-reactor operational chemical, thermal and mechanical conditions with the same or an improved level of fuel reliability and performance [61]. Fuel fabrication and performance experiments will be needed to address specific fuel pellet, cladding, gas pressure and thermal/mechanical behaviour that are impacted by higher burn-up levels beyond 62 GWd/t High burn-up fuel testing programmes will be needed along the lines of the current joint NRC/EPRI/DOE programmes [62,63] and other international projects, such as the Halden OECD high burn-up fuel irradiation tests [64].

Extended burn-up data will need to be obtained for pellet and integral fuel/gap/clad performance. Fuel pellet irradiation damage causes rim or high burn-up structure (HBS) formation, which leads to significant changes in the porosity, thermal and mechanical properties of the fuel. Initial pellet grain structure, hydrostatic stresses, point defect locations play an important role in burn-up- and temperature-dependent properties that are important parameters for fuel performance. Further research and development will be needed to assess the impact of increased fission product gas production in very high burn-up fuel. Fission product gas evolution and gaseous swelling inside the pellet has a significant impact on pellet-clad interactions (PCI). Fuel failures usually arise only through additional strains produced by fuel gaseous swelling, and the exact stress distribution inside the pellet has an important effect on fission gas release and swelling [45]. The relative importance of intra-granular swelling on PCI, stress conditions, HBS characteristics and fission gas evolution may be higher for extended/high burn-up fuels and specific fuel performance tests using highly-burned fuel specimens, rods and lead test assemblies (LTAs) should be performed.

Advances in fuel cladding technology will be crucial for obtaining very high burn-up levels. Current Zircaloy cladding products, i.e. ZIRLO, M5 and E110, may be able to reach 70-80 GWd/t burn-up with low corrosion levels and sufficient retention of their required performance properties; however, these cladding materials may not have sufficient robustness to achieve very high burn-ups (~100 GWd/t) and still endure normal operating conditions with "zero defect" low corrosion fuel performance and be able to survive accident scenarios. Fuel qualification, irradiation testing of cladding specimens and LTAs will be needed to demonstrate these advanced products, and any future cladding candidate materials. Integral fuel assembly problems such as cladding and spacer corrosion resistance, fretting wear, debris intrusion and rod bowing will have to be addressed by an integrated R&D approach that seeks to maximise fuel reliability and extend discharge burn-up levels without reducing fuel performance limits, operational and safety margins [65].

It may be possible to use novel fuel designs to reduce fuel swelling, PCI, peak pellet temperatures, increased internal gas pressure, etc., by using doped uranium oxide fuel (e.g. Zr, Gd, Th, etc. [66-68]) or using larger gas plenum volumes or annular fuel pellet designs [69] or even non-Zircaloy cladding materials such as SiC/SiC ceramic composites [70]; however, a comprehensive fuel specimen, fuel rod and LTA testing programme would be needed to verify the performance of such innovative fuel/ assembly designs. Separate effects testing, LOCA experiments, thermal-hydraulic performance and critical heat flux (DNBR) testing, etc., would also be needed to confirm that adequate performance characteristics and safety margins have not been eroded.

The adequacy of fuel performance codes for extended/high-burn-ups will depend primarily on the availability of property data for ZIRLO, M5, E110, and other advanced cladding materials being generated as part of the ANL EPRI/NRC/DOE programme [62,63], Halden reactor experiments [64], etc., and validation of these fuel performance codes against that measured data. Assessment of fundamental empirical models (e.g. Cathcart-Pawel) to predict the behaviour of very high burn-up fuel will be required. Despite considerable improvements in pellet mechanical modelling over the last decade, efforts are still needed for code improvements, including multi-dimensional mechanical models. For example, a better characterisation of the local stress (stress tensor against hydrostatic pressure) may be needed for a more comprehensive modelling of the different ways stresses effect pellet/clad and fission gas behaviour [45]. Code-to-code benchmarks and evaluations against measured data will be required as higher burn-up performance data becomes available. Co-ordinated efforts using a cadre of international fuel performance codes will serve to enhance the understanding of the impact of very high burn-ups on fuel properties, performance characteristics and behaviour during normal and transient conditions. International working groups can select appropriate tests to be used, experimental information and code assessment needs together efficiently. The recent FUMEX II co-ordinated research programme [71] is a good example of how IAEA member countries can contribute substantially to fuel performance modelling development.

