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A Study on the Application of Burnup Credit Methodology on Fuels with Burnable Absorbers

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Abstract

The management of spent nuclear fuel (SNF) is a critical challenge for the safe and efficient operation of nuclear power systems. Traditional criticality safety analyses often assume fresh fuel conditions, introducing significant conservatism that limits storage and transport capacity.

Burnup credit (BUC) methodology addresses this by accounting for the reduction in fuel reactivity during irradiation. However, current licensing practices are typically restricted to fuels that do not contain burnable absorbers (BAs), such as gadolinium, because of their complex reactivity profiles. This study investigates the impact of gadolinia-based (Gd_2O_3) integral burnable absorbers (IBAs) on the reactivity of VVER-1000 fuel assemblies and evaluates their implications for BUC methodology. Using the Monte Carlo N-Particle (MCNP) code, depletion simulations were performed for fuel assemblies (FAs) with varying uranium enrichment, gadolinia content, and IBA rod count, up to a burnup of 60 GWd/MTU (Gigawatt-days per metric tonne uranium). Results show that FAs using IBAs show consistently lower reactivity ($\Delta k_{inf} < 0$) throughout burnup than reference cases without IBAs. Furthermore, increasing the number of IBA rods and or gadolinia content will only increase $|\Delta k_{inf}|$ showing that more simulations are unnecessary. The findings suggest that neglecting IBAs in licensing basis analyses provides a conservative and practical approach, reducing computational complexity while maintaining safety margin.

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Populärvetenskaplig sammanfattning

Kärnkraftverk är egentligen bara glorifierade vattenkokare. Istället för att vattnet värms i en kastrull på spisen eller över en öppen brasa, är det metallen uran som klyvs till mindre partiklar som skapar värmen. Klyvningen och värmebildningen sker genom en process som kallas fission. Fission sker när uran drar åt sig en fri neutron, blir så pass instabil att den delas i två mindre partiklar och skickar ut fler neutroner som sedan kan klyva andra uranatomer. Målet är att upprätthålla en konstant kedjereaktion av klyvningar för att inte en olycka ska ske. Precis som briterernas kinkighet kring hur man kokar te korrekt, är uran lika kinkigt med under vilka villkor det vill klyvas. Endast uran med masstal 235, antalet protoner och neutroner i urankärnan, kan tänka sig att klyvas. Maximalt 5% ^{235}U blandas in i reaktorns bränsle, där resterande uran är av den ej klyvbara typen med masstal 238 (^{238}U).

Eftersom att koka vatten är en dyr historia för kärnkraftverk, vill operatörerna behålla bränslet så länge som möjligt i reaktorn. Det kan liknas vid hur en bilförare vill att bilen ska kunna köra så långt som möjligt. Att öka mängden av den klyvbara ^{235}U , så kallad anrikning, är ett sätt att få reaktorn att köra längre, precis som en större bränsletank på bilen. Eftersom färskt kärnbränsle är något för reaktivt av design, behöver klyvningarna regleras på olika sätt. Om mer klyvbart material läggs till kan det bli problematiskt att reglera. För att möjliggöra ökad anrikning av ^{235}U blandas också metallen gadolinium in i bränslemixen. Gadolinium har förmågan att fånga in neutronerna som klyver uran, men tappar denna egenskap efteråt. De ämnen som har denna egenskap tillhör då till kategorin brännbara absorbatörer. Detta gör att det mer anrikade bränslet blir lättare att kontrollera för operatörerna i början av dess livstid. När allt gadolinium har förbrukats kommer också en del av uranet förbrukats vilket ger en fortsatt hanterbar körning.

När bränslet inte längre kan användas i reaktorn, måste det ersättas med nytt färskt bränsle. Det gamla, använda kärnbränslet måste hanteras mycket varsamt, då det är hälsoskadligt för människor att befinna sig i dess närhet. Precis som solen kan orsaka skada på huden genom UV-strålning, kan det radioaktiva, förbrukade bränslet utsända mycket mer skadlig strålning. Speciellt farligt blir det om oväntade kedjereaktioner av klyvningar startar utanför reaktorn, då stora mängder värme och radioaktiv strålning skickas ut. Det är därför viktigt att försäkra sig om att behållare som förvarar använt kärnbränsle inte går kritiskt, inte upprätthåller kedjereaktioner, under trovärdiga omständigheter. Eftersom färskt kärnbränsle är mest benäget för kriticitet, brukar det konservativa antagandet om färskt kärnbränsle vid förvaring göras vid beräkningar för god säkerhetsmarginal.

Denna säkerhetsmarginal är väldigt konservativ eftersom använt kärnbränsle inte alls är lika benäget för klyvningar. Konceptet "burnup credit" har introducerats för att ta hänsyn till den minskade benägenheten för kriticitet i bränslet och samtidigt behålla säkerheten i förvaringen. Då kan mer effektiv packning av bränslen göras, vilket slutligen kan spara tid och pengar. Problemet med burnup credit är att det kräver kunskap om vilka ämnen som finns i bränslet när det tas ut ur reaktorn. Eftersom en mängd olika partiklar kan skapas vid klyvningar, blir det en stokastisk process som måste förutsäga bränslets materialkomposition. Ett annat problem med detta koncept är att det i dagsläget inte tillåter licensiering av bränslen som använder brännbara absorbatörer, t.ex gadolinium. Detta grundar sig i att bränslen med brännbara absorbatörer möjligen kan ha en större benägenhet för kriticitet än ett likadant bränsle utan, under vissa skeden av dess förbrukning.

Denna rapport syftar till att undersöka hur bränslen med brännbara absorbatörer kan implementeras i burnup credit metodologin för licensiering, genom jämförelse av benägenheten för kriticitet för bränslen med och utan brännbara absorbatörer. För att undersöka detta kommer ett bränsleelement att modelleras och simuleras i programmet MCNP (Monte Carlo N-Particle Code). MCNP är en kod som används för att beräkna partikeltransport genom Monte Carlo metoder. Dessa typer av metoder använder statistiska sannolikhetsfördelningar för att förutsäga en partikels beteende och kräver därför många upprepade simuleringar för att ge ett säkert resultat. MCNP kommer att beräkna multiplikationsfaktorn k_{eff} (k-effektiv), vilket är ett mått på bränsleelementets förmåga att uppnå kriticitet och jämföra det med utbränningen av bränslet. Även om inget brinner, är utbränning en industristandard för hur mycket energi som frigjorts för en viss mängd bränsle.

Ett bränsleelement med 18 olika uppsättningar av ^{235}U anrikning och mängd brännbara absorbatörer av typen gadolinium simulerades med hjälp av MCNP. Dessutom simulerades referensfall utan brännbara absorbatörer för att se hur de skiljde sig åt. Resultaten visar att samtliga 18 uppsättningar har en lägre multiplikationsfaktor k_{eff} jämfört med motsvarande referensfall under hela utbränningstiden. Dessutom visar resultaten att fler simuleringar med ökad mängd brännbara absorbatörer inte kommer att behövas eftersom storleken på skillnaden i k_{eff} , jämfört med referensfallet, endast ökar med mängden.

Rekommendationen som kan ges för licensiering av ett använt kärnbränsle, som implementerar gadolinium, för förvaring i ett transport- eller lagringssystem, är att försumma användningen av den brännbara absorbatören helt och hållet. Eftersom referensfallen utan gadolinium alltid har högre benägenhet för kriticitet än de utan, är det ett tillräckligt gott konservativt antagande som fortfarande håller säkerhetsmarginalen.

Executive summary

This thesis investigates how gadolinia-based integral burnable absorbers (IBAs) can be applied to burnup credit (BUC) methodology in criticality safety analyses for spent nuclear fuel (SNF). Traditional practices assume fresh fuel conditions, introducing significant conservatism and limits storage and transport efficiency. By taking credit for the reduction in reactivity during irradiation, BUC can reduce the over-conservatism. However, current standards exclude fuels that incorporate IBAs.

MCNP simulations were conducted to model depletion of VVER-1000 fuel assemblies (FAs) with varying ^{235}U enrichment (4% and 5%), gadolinia content (3%, 5%, 8%) and IBA rod counts (6, 12, 24). A total of 18 scenarios were analyzed up to 60 GWd/MTU burnup. Results show that FAs with IBAs consistently display lower reactivity than reference cases without IBAs, with Δk_{inf} remaining negative throughout burnup. This behaviour is primarily attributed to the residual reactivity suppression of gadolinium. Thus, for a licensing basis analysis of PWR FAs using gadolinia IBAs, neglecting the presence of absorbers was deemed bounding and conservative.

Abbreviations

TWh — Terawatt hours
GWe — Gigawatt electric
SNF — Spent nuclear fuel
BUC — Burnup credit
BA — Burnable absorber
PWR — Pressurized water reactor
BWR — Boiling water reactor
VVER — Water-water energy reactor
NPP — Nuclear power plant
FA — Fuel assembly
ppm — parts per million
MeV — Megaelectron volt
MWd/MTU — Megawatt days per metric ton uranium
 k_{eff} — Multiplication factor
 k_{inf} — Multiplication factor for infinitely large reactor
EFPD — Effective full power days
IBA — Integral burnable absorber
IFBA — Integral fuel burnable absorber
BPR — Burnable poison rod
pcm — percent mille (0.00001) MCNP — Monte Carlo N-Particle code
wt% — Weight percentage
U — Uranium
Gd — Gadolinium
ICSBEP — International Criticality Safety Benchmark Evaluation Project
LANL — Los Alamos National Laboratory

Foreword

I would like to express sincere gratitude to Westinghouse Electric Sweden for giving me the opportunity to write my master's thesis within their organization. Special thanks to my supervisor, Sergiy Chernitskiy, for teaching me MCNP and their guidance and feedback throughout this work. Another special thanks to Melker Ehrenström for introducing me to Sergiy, all the support regarding organization and censoring, and for inviting me to your home office sessions. I would also like to thank my subject reviewer at Uppsala University, Peter Andersson, for his feedback regarding the technical aspects and quality of this master's thesis.

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1 Introduction

As the world installs increasingly more intermittent energy sources, such as wind and solar, and phasing out fossil fuels, the demand for safe, stable, and low-carbon electricity generation intensifies [1]. In this context, nuclear power can become a key contributor to the energy transition. Its key advantages include low greenhouse gas emissions, high energy density, and reliable baseload power that supports grid stability in energy systems where intermittent renewable energy sources are prevalent [2].

Globally, the total generated electricity from all nuclear reactors amounted to 2667 TWh in 2024, the highest ever recorded in one year [3]. The three biggest producers were USA, China and France with around 30%, 16% and 14% respectively [4]. Installed capacity totalled 398 GWe (Gigawatt electric) distributed over 440 nuclear power plants worldwide, three more than 2023. A total of 63 reactors are currently under construction, where 29 of them are in China [3]. The international atomic energy agency high case projection for 2050 now reaches 950 GWe, 2.5 times the current installed capacity [4].

Despite the benefits and evolution of nuclear power, there are several limitations that complicate broader deployment of nuclear energy. One of the most significant challenges is the management of spent nuclear fuel (SNF). SNF remains highly radioactive for thousands of years and therefore must be handled in a safe and efficient way to ensure long-term safety [5]. Even though the SNF is not good enough as fuel any more, there is still fissile material left in it. A fissile material is a material that can undergo fission, the induced splitting of nuclei by neutrons. Consequently, there is a risk that fissile material and SNF could be arranged in a configuration that enables fission chain reactions, posing serious radiological hazards [6]. To mitigate this risk, the field of criticality safety was developed.

Criticality safety involves ensuring that no self-sustaining fission chain reaction can occur under any possible scenario. This discipline is essential across the entire nuclear cycle, including dry and wet storage, transport casks and fuel manufacturing and reprocessing facilities. To ensure safety, conservative assumptions are typically applied in criticality safety analyses. Traditionally, this means fresh fuel conditions, assuming no burnup and maximum enrichment, will be applied [7]. While this approach is appropriate for fuel manufacturing facilities, such as the Westinghouse factory in Västerås, it introduces significant conservatism when applied to SNF. In reality, the reactivity of SNF has decreased due to depletion of fissile isotopes during reactor operation. The consequence of ignoring this reduction may be overly conservative designs of storage and transport systems, resulting in reduced capacity and increased costs [8].

To address this issue, the concept of burnup credit (BUC) has been introduced. BUC accounts for the reduction in reactivity, the deviation from self-sustaining fission reactions in a reactor, that occurs as the fuel is irradiated, by considering the changing isotopic composition of the fuel [8]. However, a key limitation of current licensing practices is that burnup credit is typically restricted to intact fuel assemblies (FA) that have not used burnable absorbers (BAs) [9]. BAs are certain isotopes with the ability to absorb neutrons well and are used in reactors to control the fission process. A consequence of utilization of BAs, is a more complex burnup profile. Usually, fresh fuels without BAs will be the most reactive, but for BA fuels this will typically not be the case. This postponement of the peak reactivity on the burnup profile is what creates the complication of licensing storage and

transportation casks for these kind of fuels [10].

1.1 Goal and Purpose

The goal of this project is to explore how BAs can be incorporated into burnup credit methodologies. Specifically, the work aims to quantify the reactivity effects of depleting pressurized water reactor (PWR) FAs containing gadolinium-based integral burnable absorbers (IBAs), which are commonly used in European nuclear power plants (NPPs). By doing so, the study seeks to support the development of more realistic and efficient criticality safety analyses for spent nuclear fuel systems, ultimately contributing to more efficient storage and transport system designs.

