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LEU+ Pellets in Westinghouse Electric Sweden’s nuclear fuel factory

Criticality Safety Analysis of the nuclear fuel factory in Västerås

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Abstract

Today the maximum allowed enrichment in uranium pellets created at Westinghouse Electric Sweden's (WSE) nuclear fuel factory is 5%. The market is currently moving towards higher enrichment in its nuclear fuel for its benefits with energy security, such as higher energy density, and economics. The aim with this project is to investigate the possibility of using Westinghouse Electric Company's nuclear fuel pellets LEU+ ADOPT produced in Springfield, United Kingdom, with a weight enrichment of up to 8% safely in WSE's factory in Västerås. These pellets would be used to produce fuel assemblies with higher enrichments compared to production of today.

The analysis will be made using Monte-Carlo N-Particle Transport Code (MCNP) version 6.2. MCNP is a Monte-Carlo based program that can calculate values with the use of mathematical equations to model systems in the factory. The value calculated in this project is the neutron multiplication factor k_{eff}. With the k_{eff} value a conclusion can be made whether the system will have an uncontrolled chain fission reaction or not during an accident scenario. If the k_{eff} value is too high, modifications to the model are needed to decrease the k_{eff} value to a safe level.

In this project three different accident scenarios were analyzed, and all were unsafe with the increased uranium in the pellets before modifications were made. However, with modifications to the systems by reducing conservative margins and introducing neutron absorbers to the system a critically safe model was achieved.

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Populärvetenskaplig sammanfattning

I dagens kärnkraftverk används upp till 5% av det uran som finns i bränslet. Detta är för att det endast finns 5% uran 235 i bränslet, det vill säga bränslet är anrikat till 5%. Uran 235 är den typen av uran som genomgår en fissionsprocess, atomer som klyvs, för att utvinna energi. Kärnbränslet som tillverkas i Westinghouse Electrics fabrik i Västerås är kutsar, som små träpellets. Dessa kutsar läggs sedan i rör som svetsas ihop och dessa rör placeras sedan i knippen med mellan 100–400 bränslestavar i varje knippe. Det är dessa knippen som placeras i en reaktor och används som bränsle.

Att öka anrikningen i bränslet medför ökad energitäthet i bränslet och kan därmed vara i reaktorn under längre tid, eller så kan färre byten av bränsleknippen ske vid varje byte. Detta gör att en reaktor kan producera mer energi på mindre bränsle. Problemet med den ökade anrikningen är att detta bränsle har lättare att upprätthålla en kedjereaktion som ökar radioaktiviteten vilket ökar riskerna vid konstruktion och hantering av bränslet. Denna rapport undersöker därför om det med säkerhet skulle kunna gå att använda prefabricerade uranpellets med en anrikning på 8% i Westinghouse Electric Sveriges bränslefabrik i Västerås.

Med mindre modifieringar av de existerande modeller som finns av de system som hanterar uran i fabriken skulle det gå att hantera uran anrikat upp till 8%. Detta gjordes genom att ta bort överflödiga konservativa antaganden samt genom att introducera neutronabsorbatorer till systemet.

För att undersöka huruvida ett system är säkert eller inte används något som kallas för kriticitetssäkerhetsanalys (KSA). KSA består av att bygga upp en modell av systemet som ska undersökas i ett modelleringsprogram som heter Monte-Carlo N-Particle Transport Code (MCNP). MCNP är en kod som använder sig av Monte-Carlo beräkningar för att ta fram approximativa värden från den matematiska modell som byggts. I denna rapport är det neutronmultiplikationsfaktorn k_{eff} som undersöks. k_{eff} är ett mått på hur många neutroner som orsakar fission i en kedjereaktion mellan varje generation. Om k_{eff} är 1 kallas det för att systemet är kritiskt och kedjereaktionerna hålls konstanta. Om k_{eff} är över 1 kallas det att systemet är överkritiskt och kedjereaktionerna kommer att skena och stora mängder energi och strålning kommer produceras. Om k_{eff} är under 1 kallas det för att systemet är underkritiskt och kedjereaktionen kommer att dö ut till slut. I en bränslefabrik får inte k_{eff} överstiga vissa gränsvärden för att säkerställa att det inte kan upprätthållas en kedjereaktion i något av systemen även under olyckshändelser.

I denna rapport har tre teoretiska olycksfall analyserats. Det första teoretiska olycksfallet var när en vagn fylld med bränslestavar vältes över ett färdigt bränsleknippe i snö. Det andra teoretiska olycksfallet var när delar av bränslestavsförådet hade blivit fyllt med vatten. Det tredje teoretiska olycksfallen var när hela bränslestavsförrådet hade fyllts med vatten. Ingen modell av dessa olycksfall klarade att hålla sig under de gränsvärden som finns uppsatta om 8% anrikat uran användes istället för 5%. Detta lyckades dock ändras genom att delar av modellerna som var överdrivna togs bort för att representera verkligheten mer. Till exempel att det var för många bränslestavar i en modell jämfört med verkligheten. Det sattes även in neutronabsorbatorer i systemen som absorberade neutroner som annars skulle kunna skapa nya fissionreaktioner och därmed sänkte k_{eff}.

I de teoretiska olycksfallen där positioneringen av neutronabsorbatorer inte kunde garanteras, till exempel när stavar faller över ett golv och landar slumpmässigt, antas positioneringen av neutronabsorbatorerna hamna i en position som resulterar i det högsta k_{eff}-värdet. Däremot i de teoretiska olycksfallen där positioneringen av neutronabsorbatorerna går att bestämma, till exempel om de sitter fast, antogs positioneringen resultera i ett så lågt k_{eff}-värde som möjligt.

Dessa modifieringar resulterade i att i de teoretiska olycksfall som har undersökts i denna rapport minskade k_{eff}-värdet till nivåer något lägre än i de nuvarande modellerna.

- **I fallet med den välta vagnen** ökade k_{eff}-värdet med 8,5% när 8% anrikning användes i originalmodellen. Efter att modifikationerna gjordes, till exempel användning av neutron-absorbatorer, sjönk k_{eff}-värdet till 2,12% under originalmodellen.
- **I det delvis vattenfyllda stavförrådet** ökade k_{eff}-värdet med 8,5% när 8% anrikning användes i originalmodellen, likt det tidigare fallet. Efter att modifieringarna gjordes på denna modell hamnade k_{eff}-värdet något under originalvärdet med en minskning på 0,76%
- **I det helt vattenfyllda stavförrådet** ökade k_{eff}-värdet först med 8,4% men sjönk sedan till 4,62% under originalvärdet efter modifieringarna.

I och med att originalvärdet för samtliga teoretiska olycksfall är säkra idag innebär det att de nya modellerna också skulle kunna anses som säkra.

Från dessa resultat kan därför slutsatsen göras att för dessa system och för dessa teoretiska olycksfall kan systemen säkert hantera uran med 8% anrikning. Dock bör det noteras att det är stora delar av fabriken som ännu inte har analyserats samt teoretiska olycksfall som behöver undersökas ytterligare innan en slutsats kan dras för hela fabriken.

Executive summary

In this report the possibility of importing prefabricated LEU+ ADOPT pellets with an enrichment up to 8% to Westinghouse Electric Sweden's nuclear fuel factory in Västerås have been analyzed. An analysis of two different systems with a total of three different accident scenarios have been performed with promising results. With minor changes to the systems and the operation, the use of LEU+ ADOPT pellets would be possible in the factory. If the entire production chain, from pellet to assembly, is analyzed and proven to be critically safe with these increased enrichment levels the factory could produce fuel for the reactors of the future.

The market is currently moving towards higher enrichment in its nuclear fuel. If Westinghouse Electric Sweden could produce that type of fuel a possibility to capitalize on a new and growing market before any competitors, it could be hugely beneficial for the growth of the company. Therefore, if further analysis in this project would be possible, I believe Westinghouse Electric Sweden would prosper.

Abbreviations

BWR	Boiling Water Reactor
eV	Electron Volts
GNF	Global Nuclear Fuel – Americas
HALEU	High-Assay Low-Enriched Uranium
HEU	High-Enriched Uranium
IHECSBE	International Handbook of Evaluated Criticality Safety Benchmark Experiments
JCO	Japan Nuclear Fuel Conversion Co
k_{eff}	Neutron Multiplication factor
LEU	Low-Enriched Uranium
LEU+	Low-Enriched Uranium +
MCNP	Monte Carlo N-Particle Transport Code
mSv	Millisieverts
PVP	Polyvinylpyrrolidone
PWR	Pressurized Water Reactor
U_{235}	Uranium 235
UO_2	Uranium dioxide
WEC	Westinghouse Electric Company
WSE	Westinghouse Electric Sweden

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1. Introduction

Nuclear power plays a large roll in the electricity production of the world. In the year 2022 nuclear energy produced 9% of the world's total electricity production (World Nuclear Association, 2024). In countries with a larger amount of nuclear in its electricity mix, such as Sweden and France, nuclear produce 29.1% and 64.2% respectively (IEA, n.d.a) (IEA, n.d.b). In today's nuclear reactors the weight-enrichment in nuclear fuel is between 3 and 5% (U.S. Department of Energy, u.d.). However, the advantage of having a higher enrichment in the fuel is generating an interest for this type of fuel compared to lower enrichments. This can be seen in how different actors have started to produce uranium pellets with a higher enrichment than is currently used. For example, Global Nuclear Fuel – Americas (GNF) has been approved to produce nuclear fuel with enrichment up to 8% (Patel, 2024). Westinghouse has also produced their first production of uranium pellets, called LEU+ ADOPT™, with higher enrichment in august of 2024 at their nuclear fuel factory in Springfields, United Kingdom (Westinghouse Electric Nuclear, 2024). This indicates that the market for higher enrichment is growing.

The enrichment limit for production of nuclear fuel at Westinghouse Electric Sweden (WSE) nuclear fuel factory is 5% of Uranium isotope 235 (U_{235}). Since the market for higher enrichment in fuel is growing the limit of 5% is on the verge of being too low. Therefore, the possibility of producing nuclear fuel with a higher enrichment would be interesting for WSE. It is this interest that made this project possible.

The type of fuel used in reactors today is called Low-Enriched Uranium (LEU). LEU is maximum 5% weight-enrichment of U_{235}, while High-Enriched Uranium (HEU) has > 20% weight-enrichment of U_{235} (IAEA, 2022) (U.S. NRC, 2021 b). When enrichment increase of nuclear fuel is mentioned, it is in the range of 5-20% weight-enrichment (U.S. NRC, 2021 a). NRC is referring to this type of uranium as High-Assay Low-Enriched Uranium (HALEU) while others refer to it as LEU+. LEU+ will be the term used in this report to refer to pellets with this enrichment-level as this is the most common term when referring to higher enrichment fuel for existing reactors.

This project will investigate potential issues of criticality safety that may arise at WSE's fuel manufacturing site in Västerås due to the hypothetical use of uranium pellets with an enrichment up to 8%. These pellets would theoretically be imported from WSE's fuel factory in Springfields, United Kingdom. This can possibly create problems since the factory currently only have criticality safety analyses for maximum 5% enrichment. Criticality safety analysis is an analysis of the neutron multiplication factor k_{eff}. The k_{eff} value indicates the multiplication of neutrons that can cause another fission reaction and potentially create a critical chain reaction. K_{eff} has specific regulations set for the maximum value for a system to avoid accidents. This factor and the regulations set for each system is described in greater detail in section 2.3. Criticality safety. This project will be accomplished in the following steps:

1. Mapping the route of the imported pellets
2. Summarizing the current k_{eff} of each system
3. Perform a new criticality safety analysis with the program Monte Carlo N-Particle Transport code (MCNP) with the new LEU+ pellets. Analyze the new k_{eff} value and investigate the measures needed to decrease the k_{eff} value to be within the limitations in the systems where k_{eff} exceeds the regulations.