8.3 Spent fuel and recycling

Higher burn-ups result in increased cladding oxidation, hydriding, hoop stresses and higher fuel rod internal gas gap pressure caused by fission product gas releases from the fuel. These phenomena must be evaluated for their impact on fuel integrity during storage, transportation, retrieval, waste package placement and disposal. Mechanical properties must be evaluated for very high burn-up conditions, i.e. ductility under impact loading conditions, fracture toughness and creep. Additional experimental data will be needed to confirm the new licensing basis for dry storage conditions. The specific impact of dry storage on cladding mechanical properties will need to be assessed for all phases of spent fuel transportation handling and disposal operations [72]. Currently, the US NRC/DOE/EPRI research efforts being conducted at Argonne National Laboratory (ANL) are resolving the technical issues associated with the handling and transportation of high burn-up fuels for the 62 GWd/t peak rod limit [73].

Although a large database exists for fuel with burn-ups less than 45 GWd/t, and the database is growing for higher burn-ups, the objectives for the data collection have been driven by in-reactor fuel performance, and not necessarily by spent fuel dry storage, transportation, or disposal considerations. For example, much data exists for in-reactor irradiation creep exists, but much less information is available for post-irradiation thermal creep. Whereas irradiation conditions determine fuel performance during transient conditions, the thermal creep behaviour dependence on burn-up dominates spent fuel handling issues. Various spent fuel canister-based designs have been licensed for dry storage of spent fuel with peak rods at 62 GWd/t; however, transportation of fuel in excess of 45 GWd/t is handled on a case-by-case basis by the US NRC.

Further research and development should be performed specifically focused on burn-up-dependent properties related to dry storage effects, especially for very high burn-up conditions. Cladding temperature limits used for dry cask storage, transportation and particularly during a vacuum drying procedure should be evaluated since the cladding embrittlement characteristics change with higher burn-up. The impact of hydride formation in the fuel cladding, especially the changes in chemical and mechanical properties during dry storage conditions, needs to be evaluated for very high burn-up fuel. ANL has found that cladding hydrides are re-oriented from the circumferential to the radial direction [74]. The presence of radial hydrides can significantly reduce the ductility of cladding under hoop-stress loading conditions. High burn-up cladding, which had experienced some radial hydride

formation and thermal creep, would be most vulnerable to brittle failure at the end of dry-case storage when the cladding temperature would be at its minimum, and could experience brittle failure and fuel dispersal.

The proposed experiments at the ANL supported by the US NRC will address cladding integrity during spent fuel handling [74] for the current licensing limit; however, further experiments would be needed for very high burn-up fuel. Recently DOE, independently of the US NRC, funded the ANL to study the effects of radial hydrides on high burn-up cladding performance because of concerns regarding cladding response to fuel unloading from dry casks and transportation to permanent repositories [75]. As very high burn-up fuel specimens become available, further research and development tests should be performed to resolve spent fuel handling issues.

8.4 Safety and licensing testing

Using very high-burn-up fuels in existing commercial LWRs will require further assessment and licensing confirmatory research by each country's safety authority, (e.g. US NRC). Currently there is potential interest in using fuel with burn-ups greater than the current US NRC limit of 62 GWd/t for the "average for the peak rod". The US NRC, US DOE, EPRI, JAERI, the French research and development organisations, fuel vendors and nuclear power industry have had international programmes over the past several years to address and resolve licensing issues associated with the use of high burn-up fuel [1,76-81]. The US NRC has used the Phenomenon Identification and Ranking Tables (PIRT) process to determine what research is needed for high burn-up fuel [82]. For example, US NRC activities have focused on confirming that the current limit is adequately supported. However, the potential of having "extended burn-up" (up to 75 GWd/t) and even higher burn-ups may lead to issues that will that need to be addressed for several licensing-related topics. Issues identified in the original NRC 1998 plan [76], and the 21 August 2003 status, is shown in Table 28 [77].