1.1.1 Research questions

- How does reactivity change with burnup for a PWR FA using different configurations of uranium enrichment, burnable absorber content, and number of IBA rods?
- Is the reactivity higher at any point during burnup for a fuel assembly using IBAs compared to a FA without IBAs?
- What recommendations can be made for a licensing basis analysis in the context of BUC methodology?

2 Background

This section provides the theoretical and technical context necessary to understand the application of BUC methodology for fuels containing burnable absorbers. It begins with an introduction to the nuclear fuel cycle and the most common types of reactor, followed by an overview of the fission process, neutron cycle and the effects of BAs. Lastly, the principles of criticality safety analysis and BUC methodology are introduced.

2.1 Nuclear fuel cycle

The most common isotope used for nuclear fuel is ^{235}U . Natural uranium comprises three isotopes ^{234}U , ^{235}U , and ^{238}U with abundances of 0.005%, 0.72% and 99.27% respectively. Uranium is known to be found in at least 60 different minerals measuring 2.7 ppm of the earth's crust [11][12]. This abundance is comparable to the metal tin and almost 40 times greater than silver [12]. In high grade mines, the uranium concentration is usually less than one percent of the total crushed ore [11]. In 2024, Kazakhstan was the largest uranium miner (39% of world supply), followed by Canada (24%) and Namibia (12%) [13].

The nuclear fuel cycle can be divided into a front end and a back end, with power generation dividing them. Uranium mining and refining, isotope enrichment, and FA manufacturing encompass the front end, while radioactive waste management and reprocessing belong to the back end [14].

2.1.1 Front end

The mining techniques for extracting uranium vary depending on the ore to be processed. Generally, the ore is mined underground, in open pits, or by using a technique called in situ leaching. This technique involves pumping a solution into a permeable orebody, dissolving the desired minerals and pumping them back up to the surface where they are recovered. Open pits usually have ores with lower uranium content compared to underground mines. The final product of mining is a compound generally referred to as yellow cake (U_3O_8). Yellow cake is then transported to conversion facilities where uranium is converted into uranium hexafluoride (UF_6). This compound is chosen because its chemical properties create good conditions for enrichment (separation) of ^{235}U . There are several technologies for enrichment: diffusion, centrifugal, and laser techniques. After enrichment, UF_6 is transported to the fuel manufacturer where it is converted to UO_2 and then pressed to fuel pellets. The fuel pellets are stacked in zircalloy rods which are then assembled into bundles, called FAs. Lastly, the FAs are shipped to their designated nuclear power plant [13].

2.1.2 Back end

When nuclear fuel has been used, the waste needs to be taken care of. The SNF is either going to a reprocessing facility where useful isotopes are recovered to create new fuel or it can be put in storage. Whether interim or long-term storage. The waste remains radiotoxic for humans for thousands of years and therefore must be handled with care on a short term and long term basis [15]. Thus, this study belongs to back end research.

2.2 Reactor types

The Westinghouse Electric AB factory in Västerås manufactures nuclear fuel for more than 30 reactors in Europe, where France and Ukraine are the largest customers. It is the only place in the world that manufactures fuel for reactors of the type BWR, PWR, and VVER in the same factory [16]. Westinghouse also supplies fuel for Forsmark (BWRs) and Ringhals (PWRs) NPPs in Sweden [17].

2.2.1 BWR

Boiling water reactors (BWR) are the most common reactor type in Sweden and the second most common type in the world. Four out of six operable reactors in Sweden are BWR type and 60 operable reactors exist worldwide [3].

Inside a BWR reactor core, large amounts of heat is produced through fission, the splitting of nuclei by neutrons. Water coolant is pumped into the core to remove the heat. Out of the core comes a steam-water mixture at a pressure of 70 bars. Water droplets are removed through separators allowing only 'dry' steam to exit the reactor. The steam is then channelled directly to the turbines which drives the electric generator through expansion. Shortly afterwards, the steam is condensed back to liquid form in the condenser. The condenser is cooled by a cold source which is generally the sea or a river. Through a series of pre-heaters and pipes, the water is then directed back into the reactor core to continue the loop [18].

2.2.2 PWR

Pressurized water reactors (PWR) are the most common reactor type in the world. At the end of 2024, 313 of the 440 operable nuclear reactors were PWR type. Sweden uses two PWRs, both located at Ringhals NPP [3].

The PWR operates differently than a BWR, since no boiling occurs in the primary coolant loop. Instead of being pumped to the turbines, the 155 bar sub-boiling water is pumped to a steam generator, where the heat is exchanged to a secondary coolant loop creating pressurized steam. While the primary coolant is pumped back into the reactor, the steam made its way to the steam turbines to create electricity. After the turbines, the steam-water mixture is cooled in a condenser and pumped back into the steam generator. The main argument for using two coolant loops, instead of one as in BWRs, is to contain the radioactivity in the primary loop, thus increasing safety in the event of an accident [19]. However, the simpler design of BWRs, due to the utilization of a single coolant loop, generally results in higher thermal efficiency [18].

2.2.3 VVER

Water-water energy reactors (VVER) are a type of PWR designed by ROSATOM, the Russian state nuclear corporation, during the 1950s. There are few distinct differences between VVERs and PWRs, but they operate primarily in a similar fashion. The key discerning differences of VVERs and PWRs are the following:

- Hexagonal FAs.
- Horizontal steam generators.

- No bottom penetrations in the pressure vessel for detectors.
- High-capacity pressurizer providing a large reactor coolant inventory [19].

The hexagonal FAs are chosen because they allow for more efficient packing than a cylindrical core [20]. No bottom penetration in the pressure vessel and a high-capacity pressurizer are used for increased safety against various accident scenarios [21][22].

2.3 Fission process

In a nuclear reactor core there is a process named fission occurring. Due to electrostatic repulsion of protons, certain heavy nuclei split into two medium-heavy fragments, releasing two or three neutrons and energy. This process is induced by neutrons being absorbed into the now excited nucleus. If the excitation energy is large enough, the nucleus will deform and the repulsive force will overcome the nuclear attraction. Most nuclear fuels contain a mixture of two Uranium isotopes: ^{238}U ($> 95\%$) and the fissile ^{235}U ($< 5\%$). With every fission occurring, around 200 Mega-electron volts (MeV) of energy is released to the vicinity, mainly in the form of kinetic energy from the fission fragments and neutrons, but also from β - and γ radiation. For a reactor that operates at thermal power of 3000 MW for 30 days, this will roughly equal to 90 kilograms of fissioned fuel [23].

To induce fission by hitting fissile material with neutrons, an environment in which the probability of fission is high must be created. The likelihood of a nucleus reacting with a neutron is called cross section σ . This physical quantity represents the probability of a nuclear reaction, such as neutron scattering, neutron capture and fission, with a surface area with the unit barn ($1b = 10^{-24} \text{ cm}^2$). The fission cross section σ_f for different isotopes varies wildly depending on the incident neutron energy. For ^{235}U , the fission cross section increases with decreasing neutron energy with a maximum at thermal energies. Neutrons are released from fission with energies around 2 MeV:s and are referred to as fast neutrons. The fast neutrons must therefore be slowed down to become thermal neutrons. This process is called neutron moderation and is usually done with water. Due to its relatively low mass, high scattering cross section σ_s , and abundance, water serves as an effective moderator. When a neutron collides with a water molecule, it transfers part of its kinetic energy. After a sufficient number of collisions, the neutron reaches thermal energy and becomes a thermal neutron, capable of efficiently inducing fission in ^{235}U [23].

Heavy nuclei can be split in numerous ways, creating many different medium-heavy fission fragments. The probability of producing a certain fission fragment is defined as the fission yield. Usually, the mass number of the fission fragments falls in the 90–100 and 135–145 ranges. Figure 1 below shows the distribution of the fission yield as a function of the mass number of the fission fragments [23].

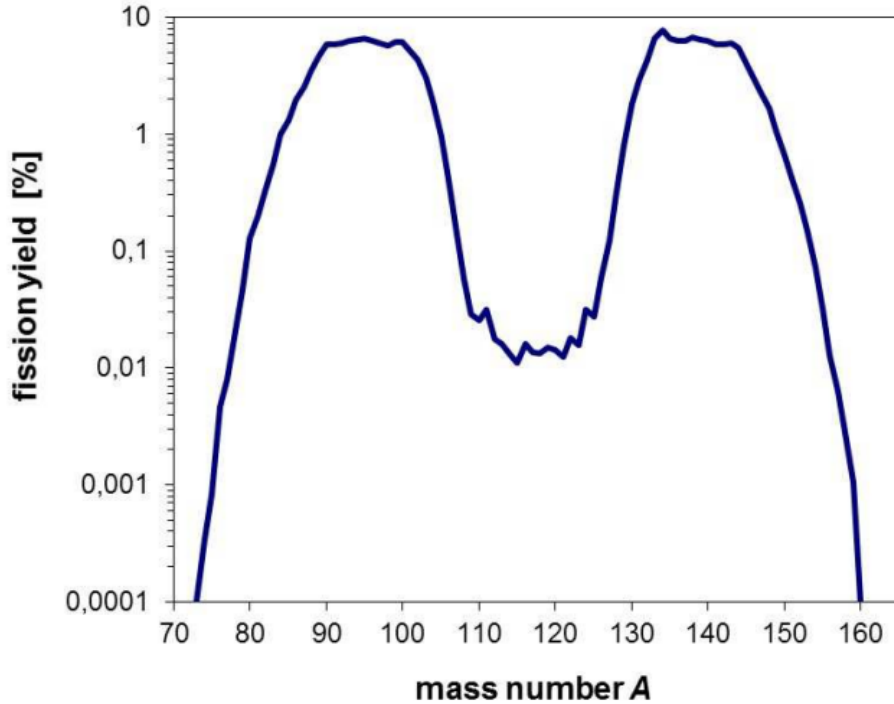


Figure 1: Fission yield distribution as a function of mass number of the fission products from ^{235}U [23]

The fission fragments are unstable, due to an excess of neutrons, and undergo β^- decay. All of the fission fragments and their decay daughter nuclei are generally called fission products. The fission products are very important for the operation of the reactor. Firstly, several fission products are very strong neutron absorbers making operation more difficult with time. Secondly, since the decay of fission products involves highly penetrative γ radiation, the dangers to health cause the handling of SNF to be problematic. Lastly, decay heat is generated, causing the need for constant cooling after reactor shutdown [24].

A continuous increase in the number of fission products with the depletion of fissile material will increase the probability of absorption in other material relative to the number of fissions. At some point, the absorption will be greater, shutting the reactor down. Therefore, finding a way to quantify and predict the state of the fission process is important [24].

2.4 Neutron Cycle and the Multiplication Factor

When handling fissile material, inside or outside the reactor, it is very important to keep the fission chain reactions to desirable levels. To illustrate the state of the reactor and eventual power fluctuations, the multiplication factor k_{eff} is introduced. It is defined as

$$k_{eff} = \frac{\text{Number of neutrons in one generation}}{\text{Number of neutrons in the previous generation}}. \quad (1)$$

During reactor operation, k_{eff} should be equal to unity between every generation of neutrons. When $k_{eff} = 1$, the fission chain reaction can sustain itself, the reactor power is constant, and the reactor is termed critical. Supercriticality is the term for when $k_{eff} > 1$ which is the condition when the neutron population increases for every generation, causing the reactor power to also increase. Similarly, during subcriticality, $k_{eff} < 1$, the neutron

population, and in effect the reactor power, is decreasing [24].

A good approach to determine k_{eff} in detail is to follow the birth of every neutron until it's eventually absorbed or leaked outside the reactor. As previously stated, neutrons are released with fast energies that need to be moderated down to thermal energies. Although the cross section for scattering is highest, some fast neutrons will induce fission in ^{238}U , introducing a few more neutrons. This addition of neutrons is named the fast fission factor ϵ and is defined as the ratio of the total number of neutrons introduced through fission to the number of neutrons arising from thermal fission [24].

In the process of reaching thermal energies, the neutrons will experience a higher probability of capture in certain actinides, elements with atomic number between 89 and 103, especially ^{238}U . These actinides have resonances in the absorption cross section for neutron energies in the 1 eV to 300 eV range. Because the energies are too low for fission, these resonances will act as absorption sinks for neutrons, decreasing the number of available neutrons. The resonance escape probability p is therefore defined as the fraction of neutrons that reach thermal energies to the number of neutrons that start moderation.

Once the neutrons are moderated down to thermal energies, around 0.0253 eV, the absorption cross section will be at its highest for various components in the reactor. Thermal capture of neutrons can occur in the moderator and other reactor components, and are therefore lost. Furthermore, there is a probability that neutrons absorbed in ^{235}U do not cause fission. Two factors can then be introduced. Firstly, the thermal utilization factor f is the ratio between neutrons absorbed in the fuel to neutrons absorbed in the reactor system. Secondly, the reproduction factor η describes the number of new neutrons for each thermal neutron absorbed in the fuel. Lastly, throughout the neutron cycle, there is always a probability of neutron leakage into the surrounding reactor components. Minimization of leakage is usually done by using reflector materials or a higher enrichment in the core extremities. However, some leakage will still occur, and therefore two non-leakage factors are introduced: The fast non-leakage probability P_{NLf} and the thermalized non-leakage probability P_{NLth} [23].