To limit the size of this project only the systems with the highest k_{eff} from the previous criticality safety analyses will be analyzed. This is to check if the systems can be modified to handle the enrichment-levels of LEU+ pellets. If the system cannot handle the enrichment-levels, the model needs to be overseen whether it is an accurate representation of the reality or if there are excessive conservative assumptions, that result in higher k_{eff} than reality, in the model. The model therefore needs to be modified and if the system still cannot handle LEU+ pellets, then the system would need to be modified.

1.2 Aim and research questions

The aim of this project is to map the route through WSE's fuel factory for imported uranium pellets and to check the possibility of using LEU+ pellets with an enrichment of up to 8% in WSE fuel factory. This would open the possibility to create higher enriched nuclear fuel assemblies and therefore WSE could satisfy parts of the markets need for higher enriched nuclear fuel.

1.2.1 Research questions

- Is it possible to safely produce nuclear fuel assemblies with imported LEU+ pellets in WSE's factory in Västerås?
- How would the LEU+ pellets travel through the factory if they were to be imported from an external pellet factory?
- Which system has the highest k_{eff} value?
- How can systems that have a k_{eff} value exceeding the regulatory limits be modified to lower the k_{eff} value to be within those limits.

1.2.2 Limitations in the scope

- Highest k_{eff}
  Only the systems and the accident scenarios with the highest k_{eff} value will be analyzed.
- Fuel types
  There are different types of nuclear fuel, for example MOX-fuel. However, in this project the only fuel that will be analyzed is Uranium dioxide fuel (UO_2-fuel) with enrichment up to 8%. This is because UO_2-fuel is the only type of fuel produced at WSE's fuel factory in Västerås.
- Economy
  This project will not have an economic point of view.

- Completed pellets
  Uranium is currently delivered to WSE as uranium hexafluoride, and is being converted into uranium dioxide, UO_2, powder to pellets. However, in this project the uranium will be delivered as completed uranium pellets and therefore excluding all the converting and pellet production.
- No recycling of pellets
  If a pellet is damaged or is not up to quality standard for delivery, the pellet is assumed to be returned to the factory in Springfield. This limitation is to exclude the recycling systems in the factory from the analysis.
- No transportation containers
  To reduce the size of this project. Only the systems in the factory and not the transportation containers will be analyzed in this project.
- Only preexisting MCNP models
  Only systems with preexisting models will be analyzed in this project to reduce the time it would take to convert a model from another simulation program to MCNP.

2. Background

2.1. Westinghouse Electric Sweden’s nuclear fuel factory

Westinghouse Electric Sweden’s nuclear fuel factory is located at Bränslegatan 1 in Västerås. This fuel factory is one of three nuclear fuel factories that Westinghouse Electric Company, WEC, owns and is the only one in the Nordics (Westinghouse Electric nuclear, n.d.a). At this facility there are different products produced. These products are:

- Boiling Water Reactor (BWR) fuel

There are different types of BWR fuel assemblies, and the products being produced at the factory is Svea-96 Optima 3 and Triton 11 (Westinghouse Electric Nuclear, n.d.b). Triton 11 is the newest product of the two and is the largest BWR fuel currently being produced at the factory.

- Pressurized Water Reactor (PWR) fuel

At the factory there are several different types of PWR fuel assemblies. WSE produces 16x16 and 18x18 fuel assemblies for the German designed KWU-reactor and 17x17 assemblies fuel for the European style of reactor (Westinghouse Electric Nuclear, n.d.c). The PWR fuel assemblies is the largest fuel assemblies by volume being produced at the factory.

- Vodo- Vodyanoi Energetichesky Reactor (VVER) fuel

VVER is a Russian reactor design where the fuel assemblies are hexagonal compared to the square fuel assemblies in BWR and PWR (Westinghouse Electric Nuclear, n.d.g). WSE produces two different types of VVER fuel assemblies, VVER1000 – fuel assemblies and VVER400 – fuel assemblies. These two fuel assemblies are created for VVER1000-reactors and VVER400-reactors respectively.

- Uranium products

WSE produces different types of uranium products where the most common product is enriched uranium powder. However, uranium granulate and pellets is also products WSE sells (Westinghouse Electric Nuclear, n.d.f).

- Control rods

The final product WSE produces at the factory is control rods for BWR reactors with them model CR 99 (Westinghouse Electric Nuclear, n.d.d).

These products are delivered to reactors all over the world (Westinghouse Electric Nuclear, n.d.c). With the products produced by WSE the total amount of carbon dioxide emissions is reduced by 200 million tons every year compared to if the same amount of energy were produced by coal plants (Westinghouse Electric Nuclear, n.d.e).

2.2. Elementary fission physics and criticality

The focus of this report is criticality safety analysis. The fundamental principle behind criticality is nuclear fission. Nuclear fission is the event when a large atom splits into two or three smaller atoms and in this process, energy is released (Bodansky, 2004, pp. 414-146). This reaction is instigated by a neutron hitting the core of a large atom to make the core split into two or three smaller atoms called fission products. In this split not only are fission products released but also neutrons and gamma radiation. The number of neutrons differs between which atom is split, on average between two or three neutrons when U_{235} is split. The steps of the nuclear fission process can be seen in Figure 1. In this report the focus will be on U_{235} since it is the only fissile material used for nuclear fuel at Westinghouse Electric's factory in Västerås.

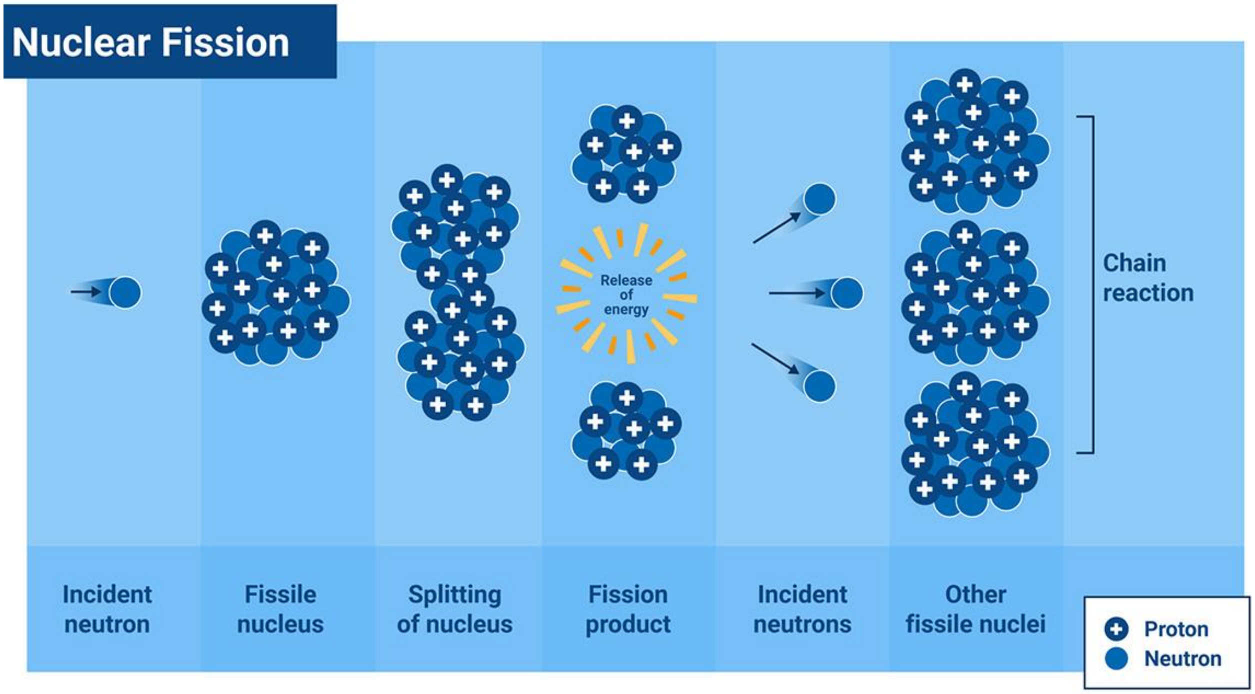

Figure 1. Fission process (Galindo, 2022).

The likelihood of a neutron hitting a nucleus and a reaction occurring is called cross section. Cross section is measured in the unit barn, b. There are different types of cross sections, the one we are mainly going to focus on is absorption cross section. The absorption cross section is the probability of a neutron being absorbed into the nucleus of an atom (Busch, 2023, pp. 32-33). This process can either lead to neutron capture or fission. Neutron capture removes the number of free neutrons in a system. The absorption of neutrons that do not cause a fission reaction is lost when a material absorbs a neutron. Materials that are especially good neutron absorbers are called neutron absorbers are introduced to a system. One type of neutron absorber is boron-10. The cross section for neutron capture for boron-10 can be seen in Figure 2.

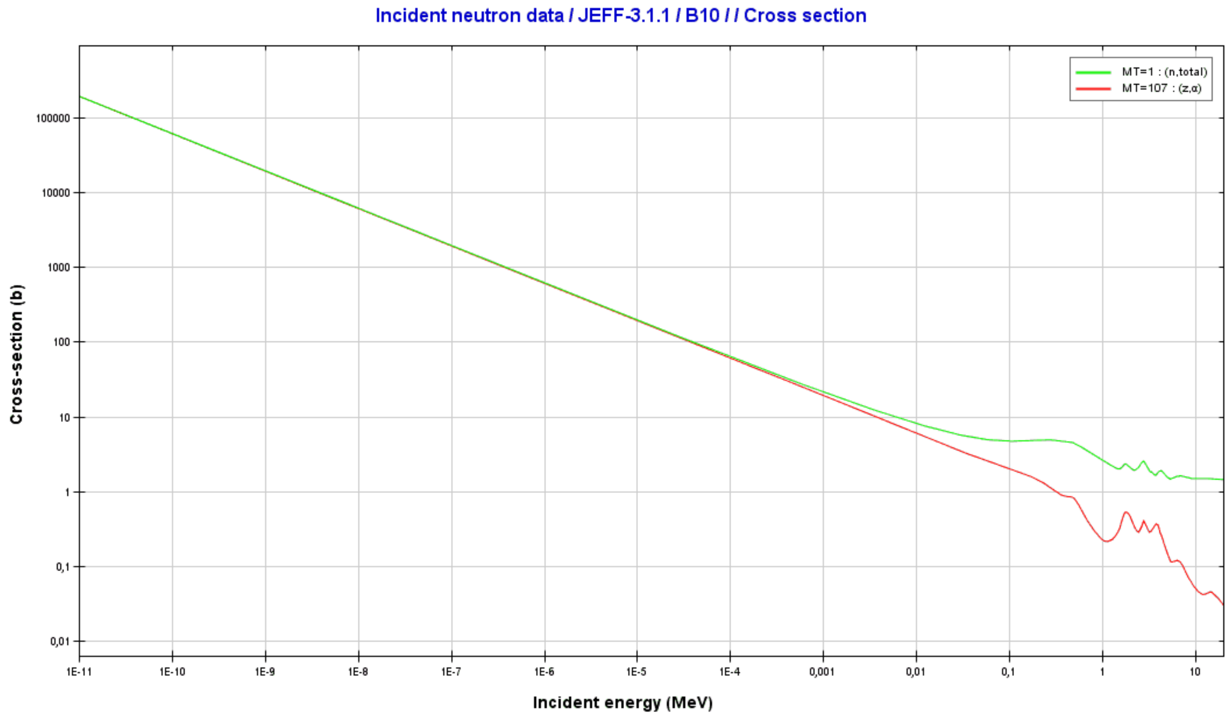

Figure 2. The cross section for boron-10. The red line is the absorption cross section, and the green line is the total cross section (Nuclear Power, n.d. a).

With a decrease of incident energy, the energy of the incoming neutron, the likelihood of an interaction occurring is increasing. The cross section of a neutron being absorbed into the nucleus of an atom and causing a fission reaction is called fission cross section (Busch, 2023, pp. 32-33). The fission cross section is the probability of a neutron hitting a nucleus and causing the nucleus to fragment into mainly two or sometimes three smaller nuclei and releasing energy (Bodansky, 2004, p. 145). For U_{235} the fission cross section is the largest for slower neutrons, neutrons with energies around 0.025 electron volts or eV, see Figure 3. These neutrons are called thermal neutrons.

