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Curium in Space

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Abstract

New technology has shown the possibility to use a miniature satellite in conjunction with an electric driven engine to make a spiral trajectory into space from a low earth orbit. This report has done an investigation of the new technique to produce power sources replacing solar panels which cannot be used in missions out in deep space. It is in essence an alternative use of curium among the many proposals on how to handle the intermediate stored used nuclear fuel or *once through nuclear fuel* as some people prefer to call it.

The idea of sending radioactive used nuclear fuel into outer space has been considered before. There was a proposal, for example, to load a space shuttle with radioactive material. This could have serious consequences to the nearby population in the event of a major malfunction to the shuttle. The improvement to this old idea is to use a small satellite with only a fraction of the spent fuel. With this method and other technological advances, it is possible to further reduce the risk of contamination in the event of a crash.

This report has looked into the nuclear energy production of Sweden and the current production of transuranium elements (Pu, Np, Am and Cm). The report has also focused on the curium (Cm) part of the transuranium elements, which is the most difficult to recycle in a fast neutron spectra. The physical property of curium reduces many of the safety parameters in the reactor as it is easily transmutated into californium, which is a high neutron emitter. Although sometimes viewed as troublesome, curium may very well find important use in new reactor techniques. *This study, however, is carried out solely to investigate if Cm can be used in space exploration, replacing Pu in RTGs.*

The energy production from the Swedish nuclear power plants generates a used nuclear fuel stream between 6-12 kg of curium each year. This report shows that the amount of curium from 3 years electric production could be used to produce 80 General Purpose Heat Sources (GPHSs that would produce heat in a safe way inside 2 Radioisotope Thermoelectric Generators (RTGs). These two Radioisotope Thermoelectric Generators could then be used to replace the solar panels, such as that used on the small satellite SMART-1, to generate enough electric power to drive the satellite far out into space. The result of this report shows the great potential of an alternative way to close the nuclear fuel cycle. To bury most of the spent fuel in a deep geological repository is maybe not the right future for nuclear power to go. The decision to bury the used nuclear fuel may have been too hasty. More consideration should be given to alternative ideas and methods such as using the intermediate stored spent fuel in new fuel for a generation IV nuclear park.

The idea of sending curium into space is, however, ahead of its need. It will only be useful on the condition that the power industry will decide that curium will never be needed as a fuel in any future nuclear power generation. The argument to make RTGs for the sole purpose to send up spent nuclear fuel compared to build RTGs to make batteries for scientific space missions will make it easier to overcome international laws and ethical questions [56]. There are many obstacles to overcome before the idea can be fully achievable, such as the separation techniques for the spent fuel stream and all the intermediate stored spent fuel that have to be implemented on an industrialized scale. Another element that is necessary to successfully close the nuclear fuel cycle is to develop a fast nuclear reactor capability to accommodate the rest of the spent fuel.

Acknowledgments

The present thesis has been an exciting and endeavouring journey, although perhaps a bit slow. There are some people I would like to mention, people who have helped to complete the thesis. Thus I would like to thank my supervisor prof. Thomas Lindblad. Many thanks are also due to Drs Magnus af Ugglas, Lars Hildingsson, Clark S. Lindsey, Sven Grahn and Prof. Christian Ekberg, for reading the manuscript and making comments and suggestions. I would also take the opportunity to thank Carin Åstrom Borinder and Torbjörn Bäck who have helped me through the administration to finalize this thesis.

Contents

Abbreviations

ADS	**Accelerator-Driven System**
AFC	**Advanced Fuel Cycle**
ATB	**Average Thermal Burn-up**
BOM	**Beginning Of Mission**
ESA	**European Space Agency**
BWR	**Boiling Water Reactor**
EMF	**Electro-Motive Force**
EOM	**End of Mission**
FBR	**Fast Breeder Reactor**
FNR	**Fast Neutron Reactor**
FR	**Fast Reactor**
GIS	**Graphite Impact Shell**
GPHS	**General Purpose Heat Source**
HLLW	**Highly Long-lived Waste**
HLW	**High Level Waste**
LEO	**Low Earth Orbit**
LLFP	**Long-Lived Fission Products**
LWR	**Light Water Reactor**
MA	**Minor Actinides (Np, Am, Cm)**
MMRTG	**Multi-Mission Radioisotope Thermoelectric Generator**
MOX	**Mixed Oxide**
OTC	**Once Through Cycle**
P&T	**Partitioning & Transmutation**
PWR	**Pressurized Water Reactor**
RFC	**Reprocessing Fuel Cycle**
RHS	**Radioisotope Heat Source**
RTG	**Radioisotope Thermoelectric Generator**
SNAP	**System Nuclear Auxiliary Program**
SRG	**Stirling Electric Generator**
TBC	**Tributylphosphate**
TPV	**Thermo-Photovoltaic**
TRU	**Transuranium (Pu, Np, Am and Cm)**
We	**Electric Watt**
Wt	**Thermal Watt**

Chapter 1

Long-lived radioactive elements

The nuclear energy program in Sweden is the second most important source of electricity after hydro power. Nuclear power has been a debatable source of energy production for a long time. After the accident at Harrisburg in 1979 there was a referendum in Sweden on when nuclear power should be phased out. The result of the referendum was that it should be phased out by the year 2010. Nuclear power is a very difficult source of energy to substitute with a more efficient and environmental friendly alternative. This is the reason why the Swedish government has not yet been able to follow up on the result from the referendum in 1980. The cause of the mistrust in the production of nuclear power arises mostly because of the nuclear catastrophes that occurred at the Three Mile Island in 1979 and the reactor melts down in Chernobyl 1986. The explosion at Chernobyl has had major consequences regarding the benefits from nuclear power on public opinion in Sweden. There are also reports indicating increased risks of cancer possibly related from the radioactive spread after the explosion [37]. The earth quake and tsunami in Mars 2011 that lead to the catastrophic events in Fukushima power plant, where the coolant water to the reactor stopped working, because the power grid and emergency generators shut down has led to further questions about the existence of nuclear power [45].

Another disadvantage of nuclear power is the spent fuel that results from the energy production. Many of the elements that results from the burning of uranium are highly radioactive and very long-lived. To ease the public opinion the nuclear reactors today are equipped with more and better safety systems. Today the long-lived radioactive elements are planned to be buried deep underground, which is not a sustainable option. The Swedish energy mix is highly dependent on the nuclear power production and until there is an equally efficient way with less problematic means of producing energy, the need to make nuclear power as safe and environmental friendly as possible is the most logical way to progress[1].

In this chapter a brief introduction will be given about the processes that produce these different long-lived radioactive elements and about the amounts that are produced. This chapter will also explain some of the techniques used to reduce or totally eliminate them.

1.1 Spent fuel

The Swedish nuclear power industry uses uranium enriched with 3-5% uranium-235 as fuel in the two types of light water reactors (LWR) operational today. The two types are pressurized water reactor (PWR) and boiling water reactor (BWR). The used nuclear fuel stream from the two different reactors are similar, the difference between the two comes from how they convert the generated heat, after splitting the atom and making power. When fission takes place in uranium-235 from a thermal neutron the heavy nucleus splits into two lighter nuclei having their different masses around 95u and 135u. The lighter nuclei are called fission products (FP). The transformation is illustrated below:

$$n + {}^{235}U \rightarrow FP + \sim 2.5n + \beta + \gamma \tag{1.1}$$

Another process that can occur inside the reactor is when the uranium-238 captures a neutron to breed plutonium-239. If this fissile product captures another neutron instead of undergoing fission, it will go on to create new elements, so-called transuranium (TRU). A series of these neutron capturing events can create new elements heavier that uranium illustrated below:

$$^{238}U + n \rightarrow {}^{239}U \xrightarrow{\beta^-} {}^{239}Np \xrightarrow{\beta^-} {}^{239}Pu \rightarrow \tag{1.2}$$

$$\rightarrow {}^{239}Pu + n \rightarrow {}^{240}Pu \rightarrow \tag{1.3}$$

$$\rightarrow {}^{240}Pu + n \rightarrow {}^{241}Pu \xrightarrow{\beta^-} {}^{241}Am \rightarrow \tag{1.4}$$

$$\rightarrow {}^{241}Am + n \rightarrow {}^{242}Am \xrightarrow{\beta^-} {}^{242}Cm \tag{1.5}$$

These two processes create the radiotoxic elements with which the nuclear power industry must deal. Radio toxicity is a measure of how hazardous a radioactive nuclide is to the health of a biological organism. Radio toxicity depends on the energy and type of radioactivity. The type of radioactivity depends on how the radioactive material decays into another element. The different radiation types of α, β, γ and neutron radiation can have different amounts of energy. The activity (A) of the decay is measured in Becquerel (Bq). One Bq, the standard SI-unit, is equivalent to $2.70*10^{-11}$ of the older unit Curie (Ci). The amount of time and the place where the element resides in the body are also important factors in deciding the level of radio toxicity of the radioisotope element. How the element enters the body, e.g. inhalation or ingestion, also greatly affects the risk of exposure. For example, plutonium is more dangerous by inhalation because it is more easily absorbed into the blood via the lungs. The equation used to measure the dose equivalent when the radiotoxic element is digested is equation 1.6.

$$H = \varepsilon\,(\mathrm{t}) * A \tag{1.6}$$

Here H is the dose equivalent that is measured in Sievert, Sv. To give a scale of toxicity, 10 Sv is an instantaneous lethal dose. The dose coefficient (ε(t)) is time dependant and the methods used to calculate ε(t) is constantly refined and are based on body-size, age and gender [39].

An important factor when comparing the hazards of the FPs and TRUs is the relative long half-life that indicates how long they will stay radiotoxic. The majority of FPs have half-lives of less than 100 years while several TRU have very long decay times of a couple of hundreds of thousands of years shown in figure 1.1.

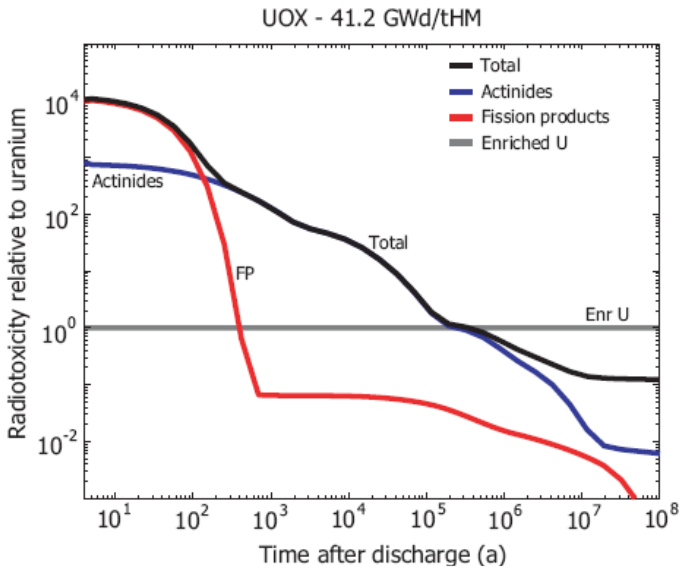

Figure 1.1. A comparison between radiotoxicity relative to uranium of the actinides[1] and FP elements from a PWR with enriched uranium [2].