Multiplying all these factors together, the multiplication factor k_{eff} in equation 1 can now be rewritten as

$$k_{eff} = \eta \cdot \epsilon \cdot p \cdot f \cdot P_{NLf} \cdot P_{NLth}. \quad (2)$$

By omitting the last two leakage terms, k_{eff} can also be written as

$$k_{inf} = \eta \cdot \epsilon \cdot p \cdot f \quad (3)$$

where k_{inf} is the multiplication factor for an infinitely large reactor. This variant of the multiplication factor is useful for characterizing the reactivity of an FA, independent of core geometry. Equation 2 and 3 are generally also referred to as the six-factor and four-factor formulas respectively [23]. A common measure for how much a system deviates from criticality is called reactivity. Reactivity is defined as

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \quad (4)$$

and is often given in pcm (percent mille = $10^{-5} \frac{\Delta k}{k}$). It can also describe how much a certain operational action or incident affects the criticality [24].

Commercial reactors generally refuel every 12-18 months, where about a fourth of all fuel assemblies are replaced. This means that every FA is irradiated in the reactor for about four years [25]. Burnup is the term generally used for the amount of energy extracted for a given amount of fuel that has been fissioned, although nothing is burning inside a nuclear reactor. It is typically measured in GWd/MTU (Gigawatt days per metric ton of uranium). Today, typical PWR burnup reaches up to 50 GWd/MTU [26]. Another way to measure fuel usage in a reactor is Effective Full Power Days (EFPD). Converting between Burnup and EFPD is given by

$$EFPD = \frac{BURNUP \text{ (GWd/MTU)} \cdot \text{Fuel Mass (MTU)}}{\text{Reactor Power (GW)}}. \quad (5)$$

EFPD measures for how long a reactor has operated at full rated power, independent of fuel mass. For example, running the reactor at half rated power for two days will equal one EFPD. This measure of fuel usage is easier to calculate and track than burnup during reactor operation, since detailed fuel mass data is not required. Thus, this has become the standard during reactor operations [25].

Facilitating an FA to last for a long time results in a design with an initial k_{eff} well above one. The value of the reproduction factor η is therefore greater than what is needed for criticality. Factors $p \cdot f$ must therefore be reduced an equal amount. During reactor operations, the reactivity balance is controlled with:

- Soluble boron in the moderator
- Higher flow of moderator from main circulation pumps (mainly BWR)
- Insertion of control rods
- Use of BAs in the beginning of the reactor cycle [24].

All of these measures have the common factor of increased absorption of neutrons. Especially important for this study are the use of BAs. They either come in the form of rods, "burnable poison rod" (BPR), integrated directly into the fuel, "integral burnable absorber" (IBA), or as a coating around the fuel, "integral fuel burnable absorber" (IFBA). A common IBA, and the type studied in this report, is gadolinium oxide (Gd_2O_3) or "gadolinia" as it is more popularly called [10]. Isotopes ^{155}Gd and ^{157}Gd , who have large absorption cross sections, transform through neutron capture to ^{156}Gd and ^{158}Gd respectively who have small absorption cross sections. This is an important property of fuels with BAs [24].

Since fresh fuel exhibits high excess reactivity, BAs are introduced to suppress this initial reactivity early in the fuel cycle. As fissile isotopes are fissioned, the gadolinium will continuously become saturated, losing its ability to attract neutrons. This process is why it is called a burnable absorber [24]. So, as the fissile material is depleted, the burnup of BAs will counteract this reactivity decrease with less absorption in the reactor system, keeping the net reactivity constant. This feature allows for higher enrichments, ultimately leading to higher burnup and longer refuelling cycles. The k_{eff} will decrease nearly linearly with

burnup for fuel without BAs. However, fuel with BAs will see a different curve where k_{eff} usually reaches a maximum when the BA is depleted. This generally occurs a third way into the lifetime of the FA [10]. Figure 2 show the general behaviour for PWR fuels with and without BAs.

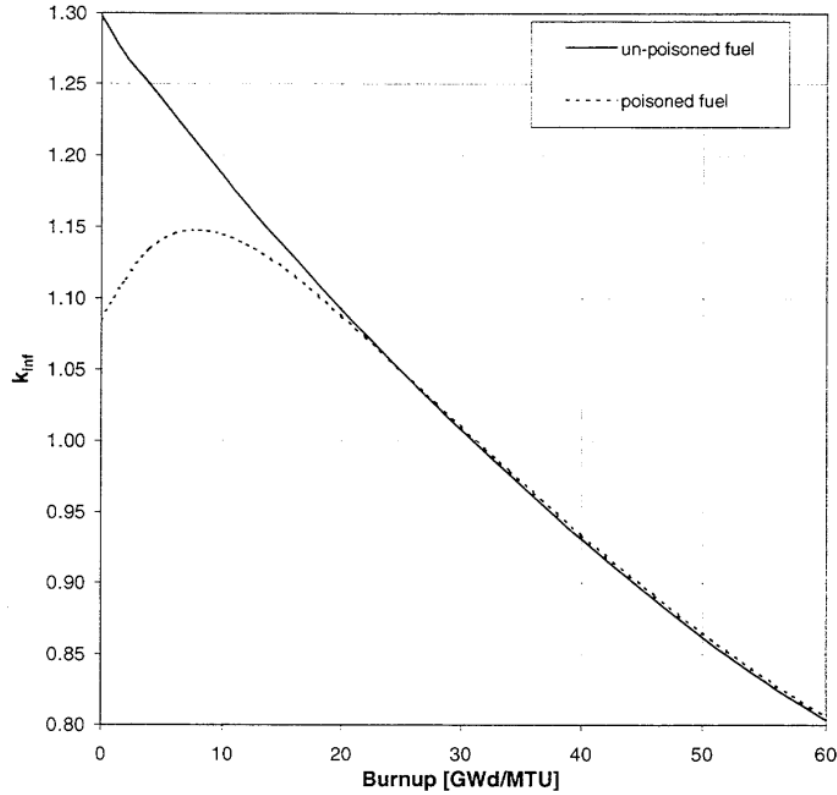


Figure 2: General behaviour of k_{inf} as a function of burnup for PWR fuel with BAs (dotted line) and without BAs (solid line) [10].

As will become clearer later in this study, the curve does not have to look like the one above. There are many factors that can change the form of the k_{inf} curve for a fuel using BAs.

2.5 Criticality safety Analysis

From the equations established for the multiplication factor k_{eff} , there exists three different scenarios for a fissile system.

- $k_{eff} < 1$ subcritical, decreasing neutron population.
- $k_{eff} = 1$ critical, constant neutron population
- $k_{eff} > 1$ supercritical, increasing neutron population [23].

As previously mentioned, the goal of criticality safety is to prevent unintentional criticality during management of fissile material outside the reactor. In other words, the fissile system should always remain subcritical. Consequently, a criticality safety analysis must always consider the following conditions:

- Normal operating conditions: Expected operating conditions and regularly recurring events during operation.

- Operational disturbance: Events representing deviations from normal operating conditions. Low probability and frequency.
- Accident/Failure: Large deviation from normal operating conditions. Very low probability and frequency [7].

Due to the difficulty to accurately model and predict many system conditions, a system should be approximated conservatively with the most simple geometry possible. Thus, k_{eff} will be overestimated. The sum of the k_{eff} , three standard deviations and all the biases and bias uncertainties affiliated with the calculation methods should not exceed the threshold 0.95. For conditions classified as accidents the threshold is set to 0.98 [7].

There are many regular conservative assumptions made when modelling a system in a normal criticality safety analysis, some of them are: Optimal moderation and reflection, Fresh fuel assumptions (maximum enrichment) and no consideration for the presence of BAs. However, the most important part is identifying the three different conditions for the system in question and model accordingly.

For the given system and condition, k_{eff} is calculated using validated Monte Carlo calculation codes. The k_{eff} should then be adjusted for bias and bias uncertainties according to benchmarks. Sensitivity studies shall be conducted where one parameter is adjusted at a time. This will show which parameter affects the system the most. More than two improbable and independent, coinciding events must be required to achieve criticality [7]. Implementation for a specific storage or transportation system is generally done by producing a loading curve. This curve usually indicates the maximum allowed fissile mass as a function of initial enrichment [27].

2.6 Burnup Credit Methodology

Burnup credit (BUC) is the concept of taking credit for the burnup of fissile isotopes in nuclear fuel when conducting a criticality safety analysis. Historically, criticality safety analyses assumed the fuel unirradiated, corresponding to maximum enrichment, since it provides a boundary for the reactivity effects for all fuels regardless of operating history. This simplifies the safety analysis significantly [28]. Ignoring the decrease in reactivity results in an excessively conservative assumption which may lead to reduced SNF storage cask capacity. For example, taking credit for burnup may increase the capacity for the number of PWR FAs in large dual-purpose casks by around 75% [29].

Using BUC will increase the complexity of criticality safety analysis since it both, adds a few extra long and laborious steps in the process, and requires consideration of operating history during burnup in the reactor. The general burnup credit calculation process can be summarized in four phases:

- Preparation: Defining problem, limits and available tools.
- Depletion: Simulate the isotopic concentration as a function of time.
- Criticality: Use the simulated isotopic concentration in a criticality safety analysis for an SNF system.
- Implementation: Present loading criterion for specific SNF system [27].

Here, the first two steps are the only unique steps to BUC, compared to regular criticality safety analysis.

The specific irradiation environment will have significant impact on the isotopic composition of the exposed fuel assembly. Creating a realistic fuel design model and operating conditions for the specified FA is therefore crucial. Fuel temperature, moderator temperature and density, soluble boron concentration, specific power, and operating history are a few parameters that need attention [9]. For example, for a high burnup fuel, an increase of moderator temperature of $1^\circ K$ during depletion will result in about 35 pcm reactivity for a generic SNF storage cask [28]. This reactivity increase, with increased moderator temperature, is a result of spectral hardening (higher average neutron energy) and production of plutonium, due to the decreased moderator density. Using a higher value for moderator temperature during BUC depletion calculations is therefore more conservative.

Certain limits for current BUC standards have been set based on supported data available. There are two types of safety analyses that can be made: Actinide-only burnup credit or limited actinide and fission product burnup credit. The set of nuclides in each type can be seen in Tables 1 and 2 below.

Table 1: Nuclides included in actinide-only BUC [9]

^{234}U	^{235}U	^{238}U
^{238}Pu	^{239}Pu	^{240}Pu
^{241}Pu	^{242}Pu	^{241}Am

Table 2: Additional nuclides included in actinide and fission product BUC [9]

^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh
^{109}Ag	^{133}Cs	^{147}Sm	^{149}Sm
^{150}Sm	^{151}Sm	^{152}Sm	^{143}Nd
^{145}Nd	^{151}Eu	^{153}Eu	^{155}Gd
^{236}U	^{243}Am	^{237}Np	

The available data only support up to 5 wt% (weight percent) ^{235}U in UO_2 fuel irradiated in a PWR up to a burnup of 60 GWd/MTU (Gigawatt-days per metric tonne of uranium). The SNF must also be cooled out of reactor for a time of 1 to 40 years [9].

The presence of BAs in a FA during irradiation will have a considerable effect on the k_{eff} as a function of burnup, as mentioned in section 2.4. The reactivity profile in figure 2 can look differently depending on the BA material, content and placement [9]. Finding conditions that conservatively bound the reactivity effects of different types of BAs are vital. For example, neglecting the existence of the absorber material in a PWR fuel using IBAs was found to be bounding and conservative, but non-conservative when utilizing IFBAs [10]. However, bounding conditions may differ depending on if actinide-only BUC or actinide and FP BUC is conducted [9].

Accounting for and accurately modelling the axial and horizontal burnup profile in the SNF assembly when calculating k_{eff} for a certain cask is important [9]. The axial burnup profile in particular is dependent on several reactor operating conditions and therefore have

a considerable variations in its appearance. Generally, the burnup is higher in the middle parts of the FA than in the top and bottom due to less neutron leakage. In addition, due to cooler moderator, and therefore higher moderator density, at the bottom than at the top, the burnup is moderately higher at the bottom than at the top [30]. Since burnup is strongly linked to reactivity, modelling and applying the correct burnup profile must be carefully considered when conducting the criticality safety analysis step of the BUC.

3 Methodology

This section describes the approach used to evaluate the reactivity effects of gadolinia IBAs on a VVER-1000 fuel assembly and to assess the implications for burnup credit methodology.

3.1 MCNP

The code used to model and perform simulations in this study is Monte Carlo N-Particle (MCNP), version 6.2. MCNP code is a Monte Carlo program used for calculating particle transport, including the multiplication factor k_{eff} . It is general-purpose, continuous-energy, generalized-geometry, and time-dependent designed to track 37 particle types (mainly neutrons, photons and electrons) over a wide range of energies. It is widely used for modelling of nuclear systems, such as reactor cores and fuel storage configurations. The Monte Carlo method describe the propagation of particles theoretically using a random walk process. Instead of solving the transport equation for particles for the average particle, like deterministic methods do, the Monte Carlo method follows the particle from birth to death sequentially, sampled using probability distributions based on nuclear data. Since the outcome of each particle is based on selection of random numbers, the simulation will need a large amount of trails to adequately describe the phenomenon appropriately [31].

MCNP can be run by sending an input file with the type .txt into the code. The input file needs a certain structure and syntax to work properly. It consist of several card blocks, each with its specific information. Three card blocks are mandatory: cell card block, surface card block and data card block. These three cards create a 3D model by defining cells, surfaces and materials used for the simulation. Figure 3 below show an example of a geometry for the simulations.