Table 28. Issues related to high burn-up fuel identified in the 1998 NRC Program Plan [77]

Issue	Status
Cladding integrity and fuel design limit	Resolved
Control rod insertion problem	Resolved
Criteria and analysis for reactivity accidents	Active
Criteria and analysis for loss-of-coolant accidents	Active
Criteria and analysis for BWR power oscillations (ATWS)	Active
Fuel rod & neutronic computer codes for analysis	Resolved
Source term and core melt progression	Resolved
Transportation and dry storage	Active
High enrichments (>5 w/o)	Deferred

The first two issues have been addressed for the existing burn-up limit; however, the required database would have to be extended for very high burn-up (> 62 GWd/t) conditions. The current database and further information gained through the PIRT process [82] can be used to prioritise what experiments are needed for very high burn-up fuel licensing, that would lead to an expanded irradiation programme up through LTAs in operating commercial LWRs, as well as testing under transient and accident conditions.

Since the achievement of improved fuel performance and reliability, as well as extended burn-ups, depends critically on the performance of the cladding, there has been a shift in the research focus at NRC and in industry from Zircaloy cladding alloys to work on niobium-bearing alloys such as ZIRLO,

M5 and E110. Research activities related to the performance of various cladding materials are under way at Argonne National Laboratory (ANL) under the auspices of the US NRC, DOE, EPRI and industry [74,75,79,80]. Confirmatory assessments for the performance of irradiated fuel with Zircaloy cladding during loss of coolant accidents (LOCA) are scheduled for completion in December 2004, based on the extended burn-up limits [79,80]. Similar assessments for fuels with ZIRLO, M5 and E110 cladding which are expected to be used for extended/high burn-up are contingent on the availability of irradiated samples to evaluate the effects of irradiation, corrosion and hydrogen uptake. Once samples are available, it is estimated that two to four years will be required to complete the assessments of ZIRLO and M5 cladding types [77]. To extend the database to very high burn-up fuels, similar experiments will have to be done at ANL and elsewhere using cladding from LTAs with much higher burn-up irradiation levels. A long-term co-ordinated international effort will be required to obtain irradiated cladding and fuel samples from very high burn-up LTAs, involving characterised fuel assemblies with well-documented fuel performance and irradiation histories. The US NRC is currently evaluating the ANL results for reconsidering the technical basis for the embrittlement criteria for advanced zirconium alloys in high burn-up applications [1,78]. The NRC has concluded that the current 17% oxidation limit does not apply to all alloys. Additional testing may be required to determine for appropriate limit for specific alloys. However, licensing criteria should remain valid for modest burn-up extensions, because corrosion, not fluence, is the major variable and has been well studied over a wide range.

With respect to reactivity insertion accidents (RIA) such as the rod ejection accident in PWRs, and the rod drop accident in BWRs, the Electric Power Research Institute (EPRI) has submitted proposed revisions to US NRC Regulatory Guide 1.77 which would be applied to all LWR fuel for burn-ups up to 75 GWd/t [83]. Many transient tests have been performed world-wide to support development of criteria to avoid clad failure or fuel dispersal during RIAs in LWRs. These tests and a scaling approach to translate their results to make them applicable to LWR conditions are described in Ref. [79]. Thresholds for cladding failure are derived for PWR and BWR with Zircaloy-2, Zircaloy-4, ZIRLO and M5 cladding, and uranium or mixed-oxide fuel. Based on an assessment of typical control rod worths in PWRs and BWRs, and the rod worth required to reach the enthalpy limits for fuel failure, it was concluded that “current operating reactors are not likely to experience cladding failure during the worst postulated RIAs” [1,79]. However, further research and development activities using long-term international co-operative research programmes will be needed to confirm that very high burn-up fuels with existing or advanced cladding and fuel designs can perform adequately during RIA events.

Work related to possible BWR power oscillations during anticipated transients without scram (ATWS) for extended/very high burn-up fuel would be performed by industry. Although there has been no industry activity related to performing BWR ATWS tests recently, future tests could be performed at the Idaho National Laboratory Transient Reactor Test Facility (TREAT). TREAT is a graphite moderated, pulse-type test reactor that has been used for transient testing of fuels and materials. TREAT could be equipped with an existing water loop, which is well suited for testing LWR fuels, and could produce rapid multiple power oscillations tailored for typical BWR ATWS scenarios. Fuel performance and/or burst behaviour could be monitored during the power peaks with its hodoscope (i.e. on-line neutron radiography) visualisation capability and subsequent post-irradiation evaluations could be used to provide information.