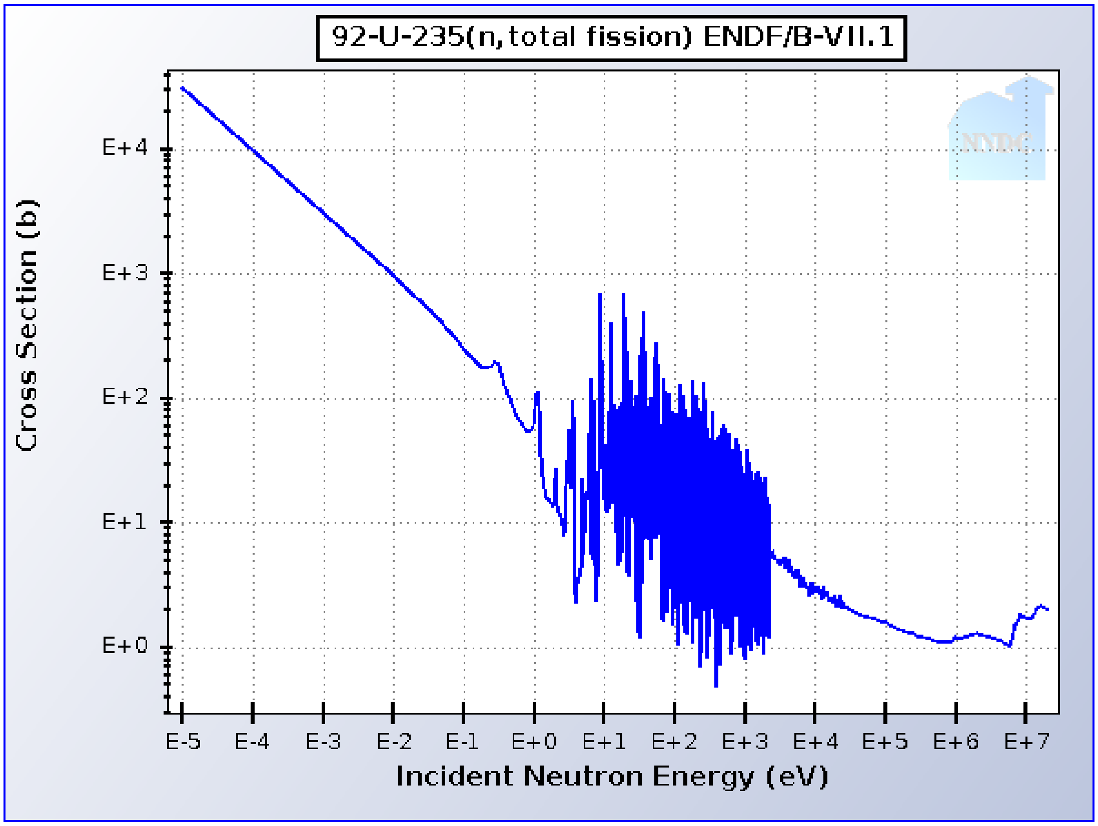

Figure 3. Fission cross section for U_{235} *(NNDC, u.d.)*

After all fission reactions, new particles are created. These particles are two or three daughter nuclei and neutrons (Bodansky, 2004, pp. 144-146). The term daughter nucleus means that two or three new nuclei are created where their total amount of protons sum up to the same number of protons as in the mother nuclei. Their total amount of neutrons can vary due to the number of neutrons emitted in the fission process. The probability of a specific nuclei being created in this process is called the fission yield for that nucleus. In this report independent fission yield is used. Independent fission yield is the nuclei being created directly after the fission reaction (JAEA, 2013). In the process of creating nuclei there is usually one of the daughter nuclei that is heavier than the other. The probability of a specific atom being created from the fission process can be seen in Figure 4.

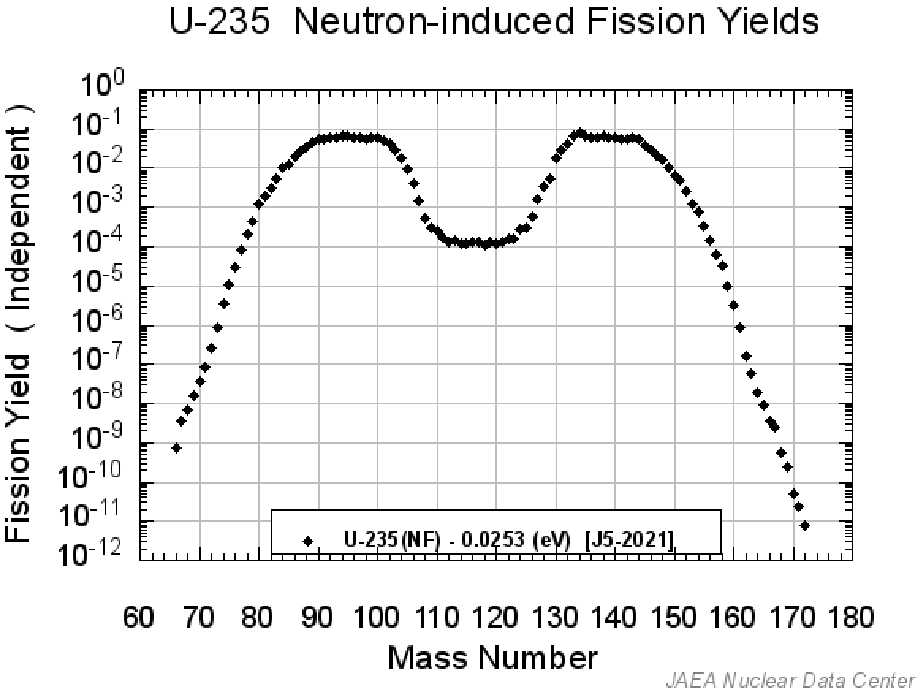

Figure 4. Plot of the fission yield for U_{235} (JAEA, 2023)

In Figure 4 it can be observed that the fission yield for atoms between the mass number 80-105 and 125-155 are greater than the rest of nuclei. This indicates that the likelihood of two daughter nuclei being created with one heavier and one lighter atom is the most likely outcome.

When U_{235} splits on average 2.43 neutrons released for every fission reaction. If all those neutrons were to create a new fission reaction the number of fissions would grow exponentially. However, that is not the case, as some of the neutrons gets lost on the way to a new fission reaction. This could be due to neutrons either being absorbed in materials and not cause fission or escaping out of the system and disappearing

2.3. Criticality safety

Criticality safety analysis is the analysis of a system to ensure that the system does not accidentally start an uncontrolled chain reaction of fission reactions (Busch, 2023, pp. 9-10). The chain reaction is measured by the neutron multiplication factor k_{eff}. K_{eff} can be calculated in different ways, and it is the proportion of neutrons that cause a new fission reaction compared to the number of neutrons that cause a fission reaction in the previous generation (Busch, 2023, p. 31). K_{eff} indicates if a system is stable, growing or decreasing in the number of fission reactions. If $k_{eff} > 1$ the system is supercritical (Bodansky, 2004, p. 153). This means that the amount of fission reactions will increase exponentially as time goes on. This is desirable during the start of a nuclear reactor to increase the energy produced in the reactor up to a certain level. After that level of energy is reached a $k_{eff} = 1$ is desirable. $K_{eff} = 1$ means that the system is critical, and the chain reaction can maintain itself in a steady state and will not increase or decrease. If $k_{eff} < 1$ the system is subcritical. This means that the fission reactions will decrease over time and eventually fade out completely. To calculate k_{eff} there is a formula called the six-factor formula constructed from the following equation (Nuclear Power, n.d.d):

$$k_{eff} = \eta * f * p * \varepsilon * P_f * P_t \qquad (k_{eff})$$

Where k_{eff} is the multiplication factor, η is the number of neutrons produced in each fission reaction, *f* is the fraction of neutrons captured in the resonance frequencies (This can be observed for neutron energy levels between 10^0 to 10^3 eV in Figure 3), p is the fraction of neutrons that avoid being captured in other nuclei and reach thermal energies, ε is the fraction that contributes new neutrons from fast neutron fission reactions, P_f is the probability that fast neutrons will stay inside the analyzed system and not leak out to the surroundings and P_t is the probability that thermal neutrons will stay inside the analyzed system and not leak out to the surroundings. In this report P_f and P_t is going to be combined to the total non-leakage factor P_{NL}, resulting in the five-factor formula (Nuclear Power, n.d.b).

To decrease k_{eff} you need to lower the value in any of the factors in the five-factor formula. The factors that will be the focus in this report is p and P_{NL}. To decrease p, neutron absorbers can be included into the system. A neutron absorber is a material that have a large cross section for neutron absorption without it leading to a fission reaction (U.S.NRC, 2021 c). One example of this is boron, B, and gadolinium, Gd (Murgatroyd & Kelly, 1977, p. 5). Boron-10, B_{10}, has a cross section for thermal neutrons at 3 813 barns while gadolinium-157, Gd_{157}, has a cross section of ~240 000 barns. In contrast tungsten-183 has a cross section of 11 barns. If B_{10} is included in a system the k_{eff} is therefore lower since the B_{10} absorbs some of the neutrons before they can cause a fission, hence p is lowered. This is the reason that B is used extensively as control rods in the form of boron carbide (Murgatroyd & Kelly, 1977, p. 9). To decrease P_{NL} an increase of leakage is needed. This can be achieved by either changing the geometry or decreasing the reflectors around the container. A reflector is a material that reflects the neutrons and therefore increase the number of neutrons that does not escape the container (U.S.NRC, 2021 d). If the geometry is changed to a shape where the mantle of the container is larger compared to the volume, then P_{NL} will decrease (Nuclear Power, n.d.c).

When neutrons are released during the nuclear reaction they are released as fast neutrons. To increase the likelihood for fission the neutrons need to be slowed down to reach thermal levels. This is done by using moderators. Moderators are materials with the properties to slow down the speed of neutrons by letting them bounce on the atoms and transferring their energy to the moderators. The lighter the atom the better the moderator. The most common moderator in WSE's fuel factory is hydrogen and oxygen or specifically water. The neutrons collide with the water molecule and transfer some of their energy and continue their journey to collide with another water molecule until it reaches the thermal energy levels. When the thermal energy is reached the chance of a fission reaction to occur increases and release new neutrons.

In a nuclear fuel factory, subcritical systems are a must to reduce the chance of an uncontrolled chain reaction to occur and large amounts of radiation to leak out into the surroundings. This is ensured by doing a criticality safety analysis of each uranium containing system in the entire factory, unless they have one of the four critically safe parameters for a system. The four critically safe parameters are if the system has an insufficient mass in a spherical container, a small enough spherical container, if the system is a cylinder with a small enough diameter or a sheet with a small enough thickness. For example, a fuel rod that is infinitely long and contain a mixture of 5% enriched uranium and water can never go critical if the diameter of the cylinder is less than 22 cm since the leakage of neutrons is too large. These parameters are determined based on the enrichment of the uranium and if it is a heterogeneous or a homogeneous mixture. For a heterogeneous mixture with 5% enrichment these parameters are:

Table 1: Critically safe parameters for a heterogeneous mixture of 5% enriched uranium according to internal documentation.

Mass (kg)	10.6
Volume of a sphere (dm^3)	23.6
Diameter of infinite cylinder (cm)	22
Thickness of infinite sheet (cm)	31.2

For an isolated system with reflectors around it that exceeds all these parameters a criticality safety analysis is required. However, if there is interaction between two closely positioned systems that interaction could increase the k_{eff} value and a criticality safety analysis would therefore be required. In internal documentation a criticality safety analysis is defined as calculations to measure how close to criticality a system is in normal use, an unlikely situation and an extremely unlikely situation. For each of these situations a limit of k_{eff} is set at $k_{eff} < 0.95$ for normal use, $k_{eff} \leq 0.95$ for an unlikely situation and $k_{eff} \leq 0.98$ for an extremely unlikely situation. For a situation to be classified as an unlikely situation two of each other unconnected low frequency problems during the normal use of the system must occur. For a situation to be classified as an extremely unlikely situation, more than two of each other unconnected low frequency problems during the normal use of the system must occur. For example, the roof is collapsing due to a huge amount of snow on it, the doors have been blocked so the snow is contained in a room and a weld on a fuel rod is faulty and all the fuel pellets fall out on the floor. Each of these problems are low frequency problems and are independent from each other. This results in a combination of problems that are extremely unlikely to occur. To further ensure that the situation is extremely unlikely a probabilistic safety analysis can be performed. If the calculated k_{eff} for the system exceeds these limitations, the system must be modified to be able to handle the situations.