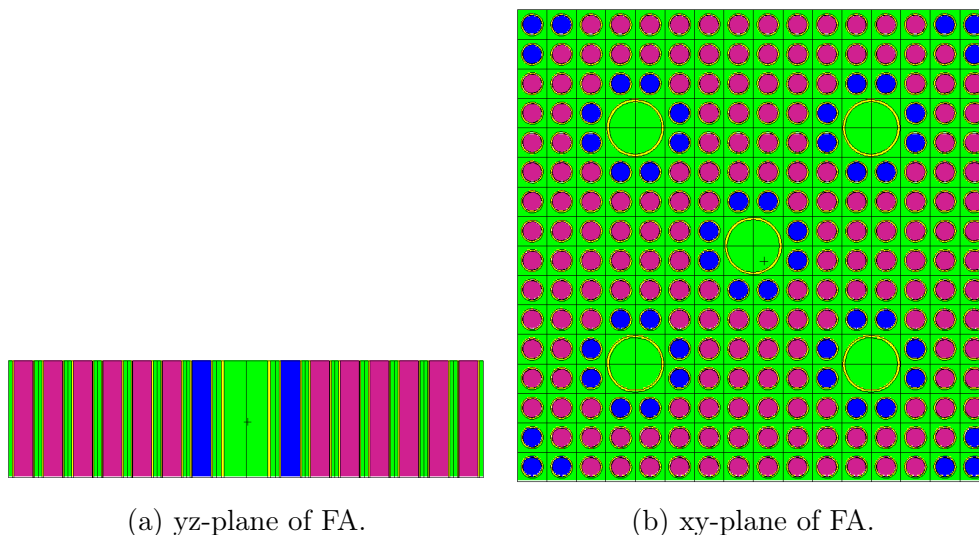


Figure 3: Example of how a MCNP-created 3D model can look like. The pink circles are fuel rods, the blue circles are IBA rods, and the five big water filled circles are guide tubes.

This FA model of the CE16x16 was created as an MCNP introduction final task.

To setup a criticality problem with burnup, a few different data cards are needed. Firstly, the source card where the criticality source, KCODE, is set up. Here, the number of source histories per cycle, the number of cycles to be skipped before tally accumulation (inactive

cycles), total number of cycles to be done are established (active cycles). The number of source histories per cycle was set to 50000, inactive cycles was set to 10, and active cycles to 60. Secondly, the criticality source points, KSRC, need to be put inside a fissile volume i.e. the fuel rods. Lastly, the BURN data card where the settings for the depletion calculations need setting up. Here, the burnup steps, fuel regions, and power are established [31].

MCNP depletion is a linked process involving steady-state flux calculations in the MCNP code and nuclide depletion calculations in CINDER90. The CINDER.dat library file contains decay, fission yield, and 63-group cross-section data not calculated by the MCNP code. MCNP runs a steady-state calculation to determine the system eigenvalue, 63-group fluxes, energy-integrated reaction rates, fission multiplicity (ν), and recoverable energy per fission (Q values). CINDER90 then takes these MCNP-generated values and performs the depletion calculation to generate new atom densities for the next time step. The MCNP code takes these new atom densities and generates another set of fluxes and reaction rates. This iterative process continues until the final burnup step specified by the user is reached [31].

3.2 Fuel design

The fuel design used for this study is a VVER-1000 design from Westinghouse. The reason for this is the availability of previous works for code validation and the existence of a validated reference MCNP input file. Furthermore, VVER and PWR fuels share identical materials and have similar operating conditions and thermal neutron spectrum, allowing for comparable depletion models. Although VVER fuels use hexagonal lattice geometry, MCNP calculates the neutron transport based on local material distribution and not on global lattice shape. So, if the moderator-to-fuel ratio is similar, and ^{235}U enrichment and burnup are comparable then the local neutron balance remains equivalent. Thus, the results of this study should be applicable to both PWR and VVER [32].

As previously mentioned, k_{eff} should be calculated using fuel design and reactor operating conditions that allow accurate representation of the physics in the system [9]. The input file models the hexagonal VVER-1000 FA with fuel pellets, with and without IBAs, guide tubes and central tube, and the water between these components. The mechanical design parameters taken into account is presented in table 3 below.

Table 3: Mechanical design parameters taken into account for a VVER-1000 FA [33].

Parameter
Rods in FA
Rod cladding \varnothing , mm
Cladding thickness, mm
Cladding material
Fuel pellet \varnothing , mm
Fuel pellet density (% of theoretical)
Fuel pellet dishing factor, %
Rod pitch, mm
Number of Guide tubes/Central tube
Material of Guide tubes/Central tube
Guide tubes/Central tube \varnothing , mm
Guide tubes/Central tube thickness, mm

The material densities and isotope compositions of the cladding, guide tubes, central tube, and water were already defined in the reference input file provided from [33]. The specific power of the FA was set to reflect the real specific power of the VVER-1000 and the height of the FA was 5 cm in the simulations, but with reflective boundary conditions. This can be regarded as an infinitely large reactor. In-core temperatures of the various components was set to normal values for typical PWR operation. Thus, 1200 °K was set as the fuel temperature and 600 °K for the moderator temperature [34]. In this study, no utilization of soluble boron and control rods were taken into account. Adding more neutron absorbing material would make the reactivity effects of IBAs less apparent.

To achieve a general consensus of the reactivity effects of IBAs with gadolinia, a number of scenarios must be tested. As described previously, one of the most lucrative uses of BAs is prolonged lifetime of fuel in the core. The usage of BAs varies wildly depending on reactor design. Number of IBA rods and their placement, and absorber material and how much is utilized [10]. Therefore, different ^{235}U enrichments, gadolinia content, and number of IBA rods are parameters that are reasonable to consider. Different configurations of the IBA rods have not been considered in this study. Parameter variations used in this study are described in table 4 below.

Table 4: Parameter variations

Parameter	Variations
^{235}U enrichment, %	4, 5
gadolinia content, wt%	3, 5, 8
Number of IBA rods	6, 12, 24

For every ^{235}U enrichment, there will be a corresponding gadolinia content of either 3, 5 or 8 percent, and for every one of those there will be a number of 6, 12 or 18 IBA rods. Thus, there are a total of 18 different scenarios that will be simulated according to table 4. For comparison, and reference, two simulations will be conducted without IBAs for both 4% and 5% ^{235}U enrichment.

Firstly, the placement of the IBAs must be considered. Since ^{235}U and gadolinia content do not affect FA geometry there will be a total of three FA designs, one for each number of IBA rods in the geometry. The FA designs are showed in figure 4 below. Due to Westinghouse limitations, the IBA rods may not be placed in the two outermost rows [33]. The FA geometry must also remain symmetric for all designs.

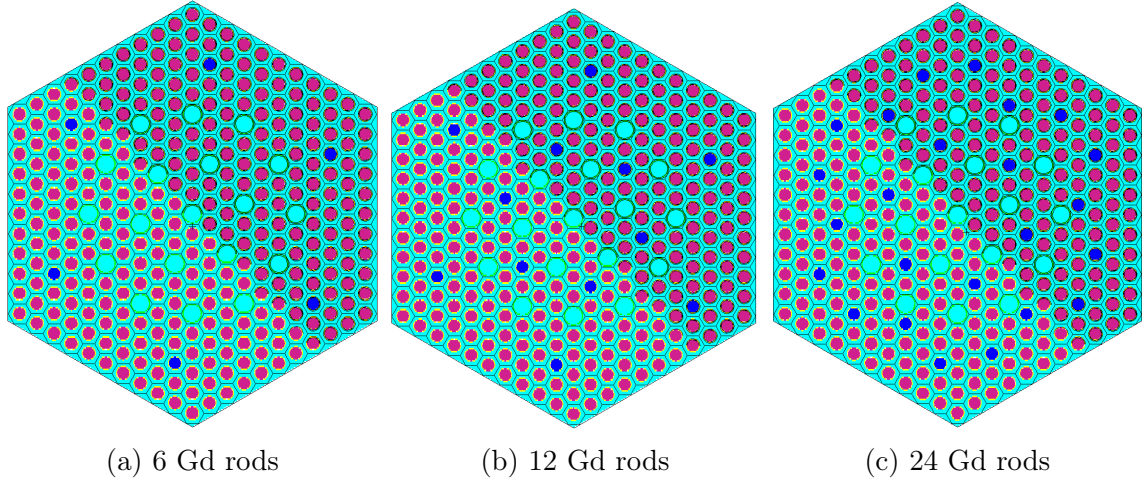


Figure 4: Fuel assembly designs - The blue circles are IBA rods, the red circles are regular fuel rods, and the hollow circles are guide tubes/central tube.

Secondly, isotope compositions for the fuel rods and IBA rods had to be calculated for all 18 scenarios. Enrichment of ^{235}U is either given as a percentage of the total amount of uranium isotopes or as a percentage of the total weight of the fuel, wt% (weight percentage). MCNP demands that isotope composition for a given material must be given as a weight percentage. The ^{235}U enrichments presented in table 4 must therefore be recalculated. Similarly, several different isotopes of gadolinium are present in gadolinia and must also be taken into account. The natural abundances of different gadolinium isotopes, which can be seen in table 5, are used in the calculation of the weight percentage. Atomic weights are also crucial for calculating isotope composition. Relevant isotopes and their atomic weights can be found in table 6 below. The mean weight of gadolinium was used instead of individual atomic weights.

Table 5: Abundances of different gadolinium isotopes in gadolinia [35].

Isotope	Abundance [%]
^{152}Gd	0.2
^{154}Gd	2.18
^{155}Gd	14.8
^{156}Gd	20.47
^{157}Gd	15.65
^{158}Gd	24.84
^{160}Gd	21.86

Table 6: Atomic weights used for different isotopes in Isotope composition calculations [35].

Isotope	Atomic Weight [u]
^{235}U	235.0439301
^{238}U	238.0507884
^{16}O	15.99491462
Gd_{mean}	157.25

Lastly, the burn time for the simulations was calculated. As stated previously, BUC methodology allows a maximum burnup for a PWR fuel to be 60 GWd/MTU. Thus, this will also

be the maximum burnup for this study. To capture the behaviour of the reactivity effects in more detail, burnup steps will be shorter up to the burnup of 20 GWd/MTU. After that point, the reactivity for all fuels decreases in a nearly linear fashion. MCNP defines burn time with EFPD, as described in section 2.4, instead of GWd/MTU. Converting from GWd/MTU to EFPD can be done using equation 5.

3.3 Code validation

A big part of burnup credit methodology is validating nuclear data and models. Computer codes calculating concentrations and depletion of isotopes important to BUC needs validation to ensure accurate criticality calculation results. Validation should also include bias and bias uncertainty of the code at a 95% probability with 95% confidence interval [9].

MCNP has undergone extensive verification and validation for k_{eff} calculations against the International Criticality Safety Benchmark Evaluation Project (ICSBEP) experimental database and dedicated Los Alamos national laboratory (LANL) test suites. Therefore, in this study, a single comparison was performed between MCNP and PHOENIX5 to confirm that the MCNP model is correctly configured for depletion calculations.

PHOENIX5 code is a two-dimensional neutron and gamma transport code developed by Westinghouse Electric Sweden for lattice burnup. The geometrical representation of a fuel assembly is detailed in PHOENIX5. This applies to the distribution and composition of individual fuel rods, with or without burnable absorbers, as well as other regions such as moderator, fuel channels, and water gaps, and control rods. For calculations of reactivity and local power distributions, PHOENIX5 utilizes various approximations to solve the transport equation in two dimensions. The standard nuclear cross section library for PHOENIX5 is based on the ENDF/B-VII.1 data set and contains nuclear data for a large number of isotopes and some materials that are essential for lattice calculations including isotope depletion [36].

The comparison can be found in figure 5 and shows that the MCNP code receive similar results to PHOENIX5. This comparison serves as a problem specific verification, while the general accuracy of MCNP for k_{eff} predictions is supported by published validation studies [37] [38].

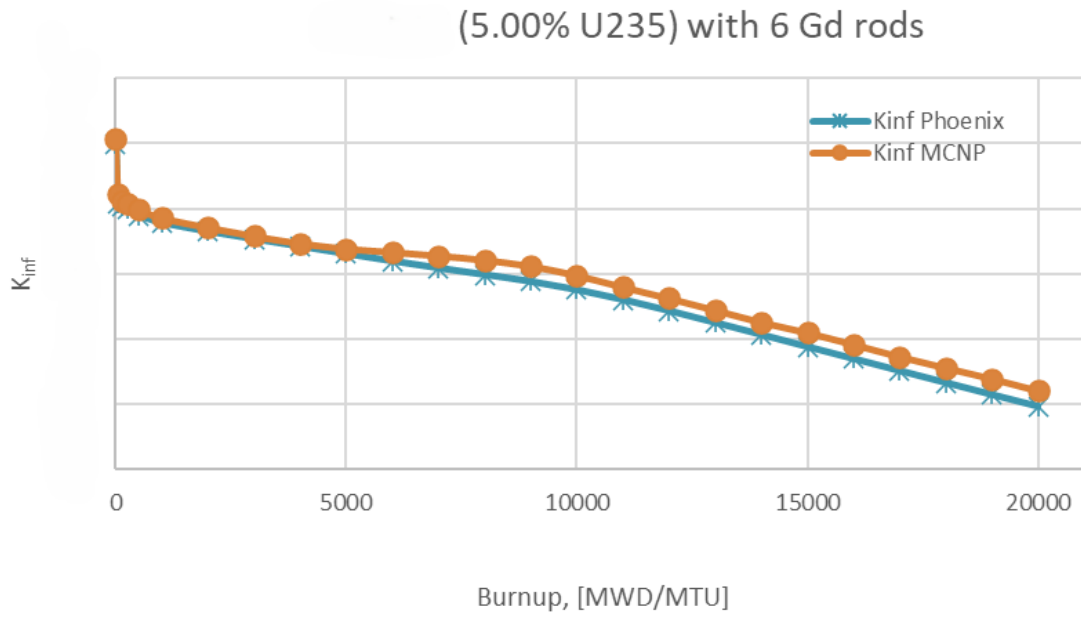


Figure 5: Comparison between MCNP and PHOENIX5. The plot shows the behaviour of k_{inf} as a function of burnup for PHOENIX5 (blue) and MCNP (orange) [33].