The diagram clearly shows why it is more important to focus on the TRUs to really reduce the long lived radioisotopes in the spent fuel. If the TRUs could be taken care of or drastically reduced the need for a long term deep geological repository would not be as great. These methods involved in such a process could help the nuclear power industry to become a more economical, socially accepted and safer system to handle the radiotoxic used nuclear fuel of nuclear power [3].

[1] The actinide elements are actinium, thorium, protactinium, uranium, neptunium, plutonium, americium, curium, berkulium, californium, einsteinium, fermium, mendelevium, nobelium and lawrencium. Ranging from atomic number 89-103 these elements are naturally radioactive and can be found in a nuclear reactor [46].

Transuranium from PWR

The TRU constitutes most of the long-lived elements in the spent fuel and from the equations 1.2 to 1.5 the transuranium elements involved are Pu, Np, Am, and Cm. Table 1.1 gives an overview of the mass of the different isotopes that are produced from an average thermal reactor loaded with uranium oxide fuel per gigawatt days per ton of heavy metal.

Nuclide	Isotope [kg]	Element [kg]	Half-Life [year]
U-234	3.7-5.2	20685(U)	2.46E+05
U-235	138-173		7.04E+08
U-236	118-136		2.34E+07
U-238	20424		2.14E+06
Np-237	15.6-16	16(Np)	2.14E+06
Pu-238	7.46-8.65	240(Pu)	8.77E+01
Pu-239	125.3		2.41E+04
Pu-240	60.8-71.2		6.56E+03
Pu-241	26.3-30.3		1.44E+01
Pu-242	17.6-19.8		3.74E+05
Am-241	9.3-11.6	12.8-16.4(Am)	4.33E+02
Am-242	3.5-4.72		7.36E+03
Cm-243	0.012	1.745(Cm)	2.91E+01
Cm-244	1.557		1.81E+01
Cm-245	0.075		8.53E+03
Cm-246	0.1		4.73E+03

Table 1.1. Actinide inventory of spent UOX fuel per GWe/year at 47.5 – 50 GWd/tHM [4].

Sweden's electric consumption for the year 2009 was 134 TWh, to which nuclear power contributed 61.3 TWh [5]. To avoid the hazards of storing these radioactive elements in a deep geological repository for over 300,000 years, we have to adopt a more sustainable way of creating energy from nuclear power. Transmutation of the TRUs could be one way of doing just that. By placing the actinides in a fast neutron flux the neutrons will split the actinides into less long-lived elements due to the higher microscopic fission cross-section in the fast neutron spectra on these elements. This will be explained in more detail in the next chapter. Special concern needs to be focused on the elements of plutonium and americium, as seen in figure 1.2, because these two elements dominate the long term radiotoxic inventory. The problem with transmutation of americium is that this will produce curium, which also has to be dealt with to bring down the long term radiotoxic inventory by more than a factor of ten [3].

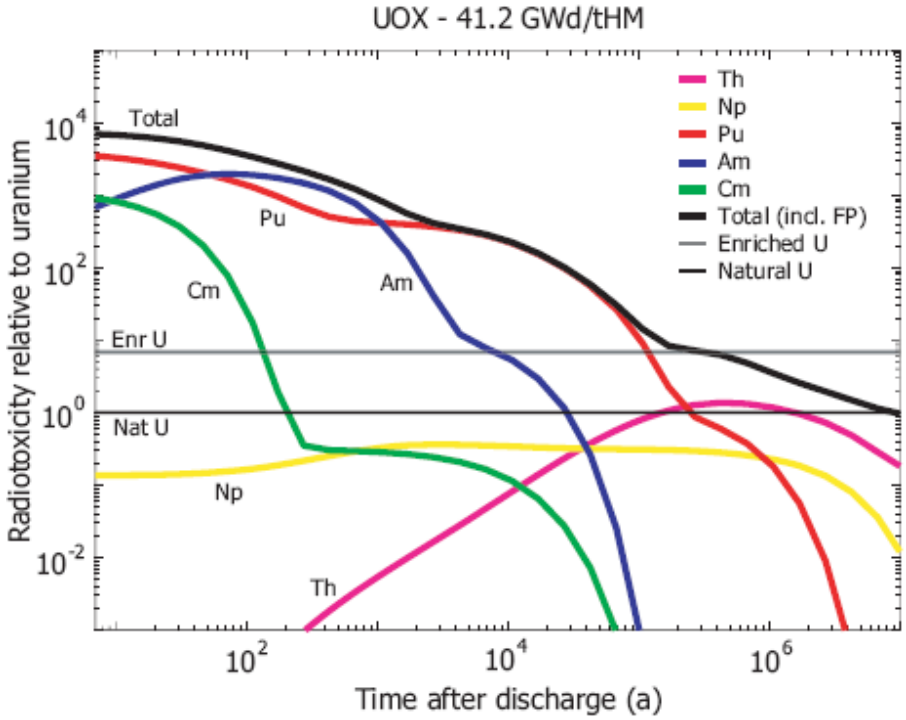

Figure 1.2. Showing the radiotoxicity of the different transuranium elements [2].

1.2 Nuclear fuel cycle for LWR (PWR/BWR)

The Nuclear fuel cycle is described by three steps “the front end”, “the service period” and “the back end” of the fuel cycle. The front end describes the start of mining the uranium ore to enrich the natural uranium with U-235 and finally producing the uranium rods that goes into the reactor. The service period describes the period when the fuel rods are in the reactor and fission of the enriched uranium takes place. The back end describes how the spent fuel is managed, whether it is contained or reprocessed, if the spent fuel is contained in a geological repository the fuel cycle is said to be open and if it is reprocessed it is said to be closed [6,7].

There are three main models on how the back end of the nuclear fuel cycle could be operated. These models are:

- Once Through Fuel Cycle (OTC)

  The once through fuel cycle is the most common approach. It does nothing to the spent fuel after the enriched uranium is burned in the reactor, the spent fuel is not processed further and is taken out as it is. After discharged from the reactor, the spent fuel is stored in a pool close by to the reactor to cool down. Here most of the activity from the spent fuel will decay. After nine months the spent fuel can be put into containers (the container provides protection from the still high radiation of the spent fuel) for shipment to an intermediate storage to cool further. After being cooled, the spent fuel will be transferred

to a permanent disposal site where it will be stored for hundreds of thousands of years [38].

- Reprocessing Fuel Cycle (RFC)

  The presently used RFC approach involves recycling most of the major actinides, plutonium and uranium, from the dissolved fuel elements and then vitrifying the rest of the TRUs and FPs together. This vitrified package will then be buried in a deep geological repository. This method has the advantage of salvaging 20% of the material as fresh uranium, but the disadvantage with this method is that reprocessed uranium contains the isotope U-232, which through natural decay creates very radiotoxic daughter nuclides. In the case of using reprocessed uranium, U-236 is another bad element that creates another radiotoxic element, namely Np 237 by irradiation.

  The strategy to reprocess the spent fuel is used today in countries like France, Germany and Japan [47], where they produce spent fuel mixed with oxide (MOX) from the recovered plutonium at an industrial scale. This MOX can then be re-used in commercial LWR as fuel with the benefit of reducing the mass of this radiotoxic source by a factor of 5. It is possible to recycle the LWR-MOX multiple times if fresh moderate burn-up, reprocessed plutonium from a LWR-UO_x is added. This method is presently the only way to recycle plutonium on an industrial scale, but when the U-235 is still very cheap it is not an economical competitive method. Another disadvantage is the fact that RFC only fissions 25% of the recycled plutonium while 10% is transformed into TRU, which does not significantly reduce the total radiotoxic inventory [6].

- Advanced Fuel Cycle (AFC)

  The AFC is the next step up in the RFC approach. While RFC only recycles the plutonium and uranium, AFC uses partitioning and transmutation (P&T) to take care of the minor actinides (Np, Am and Cm) and possibly some of the long-lived fission products (LLFP). This report will not go deeper into the different LLFPs, because their radiotoxicity is between 1000 and 100,000 times less than the TRUs. With this said, these products are not as harmful as uranium and there is ongoing research on how to take care of these products. There are many different ways of partitioning the MAs from the OTC and these will be described in a later chapter.

  Transmutation is when bombarding an element with neutrons produces less long-lived materials or stable ones, which also will be described later. AFC is the only method that fully takes care of all the elements in the high-level long-lived waste (HLLW). This would mean that there would be no need to bury the used nuclear fuel for hundreds of thousands of years, instead the storage time would be reduced to less than 1000 years. This method needs to use a fast reactor (FR), which is a reactor that has a fast neutron spectrum that can transform the TRUs. The FR system must still overcome some structural design issues before being the obvious option [6, 8].

Today Sweden is applying the once through fuel cycle, which means that after the uranium has been used for about 5 years in the reactor, the fuel is burned out and taken into an intermediate storage facility in Oskarshamn named Clab. The spent fuel has a very high temperature and high activity, which is why it has to be stored in big water pools where the water provides a shield from the radiation and cools the spent fuel. Today Clab is safekeeping 5500 tons of spent fuel with a capacity of 8000 tons. Clab is only an intermediate storage facility, because of the 300,000 years it takes for the spent fuel to decay down the same radiotoxic levels as uranium in the nature. That is why "Svensk Kärnbränslehantering AB", the Swedish company for handling of spent fuel, has decided to build a deep geological repository in Forsmark [3, 9].

OTC is a controversial method because of increased environmental awareness, increased uranium prices and the safety of storing the spent fuel material. That is why investigating other solutions are important. The Swedish government has also declared that energy production is going to be free from fossil fuels and climate neutral in the near future. These statements put extra pressure on the nuclear power industry and on selecting the fuel cycle. One of the most difficult problems with the spent fuel, which may seem quite far-fetched, is how to explain to future generations after another ice age that what is buried below is very dangerous and should not be opened. One does not have to go very far back in the history books to see that the signs guarding the tombs in the pyramids where not very effective [3].

Transmutation

The AFC model could be the answer to solve some of the problems facing the nuclear power industry and to close the nuclear fuel cycle. The technique used to reduce the inventory of spent fuel into stable isotopes or to isotopes with a short half-life is called transmutation[2].

Alpha decay is the most common way the TRUs transform into stable isotopes. The long timescale for this to take place with the TRUs is the main factor as to why it is not convenient to wait around for this to occur. Therefore, initiating transformation via artificial means has the advantage of reducing the half-life of the element and thus shortening the time that the element is radioactive. Transmutation is the process of a nucleus fission or capturing a neutron to become a new element that can in turn undergo fission or capture another neutron. The nature of the element and energy of the neutron determines if the nucleus fissions or captures the neutron. For the TRUs, the fission cross sections are higher for a fast neutron spectrum, which is why there is a need for a reactor that can provide this.