In a criticality safety analysis, the program MCNP (Monte Carlo N-Particle Transport Code) is used at WSE. MCNP is a program based on Monte Carlo calculations and is capable of transporting neutrons, electrons and photons (Los Alamos National Laboratory, 2024). With these calculations, MCNP can simulate different systems where fissile material is present. These systems can be a nuclear reactor, a nuclear fuel assembly, a pile of UO_2 powder, etc. It can model different geometries and can utilize different libraries for cross sections and material compositions. This program is therefore used to simulate normal use and different scenarios of accident scenarios to ensure the systems are critically safe.

When modelling for a criticality analysis there are internal regulations of cautionary principles implemented for the code. These cautionary principles are to always assume conservative assumptions. The assumptions should include the uncertainties for:

- Event uncertainty

This could be the uncertainty in the event occurring. For example, if the roof of the factory collapses there is not 100% guarantee that a wagon will fall over. A conservative assumption would therefore be to always assume that the wagon falls over.

- System uncertainty

When creating a part of a system, a certain tolerance is implemented for certain aspects. This could be the height of a uranium pellet or its density. Therefore, the conservative assumption would be to assume the height and density that would increase the k_{eff} value the most.

- Model uncertainty

When modeling a system, everything can not be included in the model. For example, it would take too much time to model every cable in a construction. Therefore, the conservative assumption would to be to create a model that result in a higher k_{eff} value than if every cable were included. For example, if there are cables hanging from the roof, a water reflector around the system would negate their impact on the system and would increase the k_{eff} more than if they were present.

- Parameter uncertainty

Some parameters in a system are not always consistent. For example, in a system there could be between 100-150 pellets on a tray. A conservative assumption would be that the tray is completely full of pellets rather than just 100 pellets.

- Uncertainties in the methods and tools

The uncertainties in the methods and tools used in calculating the criticality safety of a system are discrepancies from reality and could be quantified by comparing real life experiments to a model of that experiment. If there is a difference between the two, a bias is calculated and then added to the calculated value. This is further explained in Section 5.4.

All these uncertainties should be covered in the criticality analysis to ensure that the system is safe.

During a criticality analysis there are three steps that are required by company documentation during the analysis. The steps are a pre job brief, a self-assessment called STARK ("Stanna, Tänk, Agera, Reflektera, Kommunicera”) and after the work is completed a post work review is done. These steps are taken to make the process as safe as possible and to reflect over other possible problems during the analysis.

2.3.1. Assumptions

When simulating and modelling a system for a criticality safety analysis, the system cannot represent reality to 100%. Therefore a few assumptions must be made in the analysis. All the assumptions made must be conservative. By conservative assumptions mean that the assumptions made must result in a higher k_{eff} than if the assumptions were different. For example, if a system contains a neutron absorber and that neutron absorber is excluded from the model, more neutrons will be present in the model than reality. This will increase the k_{eff} and therefore result in a system that is closer to going critical than reality. This result is therefore conservative compared to reality.

According to internal documentation the conservative assumptions made in a criticality safety report is:

- Optimal moderation.
- Maximum reflection.
- Optimizing distance.
- Maximum amount of fissile material.
- Maximum enrichment in the system.
- Minimize or neglect non moderating construction materials.
- For construction materials between uranium and absorbers, reflection must be considered.
- Moderating construction materials can be replaced with the uranium compound in the system.
- Dispersion of fissile materials in a moderator smaller than 1 mm in diameter is considered as homogenous and for larger diameter than 1 mm is considered as heterogeneous.
- No or small consideration is taken for burnable absorbers.

All these assumptions are created to result in conservative results from the simulations. In this project these assumptions will be used in the simulations to ensure a conservative result.

2.3.2. Optimal moderation

Optimal moderation is when there is a perfect ratio of moderator to fissile material maximizing the k_{eff} as much as possible. If there would be an increase or decrease in the amount of moderator k_{eff} would be lower. If there is a system with a lower amount of moderator compared to the optimal amount, the system is under moderated, and over moderated if there is higher amount of moderator (Ougouag, et al., 2004).

In criticality safety analysis, optimal moderation is the baseline assumption. This is to ensure the system is critically safe no matter the amount of moderator in the system. To ensure optimal moderation in a system, multiple simulations are performed, for example with a system flooded with water with varying density. After all the simulations are performed, the simulation with the highest k_{eff} is selected as the system with optimal moderation.

2.3.3. Homogenous and heterogeneous modelling

There are two main ways that UO_2 is modelled in the factory. The UO_2 can either be modelled as a homogeneous mixture or a heterogeneous mixture. The difference between a homogeneous mixture and a heterogeneous mixture is the size of the uranium particles. According to internal documentation a homogeneous mixture is a mixture where the uranium particles is smaller than 0.1 mm. A heterogeneous mixture is a mixture where the uranium particles are larger than 0.1 cm and can be assumed to form lumps. An example of a heterogeneous mixture is a sintered uranium pellet in a bucket of water. The pellet is not mixed with water and therefore cannot moderate the neutrons in the pellet. However, it can moderate the neutrons around the pellet. In this report the focus will be on heterogeneous models.

2.3.4. Parameters that can affect criticality safety

When analyzing parameters that can affect the physical aspect of criticality safety MAGIC MERV is usually analyzed (Busch, 2023, pp. 382-383). MAGIC MERV is an acronym for the parameters. The letters stand for:

- **M**ass

The mass of the fissile material affects the production of neutrons (Busch, 2023, pp. 365-368). The increase of mass for the fissile material increases the rate of fission reactions that is possible and therefore the number of neutrons produced. The opposite is true if the mass is decreased.

- **A**bsorption (or poison)

Absorption affects the loss of neutrons from the chain reaction (Busch, 2023, pp. 368-377). If there is an increased neutron absorption in the system, the number of neutrons will decrease and k_{eff} will therefore decrease.

- **G**eometry (shape)

Geometry affects the leakage of neutrons from the system (Busch, 2023, pp. 377-380). If the geometry is correct, the volume of the system does not matter since the leakage is so large the neutrons will never reach enough fissile material to be able to sustain a chain reaction. As previously mentioned, the specific geometries that prevent criticality is shown in Table 1.

- **I**nteraction (Separation of multiple systems)

The interactions between multiple systems affects the neutron leakage (Busch, 2023, pp. 377-380). If another uranium containing system is placed near the analyzed system neutrons from that system that have escaped can reach this system. This will increase the number of neutrons in the system and therefore increase the k_{eff} value.

- **C**oncentration (and density)

Concentration affects both the production and the absorption of neutrons in the system (Busch, 2023, pp. 368-377). The concentration of fissile material in a system could both increase and decrease k_{eff} for the system. If the fissile material is mixed with a moderator at optimal moderation, a change of concentration of fissile material will affect k_{eff}.

- **M**oderation

Moderation affects the neutron absorption of the system (Busch, 2023, pp. 368-377). For U_{235} and Boron-10 the cross section increases if the neutron energy decreases. Therefore, if a moderator is introduced the energies of the neutrons will decrease and absorption will increase. This could either remove the number of neutrons from the system if there is a lot of neutron absorbers or increase the number of neutrons from the increase in fission reactions.

- **E**nrichment (or assay)

The increase in enrichment increases the fissile material in the system and that in turn increases the neutron production in the system (Busch, 2023, pp. 365-368). The increase in neutron production will therefore increase the k_{eff} for the system.

- **R**eflection

Reflection affects the neutron leakage of the system (Busch, 2023, pp. 377-380). If there is a reflector around the system, the neutrons that leave the system can be reflected back towards the system and increasing the k_{eff}

- **V**olume

In criticality safety analysis volume is the total volume of the fissile material and it could be mixed with a liquid (Busch, 2023, pp. 365-368). A change in volume would therefore change the mass or the density of the substance which in turn would affect the production of neutrons. It would also affect the geometry of the system and therefore affect the neutron leakage.

Almost all these parameters are interconnected and if one part is changed another is also changed. For example, if the mass increase either the volume or the concentration/density changes.

2.4. Why higher enrichment

In august of 2024 the first production of LEU+ ADOPT™ pellets with an enrichment up to 8% was created at Springfield nuclear fuel factory in the United Kingdom (Westinghouse Electric Nuclear, 2024). This type of fuel is expected to be commercially available in the spring of 2025.

The advantage of a higher enrichment is mostly the increase of energy that is available in each fuel assembly (U.S.NRC, 2024). The result of this is that a fuel assembly can stay in the reactor for a longer period and therefore keep the reactor running continuously without the need for new fuel. This in turn increases the energy security from the reactor. The term for energy extracted from the fuel is called burnup. Burnup is the term to describe how much energy is extracted from nuclear fuel (Bodansky, 2004, p. 206). The unit used to describe burnup is usually mega- or gigawatt days per metric ton of heavy metals (MWd/t or GWd/t).

The average burnup has increased from 40 GWd/t to over 60 GWd/t in the last 50 years (World Nuclear Association , 2021). If the enrichment increased even further, the possible burnup would also increase as well as the operation time for the reactors. This would also decrease a reactors dependence on deliveries.

2.5. Previous incident

The importance of criticality safety analysis is to prevent accidents. One such accident scenario is the Tokaimura criticality accident scenario in 1999 (World Nuclear Association, 2020). This accident scenario occurred at a fuel preparation plant ran by JCO (formerly Japan Nuclear Fuel Conversion Co.). At this facility uranium with enrichment levels up to 20% were processed. On 30th of September there were three workers that was preparing fuel for an experimental fast breeder reactor. The fuel was enriched to 18.8% U_{235}. The three workers had no proper qualifications and little knowledge of the risks of uranium enriched to this level. Previously the workers had only worked with fuel with an enrichment of <5%.

The process that the workers worked with consisted of dissolving uranium oxide powder in nitric acid in a dissolution tank that then would go through a process of mixing before entering a precipitation tank (World Nuclear Association, 2020). This precipitation tank did not have a critically safe geometry or volume for this high enrichment. The three workers did not understand the process and therefore skipped the first steps and poured multiple buckets of uranium oxide powder and nitric acid in the precipitation tank and did a manual stir instead. Thus, a critical mass was reached, and criticality occurred. This criticality continued for 20 hours. Due to the extensive time that the criticality occurred the radiation emitted from the factory impacted other factory workers and surrounding habitants. There was also production of fission products that resulted in a continuous release of radiation even after the criticality was stopped.

Two of the three workers died due to the large dose of radiation they received, 16 – 20 and 6 – 10 greys respectively (World Nuclear Association, 2020). They both died due to acute radiation syndrome, one after 12 weeks and the other was kept alive for 7 months. The third factory worker survived after receiving 1 – 5 greys. The other factoryworkers received a larger, but not dangerous levels of radiation. None, except for the three workers in the room of the accident, received a dose over 50 mSv, the maximum annual dose allowed for nuclear workers.

3. Method

The first step in this project is to get acquainted with the layout of the factory. This is achieved by reading the necessary documentation of the factory and thereby getting an understanding of how the uranium pellets are transported through the production line. Afterwards consultation with experts in the factory for each specific area of the factory is done. After a general understanding of where the pellets would enter the factory and how they would be processed into fuel assemblies, a more in-depth mapping of each system began. In this analysis each k_{eff} of the existing systems with the current enrichment levels is documented and used as a baseline for the order of the analysis that was made in this project. To increase efficiency, the system with the highest k_{eff} is analyzed first and modified to be able to handle LEU+ pellets. This is done to check if these systems can handle a higher enrichment than 5%.