4 Results

In this chapter, results from the simulations of 18 different FA configurations, with IBAs, together with the two reference configurations, without IBAs, are presented.

4.1 4% ^{235}U enrichment

This section presents the simulation results of the FA configurations with a 4% ^{235}U enrichment with either 3%, 5% or 8% of gadolinia content. Figures 6, 7, and 8 below show the reactivity effects as a function of burnup for the three FA designs presented in figure 4 above. The reference case without IBAs (black line) is also plotted for comparison.

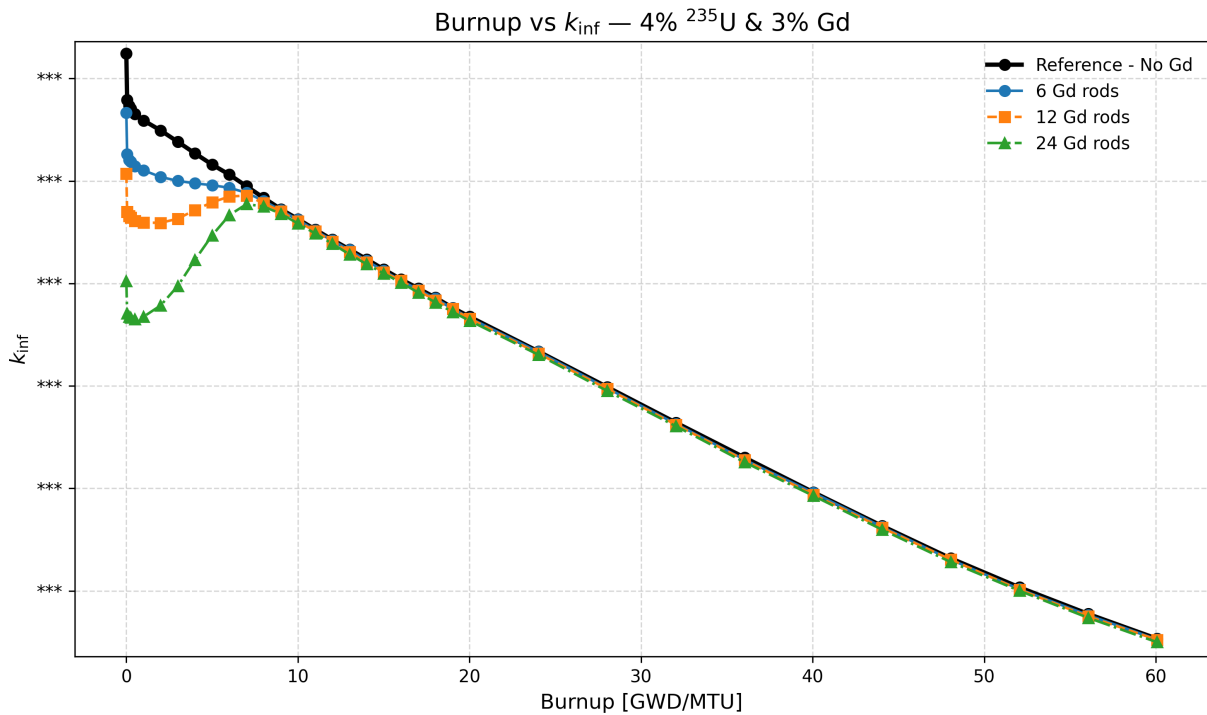


Figure 6: k_{inf} as a function of burnup for an FA with 4% enrichment of ^{235}U and 3% gadolinia content (The y-axis has been censored due to proprietary information).

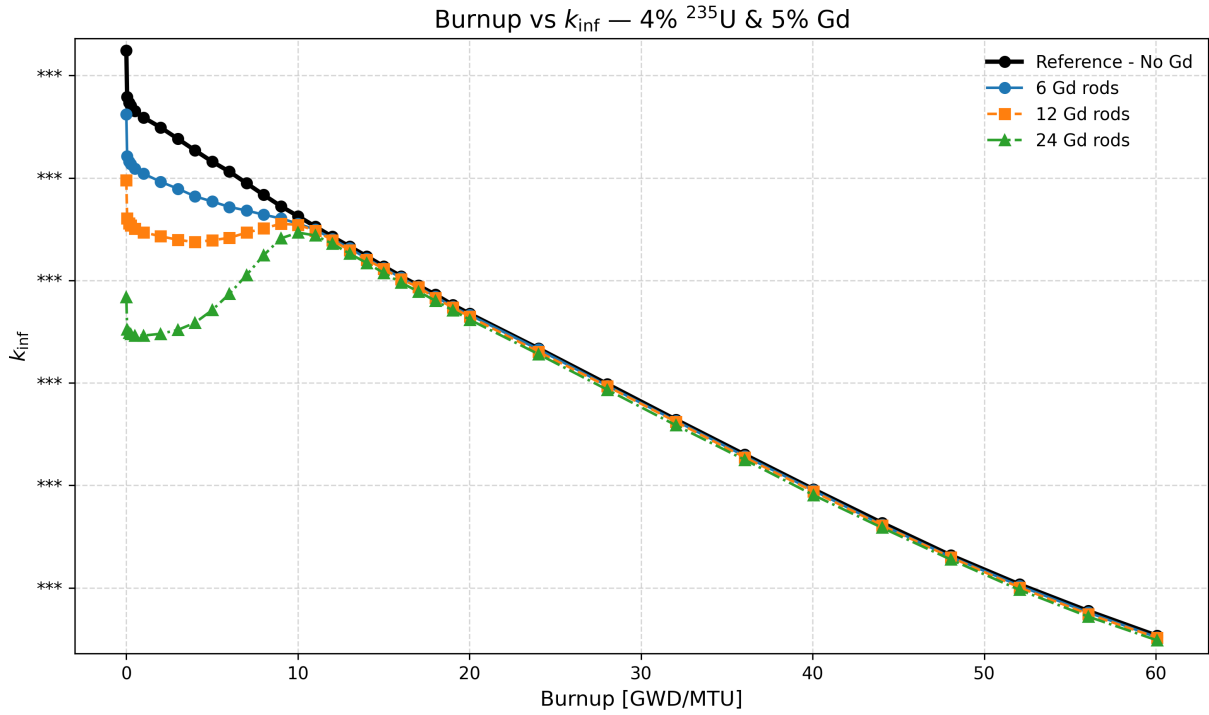


Figure 7: k_{inf} as a function of burnup for an FA with 4% enrichment of ^{235}U and 5% gadolinia content (The y-axis has been censored due to proprietary information).

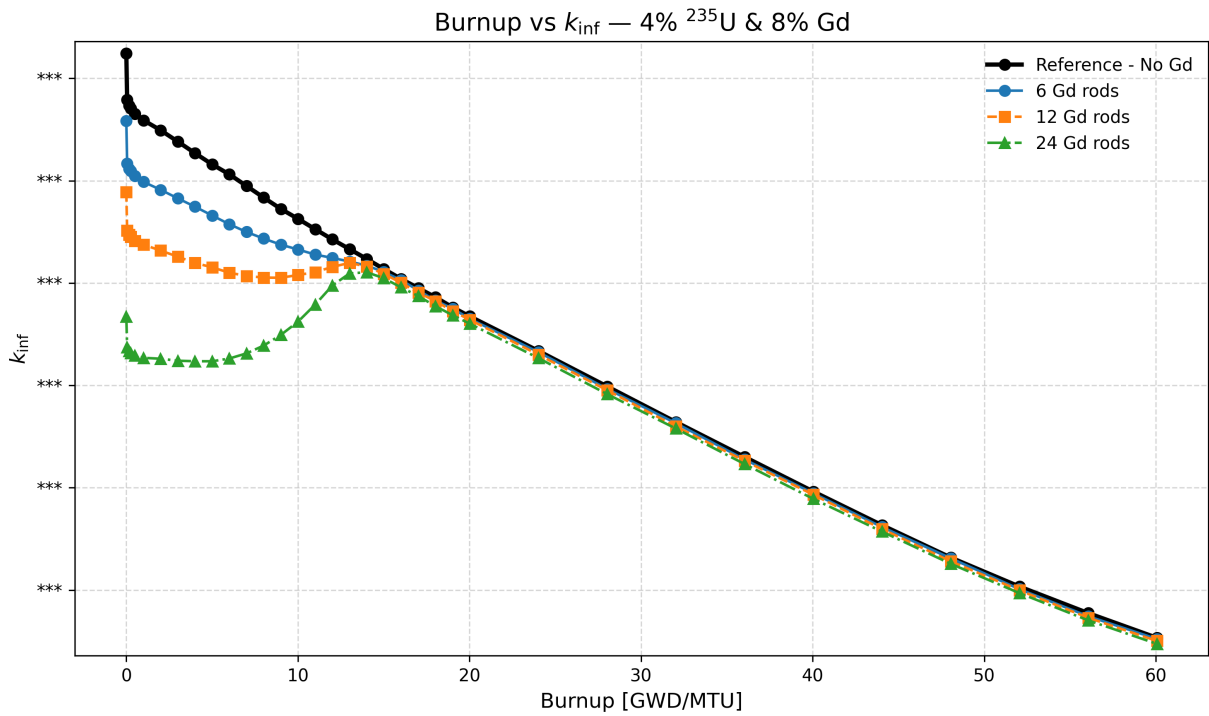


Figure 8: k_{inf} as a function of burnup for an FA with 4% enrichment of ^{235}U and 8% gadolinia content (The y-axis has been censored due to proprietary information).

Figures 6, 7, and 8 show some trends that explain the effects of both the gadolinia content and the number of rods. First, when the gadolinium is depleted, all three figures show that the reactivity decreases, for all cases, following the reference case in a similarly linear fashion. If they are above or below the reference case is hard to discern and will be investigated further down. Secondly, a higher gadolinia content contributes to a longer depletion time for BA. Figure 6 shows that gadolinium depletes around 8 GWd/MTU, 10 GWd/MTU for figure 7, and 14 GWd/MTU for figure 8. It also slightly decreases the initial reactivity of the FA. Lastly, a larger number of IBA rods will cause a larger initial decrease in reactivity. However, for the cases with 24 IBA rods, peak reactivity does not occur at 0 GWd/MTU, but rather when the gadolinium is depleted. Thus, these are good examples of why fresh fuel approximations cannot be applied to fuels using BAs. For cases with 6 IBA rods, the reactivity decreases in a more linear fashion until it is depleted. Designs with 12 IBA rods have an observable upswing in reactivity at the end of the gadolinium lifetime but never as much as the cases with 24 IBA rods.

To see how much the reactivity differs from the reference case, Δk_{inf} was plotted as a function of burnup. Since fuels utilizing BAs generally have a lower reactivity than fuels without, $\Delta k_{inf} = k_{inf,BA} - k_{inf,ref}$ in figures 9, 10 and 11. Thus, discerning any point where $k_{inf,BA} > k_{inf,ref}$ will become easier. Considering that the reactivity difference is close to zero after the depletion of gadolinium, three additional lines were plotted to showcase this in more detail. Finally, the uncertainties associated with the Monte Carlo method were plotted on the zoomed lines as error bars. The error bars show one standard deviation for Δk_{inf} and were calculated using error propagation for subtraction [39]. k_{inf} calculations with applied MCNP settings generally result in a standard deviation of around 35–40 pcm, consequently resulting in a standard deviation for Δk_{inf} of around 50–55 pcm.

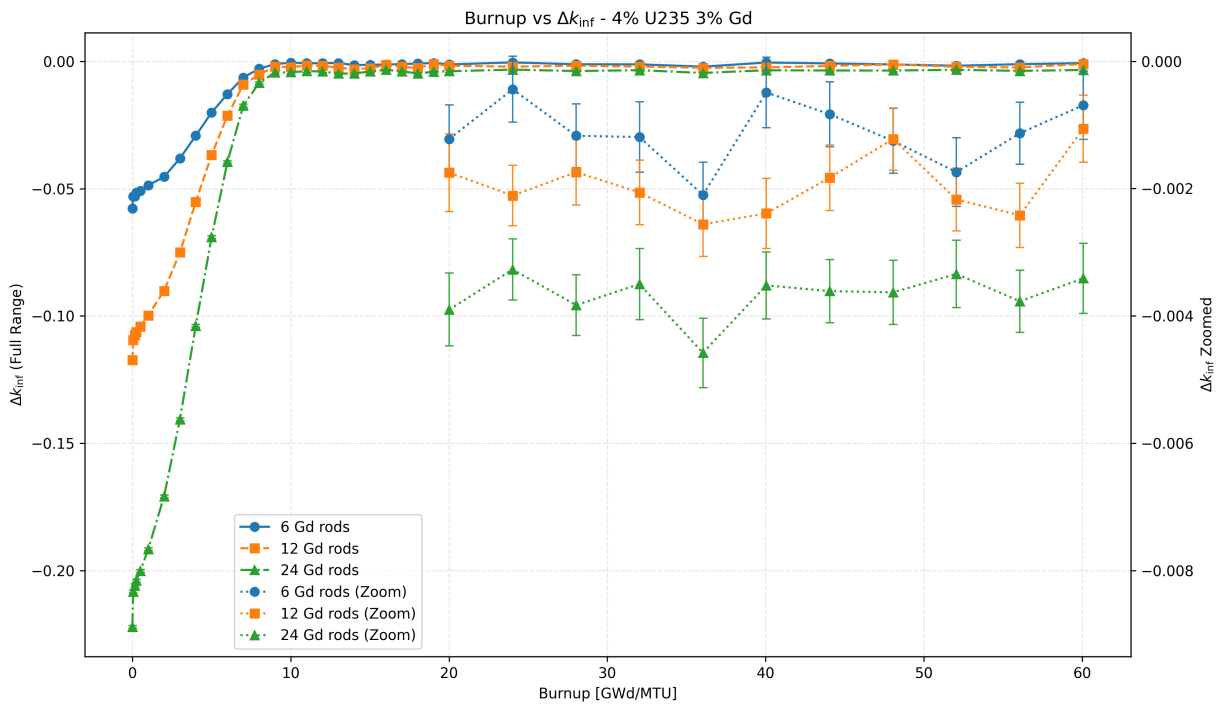


Figure 9: Δk_{inf} as a function of burnup for an FA with 4% enrichment of ^{235}U and 3% gadolinia content. After 20 GWd/MTU, the same values are plotted again but zoomed in for the sake of clarity. The error bars show one standard deviation for Δk_{inf} .