There are three different industrial techniques competing today to become the next generation reactors (generation IV) that provide for transmuting the spent fuel and closing the nuclear fuel cycle. The different techniques are fast neutron reactors (FR), subcritical accelerator-driven system (ADS), and using a fast spectrum of neutrons in a BWR. As explained above, it is not enough to use a thermal neutron spectrum in a LWR to close the nuclear power fuel cycle. That is why AFC is dependent on the research of these three techniques to create a full neutron spectrum.

[2] Transmutation was first discovered in 1901 by Frederick Soddy, along with Ernest Rutherford, when they observed that radioactive thorium transmuted into radium. Later in 1917, Rutherford showed how transmutation could be used artificially to create oxygen from nitrogen by bombarding the nitrogen atom with a neutron flux [10].

There are different aspects to consider when choosing between these three techniques such as, transmutation/incineration yield, fuel fabrication/re-fabrication, availability of reactor types, reduction in radiotoxic inventory, safety parameters, time to develop and economic feasibility. These three types will be discussed in a more thorough way with their disadvantages and advantages below.

Fast reactor

The fast neutron reactor or simply the fast reactor is an old concept; in fact, the first nuclear reactor to produce electricity was the fast reactor EBR-1 in the United States, which was already in use 1951. A fast reactor has a fast neutron spectrum that requires a more fissile fuel than the thermal reactor. The neutrons should not be slowed down by a moderator such as water; instead sodium and liquid metal coolant are used[3] due to the fact that the fuel in the FR needs fast neutron to fission. The fission of the fuel produces more neutrons than a thermal reactor, which could be used to transmute spent fuel or to generate more fuel. A generator that breeds more fuel than it uses up is also referred to as a fast breeder reactor (FBR). The ratio between the fission cross section and the absorption cross section from fast neutrons on TRU and plutonium is higher in a FR than in a thermal reactor, which is why the FR is a very good way of closing the fuel cycle [11, 12]. There are fast reactors that have used liquid sodium as coolant for the reactor core, such as the Super-Phenix in France. If sodium is exposed to air there will be an explosion, this puts high constructional safety requirements on the design of a FR cooled with sodium. There are solutions with a double-wall concept, where the idea is to build an extra confinement wall outside the primary system. The gap between the two walls would be filled with nitrogen, which does not react with sodium, to insure that a leakage would not have serious consequences. Another problem with the FR is the material used in the construction of a FR, which needs to be compatible with the coolant material, which has corrosion effects on stainless steel. Radiation damage to the structural material of the fast reactor is a big concern for its reliability and lifespan. This area requires more research and development [40].

Accelerator-driven system

A subcritical accelerator-driven system uses a linear or cyclotron accelerator to create a fast proton flux which will impinge on a heavy metal target that will undergo spallation and emit tens of neutrons. The neutrons that come out from the spallation will hit the target (e.g. minor actinides) and transmute the elements. The sub-critical reactor can be designed to work both in a thermal or fast neutron spectrum. The accelerator needs 15-30% of the generated energy from fission in the sub-critical reactor to be operational. The major cost of the ADS will be the construction of the accelerator that is approximately 30% of the total cost. This is an expensive endeavour and that is why the ADS will mostly be used to fission TRU. This technique is very promising because of the reactor's sub-criticality there is no risk for the nuclear chain reaction to

[3] State of the art research suggests that lead could be one possibility as coolant, which is now under development by Janne Wallenius and the Electra project. The test reactor will use a new fuel mixture of plutonium, zirconium and nitride together with a fast neutron spectrum using liquid lead as coolant [48].

increase exponentially. This is due to the subcritical reactors need of constant insertion of neutrons, which easily can be switched off. The sub-critical reactor could therefore be loaded with merely minor actinides or in a mixture with other actinides to reduce the amount of high level waste that would be stored in a geological repository by a factor of 100. The problem with ADS is that the bigger the spallation target is, the less efficient the proton source becomes, which is why an industrial ADS is limited to less than 1000 MWth. Another problem is that absorption rods will not have any effect on the reactor core, which is why passive mechanisms would be needed to shut down the proton beam in case of an accident. The advantages of an accelerator-driven system drive many researchers to further development of the sub-critical reactor, the spallation target and accelerator to make ADS an industrial possibility [11, 13, 35].

Transmutation in BWR

This idea takes advantage of the fact that there is a big void in the upper part of the BWR, which means that the neutrons do not get moderated i.e. slowed down by the water there. This together, with the use of hafnium as a cladding material, which has a stopping effect to the thermal neutrons but not the fast neutrons, would make it possible to harden the neutron spectra. This would make it possible in theory to load a reactor with TRU. Transmutation in a BWR has the great advantage that the reactors already exist. Instead of having to build new advanced reactors, one can modify the existing ones for the fuel cycle to be closed. Regrettably it is more complicated than this, for example, the absence of water results in no moderation of the fast neutrons in the air. Many of the inherited safety parameters in the reactor will decrease by adopting a fast neutron spectrum in the BWR due to the higher energy of the fast neutrons [14].

There are still many problems and challenges to overcome before generation IV reactors will be produced on an industrial scale. The constructional issues of the reactors are not the sole problem, but one needs to clarify the targets of how to reprocess the used nuclear fuel as well as how to manufacture the new fuel. The following chapter will deal in more detail with the difficulties of creating fuel and reaching the targets with TRU and with the different methods for partitioning and separation.

Chapter 2

Targets and fuel production for generation IV

In order to close the fuel cycle and make nuclear power an environmentally acceptable option in the mix of energy production methods, the nuclear power industry needs to take care of all the TRUs in a safe way. This chapter will explore the ongoing investigation into the target and fuel processes for TRUs. To be able to create targets and make fuel, there is a need to partition and separate the individual elements. This will be explained in greater detail below. Firstly, though, this chapter will explain some of the engineering difficulties that lie ahead with burning TRUs in a mixture of different reactors.

2.1 Different uses of generation IV reactors in burning TRUs

In the previous chapter it was explained that the only way to make nuclear power sustainable was to adopt the AFC model and this model needs a fast neutron spectra. Today the RFC model works fine where plutonium and uranium are remade into MOX from the OTC and brings down the amount of HLW, but does not really reduce the radiotoxic depository. The difficulties with the RFC model is that the MOX fuel generates eight times more MAs (Np, Am and Cm) than UOX fuel, which will be the long term disposal problem [15]. As there are many LWR operational today and more LWR are being built or planned around the world, there are a couple of examples of how the mixture of thermal reactors together with a fast neutron spectrum could look like. Below are some of these different fuel cycles that this report has studied in greater detail:

1. **THERM** This fuel cycle uses 100% of thermal reactors where some of them will be used for the multi-recycling of Pu and MAs in the MOX fuel compositions.

2. **MIX 1** This fuel cycle uses 70% thermal reactors loaded with UOX and 30% of the composition multi-recycles the actinides with a FR.

3. **MIX 2** This fuel cycle uses 70% thermal reactors for the UOX, 10% for the thermal reactor loaded with MOX and 20% FRs for the Pu and MAs.

4. **FAST** A system that consists of 100% FRs of the FBR type.

5. **Double-strata** Here the first stratum is a conventional fuel cycle for a PWR and the second stratum is a partitioning & transmutation setup, which uses a sub-critical reactor together with an accelerated-driven system.

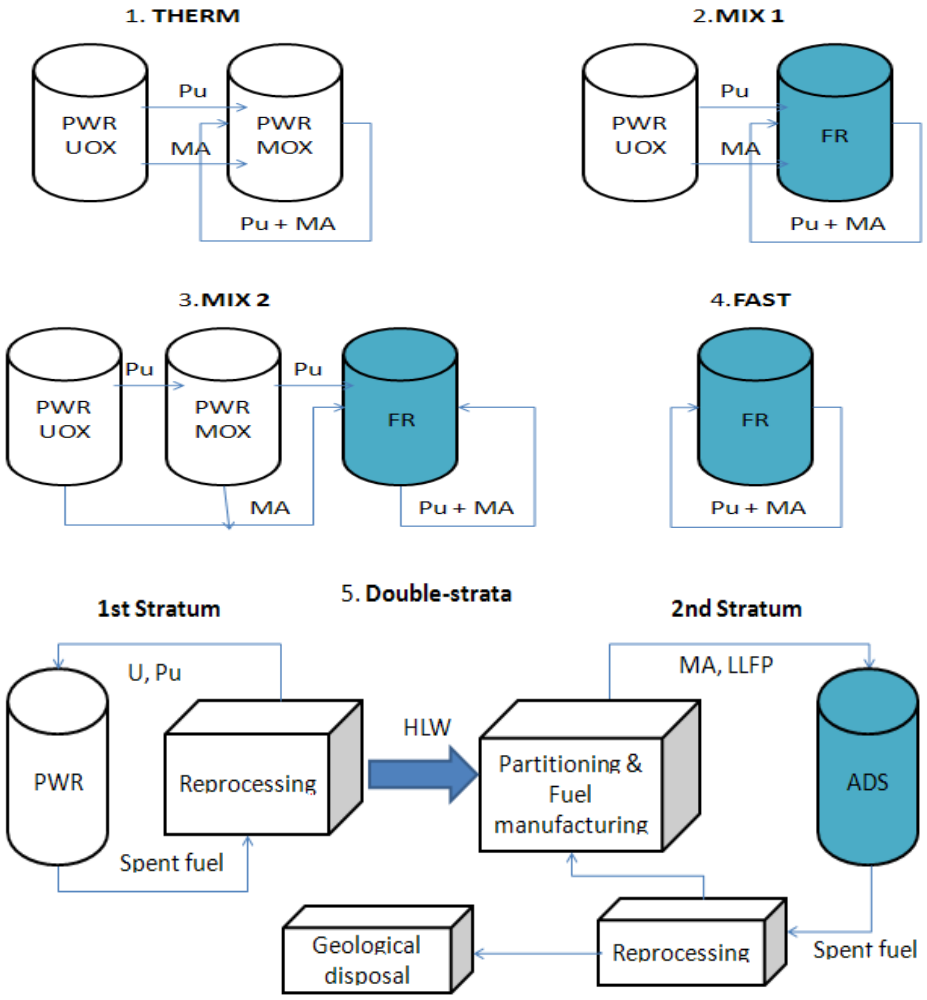

Figure 2.1 Showing the different compositions of the potential reactor parks [4].

With the research conducted so far on these different models, it can be concluded that simultaneous management of MAs and plutonium can make the nuclear industry sustainable. Sustainable meaning that there will be equilibrium between production and consumption of the MAs. The radiotoxic inventories would be dramatically reduced, with the only remaining radiotoxic inventories coming from the partitioning and reprocessing stage. The partitioning and reprocessing facilities would have to be in close vicinity to the reactors to reduce the transport of the radioactive materials.

The **THERM** alternative suggests that different PWRs would be dedicated to different fuels. This is because the Pu concentrations in the MOX fuel would be too high after the second recycle. Therefore as explained in Chapter One, a transmutation in the BWR would be preferred to handle the MAs. However, this scenario needs much more research and development.