The analysis is done with the program MCNP6.2 by modifying preexisting models of each system with a different enrichment. This will then calculate the k_{eff} of the system and thereby show if the system is within the limitations for k_{eff} or if any further modifications to the model are needed. When calculating k_{eff} with MCNP a mean value for the normally distributed k_{eff} is calculated with a standard deviation (s). Therefore, to have 95% confidence interval, three standard deviations are added to the mean k_{eff} before checking if the system is within the limitations mentioned in Section 2.3. If a system is within the k_{eff} limitations the work will continue with the next system. However, if the system was not critically safe, modifications to the model were done. These modifications could be:

- Reducing excessive conservative margins.

For example, modelling the system more accurately to decrease the k_{eff}. In the current models some of the geometries in the system might be more general and therefore differ from reality. A theoretical example of this is a cylinder of optimal moderated UO_2 powder. If the cylinder is critically safe with a diameter of 23cm then it will be critically safe if the diameter is smaller. There is no need to model it for a smaller diameter if it still manages to be within the limitations. However, if the enrichment in the system is increased and the model no longer is within the limit of k_{eff}, the model might need to be altered to reflect reality better by redesigning the modelled geometries.

- Increasing neutron absorption.

The way that the increase in neutron absorption could be done is by installing boron carbide plates around the system to decrease interaction between different systems. It could also be by incorporating neutron absorbers into the system, for example putting boron shields between racks in a storage unit. This would decrease the neutron interaction between the racks and therefore reduce the k_{eff} of the system.

- Reduce the internal interaction between the uranium in the system

If the distance between the uranium products within the system is increased the interaction within the system is decreased leading to a lower k_{eff}. This could be performed by increasing the spacing within in a system, for example, increasing the spacing in the fuel assembly storage.

- Reduce the amount of uranium in a system.

If the system still is above the limit, a reduction in the amount of uranium in the system could be necessary. This could be done by, for example, reducing the amount of fuel rods on a cart. Since the amount of uranium is decreased the amount of fissile material is decreased in the system and therefore k_{eff} will be lower.

- Reducing the enrichment in the pellets.

If the system still is not critically safe another solution is to reduce the enrichment in the pellets. Since the factory is currently critically safe the new enrichment would hopefully be higher than the 5% limit of today. However, it might not be as high as the theoretical LEU+-pellets used in this project.

After the final modifications to each model, the water density was varied to check if the system is optimally moderated. An explanation of what optimal moderation is can be found in Section 2.3.2 and one example of this is the peak in Figure 11

The analysis of each system was then summarized in this report with the k_{eff} of the system and the modifications that were done to each system to achieve a k_{eff} value under the limitations.

The results were validated against to International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE). IHECSBE is a collection of practical experiments done with a measured k_{eff} value and standard deviations. Each experiment has a premade model that will be updated with the newest database for cross section and other physics parameters. To validate for enrichments up to 8% three different experiments with enrichment above 5% and below 10% were investigated and each of the experiments had 3 to 4 different variations resulting in a total of 10 different calculations. These experiments were chosen to increase the current validation to include experiments above 5% and were performed in situations similar to the factory. For example, in room temperature. These experiments will be compiled together with the preexisting validation experiments for the MCNP 6.2 program that WSE have performed. The updated list of experiments will then be used to calculate the bias (B) of the code that will be included in the results to ensure that the k_{eff} for a system is lower than the limitation.

4. Mapping the route of prefabricated pellets through the factory

Depending on how the LEU+ pellets would be delivered to the factory the starting point of this mapping is different. They can either be delivered unsorted in boxes of pellets that need to be sorted and put in transport container 6. Currently the pellets are sorted in a system that have other functions as well. However, in this project the pellets are assumed to be ready for charging when they arrive at the factory and therefore a system of sorting the pellets need to be constructed if they arrive unorganized. The possibility of the need to build a system to inspect if the pellets have been damaged in the transport should also be considered. There is currently a system for this and could therefore be reused for this purpose. From the inspection the pellets can either be put directly into the system 356 or system 357[1]. From system 357 the pellets continue the route to system 358. Here the pellets are put into the fuel rods. This system is connected to system 411-1.

From system 411-1 the fuel rod is transported to system 411-6. From system 411-6 the fuel rod goes through system 412-1. From this point the fuel rods are completed, and the next step is to either put them in system 413 – fuel rod storage or to put them together to completed fuel assemblies. Depending on what type of reactor the fuel is meant to be used in, the fuel rods can either go to system 414 or system 421.

All the transportation of the fuel rods is with system 416.

If the fuel rods are meant to be used in a BWR reactor the fuel rods first get assembled into a fuel assembly in system 414. Later the fuel assembly will be placed in system 415.

The PWR systems have been removed from this report since they have not been analyzed in this report.

The source for this mapping is from internal documentation. An illustration of the mapping can be seen in Figure 5.

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[1] The names for every system in this mapping, except those analyzed, have been removed from this report.

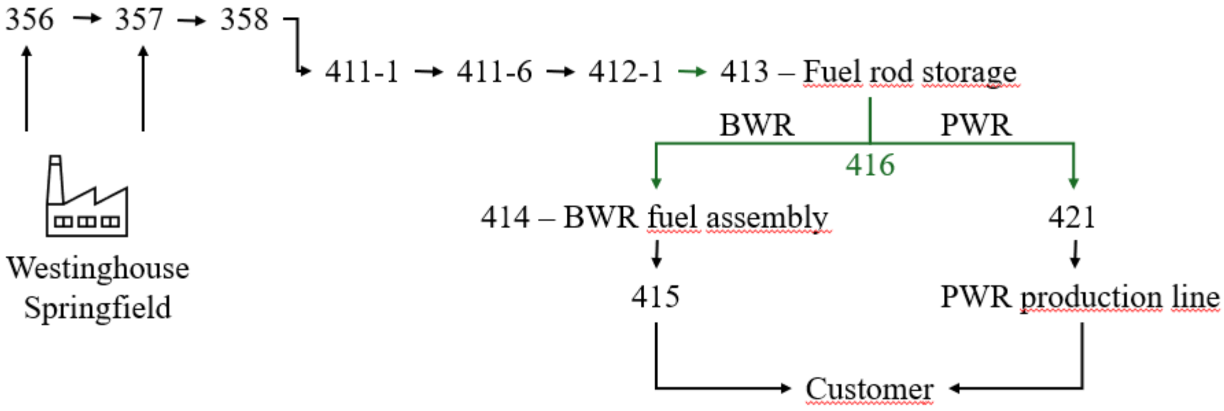

Figure 5. Pellet mapping in the factory

5. Modelling

5.1. How does MCNP work?

The modelling program used in this project is the code Monte Carlo N-Particle (MCNP), version 6.2 (Los Alamos National Laboratory, 2024). MCNP is a Monte-Carlo based program. A Monte-Carlo program uses a model of equations with random variables to calculate approximative values (Harrison, 2010). In this project the calculated value is k_{eff} and its standard deviation. For a Monte-Carlo program to be effective, a large sample size is necessary since the variables are random and it takes multiple results to be able to notice a trend or pattern. The results from MCNP in this project are normally distributed, and a standard deviation is used to determine the uncertainty of the result.

MCNP is a program-code that use an input file divided into three different parts to describe cells, surfaces and materials constructed from these materials and surfaces. These parts come together and create a 3D-model. An example of a MCNP model can be seen in Figure 6. This model is of a pellet with a diameter two and the height one and is made of water. In the materials the library for the cross section at a specific temperature for each atom is added.

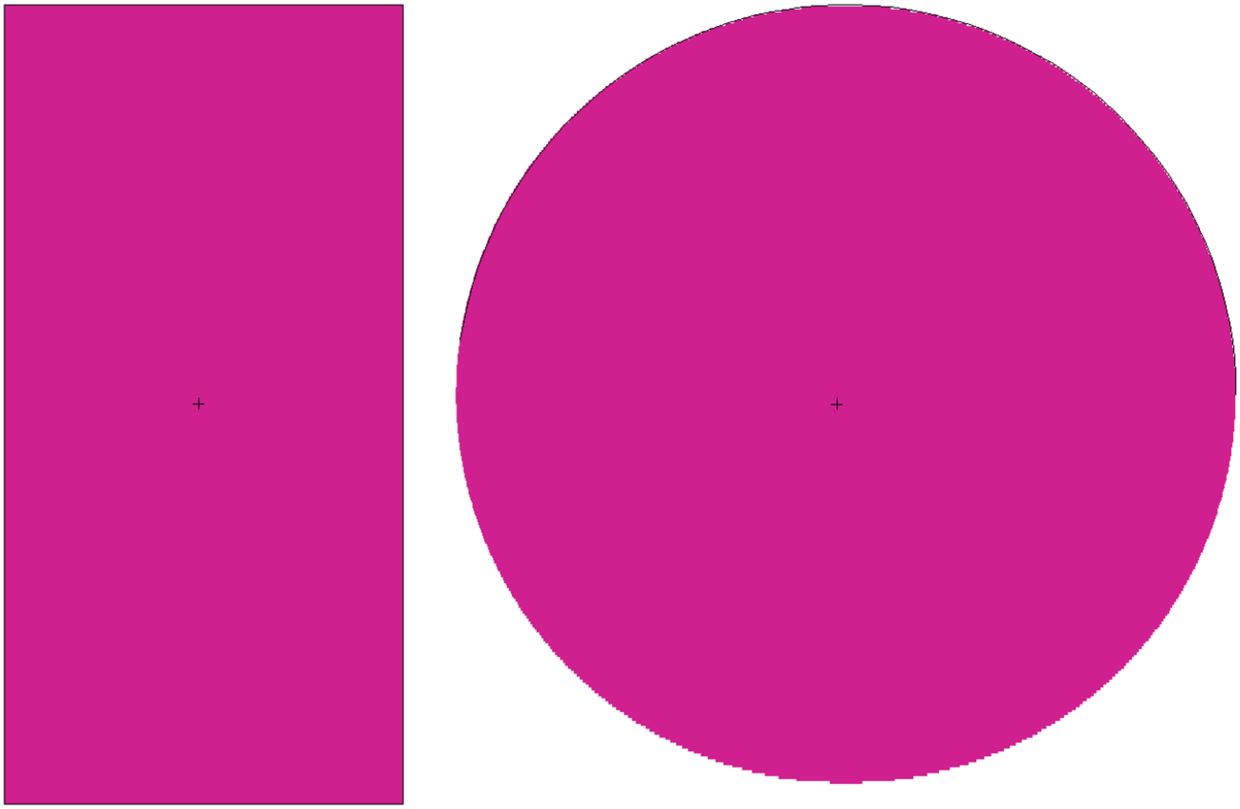

Figure 6. A pellet modeled in MCNP. Side view, left, and top view, right.

5.2. System 414

System 414 is the fuel assembling station for BWR and VVER fuel assemblies. In this report one accident scenario from this system will be analyzed both as a parameter study of the enrichment in the system and to be modified to be able to accept LEU+ pellets. The accident scenario is a cart transporting fuel rods falling over a TRITON-11 fuel assembly, where everything is covered in water with a density matching the density of snow. As mentioned in the method, to have a 95% confidence interval, three standard deviations are added to the mean k_{eff}. To include for the bias of the code B is also added to the mean k_{eff}. For this specific model an additional "punishment" is added to increase the margin to the k_{eff} limitations called R_x. This results in a k_{eff} represented by the following equation:

$$k_{eff} = k_{eff\,mean} + 3s + B + R_X$$

5.2.1. Parameter study

The parameter study is increasing the enrichment in the pellets from 5% up to 8% in increments of 0.5%. This was done to observe the effects enrichment has on the k_{eff} value. The enrichment that causes the highest k_{eff} value will be used in the following simulated accident scenarios to ensure that all enrichment levels analyzed can be used in the system.

The results can be seen in Table 4. The k_{eff} increases with increased enrichment. Therefore, for the following simulations the modeled enrichment will be the highest possible enrichment at 8%. If the k_{eff} value of the system is within the limits mentioned in section 2.2. (Elementary fission physics and criticality), all other enrichments below the modeled enrichment will therefore be safe to be used in the system.