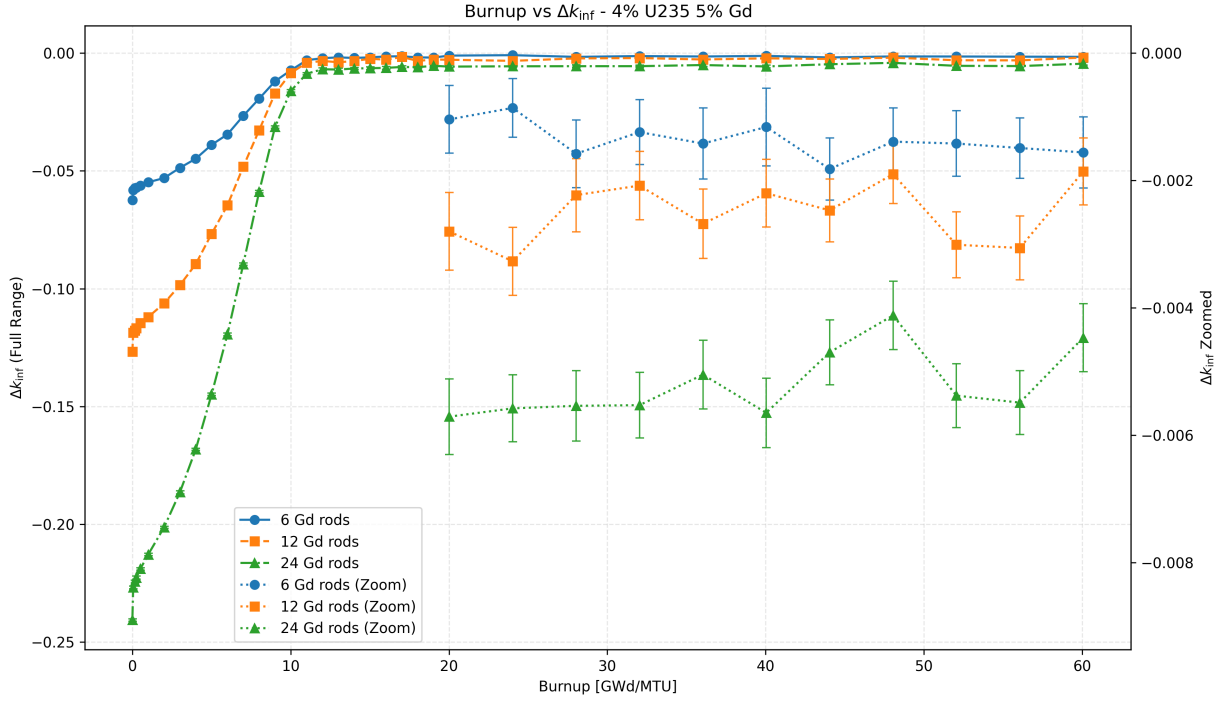


Figure 10: Δk_{inf} as a function of burnup for an FA with 4% enrichment of ^{235}U and 5% gadolinia content. After 20 GWd/MTU, the same values are plotted again but zoomed in for the sake of clarity. The error bars show one standard deviation for Δk_{inf} .

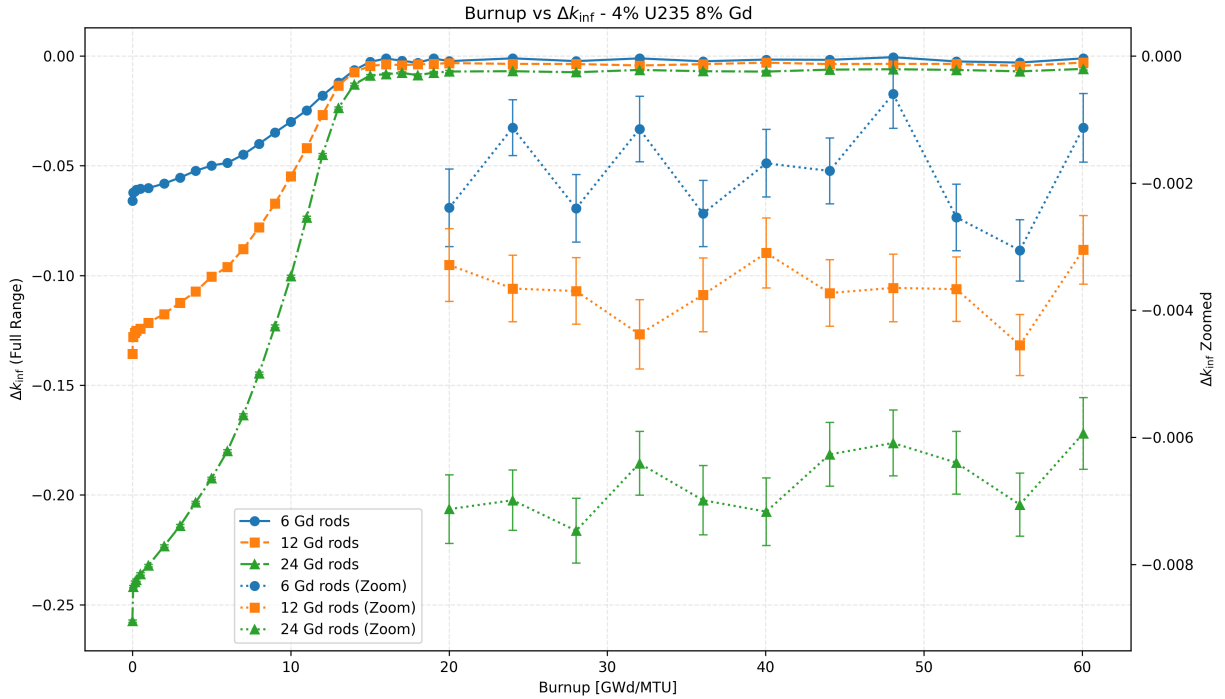


Figure 11: Δk_{inf} as a function of burnup for an FA with 4% enrichment of ^{235}U and 8% gadolinia content. After 20 GWd/MTU, the same values are plotted again but zoomed in for the sake of clarity. The error bars show one standard deviation for Δk_{inf} .

Figures 9, 10 and 11 show more clearly when gadolinium is depleted for each design i.e. flattening of the curves. Most importantly, at any point during FA burnup, $\Delta k_{inf} < 0$.

Both gadolinia content and the number of IBA rods have an effect on Δk_{inf} . Firstly, Δk_{inf} generally increases with the number of rods. The reason for this and the uncertain zig-zag behaviour will be discussed later. Secondly, the gadolinia content generally increases slightly Δk_{inf} when comparing between designs with the same number of rods. However, this effect is not as apparent as the effect of the number of rods.

4.2 5% ^{235}U enrichment

This section will present the simulation results from the FA configurations with 5% ^{235}U enrichment with either 3%, 5% or 8% of gadolinia content. Figures 12, 13, and 14 below show the reactivity effects as a function of burnup for the three FA designs presented in figure 4 above. The reference case without IBAs (black line) is also plotted for comparison.

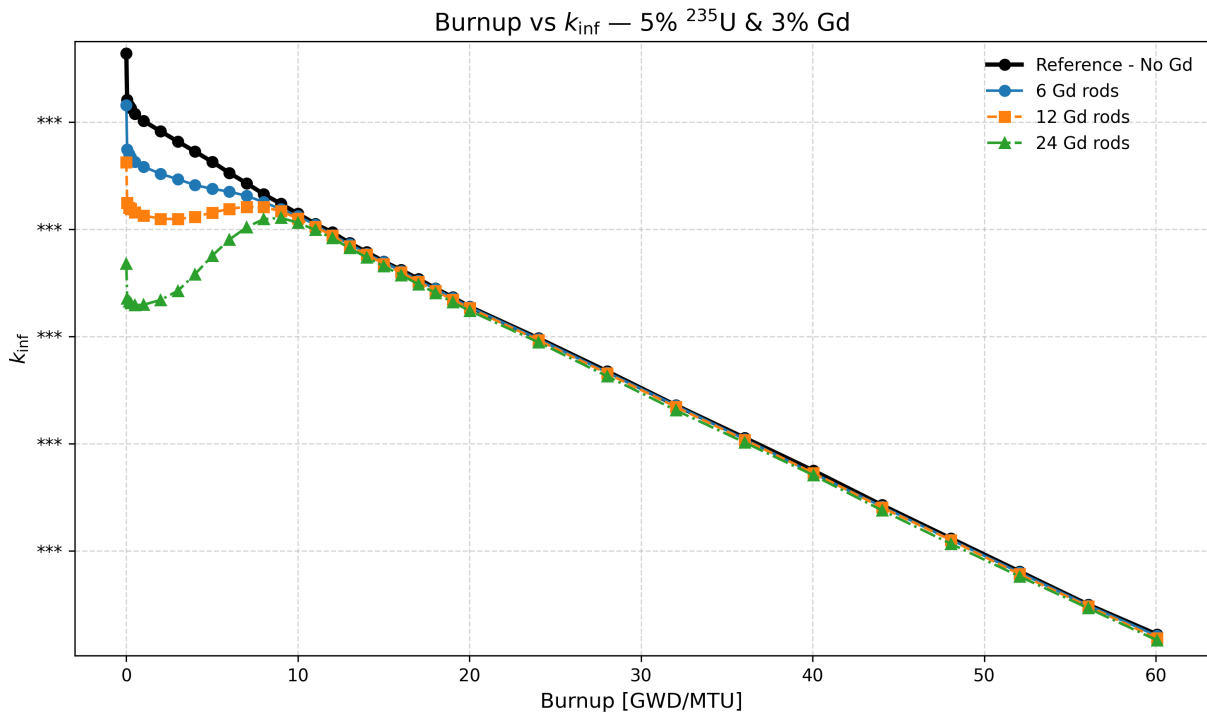


Figure 12: k_{inf} as a function of burnup for an FA with 5% enrichment of ^{235}U and 3% gadolinia content (The y-axis has been censored due to proprietary information).

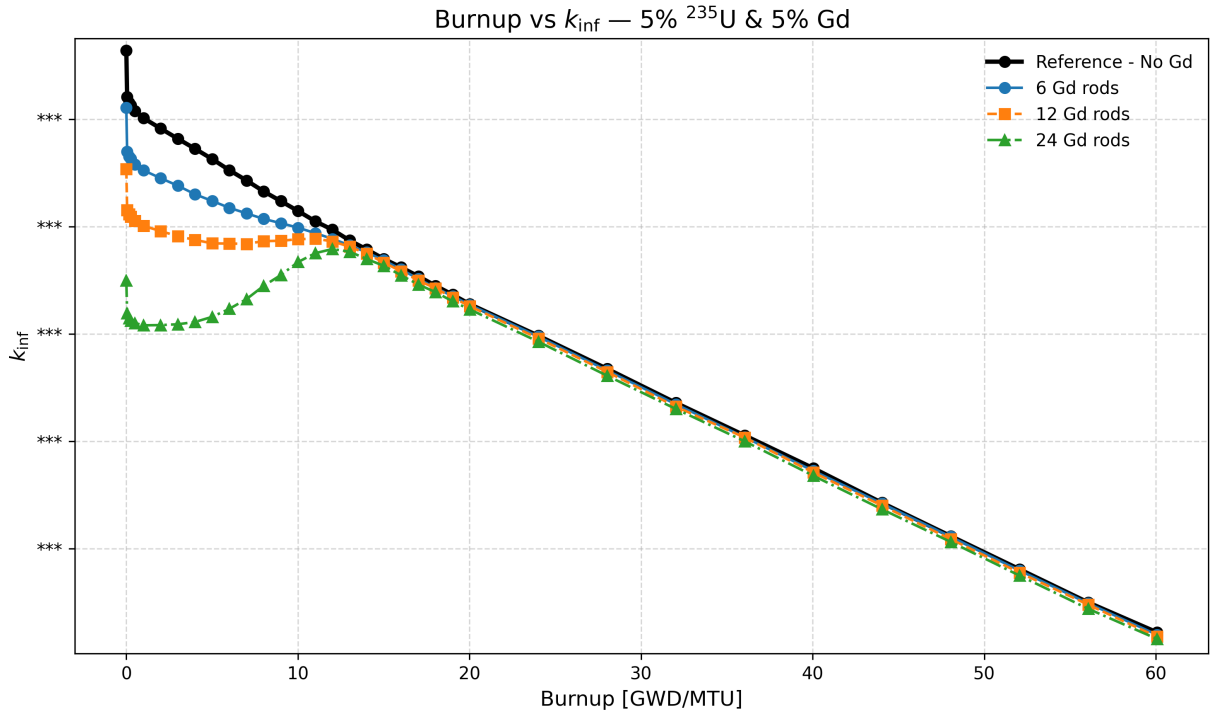


Figure 13: k_{inf} as a function of burnup for an FA with 5% enrichment of ^{235}U and 5% gadolinia content (The y-axis has been censored due to proprietary information).