The **MIX 1**, **MIX 2** and **FAST** reactor fuel cycles all use a fast neutron spectrum reactor (FNR), which could use either the FR or ADS methods to burn the MAs. In these FNR schemes, it is proposed to recycle neptunium homogeneously and mix it in the core. The americium part of the processed used nuclear fuel would be placed in the peripheral ring of the core in the form of targets. The curium part would be separated out and stored for a century to let the Cm 243 and Cm 244 decay into Pu, which would in turn be mixed with standard MOX. The rest of the Cm, mainly Cm 245, would be mixed in with the americium targets [16, 17].

The **double-strata** fuel cycle has the advantages of mitigating the reactivity safety parameters due to the use of a sub-critical reactor together with ADS. The other advantage of using ADS is the flexibility of design due to the non-criticality condition of the reactor. From the reprocessing facility in the second stratum only radioactive used nuclear fuel without long-lived nuclides will be sent to the geological repository for final storage for less than 500 years [41].

However, all these strategies need more research and development before any of the models would become a reality. Most of these results has come from the research of Am and Np in different reactors (FBR, PWR) in homo- and heterogeneous recycle modes.

2.2 Separation and Partitioning techniques

For separation and partitioning techniques to work and make AFC possible, good separation techniques need to be developed on an industrial scale for the MAs. There are two different techniques on how to separate the actinides, depending on the type of fuel, composition and purpose with the separation. The two methods are the aqueous and the pyrochemical, non-aqueous, methods.

Aqueous

The only aqueous technique used in full scale is called the PUREX process which is a hydrometallurgical technique. It dissolves the irradiated fuel in nitric acid and then mixes the aqueous liquid with an organic solvent (e.g. kerosene) containing tributylphosphate (TBP) which in turn extracts the selected metals into the organic phase. The process is called liquid/liquid solvent extraction and relies on the differing distributions for various elements and oxidation states between the two fluid phases. With a concentration of 30% TBP the process can separate a mixture of plutonium and uranium from the used nuclear fuel and with some adjusting of the valence of plutonium it may be partitioned into an aqueous solution. By changing the concentration of TBP it is also possible to separate out neptunium [18]. Neptunium is not a very big problem, because it can easily be recycled into the MOX fuel or be kept separated until transmutation is a viable option. There has been discussions on what to do with the Am and Cm part of the TRU. The idea is to separate them together with the lanthanides (fission products) that have three valence electrons (Ln(III)), because Am and Cm are also in a trivalent oxidation state after the PUREX process (Am(III) & Cm(III)). The group name of Am(III) & Cm(III) is An(III),

which is an acronym for actinides(III). The method called DIAMEX process, using a **DIA**mide **EX**tractant, separates the Ln(III) and An(III) from the rest of the FPs and nitric acid. The next step uses more diluted nitric acid that binds An(III) instead of Ln(III). This method is called SANEX (selective actinide extraction). Finally, by oxidation of the Am(III) to Am(IV), the Am(IV) can be separated from the Cm(III) by a TBP solvent [19]. The flow chart of this method is shown in figure 2.2.

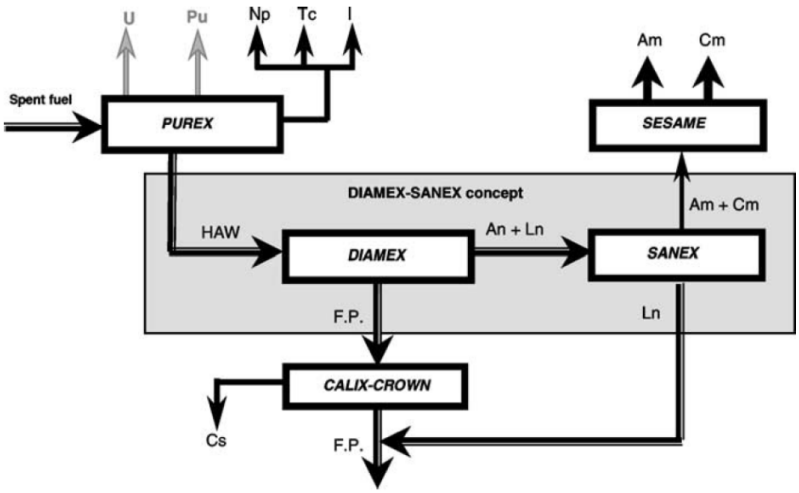

Figure 2.2. This diagram shows the separation processes involved in extracting Cm [19].

These separation methods are still at a testing stage and need more refinement before being implemented on an industrial scale. The americium could, after the separation from curium, be recycled with LWR-MOX or FR-fuel or be burned with an ADS. The curium part is still a mystery to be solved, because of its very high radioactivity. Curium is such a high heat and neutron source that it is very hard to recycle it in the LWR- or FR-MOX fuel. In the LWR the production of californium is a strong neutron emitter that creates problems to the new fuel creation. The PUREX process is a good method of separating out the actinides from the used nuclear fuel, but there needs to be more research on the process to make the used nuclear fuel stream "non-TRU" [18, 20].

Pyrochemical

Pyrochemical processes are conducted at very high temperatures, often above 500^{o}C and use distillation, electrolysis or oxidation as a means of separating the selected elements. Many pyrochemical processes have been studied, even before the PUREX methods, but not that many have been taken beyond the research and development stage. Some of the advantages with the pyrochemical process over the PUREX process are that it can avoid the presence of moderating materials lowering criticality risks, is more compact and produces smaller waste volumes. However the relatively low efficiency in recovering the minor actinides makes it less suitable if the purpose is to recycle americium or separate curium for a possible use in RTGs. This technique is also preferred when thinking of using shorter cooling times. The method has only been tested in a model scale with the use of a large bulk of alkali metal salts (LiCl-KCl, LiF-

CaF2 or NaCl-KCl to separate the actinides. This method could work well as a pre-separation step before PUREX or as an autonomous process to obtain dry reprocessing with shorter cooling times. If this could work on an industrial scale it may shorten the time interval for a complete recycle sequence [20, 21].

2.3 The problem with Cm (radiation and amounts produced)

The implementation of the advanced fuel cycle compared to the reprocessing fuel cycle would not drastically raise the dose exposure to workers. The part of the recycle process where the most change will be is in the reuse of the fuel and in the fabrication of the targets. The separation processes suggested for use in the advanced fuel cycle, as mentioned above, for the curium would imply the need to reinforce the shielding. This is mostly due to the very high α, γ and neutron radiation from the radioactive decay from these elements. The radiation shielding would in fact need reinforcement throughout all the recycling facilities for Am and Cm. That is why these partitioning facilities for the MAs should be built in close vicinity to the already existing reprocessing plants for U and Pu so as not to increase the costs due to even more shielding.

The long-term (300,000y) radiotoxic inventories would be reduced but with the cost of the short-term exposure from the partitioning process. If this model would be a reality it would not have to increase the overall impact on the environment, but set higher operational requirements on the nuclear facilities concerned.

The already existing mined uranium would extend the fuel base by a factor of 100 in the advanced fuel cycle. If both the Pu and MAs would be recycled, this being 13,000t to 16,000t based on future needs of 62,500t to 82,800t worldwide in the year 2015. Only these figures alone should be enough for the government to rethink the geological disposal option and try to make a better mixture of new nuclear power production to create the AFC model [22].

Amounts and Suggested ideas

A LWR nuclear park that produces 61.3 TWhe will have to treat the annual inventories after 10 years discharge of 107 kg neptunium, 109 kg americium and 6 kg - 12 kg of curium [23]. The variation of 6 kg is due to the fact that different isotopes of curium have different half-lives and alpha decays into other elements at different rates, for example Cm 242 has a half-life of 162.8 days compared to 18.1 years for Cm 244. There are strategies today that suggest having an interim storage of the curium isotope Cm 244, because of its very high activity, which makes it hard to handle as a target [22]. After a century or so the Cm 244 will have decayed into its daughter nucleus Pu 240.

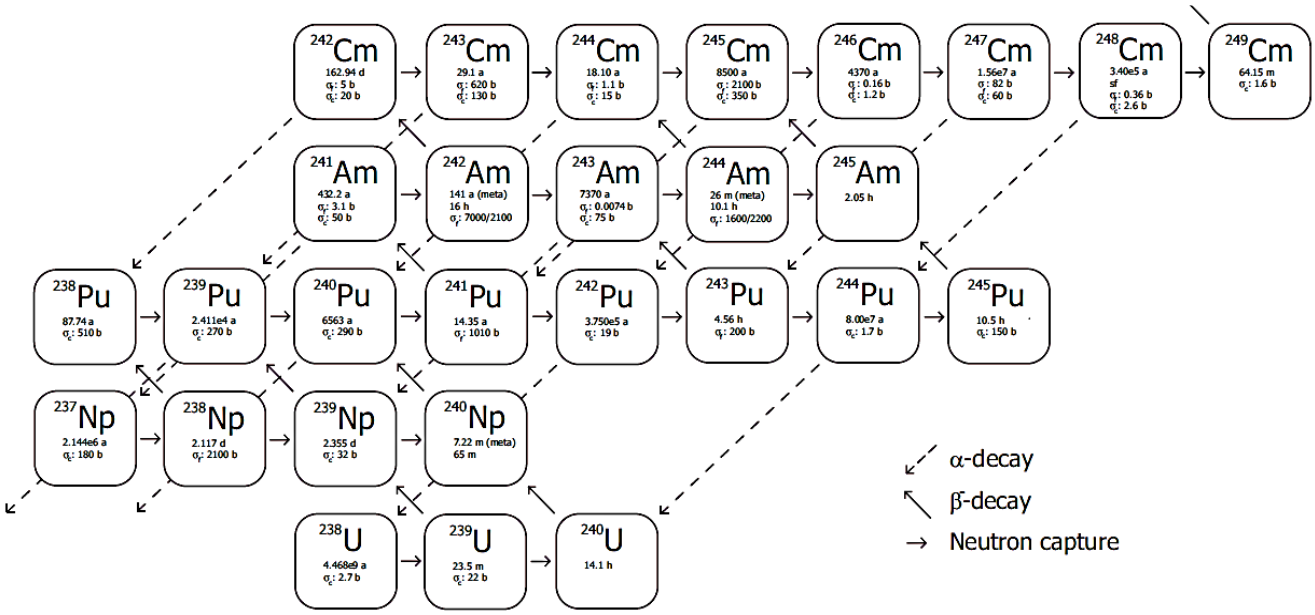

Figure 2.3. The decay chains for the Minor Actinides (Np, Am, Cm) [2].