A panel of experts on source term was assembled by the US NRC to evaluate the applicability and adequacy of the source term in NUREG-1465 to reactors with a maximum assembly burn-up of 75 GW/t. The panel concluded that the revised source term is generally applicable for burn-ups greater than the 62 GWd/t limit up to 75 GWd/t [84]; however, some evaluation may be required for very high burn-up fuel conditions.

Chapter 9

SUMMARY AND RECOMMENDATIONS

Historically, LWR average discharge burn-ups have steadily increased with time, the accumulation of experience and with technological improvements in fuel assembly design and materials. One of the main driving factors has been to decrease the fuel cycle cost. Another has been the increased operational flexibility that high burn-ups allow. Average discharge burn-ups are currently in the region of 50 GWd/t, and for burn-ups up to and at least a little beyond this figure, there is a clear economic incentive to continue the past trend.

For very high average discharge burn-ups in the range 60-100 GWd/t considered in this expert group study, the fuel cycle cost assessment has shown a less clear-cut economic incentive. The case for continued increase in burn-ups is only clear with an undiscounted economic model, and then only provided that back-end unit costs do not rise too steeply with burn-up. Discounted economic models show a benefit from increased burn-ups only with an optimistic relation between initial enrichment and average discharge burn-up and burn-up-independent back-end unit costs. Since there is no single economic model that applies to all utilities, depending on the local circumstances, some countries or utilities may see a benefit from very high burn-ups while others may not.

Attaining very high burn-ups will necessitate technological developments in almost every aspect of the fuel cycle. Most of these are considered achievable if there is sufficient incentive for the industry. In such a conservative technological area, future progress towards very high burn-ups can be expected to be made in incremental steps, just as has happened historically. However, this report has identified several technological barriers to very high burn-ups. The most significant is the 5.0 w/o criticality limit that currently applies in fuel fabrication plants. Relaxing this limit is not just a technological issue, it will also require significant investment decision by fuel fabricators. Even without necessarily wanting to go to as high a burn-up as 60 GWd/t, BWR utilities are already calling [7] for the 5.0 w/o limit to be relaxed as it is already proving to be a problematic constraint even at current burn-ups. The successful relaxation of this limit to, say, 6.0 or 7.0 w/o may determine the highest practical average discharge burn-ups that will eventually be attainable.

Other technological areas where further development will be required for very high burn-ups include fuel assembly design, fuel assembly materials, in-core reactor physics behaviour and fuel thermo-mechanical behaviour. There are also implications for the back-end of the fuel cycle. Where a once-through fuel cycle is chosen, there may be implications from the higher decay heat output and neutron output of irradiated fuel assemblies on transport and/or interim storage. For a reprocessing cycle the elevated decay heat and neutron outputs are likely to have significant technological ramifications. Several recommendations regarding future technological direction follow from this expert group study:

- There is now an urgent need for a systematic review of the 5.0 w/o criticality limit in fuel fabrication plants. Even at current burn-ups, BWR utilities are starting to see this constraint as a problem and a relaxation would help to improve operational flexibility. A key objective of work in this area would be to identify the absolute enrichment limit which fuel fabrication

plants could be licensed to and what changes would be needed to achieve it. This would not only identify the technological requirements in more detail, but may also point to a practical limit for LWR enrichment and therefore average discharge burn-up. There may be scope for revising the degree of pessimism assumed in fuel fabrication plant criticality assessments. Within the OECD/NEA, the area of criticality assessments falls within the remit of the Nuclear Science Committee and of the Committee for the Safety of Nuclear Installations, and there may be strong grounds for a collaborative activity to address this question. The fuel fabricators may be limited in what they could achieve in this area by themselves, and a collaborative activity with the OECD/NEA could make a valuable contribution.

- The areas of core design, fuel assembly design, fuel assembly materials and back-end fuel treatment will all require significant technological development if very high burn-ups are to be realised in practice. There may be a useful role for the Nuclear Science Committee to organise an international workshop or information exchange meeting on very high burn-ups. The objective would be to encourage fuel vendors to submit their own assessments of the technological requirements of very high burn-ups, which may help to stimulate awareness and developments in the field. The same forum could also include the technological developments needed at the back-end of the fuel cycle for very high burn-ups.