5.2.2. Modified model

The first modification to the original model was to increase the enrichment to 8%. This increase in enrichment and no other modification to the model increased the k_{eff} value to 8.77% above the systems previous k_{eff}. Since this value is above the limit of 0.95 modifications to the model are needed.

The modifications to the model were iterated to find a model where the k_{eff} value would be within the limitations. To decrease the simulation time during the iteration, a smaller neutron sample size was used for the calculations. When the model was near the final configuration a larger neutron sample size were used to decrease the statistical uncertainty compared to the calculations with the smaller neutron sample size.

The final modifications were to reduce the number of fuel rods in the cart to represent reality and add a few boron rods to the system.

5.3. System 413

System 413 is the interim storage for fuel rods before the rods are assembled into finished fuel assemblies. In this storage both finished fuel rods and empty zircaloy tubes is stored. In the current model all the compartments are filled with finished fuel rods.

The two accident scenarios are one unlikely situation with a limit of $k_{eff} \leq 0.95$ where the storage is partly filled with water and one extremely unlikely situation with a limit of $k_{eff} \leq 0.98$ where the storage is completely water filled. in system 413 k_{eff} is calculated the same way system 414 is calculated except for R_x. This results in a k_{eff} represented by the following equation:

$$k_{eff} = k_{eff\,mean} + 3s + B$$

The Modifications to system 413 was to introduce the zircaloy tubes into the system and to add neutron absorption plates in the form of boron carbide into the system as well. This configuration was used in both the partly water filled storage and the completely water filled accident scenarios.

5.4. Validation

The experiments for this validation can be seen in Table 2.

Table 2 Experiments used in validation. the enrichment of each experiment and the number of cases taken from each experiment.

Experiment number	*Case*	*Enrichment*	$k_{eff\,mean} \pm s$ *(calculated)*	$k_{eff\,mean} \pm s$ *(benchmark)*	$k_{eff\,mean} \pm s$ *(Normalized)*
LCT 023	1	10 %	0.9964±0.0012	0.9972±0.0011	0.9992±0.0016
	2	10 %	0.9977±0.0011	0.9995±0.0011	0.9982±0.0016
	3	10 %	0.9994±0.0012	1.0013±0.0011	0.9981±0.0016
LCT 025	1	7.5 %	1.0042±0.0042	1±0.0041	1.0042±0.0058
	2	7.5 %	0.9921±0.0011	1±0.0044	0.9921±0.0045
	3	7.5 %	0.9966±0.0011	1±0.0047	0.9966±0.0048
LCT 078	1	6.9 %	0.9978±0.0002	0.99954±0.00006	0.9983±0.0002
	5	6.9 %	0.9959±0.0002	0.998±0.00006	0.9979±0.0002
	11	6.9 %	0.9981±0.0002	0.99938±0.00006	0.9987±0.0002
	15	6.9 %	0.9987±0.0002	0.99947±0.00007	0.9986±0.0002

These values were input into the preexisting internal validation document. This increased the total number of tests. The calculations for the bias, B, were done according to internal documentation. The bias is a value that corelates to how the code represents reality. If a code continuously underestimates the k_{eff} value for an experiment, a bias for the code is introduced to make the code align with the actual value. For example, in Table 2 for experiment 2 in LCT 025, the benchmarked k_{eff} value is 1 and the calculated value is 0.99214. This means that the code has underestimated the

k_{eff} value by 0.00786, if this was the case for every calculation a bias need to be introduced to the calculations to compensate for the underestimations for the code to represent the true value of k_{eff}.

From the parametric calculations B was calculated and the upper subcritical limit, USL, is also calculated. USL is the highest k_{eff} value that is allowed in the system. The USL can be calculated using equation (USL).

$$USL = K_L - dm \tag{USL}$$

where K_L is the lower tolerance limit and dm is an additional administrative margin. Since the distribution of the results are not normally distributed B needs to be calculated non-parametrically. To calculate B the first variable needed to be calculated is K_L. To calculate K_L equation (K_L) is used.

$$K_L = k_{eff}^{min} - uncertainty\ k_{eff}^{min} - NPM \tag{K_L}$$

Where k_{eff}^{min} is the minimum of all the normalized k_{eff} and the uncertainty for that k_{eff}, NPM is non-parametric margin. Depending on the percentage of the degree of confidence, β, the NPM varies in value. To calculate β equation (β) is used.

$$\beta = 1 - q^n \tag{β}$$

where q is the desired fraction of the population and n is the number of total experiments. To calculate B non-parametrically equation (BUSL) is used.

$$B = 1 - dm - USL \tag{BUSL}$$

Equation (USL) and (B) is then combined to create equation (B)

$$B = K_L - 1.0 + dm \tag{B}$$

All the parameters needed to calculate B non-parametrically can be seen in

Table 3: All the parameters needed to calculate B[2].

Min k_{eff}	A
Min k_{eff} s	C
β	D

---

[2] The values in this table have been removed for secrecy

NPM	E
K_L	F
dm	G
USL	H
B	I

As can be seen in Table 3 B is calculated to I and will be added to every calculated k_{eff} value to ensure that the measurements are as close to reality, or lower, as possible.

6. Results

6.1. Current k_{eff} values

For every system that was considered in this report the k_{eff} will be listed in the Table 4.

Table 4: List of all the highest k_{eff} values for each limit available for all the systems the pellets would go through. In this report the values have been removed due to secrecy.

System	*Normal use ($k_{eff} < 0.95$)*	*Unlikely situation ($k_{eff} \leq 0.95$)*	*Extremely unlikely situation ($k_{eff} < 0.98$)*
356			
357			
411-1			
411-6			
412-1			
413		Y	Z
414		X	
415			
415 (extra boron)			
416			

The values X, Y, Z represent the k_{eff} values + 3 standard deviations + Bias + any accident scenario specific additions.

6.2. System 414

6.2.1. Parameter study

Results for each simulation with enrichment from 5% to 8% in 0.5% increments can be seen in Table 5 and Figure 7:

Table 5: The k_{eff} value plus 3 standard deviations plus B plus R_x for each simulation in the parameter study.

Enrichment	$k_{eff\,mean}+3s+B+R_x$
5%	X[3]
5.5%	1.0189*X
6%	1.0355*X
6.5%	1.0501*X
7%	1.0633*X
7.5%	1.0747*X
8%	1.0854*X

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[3] The value X have been removed due to secrecy

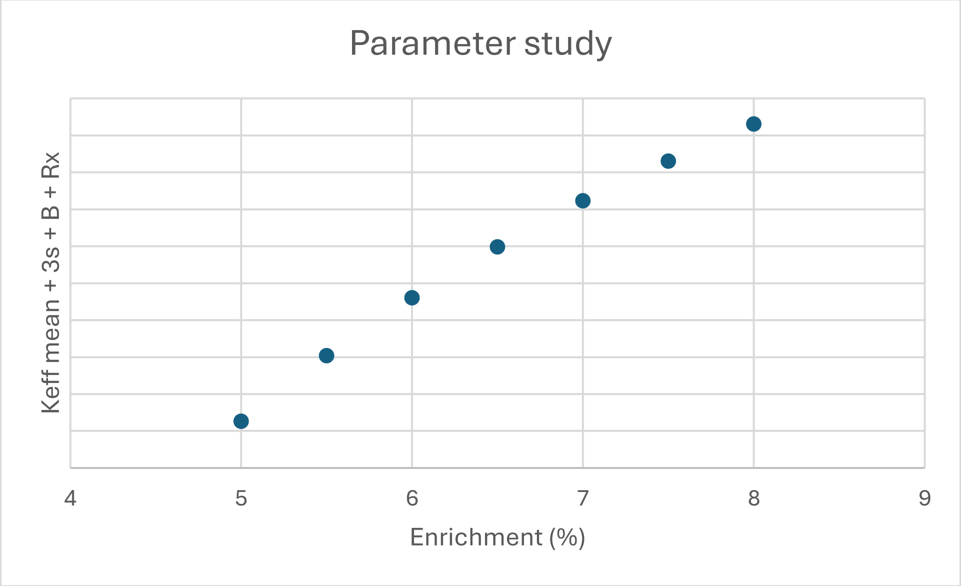

Figure 7. The k_{eff} value plus 3 standard deviations plus B plus Rx for each simulation in the parameter study.[4]

6.2.2. Modified model

The results of each modification to the model of system 414 can be seen in Table 6 and Figure 8:

Table 6: The k_{eff} of each model modification for system 414

Modification	$k_{eff\ mean}+3s+B+R_x$
Original model	X
8% enrichment	1.0854*X
Modified model	0,9788*X

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[4] The values on the Y axis have been removed due to secrecy

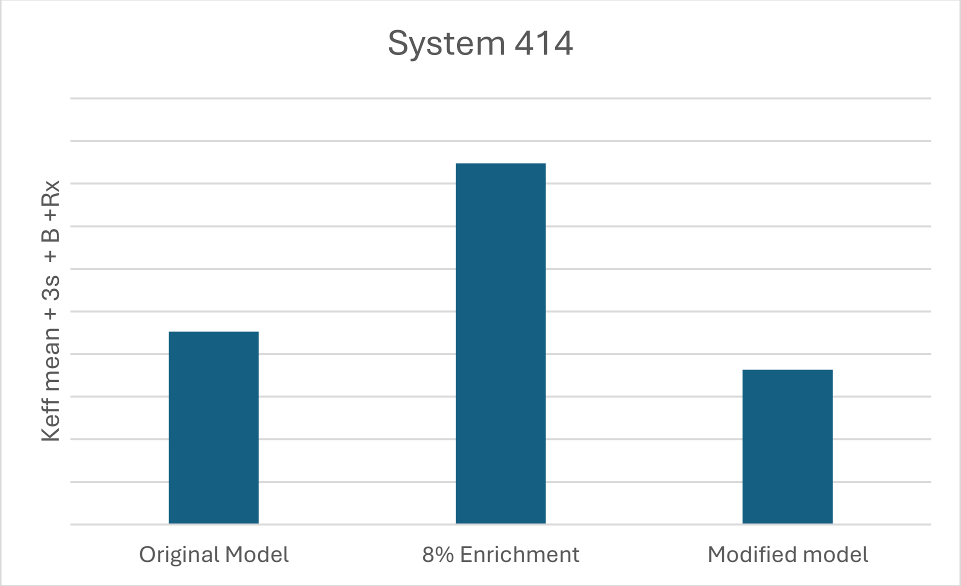

Figure 8. The keff of each model modification for system 414[5]

To check if the system is optimally moderated the water density was changed. The result from the varying water density can be seen in Figure 9:

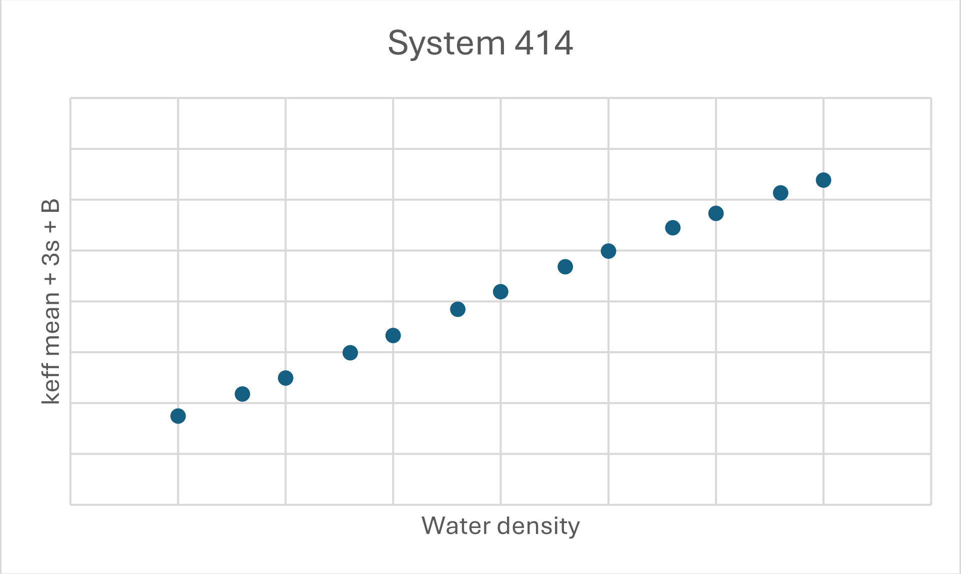

Figure 9. $k_{eff\ mean} + 3s + B$ *for varying water density for the partly water filled system.*[6]

The calculations for system 414 were made at the highest water density for this specific accident scenario.