The partitioned MA fractions in the used nuclear fuel (the different groups of nuclides or even the individual nuclides) need to be stored for a very long time before the necessary special reactors will be developed to provide transmutation. To achieve a maximum radiotoxic inventory decrease, all of the curium needs to be taken care of. The Cm 245 (half-life 8,530y) that is a very fissile material poses no problem. Cm 242 alpha decays very fast, because of the short half-life of 160 days. The problem is Cm 244 with a half-life of 18.1 years that is also highly active and is very difficult to make targets from. In fact the neutron dose rates around the blending glove box would increase with a factor of 100 [22]. That is partly why it is suggested to let all the curium sit in an interim storage room for around 100 years to let the Cm 244 alpha-decay to Pu 240 and then recycle the mixture of Pu 240/Cm 245. As mentioned earlier, curium would require a lot of further investigations before it could be stored safely until decaying into Pu. The neutron and heat emission from the Cm that will be stored separately will require much more research and development before that can be a reality, if even possible. The problem with separating out the Pu part from the MAs is the proliferation risk, because of Pu's high fissile properties. Therefore this kind of separate storage would require a highly secure facility [42]. A benefit from separating the Cm is that the heat load and cooling time would be reduced, implying that the other elements could be processed into fuel faster after discharge [22].

In the AFC strategy the separation of MAs, LLFP and Pu creates many new radiotoxic forms that could be taken care of accordingly to the processes explained. The developing of better and safer ways of storage and separation is an issue requiring more research and development. If achieved and implemented into the nuclear park there are still problems with the Cm part to really reduce the long term hazard. The next chapter will explore these other possibilities in a technical way to give some other options to the curium problem.

2.4 Radiation and Heat Production

A table of the heat production from 1 gram from the different curium isotopes -242, -243, -244, -245 and -246 is shown below. The number of atoms (N) is calculated by Avogadro's number divided by the isotopic mass number, because the mass number of these isotopes are almost the same the number of atoms contained in 1 gram is 2.478 * 10^21 for the different isotopes. The half-life and the Q-values of the different isotopes were found at "www Table of Radioactive Isotopes" [36]. The decay constant is calculated from the well-known equation $\lambda = \ln(2)/T_{1/2}$, where one year is 365.25 days, and finally the Rate of Decay is calculated by multiplying the decay constant with the number of atoms.

Isotope	Half-life (Y)	Decay constant (λ)	Rate of Decay (R) in Bq	Q-value (keV)
Cm 242	0.446	4.925*10^-8	1.220*10^14	6215.56
Cm 243	29.1	7.553*10^-10	1.872*10^12	6168.8
Cm 244	18.1	1.214*10^-9	2.996*10^12	5901.65
Cm 245	8500	2.586*10^-12	6.356*10^9	5623.5
Cm 246	4730	4.647*10^-12	1.138*10^10	5474.8

Table 2.4. Some of the constants and values for the different curium isotopes.

By using the equation for radioactivity (eqn. 2.4), which gives the number of radioactive decays per second, and then multiplying this result by the Q-value, a diagram can be produced that shows the heat production with time. In Diagram 2.4 a plot of the heat production from Cm 244 transforming into Pu 240 by alpha decay is illustrated. The diagram 2.4 also shows the heat production from the daughter nuclide, Pu 240, which is illustrated in the diagram as the green line. Pu 240 has a half-life of 6563 years and a Q-value of 5256 keV and will therefore produce a heat output for many years after the decay of Cm 244. The red line gives the total heat production from the two elements over time. The heat production from the daughter nuclide is calculated from the Bateman equation 2.5. The diagram 2.4 has a logarithmic base 2 heat power scale and starts at -8 ($2^{-8} = 0.004$) watts and goes up to 2 ($2^2 = 4$) watts. The heat production from Cm 244 is decreasing in a linear curve, which is expected in a logarithmic base 2 diagram because of the half-life. Even though the daughter nuclide Pu 240 continues generating energy over a much longer time the half-life is much bigger and therefore the energy is very low.

$$A_{Cm244} = \lambda_{Cm244} * N_{Cm244} * e^{-\lambda_{Cm244} * t} \qquad \text{(eqn. 2.4)}$$

$$A_{Pu240} = - N_{Cm244} * \frac{\lambda_{Cm244} * \lambda_{Pu240}}{\lambda_{Pu240} - \lambda_{Cm244}} * e^{-\lambda_{Pu240} * t} - e^{-\lambda_{Cm244} * t} \qquad \text{(eqn. 2.5)}$$

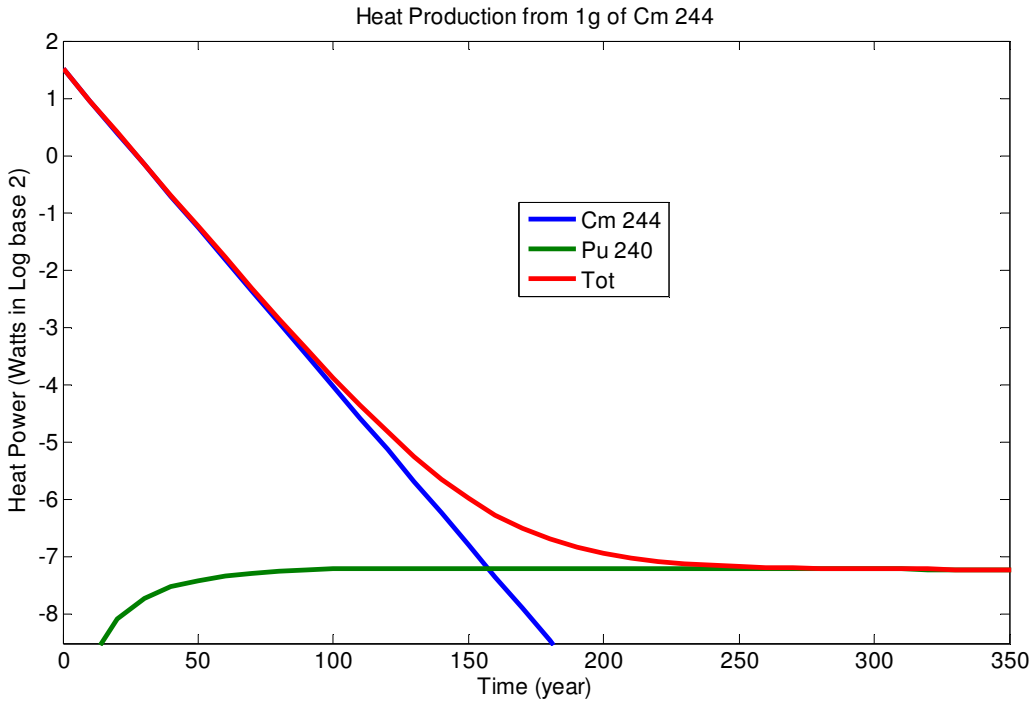

Diagram 2.4. Heat production from 1 gram of Cm 244 and its daughter nucleus Pu 240 together with the total heat production in 350 years.

The different curium isotopes have different half-lives and different Q-values and will thus have different heat production from that shown in Diagram 2.5. In diagram 2.5 the heat production from the isotopes Cm 242, Cm 243, Cm 244, Cm 245 and Cm 246 are shown. The heat production from Cm 242 is dominating the total heat production in the beginning of the decay as it starts generating a heat output of 122Watts. The high heat release from Cm 242 comes from its very short half-live and only after a couple of years the Cm 242 has decayed into Pu 238, which has a half life of 87.7 years and a Q-value of 5593. The yellow line shows the total heat production from the five different isotopes.

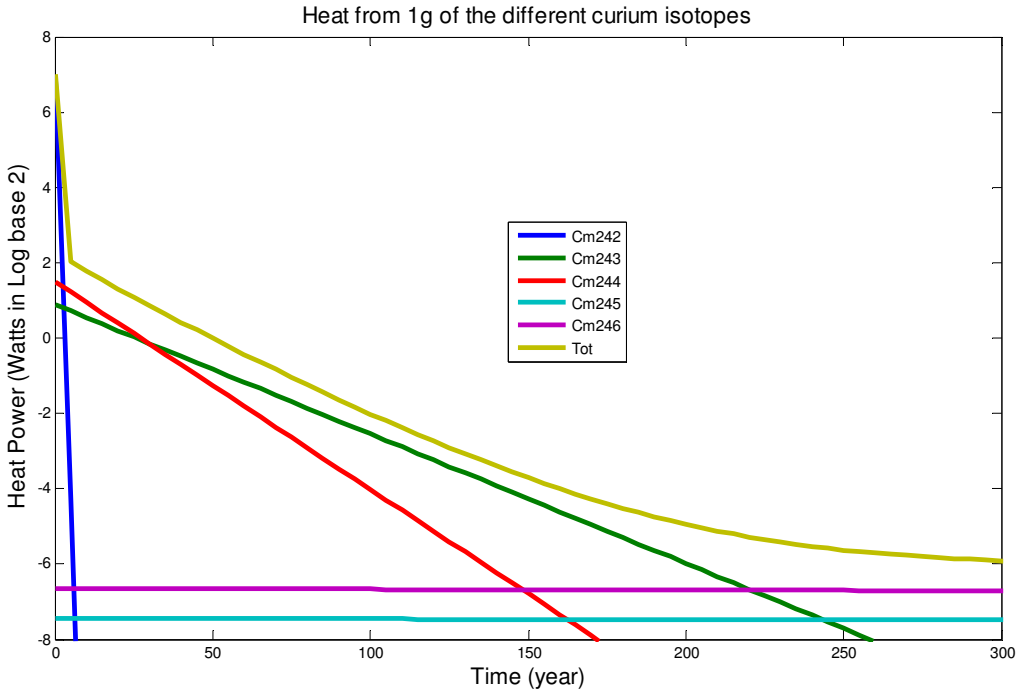

Diagram 2.5. Heat production from 1 gram of the different isotopes of curium vs. time.

The two curves from the isotopes of Cm 245 and Cm 246 show them generating only around 0.008 W_t mostly due to their long half-lives. This makes their power output very long, but also obviously making the curium radioactive for many years.

By studying Diagram 2.5 it is clear that the three isotopes of Cm 242, Cm 243 and Cm 244 are decaying much faster than the isotopes of Cm 245 and Cm 246, generating a higher heat output. These three isotopes could be useful for generating energy for an intermediate time period and because of their alpha-decay into their daughter nuclide of plutonium (e.g. in Diagram 2.4 for Cm 244), the total heat output lasts for a longer time. The Cm 245 and Cm 246 would not be very useful for generating heat due to their stable atomic structure resulting in the very long half-life. From Table 1.1, which shows the different amounts of curium isotopes produced from an average thermal burn-up reactor, and Diagram 2.5, it is reasonable to argue that Cm 244 would be the isotope best suited as a power source in a space satellite.

Chapter Three will investigate the possibilities for sending spent fuel into space and also explore the advantages of using Cm 244 as an energy source to power a spacecraft.

Chapter 3

The use of curium in space

The idea of using spent fuel in space has been under consideration for a long time. This could entail gathering up all of the TRUs and, with the help of a big rocket, transport the used nuclear fuel into space. A rocket of that size fully loaded with these amounts of radioactive material could have serious consequences if the rocket would explode or in some other way have an accident [24]. That is why this paper suggests the use of the curium part only of the TRUs, which is the most prominent problem for the AFC as explained in previous chapters. By only sending up the curium part of the TRUs the payload mass would decrease significantly. Another feature of this strategy is to use a small satellite that could be put in a low-orbit around the earth and the heat output from the curium could then be used to power the spiralling of the satellite out into deep space. Using a small satellite has also the advantage that it can piggyback on a rocket with a large primary payload, minimizing the costs. In fact, the same rocket might launch more than one satellite loaded with curium and thus further lower the costs.