- Current nuclear data libraries and nuclear lattice and whole core code systems cannot presently be considered validated for burn-ups in the range 60-100 GWd/t considered here. Extending the validation of the nuclear design codes to very high burn-ups can only be done by the utilities with lead test assembly irradiations followed by larger-scale irradiations to very high burn-ups. The fuel vendors will, however, be reliant on accurate nuclear data libraries, and it may be beneficial for the Nuclear Science Committee to consider asking its nuclear data evaluators to review the adequacy of LWR nuclear data for very high burn-ups.

- The adequacy of depletion/decay chain modelling and of few-group parameterisations in nuclear design codes needs to be verified for very high burn-ups. This is a topic that fits within the remit of the OECD/NEA Nuclear Science Committee's Working Party on Reactor Systems (WPRS).

- The adequacy of neutron transport codes for calculating reactor pressure vessel neutron fluence needs to be demonstrated at very high burn-ups. This is also a topic that falls within the WPRS remit.

- New criticality benchmark experiments will be needed to validate criticality assessment methods for fuels with high initial enrichments.

- Research and development is needed to clarify the various factors which play a role in the formation of the high burn-up structure (HBS) and the threshold at which HBS occurs.

- Experimental and theoretical work is needed to clarify fission gas swelling behaviour of fuel at very high burn-ups.

- Research and development is needed to improve methods for estimating volatile fission product release from failed fuel.

- Research and development is needed to qualify the reactivity insertion accident (RIA) behaviour of high burn-up fuel and advanced cladding alloys.

- Research and development may be needed to obtain a better understanding of advanced cladding behaviour in loss of coolant accidents (LOCA) at high burn-ups.

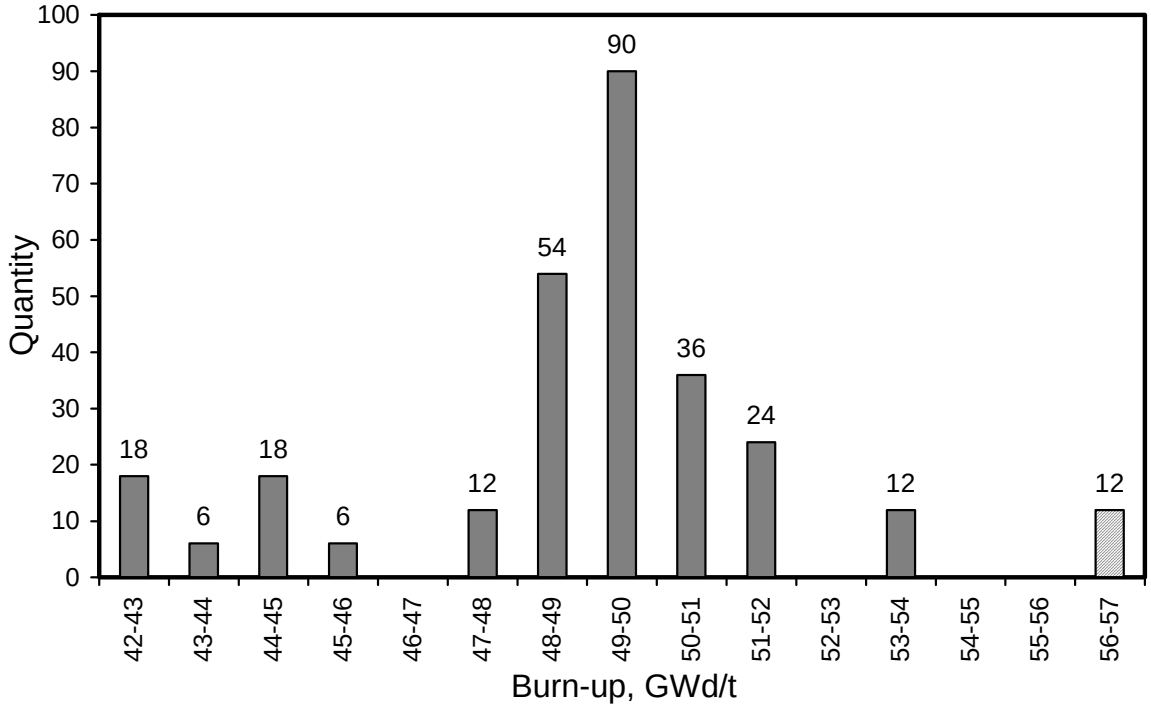
- Further research and development is needed to clarify burn-up dependent properties related to dry storage effects at high burn-ups and to better establish the properties of high burn-up spent fuel.