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[5] The values on the Y axis have been removed due to secrecy

[6] The values on the X and Y axis have been removed due to secrecy

6.3. System 413

6.3.1. Partly water filled storage

The results of the original model and the different modifications made to system 413 with a partly water filled storage can be seen in Table 7 and Figure 10:

Table 7: The k_{eff} of each model modification for system 413 with a partly water filled storage.

Modification	$k_{eff\ mean}+3s+B$
Original model	Y[7]
8% enrichment	1.0852*Y
Modified model	0.9924*Y

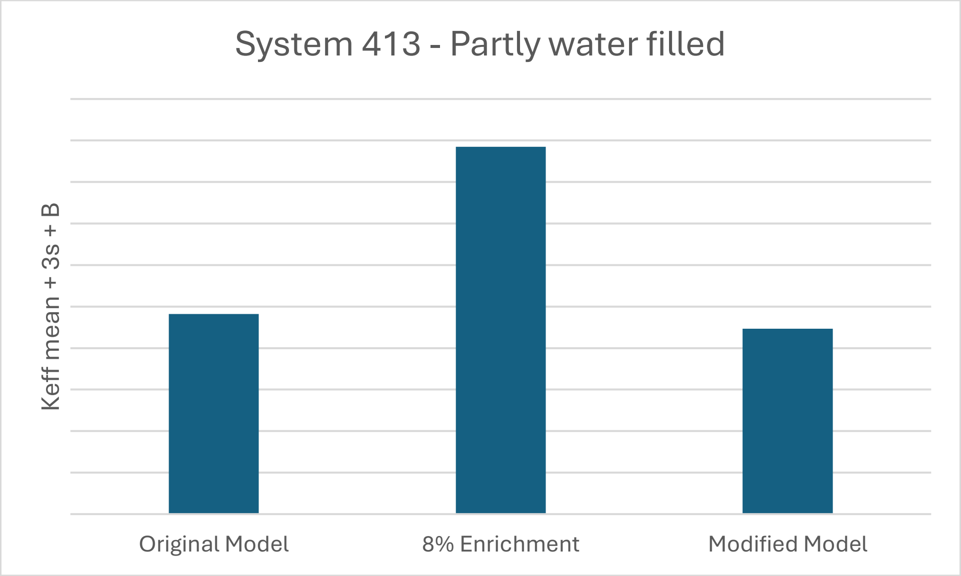

Figure 10. The k_{eff} of each model modification for system 413 with a partly water filled storage.[8]

To check if the system is optimally moderated the water density was changed. The results from the varying water density can be seen in Figure 11 and Figure 12:

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[7] The value Y have been removed due to secrecy

[8] The values on the Y axis have been removed due to secrecy

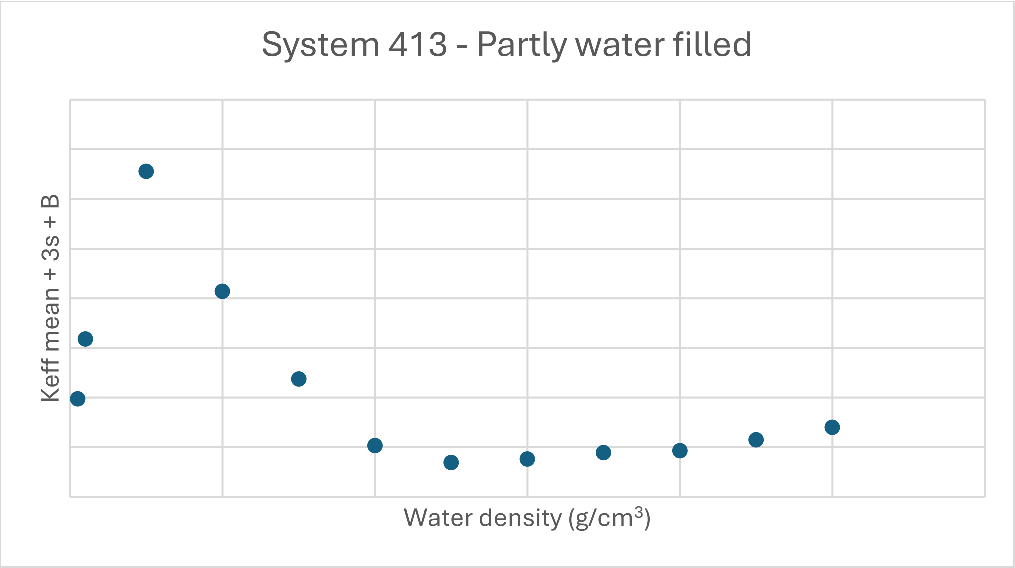

Figure 11. k_{eff} mean + 3s + B for varying water density from X to Y for the partly water filled system. [9]

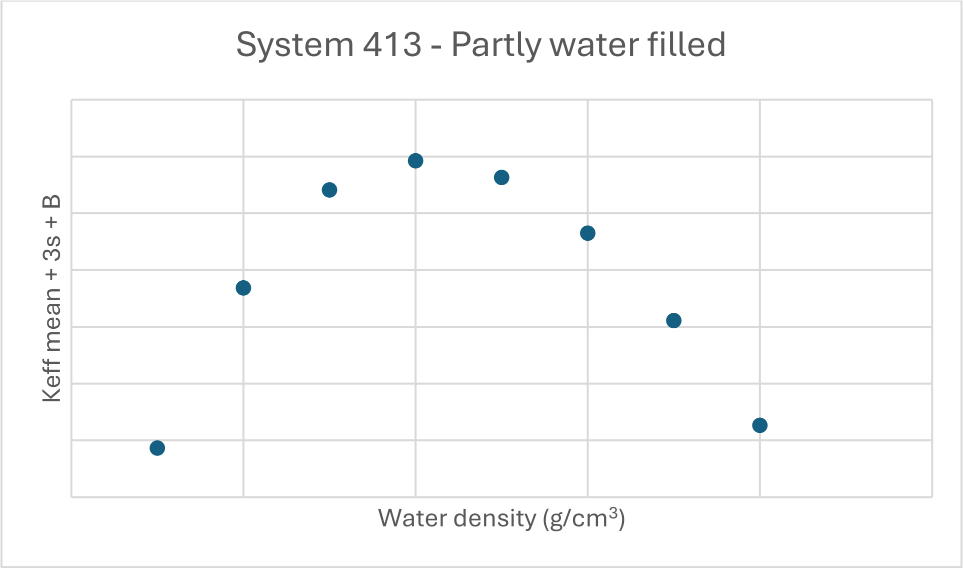

Figure 12. A zoomed in graph over the peak in Figure 10.[10]

6.3.2. Completely water filled storage

The results of the original model and the different modifications made to system 413 with a completely water filled storage can be seen in

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[9] The values on the X and Y axis have been removed due to secrecy

[10] The values on the X and Y axis have been removed due to secrecy

Table 8 and Figure 13:

Table 8: The k_{eff} of each model modification for system 413 with a completely water filled storage.

Modification	$k_{eff\,mean}$+3s+B
Original model	Z[11]
8% enrichment	1.0835*Z
Modified model	0.9558*Z

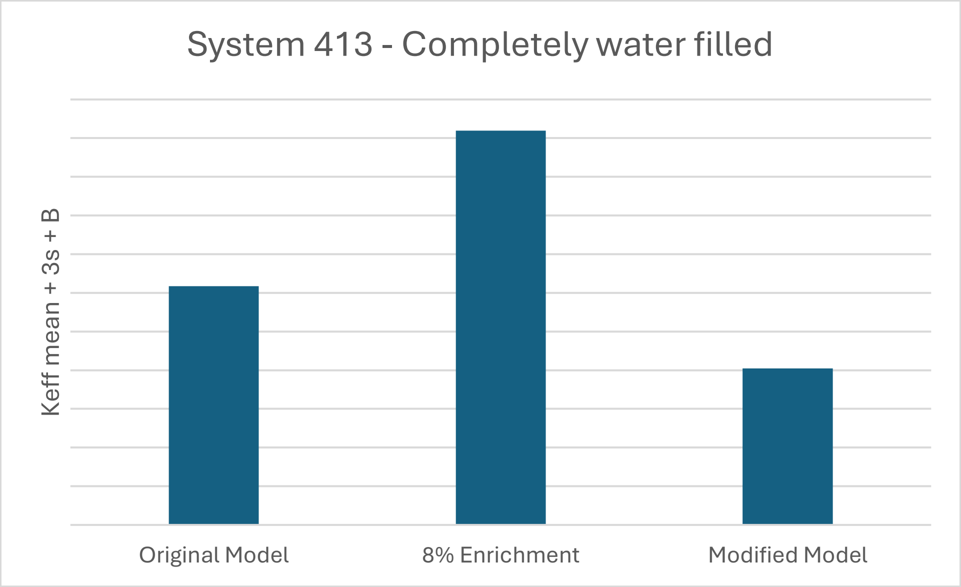

Figure 13. The k_{eff} of each model modification for system 413 with a completely waterfilled storage.[12]

To check if the system is optimally moderated the water density was changed. The results from the varying water density can be seen in Figure 14 and Figure 15:

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[11] The value Y have been removed due to secrecy

[12] The values on the Y axis have been removed due to secrecy

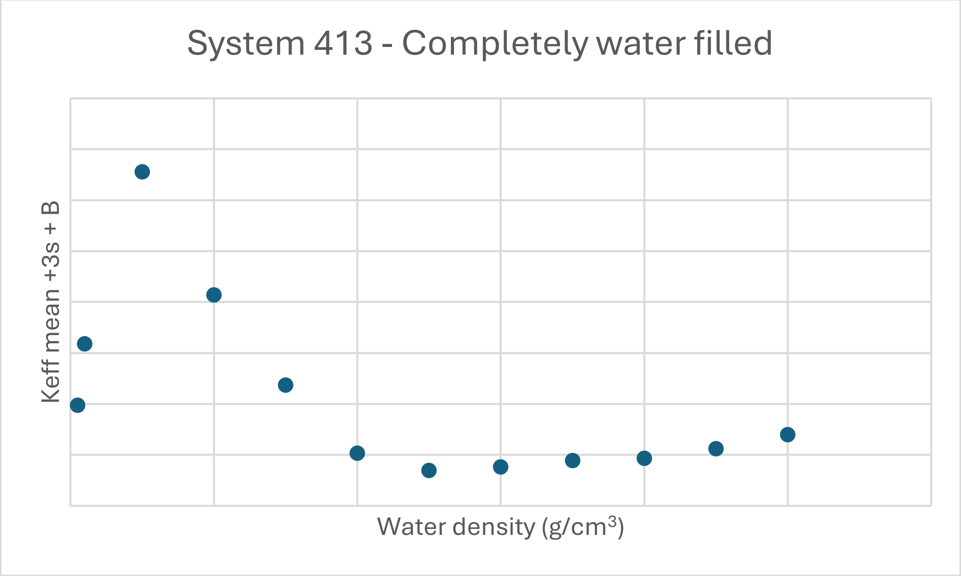

Figure 14. $k_{eff\ mean} + 3s + B$ *for varying water density from X to Y for the completely water filled system.*[13]