3.1 Design of satellite

For this mission to work there are a couple of restraints to the design of the satellite. The satellite needs to be small, robust, cheap and safe. That is why this paper will look deeper into the SMART-1 project that utilized some of the ideas that could be used for a disposal project like that suggested by this paper. (SMART is the acronym for **S**mall **M**issions for **A**dvanced **R**esearch in **T**echnology.)

SMART-1

SMART-1 was a project initiated by the European Space Agency (ESA) in an attempt to demonstrate manufacture and implementation of a small and inexpensive satellite. The design was developed by Rymdbolaget in Solna and then built by Saab Ericsson Space in Linköping [25].

The SMART-1 satellite has the dimensions of a one cubic meter box weighing 367 kg, of which 84 kg was the propellant gas Xenon. Its main goal was to demonstrate electric propulsion system

for deep space. This was done with an ion engine powered by two electric solar panels. An Ariane-5 rocket put SMART-1 in an elliptical low earth orbit (LEO), from where it used the ion engine to spiral its way out. During 100 days of spiralling, SMART-1 was in the Earth's radiation belt. The radiation caused problems with the engine, the main computer and the star trackers, but none of were fatal to the spacecraft.

Another instrument (OBAN) on board SMART-1 would test a new autonomous navigation system that could guide future satellites with the help of star trackers on the side of SMART-1.

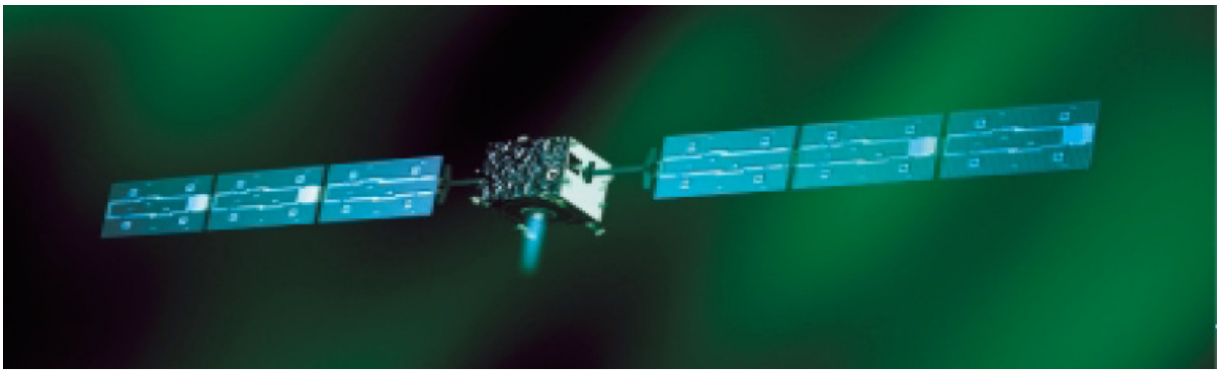

Figure 3.1 SMART-1 in the radiation belt [25].

In Figure 3.1 the blue jet is the ion gas being neutralized by electrons.

The energy production on SMART-1 consisted of two solar panels that deliver 2 kWe (fig. 3.1). The downside of using solar arrays is that they do not work very well when far away from the sun, the efficiency falls by distance squared. The energy production will also cease when the sun is eclipsed by a planet or moon. Micrometeoroids in space could also be a problem if they hit the solar arrays.

The total cost of building and launching the SMART-1 satellite was €110 million [25, 26].

3.2 Energy production

The development of nuclear power for space applications started shortly after the end of World War II. The two different approaches that raised the most interest were the radioisotope thermoelectric generator (RTG) and a small nuclear reactor controlled autonomously.

The first RTG was launched in 1961 by the U.S. and has been steadily improved since. The RTG is ideally suitable for autonomous missions because of its long life time, compact size, and high reliability and sturdiness. RTGs are not affected by radiation, distance from the sun or other environmental effects, which make the RTGs perfect for long missions out of the solar system.
The radioisotope thermal generator consists of two parts. The first being the radioactive heat source and the second is the thermoelectric generator [27]. The RTG and the two parts it consists of will be further investigated in the next part.

The System Nuclear Auxiliary Program (SNAP-10A) showed in 1965 the viability of an automatically controlled nuclear reactor, which was maintained in a subcritical condition to prevent any accidents to occur in case of a fire, explosion or projectile [27]. The SNAP-10A weighed 435 kg, including 37 fuel rods in the reactor. A reactor of this size loaded with curium/uranium/plutonium could have devastating consequences if it malfunctioned in any way during the launch. This is why an automatically controlled reactor is not the best option to power a satellite in a safe way. Creating fuel from only curium would also be impossible due to the low fraction of delayed neutrons [35]. This type of reactor would also need autonomous control for a very long time, until the satellite is outside the solar system and placed in at a final destination. An autonomous system that would not shut down during these long timescales would have to go through many and very expensive tests before being approved. These problems and risks make the automatically controlled nuclear reactor a bad choice for the kind of project this report is suggesting.

Radioisotope Heat Source

The radioactive heat source consists of a material that undergoes natural decay and in this process releases thermal power. Figure 3.2 shows the schematic of the standard module used in all new radioisotope thermoelectric generators used by the U.S. namely the General Purpose Heat Source (GPHS). The fuel used is Pu 238 in oxide form, which has a half-life of 87.7 years. Thus the reduction in thermal power each year is only 0.8%, making the fuel ideal for long term missions.

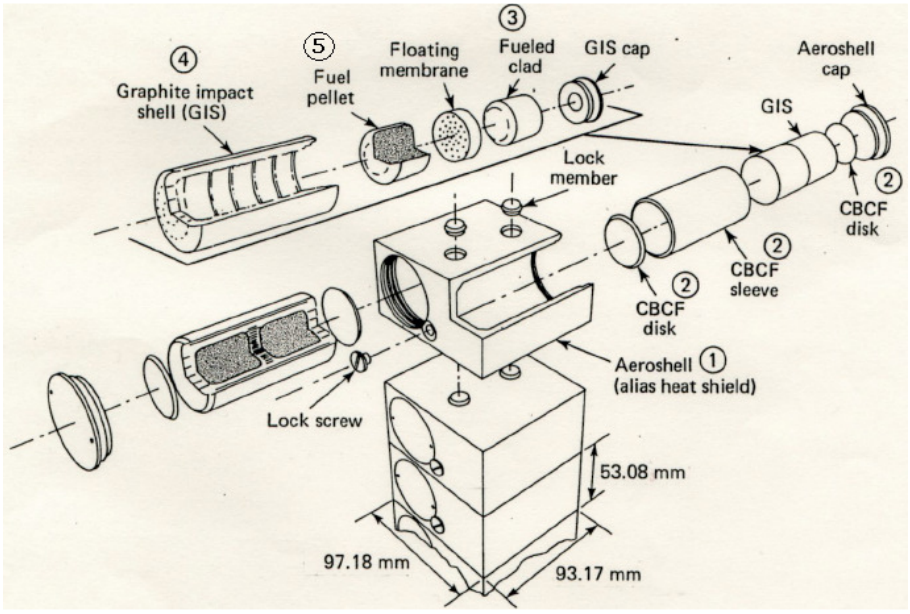

Figure 3.2. Diagram of the General Purpose Heat Source -modules stacked on top of each other [28].

Figure 3.2 shows a cutaway design of a GPHS-RTG and one can see the many safety precautions made to the fuel pellet and packaging to secure the system in case of a possible accident. Each module consists of five main elements, the fuel (5), the graphite impact shell (4), the fuel cladding (3) the carbon-bonded carbon fibre insulation (2) and the Fine Weave Pierced Fabric aeroshell (1). The fuel is encapsulated in a 0.55mm welded iridium alloy clad to resist oxidation

in case of a crash. The cladding is also chemically compatible with the fuel and the graphite components can survive high temperatures in case of re-entry into the atmosphere. The fuel and clad is also called a "fueled clad". The fueled clads are then placed two and two into the cylindrical graphite impact shell, where they are protected from impact accidents. The graphite impact shell are put in the next cylindrical case, namely the carbon-bonded carbon fibre, this is to insulate the graphite impact shell inside the aeroshell and limit the peak temperature, but also maintain a sufficient temperature to ensure its duplicity during inadvertent re-entry and impact. The Fine Weave Pierced Fabric is very safe and has high safety margins in case of impact on hard surfaces, thermal stress associated with re-entry and postulated launch vehicle explosion. Where safety is the principal design for the General Purpose Heat Source they must also go through substantial testing before being approved for launch [28].

Every individual module consists of four fuel pellets, where the dimensions of the pellets are 2.76 cm in diameter and 2.76 cm in length. These modules can now be packed on top of each other to generate heat to the thermoelectric generator. The general criterion for the fuel is that:

- The half-life of the isotope is sufficiently long to produce heat energy for the full time of the mission.
- The safe handling of the type and quantity of the emission to produce the fuel pellets can be achieved.
- The specific power (heat per mass) and power density (heat per volume) is high enough.
- It is noncorrosive, water-insoluble, has good engineering properties and is chemically stable at high temperatures.
- The production cost is low.
- The proximity to sensitive instrumentation and biological systems.

[28, 29, 30]

Thermoelectric generators

There are a couple of different conversion units that use heat to produce electricity. The one that has been used for all of the radioisotope thermoelectric generators used by the U.S. is the solid-state thermoelectric power generator that exploits the Seebeck effect[4] to produce electricity. The Seebeck effect emerges when there is a temperature difference between two different conducting materials. The effect causes a voltage that produces a current loop and a magnetic field in the conductors if they are connected in a closed loop (fig 3.3).

[4] In 1823 The German physicist Seebeck discovered that two dissimilar conductors in a closed loop heated in one of the junctions would cause a compass needle to deflect due to the presence of a magnetic field. Later it was found that the temperature difference in fact created a current, which then created a magnetic field. Seebeck had discovered the thermoelectric effect! [49]

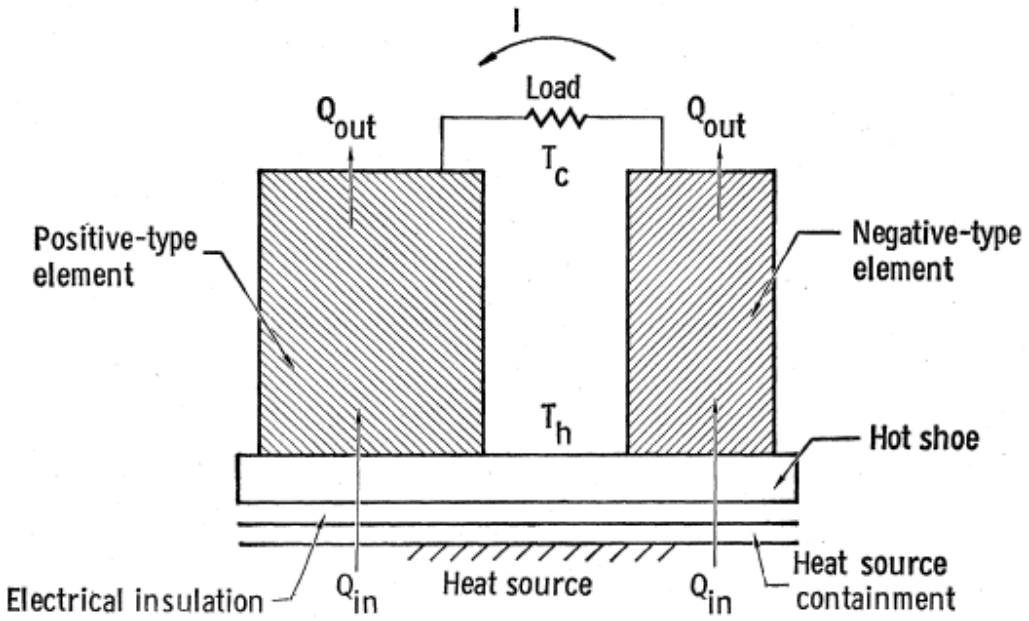

Figure 3.3. A graphical view of thermoelectric energy conversion by the Seebeck effect [31].