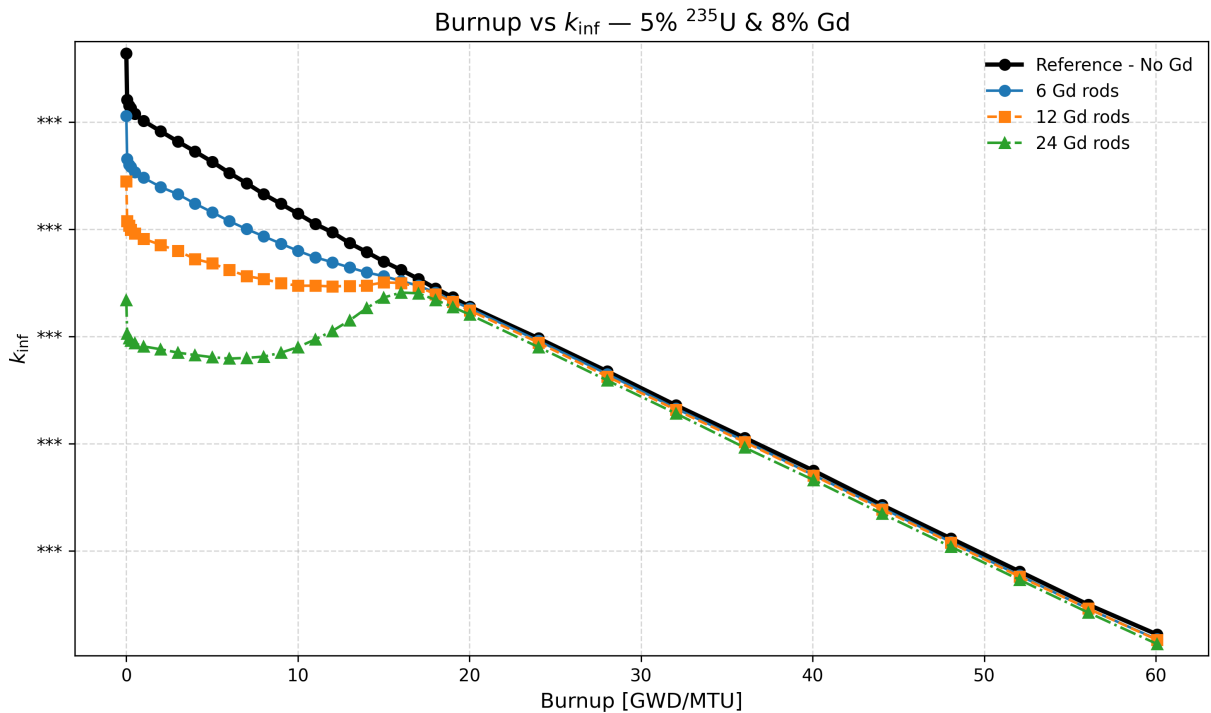


Figure 14: k_{inf} as a function of burnup for an FA with 5% enrichment of ^{235}U and 8% gadolinia content (The y-axis has been censored due to proprietary information).

Figures 12, 13 and 14 plotting the different cases with 5% ^{235}U enrichment show different behaviours than those with 4% enrichment. The effects of BAs explained in section 4.1 are

apparent in these scenarios as well. When gadolinium is depleted, the reactivity decreases with the same rate as in the reference case. A higher gadolinia content contributes to a longer depletion time for gadolinium, regardless of the number of IBA rods. Figure 12 shows that gadolinium depletes around 10 GWd/MTU, 13 GWd/MTU for figure 13 and 18 GWd/MTU for figure 14. This shows that a higher enrichment of ^{235}U paired with a higher gadolinia content will yield longer depletion times for gadolinium. Similarly, a larger number of IBA rods will cause a larger initial decrease in reactivity.

To see how much the reactivity differs from the reference case, Δk_{inf} was plotted as a function of burnup. The plots in figure 15, 16, and 14 were set up the same way as in section 2.4.

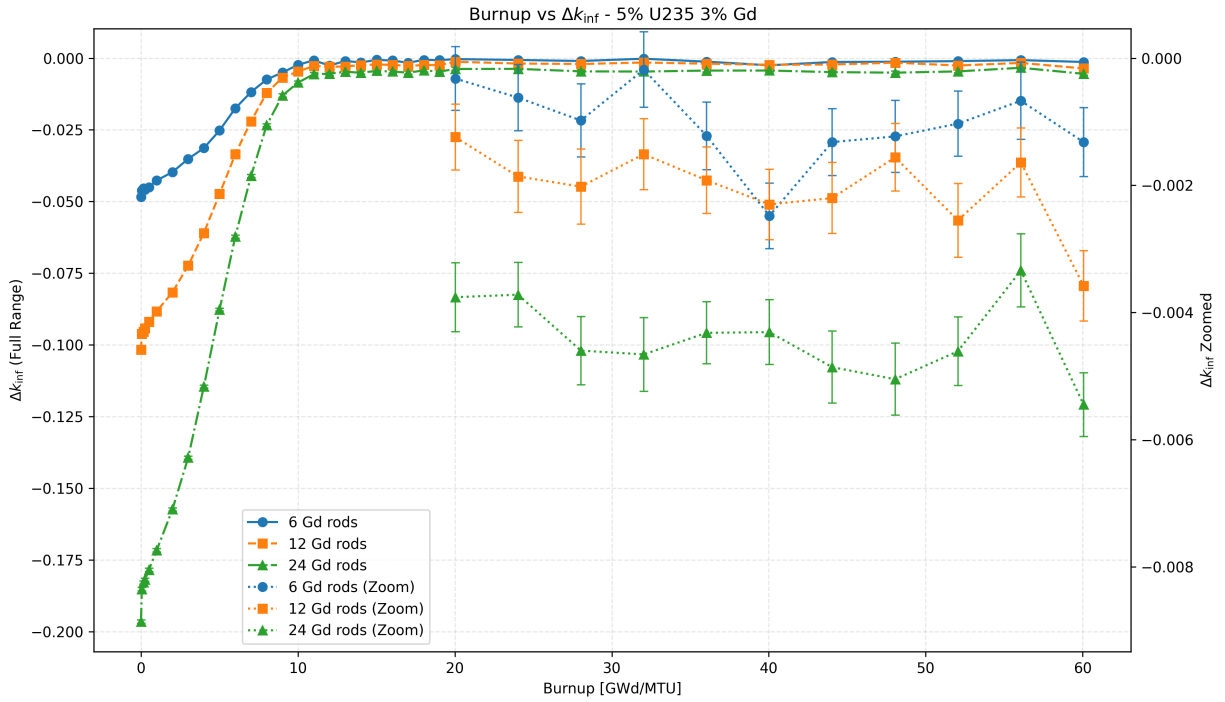


Figure 15: Δk_{inf} as a function of burnup for an FA with 5% enrichment of ^{235}U and 3% gadolinia content. After 20 GWd/MTU, the same values are plotted again but zoomed in for the sake of clarity. The error bars show one standard deviation for Δk_{inf} .

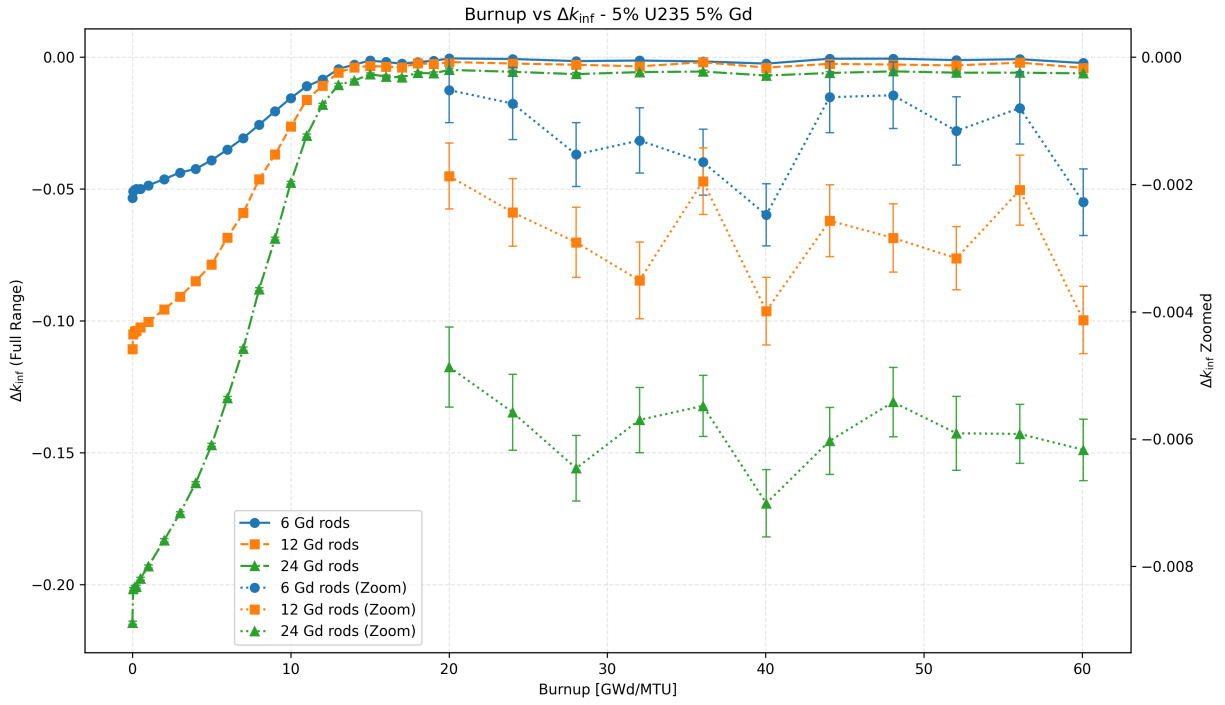


Figure 16: Δk_{inf} as a function of burnup for an FA with 5% enrichment of ^{235}U and 5% gadolinia content. After 20 GWd/MTU, the same values are plotted again but zoomed in for the sake of clarity. The error bars show one standard deviation for Δk_{inf} .

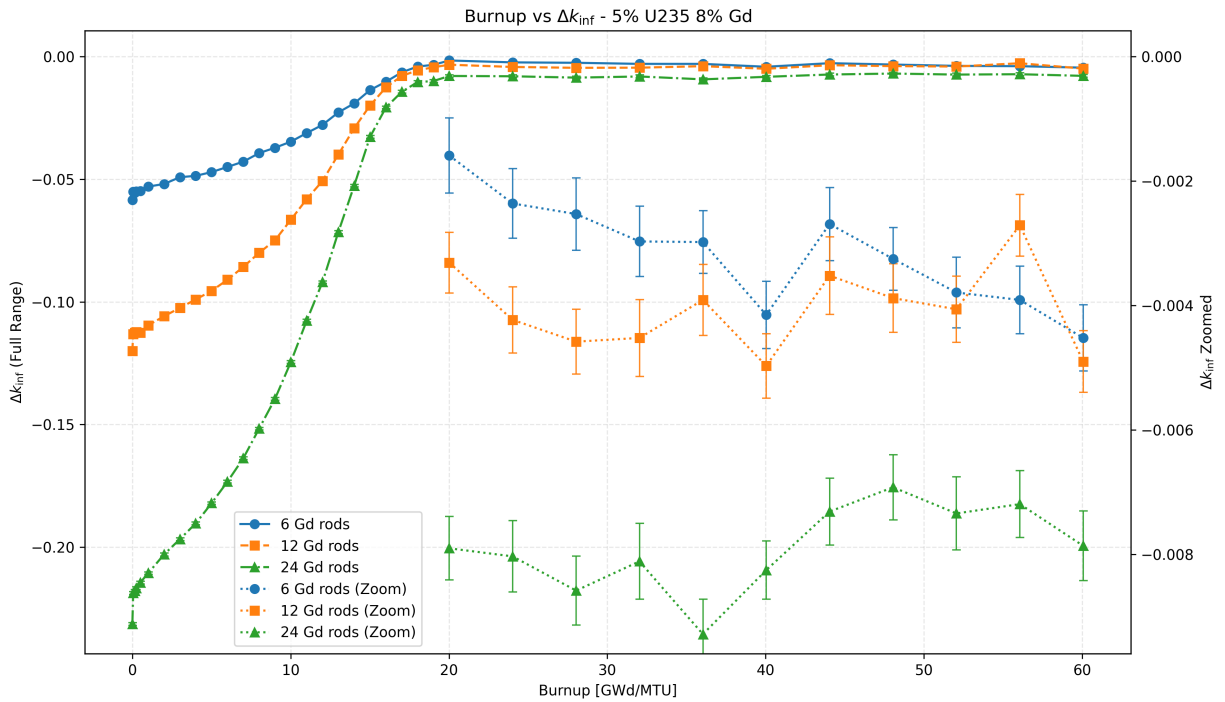


Figure 17: Δk_{inf} as a function of burnup for an FA with 5% enrichment of ^{235}U and 8% gadolinia content. After 20 GWd/MTU, the same values are plotted again but zoomed in for the sake of clarity. The error bars show one standard deviation for Δk_{inf} .