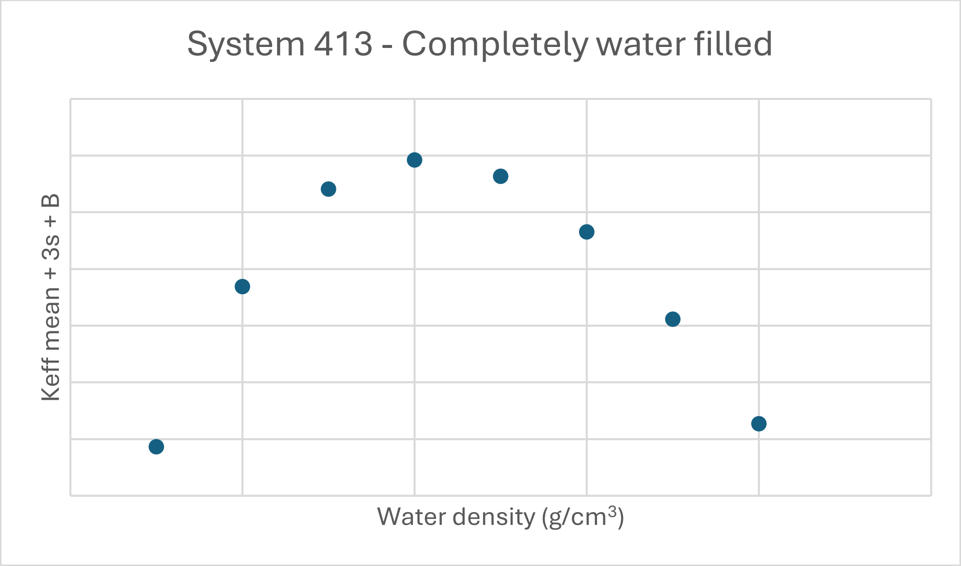

Figure 15. A zoomed in graph over the peak in Figure 14.[14]

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[13] The values on the X and Y axis have been removed due to secrecy

[14] The values on the X and Y axis have been removed due to secrecy

7. Discussion

7.1. System 414

7.1.1. Parameter study

From the results in Table 5 we can observe an increase in the k_{eff} for increased enrichment. This is as expected since the increased enrichment corresponds to increase of fissile material. This result therefore lays foundation for the enrichment in the other models.

The enrichment is one parameter that was analyzed. However, in the previous criticality safety analysis of this accident scenario the optimal distancing for fuel rods was also analyzed. This parameter was not analyzed in this report due to time restrictions. The distancing could influence k_{eff} since the spacing of the fuel rods would change and that would affect the moderation of the interacting neutrons. This is something that would need to be analyzed before the system could be deemed safe for a higher enrichment.

7.1.2. Modified model

The modification needed to reduce the k_{eff} for the given accident scenario was a reduction in the excessive conservative margins set for the model and the introduction of boron carbide rods. The original model was modeled with an excess number of fuel rods. The calculations were made on the assumption that the wagon was filled with fuel rods for TRITON-11. In the new model there is a conservative estimation of the model with fuel rods and boron carbide rods. The reason for the extra fuel rods compared to reality is to have perfect symmetry for the positioning of the fuel rods around the TRITON-11. The carbide rods are identical to the fuel rods except that the uranium is removed and replaced with boron carbide instead. These rods will be re used and placed on new carts when the fuel rods have been used. The positioning of the boron rods is a parameter that would need to be analyzed to find the most conservative positioning for them. In this model, the four boron carbide rods have been placed on the perimeter evenly spread out. This is a modification to the system that would need to be suggested to the factory before the implementation of 8% could be made.

The accident scenario with a the higher k_{eff} limit of 0.98 had a smaller margin to the limit than the accident scenarios analyzed in this report. However, this accident scenario was not analyzed in this report due to the restricted time frame for this project. This specific accident scenario did not have a preexisting MCNP model. This would require a conversion from an older simulation program to MCNP, and that conversion would be too time consuming to be able to fit in this report.

There are several other accident scenarios for this system that were disregarded since their k_{eff} were lower compared to the accident scenarios written in the mapping of the factory. However, these accident scenarios could get a larger increase of k_{eff} compared to the accident scenarios analyzed in this project and would therefore be of importance for the future analyzing in this matter. The other accident scenarios should therefore be analyzed before a conclusion could be made for the possible use of LEU+ pellets in system 414.

From the analysis of the water density a peak in Figure 9 k_{eff} can be seen at the water density of x g/cm^3. This result indicates that the system is below the limit set for this accident scenario up to the limit of x g/cm^3 and the system is therefore critically safe with this configuration.

7.2. System 413

7.2.1. Partly water filled storage

The results for the partly water filled storage in Table 7 increased for the unmodified model with 8% enrichment, as expected. With the introduction of boxes filled with empty zircaloy tubes and boron carbide, the k_{eff} was lowered below the limitations for this accident scenario.

The positioning of the empty zircaloy tubes was assumed to be placed in certain positions in the model. However, the positioning of these boxes is not correlated to the positioning in the factory. The positioning was chosen in discussion with experienced criticality analysists to result in the highest possible k_{eff}. However, the positioning of the zircaloy tubes have not been analyzed due to a lack of time and would therefore need to be analyzed further.

As can be seen in Figure 11 and Figure 12 the water density analysis for this system indicates that the optimal moderation occurred at y g/cm^3 instead of the optimal moderation for the model with 5% enrichment. However, the change in k_{eff} between the two different water densities are small and the system is still critically safe in this configuration.

The boron carbide plates are a modification to the system that would need to be suggested to the factory before any implementation of LEU+ pellets to the factory could be done.

7.2.2. Completely water filled storage

The results for the completely water filled storage in Table 8 were interesting. In both scenarios the k_{eff} increased, as expected. However, after the modifications made to the system, the k_{eff} decreased more in this model compared to the partly water filled storage. This is interesting since

An aspect that could decrease k_{eff} is the increase of water in the system. This increase of moderator would moderate the neutrons more and more neutrons reach thermal energies. The absorption in the boron carbide is therefore increased, since the absorption cross section for boron carbide is higher for lower energy neutrons as can be seen in Figure 2. This therefore decrease the number of thermal neutrons in the system. Since the fission cross section for U_{235} is larger for thermal neutrons, as can be seen in Figure 3, the decrease in the number of thermal neutrons would therefore decrease k_{eff}.

The optimal moderation for this system was increased slightly to z g/cm^3 compared to the optimal moderation for the model with 5% enriched uranium. The change in k_{eff} for the two different water densities is insignificant and the system can be assumed to be safe for the use of LEU+ pellets with this configuration of the model.

The boron carbide plates are a modification to the system that would need to be suggested to the factory before any implementation of LEU+ pellets to the factory could be done.

7.2.3. General for system 413

In the model used for both accident scenarios there are two parameters that was assumed to be fixed. These parameters are the positioning of the boxes filled with empty zircaloy tubes and the positioning of the boron carbide plates. If these parameters were changed the k_{eff} of the system would change and therefore the possibility of a more optimized model should be analyzed.

The changes in the positioning of the boxes filled with empty zircaloy tubes could both increase and decrease the k_{eff} for the system. A further analysis of the positing of the boxes should be performed and the positioning of these boxes should therefore be administered to decrease k_{eff} as much as possible. Further discussion with the system supervisor would be needed to set regulations for the positioning of the boxes.

The positioning of the boron carbide plates should also be analyzed further. The configuration in this report is to place the boron carbide plates closest to the center of the racks to decrease the k_{eff} as much as possible. However, the positioning of the boron carbide plates might be suboptimal, and further analyses should be performed to ensure the optimal positioning for the plates.

These two parameters need further investigation to determine the true criticality safety of the system. If the changes to the model would increase the k_{eff} value of the system, further modifications would therefore be needed. This is a necessity to ensure the safe handling of LEU+ pellets in this system.

For this system there are several more accident scenarios that have been calculated for an enrichment of 5%. However, in this report these accident scenarios have been disregarded because of their lower k_{eff} values compared to the accident scenarios that were included in this report. Although, since the dynamics of a system could change with enrichment changes there is no guarantee that the other accident scenarios would not result in a higher k_{eff} value compared to the accident scenarios in this report. However, these accident scenarios should be investigated further to be able to make a conclusion if this system is safe for LEU+ pellets.

7.3. PWR fuel

The systems analyzed in this report, which is used for a specific fuel type, have been for BWR or VVER fuel assemblies. However, since the PWR assemblies are larger and therefore contain more UO2, they are more reactive. The reason for the lack of analysis on the systems containing PWR fuel assemblies is due to the lack of preexisting MCNP models for the systems containing PWR fuel assemblies and the lack of time for this project. Any conclusions about the feasibility of LEU+ pellets in the factory can therefore not be applied to PWR fuel.

7.4. Future studies

The future studies that can be motivated from this report is to analyze the rest of the systems that the pellets would travel through. There are several more systems that have not been analyzed yet. Since the systems analyzed in this report all needed both a reduction in the excessive conservative assumptions and the introduction of neutron absorbers, there is a high probability that other systems would need it as well. A future study of the other systems would therefore be needed before any conclusions about LEU+ pellets in the factory could be done.

For every system there are several accident situations analyzed. In this report only three accident situations have been analyzed since they had the smallest margins to the k_{eff} limitations and a preexisting MCNP model. Since the dynamics of a system is changed when the enrichment is changed, additional analysis of the rest of the accident situations should be done.

In the factory there is the possibility to recycle uranium pellets or uranium powder that have been damaged or contaminated. In this report, this part of the factory has been assumed not to be able to accept LEU+ pellets or powder. This should therefore be a potential future study on whether recycling is possible at this facility. The same can be said about the usage of LEU+ UO_2 powder in the factory and what systems need to be modified to be able to manage LEU+ powder.

In this report the economics have been disregarded. That part of this project would also be interesting to further investigate to see if the modifications made in this project would be feasible or if other cheaper modifications would be better suited for the factory. These modifications could be to reduce the amount of uranium in a system, a solution that was prioritized lower in this project to be able to continue the production in the same manner as today.

In this report, some systems have no calculations made on them since they have critically safe parameters. However, these parameters are calculated based on an enrichment of 5%. These parameters will change if the enrichment is increased to 8% and would therefore be re calculated. This might change the need for a criticality safety analysis for these systems.

7.5. Error sources

One source of error in this project is the validation. The validation used at WSE has a total of X experiments at a variety of enrichment levels up to 5%. In this report the number of experiments over 5% is 10 and the bias is therefore heavily influenced by the experiments with lower enrichment. For the validation to properly represent the higher enrichments used in this study, a larger sample size of higher enrichment experiments is needed. However, the number of experiments that can be validated against above 5% is significantly lower than under 5%. Therefore, the number of experiments with enrichment up to 8% would need to increase for the validation.

In the accident scenario analysis for system 414, the placement of the boron rods was not analyzed further. This means that there could be a configuration with a more conservative positioning of these rods which would result in a higher k_{eff} than calculated. This could result in a wrong conclusion if a further analysis is disregarded.

One of the main reductions in the k_{eff} value was to reduce the conservative margins. In theory, this would not result in a higher k_{eff} value since the conservative margins are excessive compared to reality. However, these conservative margins also include a sort of extra margin to the k_{eff} limit for situations that have not been thought of or calculated for. This would therefore increase the risk for a criticality accident to occur if this accident scenario would occur. With that said, there are still conservative assumptions made in these models. They are not as conservative as the assumptions made in the previous analysis.

8. Conclusion

The pellets would get transported through the factory according to the mapping written out in Section 4. Mapping. In this section all the k_{eff} values have been mapped as well as the systems with the highest k_{eff} and a preexisting MCNP model. For an accident scenario with the limitations of 0.95 the system with the highest k_{eff} is system 413 with a k_{eff} of Y, and with the limitations of 0.98 it is system 413 with a keff of Z.

The modifications made to the model, for the system to be able to handle LEU+ pellets without exceeding the regulations for k_{eff}, have been:

- Reducing the excess conservative margins
- Introducing neutron absorbers to the system

With the models and the amount of analysis possible within the time limit, the systems 413 and 414 would be able to accept LEU+ pellets with small modifications to the systems. However, there are still a lot of accident scenarios and parameters that were not analyzed in this report that would need updated analysis before any real-life conclusions could be drawn.

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