The thermoelectric electro-motive force (EMF) that is created over the junction is in the order of μV to several mV. The thermoelectric electro-motive force created depends on the material used and an estimate of the voltage can be made by eqn. 3.1 with a hot side (T_h) and a cold side (T_c).

$$V_{junction} = \int_{T_c}^{T_h} (\alpha_p (T) - \alpha_n (T)) \cdot dT \qquad \text{(eqn. 3.1)}$$

Seebeck investigated many different materials and ordered them in the product, ασ, where α is the Seebeck coefficient and σ is the electrical conductivity. Equation 3.1 describes the thermoelectric voltage created over a temperature difference (T_h-T_c) between the materials p and n. The Seebeck coefficient is measured in volts per degree or more often in μV per Kelvin.[5] In figure 3.3 Q_{in} and Q_{out} represents the heat flow of the thermoelectric device, where Q_{out} needs to be rejected to keep that side in T_c. These devices are thermally connected in parallel and multiple junctions are connected in series electrically to give a conversion device with the required voltage and power. In a radioisotope thermoelectric generator using a thermoelectric conversion device, only 6.5-7% of the heat produced by the radioisotope is converted to electricity. In space, radiation is the only option (convection or conduction cannot be used for obvious reasons) to dissipate the excessive heat of 92.5-93%. This puts high restraints on the engineering design of the satellite. The fact that the heat from the radioisotope decreases exponentially with time from natural decay led to the use of a shutter. The shutter controls the radiating surface of the heat source so that there is a constant heat output and electricity production from the beginning of the mission (BOM) until the end of mission (EOM) [30, 31, 49].

[5] The best conductors to use in thermoelectric purposes are those metals with a high Seebeck coefficient, most metals possesses coefficients around 10 μVK^{-1}. Seebeck coefficients around these numbers give associated generating efficiencies of less than 1%. In 1930 the development of synthetic semiconductors with a Seebeck coefficient of more than 100 μVK^{-1} can provide efficiencies of 5% [49].

Alternative energy conversion systems

The currently used radioisotope thermoelectric generator is the GPHS-RTG, which includes 18 General Purpose Heat Source modules to produce 285 watts of electric power. The next generation of radioisotope thermoelectric generator is called the Multi-Mission Radioisotope Thermoelectric Generator (MMRTG), which makes use of the same heat conversion technique but only uses 8 General Purpose Heat Sources and will have a >6% conversion ratio that produces ≥110We at beginning of the mission. The weight of the Multi-Mission Radioisotope Thermoelectric Generators is estimated to be less or equal to 45 kg. The Multi-Mission Radioisotope Thermoelectric Generator is based on the GPHS-RTG and will therefore be much easier to develop and certify than other designs.

One alternative power conversion design makes use of a Stirling electric generator and is therefore called the Stirling Radioisotope Generator (SRG). Unlike the Multi-Mission Radioisotope Thermoelectric Generator, the Stirling Radioisotope Generator uses a mechanical system to produce electricity, which could be a problem in environments with high radiation. The mechanical system in the Stirling Radioisotope Generator uses helium that expands from heat entering the hot side and then rejects waste heat into the cold side. The fast change in pressure causes a piston to move back and forth, creating electricity. The Stirling Radioisotope Generator will have a conversion ratio over 20% and only use two General Purpose Heat Source-modules with the same electric production as the Multi-Mission Radioisotope Thermoelectric Generator. The advantage with the Multi-Mission Radioisotope Thermoelectric Generator is that it has no moving parts and is therefore the more robust choice. It is based on a reliable technique and the higher predicted electric output at beginning of the mission. The advantages with the Stirling Radioisotope Generator is the higher conversion efficiency, which results in a lower fuel mass, less radiation and lower unit mass.

Other conversion technologies such as the thermionic, thermo-photovoltaic (TPV) and Brayton conversion systems need further research to improve conversion ratios and mass. Maybe there is also a possibility to use a hybrid of these ideas [30, 32].

3.3 Ion thrust

To propel the satellite “forward”, the ion thrust engine is a great choice compared to the conventional chemical engine. The ion engine employs the early work of the American physicist E.H. Hall way back in 1879. He discovered that a current flowing in a magnetic field will create an electric field in a transverse direction to the current flow [43]. Equation 3.2 shows how a current with velocity (v) and charge (q) in the x-direction going through a magnetic field (B) in the z-direction creates a Lorentz force (F_L). The Lorentz force deflects the electrons and a charge difference is created over the semiconductor. This charge difference gives rise to the electric field (E_y), which produces a force ($F_y = qE_y$) that cancels out the Lorentz Force (eqn. 3.3). These equations put together give the steady-state Hall field $E_y=v_xB_y$.

$$F_L = -q\, v_x\, B_z \qquad \text{(eqn. 3.2)}$$

$$F_y - F_L = 0 \qquad \text{(eqn. 3.3)}$$

The ion engine requires an ionized material that can be accelerated through the electric field to create thrust. Ionizing the propellant is energy consuming, which is why the propellant material should have a high mass/ionization energy ratio. Xenon gas is the most commonly used propellant, because of the relatively high mass number (131) and low ionization energy (1170.4 kJ mol^{-1}) needed. Another benefit with Xenon gas is that it does not give rise to much erosion to the satellite. The ion engine used for the SMART-1 needed 1350 W of electric power to give a very small thrust of 0.07N. With the vacuum of space a very small but continuous acceleration can build up a great velocity over time and even send a satellite out of the solar system. The ion engine shoots out the propellant gas much faster than a chemical engine for a given amount of energy and thus can go on for a longer time in outer space, making it the most efficient propulsion system for in-space transportation. The drawback is that absolute thrust is low. So an electric propulsion system cannot launch a vehicle from the earth or other planet or moon, for that matter. [25].

3.4 Space laws and Navigation system

In space (generally recognized to start at 100 km above sea level) there are laws and treaties that are recognized to different levels by different countries. These laws for space were developed after the first satellite “Sputnik 1” was launched into space by the Soviet union 1957. The most widely adopted space treaty among countries is that of the Committee on the Peaceful Uses of Outer Space (COPUOS), which is an international forum for the development of international space laws. COPUOS has been able to conclude five international treaties and agreements and these are:

1 The Declaration of Legal Principles Governing the Activities of States in the Exploration and Uses of Outer Space (General Assembly resolution 1962 (XVIII) of 13 December 1963)

2 The Principles Governing the Use by States of Artificial Earth Satellites for International Direct Television Broadcasting (resolution 37/92 of 10 December 1982)

3 The Principles Relating to Remote Sensing of the Earth from Outer Space (resolution 41/65 of 3 December 1986)

4 The Principles Relevant to the Use of Nuclear Power Sources in Outer Space (resolution 47/68 of 14 December 1992)

5 The Declaration on International Cooperation in the Exploration and Use of Outer Space for the Benefit and in the Interest of All States, Taking into Particular Account the Needs of Developing Countries (resolution 51/122 of 13 December 1996)

[51]

The treaty on "Use of Nuclear Power Sources in Outer Space" is the most relevant treaty to this thesis and the general goals for radiation protection and nuclear safety states:

"States launching space objects with nuclear power sources on board shall endeavor to protect individuals, populations and the biosphere against radiological hazards. The design and use of space objects with nuclear power sources on board shall ensure, with a high degree of confidence, that the hazards, in foreseeable operational or accidental circumstances, are kept below acceptable levels" [52].

Propulsion in LEO with an ion engine powering a spiral path would be the lowest energy consuming course to manoeuvre the satellite out into space to get rid of the curium. To be able to spiral away from the Earth's gravitational pull, SMART-1 used the ion thrust for one third to one half of every orbit two times every week, for sixteen months. Image 3.4 shows how SMART-1 expands these elliptical circles to finally "jump" over to an orbit around the moon, which was the final destination of the SMART-1 mission.

As mentioned earlier in this report, one of the objectives with the SMART-1 mission was to test a new autonomous navigation system, which would be the cheapest choice for navigating these kinds of missions. There are other solutions to navigate the satellite out of the Earth's gravitation well that are mentioned below.

The satellite could be governed by an outside contractor such as the Swedish space corporation. There are many contractors that have much experience with full responsibility for managing, controlling and steering these types of satellites. They control the path of the satellite by the use of one or several ground based stations. This is conducted to adjust for the small deviations from the given direction due to various disturbance factors. The adjustments are done when the satellite is passing over the ground based stations by activating the small rocket engines on the sides of the satellite.

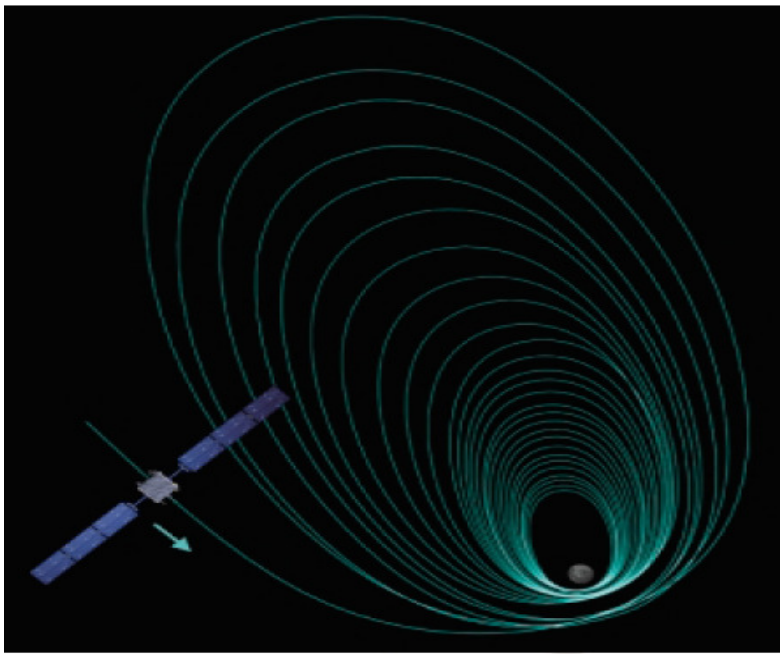

Figure 3.4. The elliptical path of SMART-1 [25].