Similarly as in scenarios with 4% enrichment of ^{235}U , the figures 15, 16, and 17 show more clearly when gadolinium is depleted. They also conclude that $\Delta k_{inf} < 0$ at all points during

FA burnup. Here also, both gadolinia content and the number of IBA rods have an effect on Δk_{inf} . Increasing the number of rods generally increases the Δk_{inf} . Having a higher gadolinia content increases Δk_{inf} slightly when comparing between designs with the same number of rods. However, this effect is not as obvious as the effect from the number of rods. Generally, the effects on Δk_{inf} are very similar to the ones in section 4.1. Any effects from the enrichment of ^{235}U , for example when comparing between figure 11 and 17, are hard to perceive due to data point fluctuations.

5 Discussion

The results from all 18 simulations clearly demonstrate that the reactivity of VVER-1000 FAs incorporating IBAs is consistently lower than that of the same FAs without IBAs. Since it is the gadolinia content that mainly affects depletion time and the number of IBA rods that affect reactivity decrease, simulating scenarios with a larger number of IBA rods or higher gadolinia content is redundant.

As mentioned in section 2.5, criticality safety analyses use models that predict system behaviour conservatively. Thus, when conducting BUC calculations, the same principles should be applied. Given the vast number of possible combinations for loading BAs in a reactor, including variations in BA type, isotopic composition, rod count, and spatial arrangement, a recommended approach for licencing basis analysis would be to group BA types based on their reactivity effects and characteristics. For the case of this study, only recommendations for PWR fuels using gadolinia IBA rods can be given. Since all 18 simulated cases had lower reactivity than the reference case without IBAs, a bounding and conservative approximation would therefore be to neglect the existence of IBAs. After the depletion of gadolinium, the reactivity decrease of the IBA cases follows the reference case without IBAs closely. Considering that most reactors discharge fuel assemblies at an average burnup of approximately 50 GWd/MTU, and that the observed difference in infinite multiplication factor $|\Delta k_{inf}|$ remains below 0.01 (1000 pcm) for all cases, this simplification introduces only a minor conservatism for most SNF. Therefore, ignoring IBAs in BUC calculations for licencing purposes appears to be a reasonable and defensible approach.

Neglecting IBAs with gadolinia have important implications for both criticality safety analysis and operational economics. The conservative treatment of IBAs ensures that criticality margins are met without requiring comprehensive modelling of many different IBA configurations. Consequently, this approach reduces computational complexity and may accelerate licencing processes, which is very valuable in an industry plagued by long licence processing times.

5.1 Residual reactivity suppression

As seen in the figures showing Δk_{inf} as a function of burnup, increasing the number of rods and the gadolinia content will increase Δk_{inf} after gadolinium was depleted. This phenomenon can be explained simply with the change in ^{235}U density in the IBA fuel rods. The wt% of ^{235}U decreases as the gadolinia content increases. Lower fuel density results in lower reactivity compared to fuels without IBAs. Theoretically, a higher gadolinia content or a larger number of IBA rods should result in a more negative residual reactivity suppression. Not only is there a lower fuel density of IBA rods compared to regular fuel rods, causing a slightly lower reactivity, but the remaining gadolinium isotopes in the fuel still have small neutron absorption cross section further decreasing reactivity [40]. Compared to IFBA, the type of fuel that puts the BA as a coating around the fuel pellet, where no displacement of ^{235}U occurs and consequently no negative residual reactivity suppression transpire. Another study showed that a PWR fuel using IFBA rods may have a positive Δk_{inf} after the IFBA burned out. The same study showed that PWR fuels using IBA rods of type Gd_2O_3 (gadolinia), Er_2O_3 (erbium oxide) or $\text{Al}_2\text{O}_3\text{-B}_4\text{C}$ (aluminium oxide boron carbide) always have a negative Δk_{inf} , which is in line with the findings in this study [10].

5.2 Radial burnup effect

At some points in the figures, the Δk_{inf} is smaller for the scenario using 12 IBA rods than for the scenario using 6 IBA rods. This phenomenon can be seen in figure 9 at 48 GWd/MTU, figure 16 at 40 GWd/MTU, and figure 17 at 56 GWd/MTU. Due to uncertainties caused by the model and MCNP methods more of these instances could have happened.

A possible explanation for this is the radial heterogeneities in the neutron field and the isotopic composition of the material. Since neutrons coming from the outside will more likely be captured by the gadolinium than fissioned in ^{235}U , the neutron flux will be strongly suppressed near the pellet periphery. This will cause the gadolinium in the periphery to burn out faster and remain longer in the centre. The radial burnup effect will also cause ^{235}U to burn out non-uniformly. All of this will affect local reactivity in the pellet, isotopic evolution, and cross sections used in the next burnup steps.

For the simulations used in this study, the IBA fuel pellets were treated as a homogeneous region. A consequence of this is that MCNP will average out and misrepresent the radial burnup effect. To check whether the use of homogeneous regions has a great impact on Δk_{inf} , new simulations were conducted for scenarios with 6, 12 and 24 IBA rods, using 5% enrichment of ^{235}U and 8% gadolinia content. Only these three were chosen because the results already show what was needed for this study. Consideration for the radial burnup effect is done by modelling the IBA fuel pellet using radial segmentation. Radial segmentation involves dividing the volume of the IBA fuel pellet into shells with equal volume. Westinghouse Electric Sweden uses either five, eight, or 10 zones depending on the detail needed. The IBA fuel pellet was therefore divided into 5 segments that can be seen in Figure 18 below. The simulation was then conducted with MCNP as usual, and the results can be seen in figure 19.

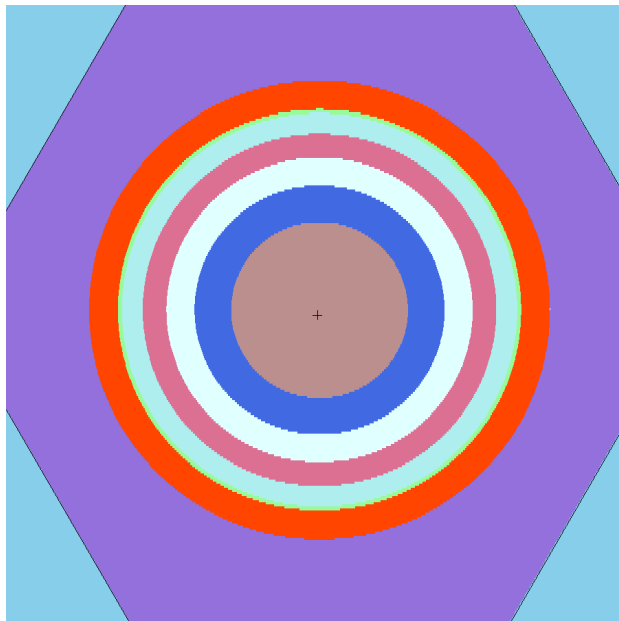


Figure 18: IBA fuel rod seen from the xy-plane. The fuel pellet is divided into 5 segments with equal volume.

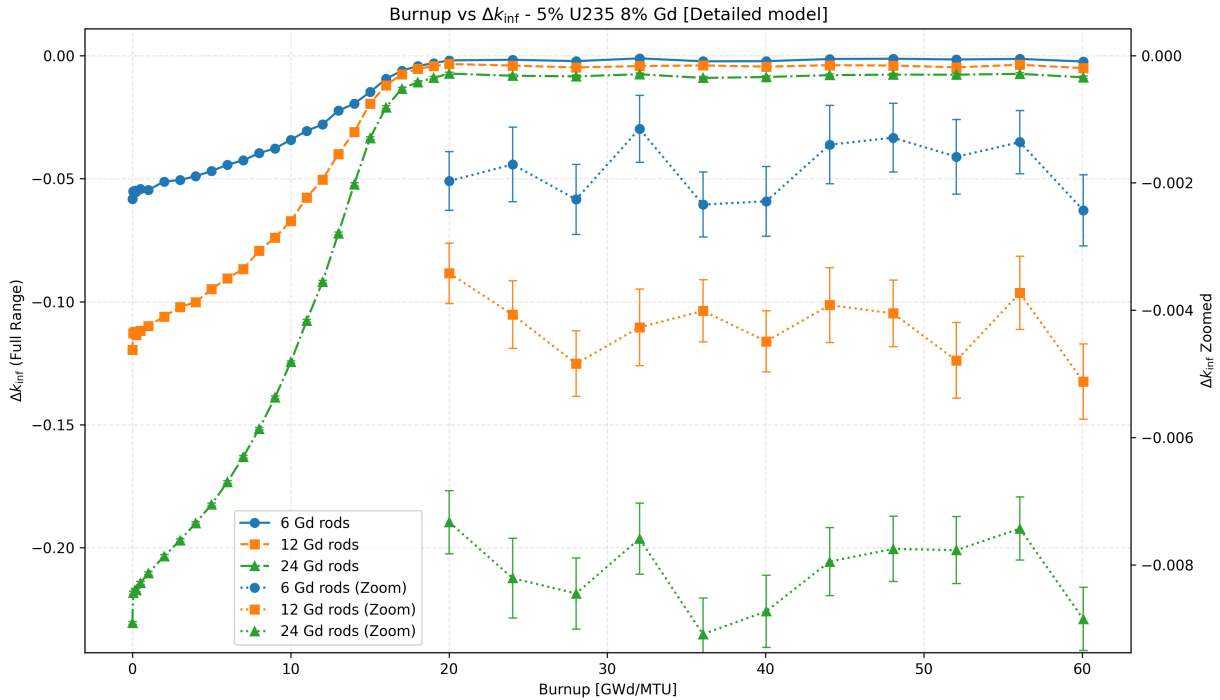


Figure 19: Δk_{inf} as a function of burnup for the more detailed FA with 5% enrichment of ^{235}U and 8% gadolinia content. After 20 GWd/MTU, the same values are plotted again but zoomed in for the sake of clarity.

Comparison of figure 19 with figure 17 in section 4.1 shows that the results of the detailed model using radial segmentation become more well-behaved. No cross-overs exist, and the lines for each scenario also more clearly exist in its own region. This better represents what should theoretically happen, as described in the previous section. However, since both the detail model and the original model have $\Delta k_{inf} < 0$ in any burnup, it does not matter which model you use to see the reactivity effects and behaviour. Consequently, for studies that require considerable detail, it is better to use models that utilize radial segmentation from the start.

5.3 Uncertainties and sources of error

In addition to the aforementioned residual reactivity suppression and the radial burnup effect, there are several uncertainties and sources of error when simulating the isotopic depletion of an FA. One common uncertainty is the inherent statistical uncertainty of Monte Carlo methods. The random walk process requires many iterations and histories to produce precise results. Although the simulations used in this study have a large amount of both, there still exists some small statistical uncertainty as shown by the error bars in the figures. This does not affect the results of the study but explains some of the zig-zag behavior in the figures.

Another source of error are the IBA rod placements used for the FA designs. In this study, no consideration for the many different configurations that the IBAs could be arranged was taken. The placement can affect local neutron flux and burnup distributions which are difficult to predict before simulation. In fact, other configurations may have a higher reactivity than those chosen. Whether this will have a considerable effect on the results is hard to say but the residual reactivity suppression of gadolinium will still exist, theoretically resulting

in $\Delta k_{inf} < 0$.

5.4 Future studies

There are several future studies that can be conducted based on the content and findings of this study. This thesis studies the reactivity effects of a VVER-1000 fuel with different configurations of gadolinia IBA rods. Firstly, VVERs and PWRs are not the only reactor types that use burnable absorbers. For example, BWRs also use BAs to extend the fuel life time in the reactor but since they work and are designed differently a separate BUC analysis has to be conducted for these reactors.

Another future study would be to investigate the reactivity effects of the different types of BA rods. Both in a PWR and BWR type reactor. As previously mentioned, there are three different types of BA: IBA (integral burnable absorber), IFBA (integral fuel burnable absorber), and BPR (burnable poison rods). Different materials are also used in these BAs. For example, other commonly used materials for IBAs are Er_2O_3 or $\text{Al}_2\text{O}_3\text{-B}_4\text{C}$. Thus, many combinations of reactor, BA type and material can be investigated. However, it is still important to consider the in-core conditions and geometry of the types of reactors using different configurations of BAs.

Lastly, isotopic depletion represents only the first step in establishing a licensing basis analysis for spent fuel management. After depletion, the isotopic composition and residual reactivity of an FA will be input into a criticality safety analysis for transportation and storage systems. One of the primary advantages of using BUC is the potential for increased packing efficiency. Optimized space usage in storage and transport casks can lead to substantial economic benefits by reducing the number of required casks and reducing material, fabrication, and logistic costs. However, quantification of these benefits would require an extensive assessment. Hence, a future study would be to conduct an economic feasibility study comparing designs including BUC with current industry standards. Such an analysis would require comprehensive modelling and simulation of storage and transport configurations along with cost analysis for manufacturing, licensing, infrastructure, and logistics.

6 Conclusions

This study investigated the application of BUC methodology for PWR fuels using gadolinia-based IBAs. Through 18 depletion simulations for an FA with different configurations of ^{235}U enrichment, gadolinia content, and number of IBA rods, the results consistently showed that the presence of IBAs reduces reactivity compared to reference cases without IBAs. Meaning Δk_{inf} was always negative during the entire burnup time and was a result of the residual reactivity suppression of gadolinium.

A higher amount of IBA rods and or a higher gadolinia content would increase $|\Delta k_{inf}|$ in the end of FA burnup. Thus, further analysis for more scenarios would be redundant. Recommended actions for a licencing basis analysis for a PWR fuel using gadolinia-based IBAs would be to neglect the existence of IBAs entirely. This assumption would ensure reasonable conservatism, reduce complexity, and save licencing process times.

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