Appendix A

FUEL BURN-UP EVOLUTION IN THE KOLA-3 AND KOLA-4 NPPS

The KOLA-3 nuclear power plant in the Russian Federation is the lead plant for high-enriched fuel assemblies for VVER-440 reactors. In 1986 fuel assemblies enriched to 4.4 w/o were loaded in KOLA-3 for the first time. From 1991, 4.4 w/o assemblies were being loaded routinely for five annual irradiation cycles, and by 2003 cumulative experience had been gained with 276 fuel assemblies of 4.4 w/o having been irradiated over the five-year life cycle. At the present time, extensive experience of operating fuel assemblies for four and five years up to average assembly burn-ups of approximately 50 GWd/t has been accumulated. Building on this experience, twelve fuel assemblies of 4.4 w/o initial enrichment were loaded for a sixth irradiation cycle in 2001. At discharge, these assemblies had accumulated an exposure of 1 871.4 full power effective days, with average assembly burn-ups of ≈ 57 GWd/t. Seventy-two additional fuel assemblies intended for five-cycle irradiation were loaded in 2002 as part of the eighteenth fuel loading. Figure A.1 shows the distribution of discharge burn-ups attained by the 276 lead assemblies over five operating cycles. The twelve assemblies irradiated for a sixth cycle appear twice, having contributed once to the database after five years of irradiation and again after six years.

Figure A.1. Number of assemblies with 4.4% enrichment discharged out of the KOLA-3 VVER-440 reactors after five to six years of operation versus burn-up (shaded column indicates assemblies irradiated for six years)



These are the twelve assemblies represented by the hatched bar at 56-57 GWd/t. There are currently thirty-six assemblies of 4.4 w/o enrichment being irradiated for a sixth operating cycle. These use the second-generation assembly design described in Chapter 2.

Other developments with VVER-440 fuel designs include the first use in 1998 of urania gadolinia (UO_2/Gd_2O_3) fuel in the KOLA-4 plant in the working (i.e. non-moveable) fuel assemblies (WA) fuel assemblies intended for five-year irradiation cycles. At the time of this writing (2004, sixteenth fuel loading), 168 WA with urania gadolinia fuel are being irradiated in the core of KOLA-4. Twelve of these WAs are in their fourth year of irradiation. After three years of operation (2002) the maximum burn-up in these assemblies was 38.66 GWd/tU. All of the assemblies are still leak tight. During the previous (fifteenth) fuel loading there were 102 WA with urania gadolinia, of which twelve WAs were in operation for the third year, twenty-four WA were in operation for the second year and sixty-six WA were in operation for the first year.

Appendix B

RELATIONSHIPS BETWEEN THE DISCHARGE BURN-UP, BATCH FRACTION, REACTOR POWER, CAPACITY FACTOR, REACTIVITY RESERVE FOR COMPENSATING THE BURN-UP IN CASE OF THE EQUILIBRIUM CYCLE

This appendix presents some basic equations for a batch-loaded core based on the linear reactivity model and derives several different relations for the thermal energy output.

Definitions

B_D Average discharge burn-up

M Mass of the heavy isotopes in a fresh fuel assembly

N_f Number of fuel assemblies in the core

χ Batch fraction

P Thermal power of the reactor taking into account the capacity factor

T Cycle length, the time interval between two refuellings

ΔR_B Reserve reactivity for compensation of the core burn-up accounting for the finite batch fraction (not taking into account the burnable poison)

Δk The theoretical (extrapolated) excess reactivity of the core at zero burn-up without boron and burnable poison, but taking into account the reactivity effects of the moderator temperature, the power, the xenon poisoning at nominal power

$\partial k/\partial B$ Absolute value of the burn-up derivative of the reactivity

Relationships

The thermal energy output PT of a cycle is given by:

$$PT = B_D M N_f \chi \tag{B1}$$

The above relationship is exact, corresponding to the definition of the burn-up and energy conservation for the equilibrium cycle.

$$B_D = \frac{2}{x+1}\frac{\Delta k}{\partial k / \partial B} \tag{B2}$$

The above relationship is obtained from the criticality condition at EOC by using the crude assumptions that the average burn-up differences between the assembly batches, loaded in different cycles into the core, are the same, and furthermore that excess reactivity is a linear function of the core

average burn-up. It illustrates the dependence of discharge burn-up on the batch fraction: for a single batch core ($\chi = 1$) the discharge burn-up attainable is just half that attainable with a continuous refuelling scheme ($\chi = 0$).

From (B1) and (B2):

$$PT = \frac{2x}{x+1}\frac{\Delta k}{\partial k/\partial B}MN_f \tag{B3}$$

$$B_D = \frac{2\Delta k}{\partial k/\partial B} - \frac{PT}{MN_f} \tag{B4}$$

Defining:

$$\Delta R_B = \Delta k \frac{2x}{x+1} \tag{B5}$$

Finally, from (B3) and (B5):

$$PT = \frac{\Delta R_B}{\partial k/\partial B}MN_f \tag{B6}$$

Appendix C

ECONOMIC SAMPLE CASE SPECIFICATION

This appendix records the full specification of the fuel cycle economics sample case discussed in Section 7.4:

1. Reactor type PWR. Four-loop 1 200 MWe. Ninety-tonne HM inventory.

2. Economic lifetime 30 years fixed, load factor 0.85.

3. Table C1 shows the combinations of reload fraction, cycle length and initial enrichment assumed in the optimistic scenario, which are chosen to give assembly average discharge burn-ups of 45, 55, 65, 75, 85 and 95 GWd/t. These enrichments are based on an extrapolation of published data from Westinghouse [4].

4. Table C2 shows the combinations of reload fraction, cycle length and initial enrichment assumed in the optimistic scenario, which are chosen to give assembly average discharge burn-ups of 45, 55, 65, 75, 85 and 95 GWd/t. These are based on calculations for four-batch refuelling schemes, with a correction for the increased burnable poison loading needed at very high burn-ups, the correction being based on 3-D whole core calculations carried out using the CMS code suite [3].

5. Tails assay 0.3 w/o fixed.

6. Fuel masses specified by power, load factor, cycle length and average burn-up.

7. Currency US dollars.

8. Economics model sensitivity range: Undiscounted, 5% and 10% applied to all fuel cycle cost components.

9. Lead times of 2 years for uranium ore, 1.5 years for conversion, 1.0 year for enrichment and 0.5 year for fuel fabrication fixed for all cases.

10. Uranium ore cost 50$/kgU fixed.

11. Conversion cost 7$/kgU fixed.

12. Enrichment cost 90$/kgSWU fixed.

13. Fabrication base cost 300$/kgU independent of burn-up. Also, burn-up dependent fabrication cost as a sensitivity case, using Table C3.

14. Spent fuel transport cost 230$/kgU delayed five years after discharge, following the 1994 OECD/NEA economics study (assuming 1$ = 1 Euro). Also, burn-up dependent transport cost as a sensitivity case using Table C4.

15. Spent fuel encapsulation and disposal cost 610$/kgU deferred 40 years after discharge following the 1994 OECD/NEA economics study (assuming 1$ = 1 Euro). Also, burn-up dependent encapsulation/disposal cost as a sensitivity case, using Table C5.

Table C1. Fuel cycle parameters for optimistic scenario

Burn-up	Initial enrichment (w/o)	Cycle length (months)	Refuelling fraction
45	3.841	12.0	0.268
55	4.481	12.0	0.218
65	5.102	12.0	0.186
75	5.785	12.0	0.161
85	6.495	12.0	0.142
95	7.258	12.0	0.127

Table C2. Fuel cycle parameters for pessimistic scenario

Burn-up	Initial enrichment (w/o)	Cycle length (months)	Refuelling fraction
45	3.8	11.4	0.25
55	4.5	13.9	0.25
65	5.4	16.4	0.25
75	6.5	19.0	0.25
85	7.5	21.5	0.25
95	8.5	24.0	0.25

Table C3. Burn-up dependent fabrication cost (linear escalation)

Burn-up	$/kgU
45	300
55	330
65	360
75	390
85	420
95	450

Table C4. Burn-up dependent spent fuel transport cost (escalation proportional to burn-up)

Burn-up	$/kgU
45	230
55	280
65	330
75	380
85	430
95	480

Table C5. Burn-up dependent encapsulation/disposal cost (escalation proportional to burn-up)

Burn-up	$/kgU
45	610
55	745
65	880
75	1 015
85	1 150
95	1 290

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