Communication with the satellite is done via different radio frequencies. These frequencies are limited and need to be shared between satellites. To make sure that all the systems work and that there are no disturbances, there is a need for frequency coordination. This service and protection from intrusion is also provided by different contractors [33].

The final destination of the curium would be debatable, on one hand it would be a good idée to calculate a spiral path that would send the curium into the sun and get rid of the highly radioactive curium once and for all. This approach could be a good choice if the curium would never be needed and a poor choice if curium is someday found to be a useful material for future applications. Other ideas for the final destination of curium could be in an orbit around some close moon or planet such as Mars or in one of the 5 Lagrange points[6] such as those for the earth-moon system or the earth-sun system.

[6] There are 5 Lagrange points between a system of two objects where one of the object is orbiting the other in a steady orbit. The Lagrange points are named L1, L2, L3, L4 and L5 after the Italian-French mathematician Joseph Louis Lagrange. There are five different positions where a third object with negligible mass is in equilibrium with the two objects' gravitational pull. It is only L4 and L5, situated with 60 degrees angels away from the two objects that are stable. The other points, though, require minimal station-keeping fuel for spacecraft to remain stable. It is also possible for multiple objects to be placed in the points L4 and L5 [50].

Radiation damage to the electronics

During the satellites spiral trip to its final destination, the electronics on board will be submitted to external radiation from the sun, cosmic rays and internal radiation from the RTGs, which could cause the satellite to malfunction and in worst case crash. The spiral path chosen for this mission will further add to the external radiation part due to the fact that the satellite would have to be in the Van Allen belts for a much longer time than it would be in a more direct path.

The Van Allen belts is made up of two doughnut shaped clouds of ionized particles with the first belt located at 3000 km to 8000 km above the earths geomagnetic equator and the second belt at 15000 km to 25000 km. The inner belt is for the most part made up of highly energized protons and electrons where the upper belt is mostly made up of even higher energized electrons. The particles are trapped by the earth's magnetic fields where they bounce back and forth in a spiral manner of the poles. The energized particles originate from solar winds and cosmic rays and are lost by collisions with the atmosphere at the poles (i.e. Aurora Borealis and Aurora Australis) or in some other way managing to the escape the magnetic fields into space. The inner zone has intensity values of $2*10^9/cm^2$ sec amounts of unidirectional electrons with energy greater than 20 keV, $1*10^7/cm^2$ sec omnidirectional electrons with energies over 600 keV and $2*10^4/cm^2$ sec omnidirectional protons with energies over 40 MeV. The outer zone has intensity values of $\sim1*10^{11}/cm^2$ sec amounts of omnidirectional electrons with energy greater than 20 keV, $\sim1*10^8/cm^2$ sec omnidirectional electrons with energies over 200 keV and $\sim10^2/cm^2$ sec omnidirectional protons with energies over 60 MeV [53, 57].

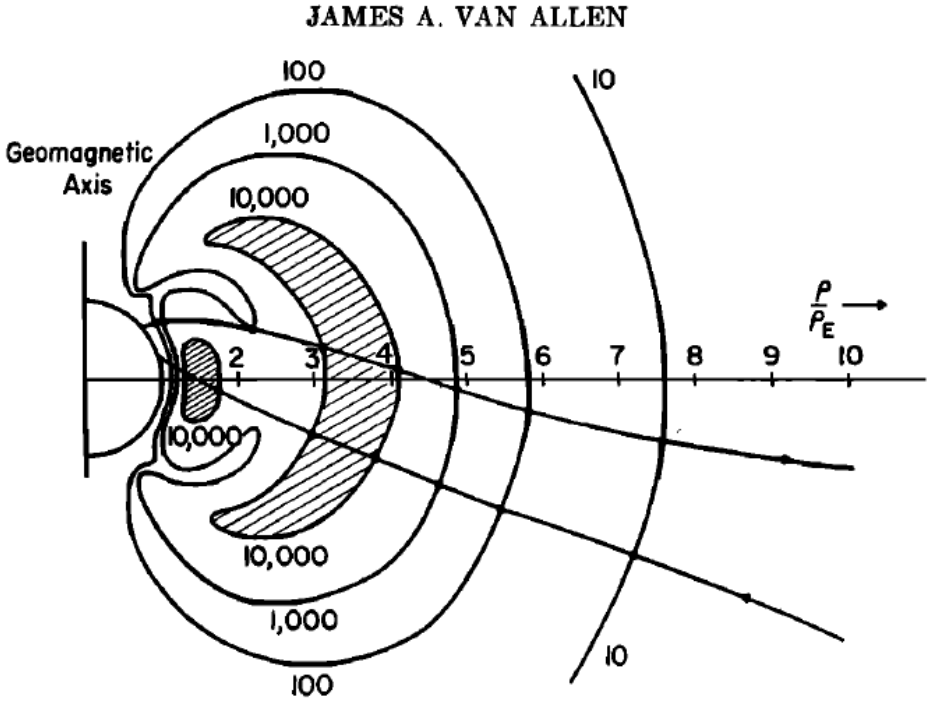

Figure 3.5. The van Allen Belts. Cross section picture of the intensity zones ranging from 10 to 10000 counts made by an Anton 302 Geiger tube. The different intensity regions mapped by Explorer IV and Pioneer III. The linear axis is measurement in earth radius (6371km). The trajectory of Pioneer III is also illustrated in the picture. The two shaded areas are showing the inner and outer Van Allen Belts [53].

The neutron radiation from the RTGs and the electrons and protons radiation from the Van Allen belts are the major sources of radiation damage on the satellites electronics. Stopping the highly energized electrons and protons requires a heavy density material such as Tungsten, Tantalum or Lead. To shield the electronics from neutron radiation the use of Hydrogen-containing polymers mixed with up to 20% of Boron additives has the effect that the hydrogen in the polymer slows down the neutron so the isotope of Boron, ^{10}B, can absorb the thermal neutrons [54].

3.5 Failed/Successful missions with curium

In 1970 the Apollo 13 mission suffered a severe technical event and nearly ended in a disaster but it was not in vain. The Apollo 13 mission was the third Apollo mission planned to land on the moon. Halfway to the moon an oxygen tank ruptured and a small explosion took place, causing the mission to be aborted. The crew managed to return safely to Earth and crash the shuttle in the Pacific Ocean. This "successful failure", as the mission was named, showed that a SNAP 27 Radioisotope Thermoelectric Generator, intended for an experiment to be left on the Moon, could withstand the enormous pressure associated with the re-entry of the lunar module. After the module splashed in the Tonga Trench it sunk down to a depth of 6.5 km where it still lays today. No measurement of any contamination has been made until 2008, so the SNAP 27 is assumed to be intact [30]. This accident and other extensive tests prove the stability of the GPHS modules and there are no concerns of the safety of these modules.

Successful missions with curium

The U.S has launched two successful missions where they used curium in the RHS, SNAP 11 (also called Surveyor) and Snap 13. The Snap 13 mission was intended to show that the heat from the Cm 242 isotope could generate electricity from a thermionic energy converter unit. The Snap 11 mission used a thermoelectric converter where the whole generator weighed 13.6 kg and was design to produce a total of 25 Watts. Both missions only had a designed lifetime of a quarter of a year, because of their short half-life of 160 days. The Snap 13 mission was designed to produce 12 Watts and the generator had a weight of 1.8 kg, length of 10 cm and a diameter of 10cm. These are the only known missions that have used curium as a power source for a space mission [31].

The next chapter will look at the isotopes and their quantities that would be best suited to use in a curium produced Radioisotope Thermoelectric Generator for powering a spacecraft on a spiral trajectory into deep space. Such a system eliminates the problem of geological disposal of curium.

Chapter 4

Results

This chapter examines whether the different techniques discussed in previous chapters can use curium in RTGs for space missions. The first chapter outlined the problems with nuclear power production today and explained that spent fuel production is the biggest problem. Chapter 2 continued to break down the used nuclear fuel problem and highlighted the main issues of handling the spent fuel. Chapter Two argued that for the time being the challenge for a future sustainable nuclear power cycle (including fast burners), was the curium problem. It also described the difficulties in producing mixed fuel or targets from curium. Chapter Three described propulsion techniques used in different types of space missions and focused in particular on the use of small satellites that operate with ion engines to spiral out into deep space starting from low earth orbit.

4.1 Calculations

Most of the curium in the used nuclear fuel consists of the isotope Cm 244 with a half-life of 18.10 years. From the calculations in Section 2.4, this would be the best suited isotope to use as fuel in the RTG. Here the Q-value for Cm 244 is 5901.61 keV [36] and N_A is 6.023 * 10^23.

$$\lambda = \ln(2) / T_{1/2} \qquad \text{(eqn. 4.1)}$$

$$N = m_{Cm\ 244} * N_A / M \qquad \text{(eqn. 4.2)}$$

$$P = \lambda * Q * N \qquad \text{(eqn. 4.3)}$$

Here:

λ : Decay constant, $T_{1/2}$: Half-life, N : Number nuclides, m(Cm 244) : Mass of Cm-244 [g], N_A : Avogadro's number, M : Molecular mass, Q : Q-value, P : Power.

Using the equations 4.1 to 4.3 in the present case with 1 gram of Cm 244 we have that:

$\lambda = \ln(2) / 18.10*365*24*3600 = 1.214*10\text{^-}9\ s^{-1}$

$N = 1*10^{-3} * 6.022*10\text{^}26 / 244 = 2.468*10\text{^}21$ atoms

$P = 1.214*10\text{^-}9 * 5901.61 * 10\text{^}3 * 1.6*10\text{^-}19 * 2.468*10\text{^}21 = 2.834\ Js^{-1}$

This isotope can produce 2.834 Watts of thermal energy per gram calculated from the equations 4.1 to 4.3

From Section 2.3 it was said that a LWR nuclear park that produces 61.3 TWhe will have to treat the annual inventories after 10 years discharge of 107 kg neptunium, 109 kg americium and 6-12 kg of curium each year. The isotope Cm 244 constitutes ~90% (89.28) of the total amount of curium generated. After 3 years the minimum amount of 18kg curium in the spent fuel could generate a total of 51.012 kW of thermal energy with the previous calculated thermal energy from 1 g of Cm 244. With the use of a Multi-Mission Radioisotope Thermoelectric Generator with a conversion ratio of >6% would give an electric energy of 3.060 kWe. This would be more than enough to drive an ion engine that required an electric power of 1.35 kWe [31].

The pellets used in the General Purpose Heat Source has a volume ($\pi \bullet r^2 \bullet h$) of 16.51 cm^3, where r is 1.38 cm and h is 2.76 cm. The density of curium in room temperature is 13.51 g/cm^3, therefore 223g would be needed to produce one pellet. The 18 kg of Cm 244 from the 3 years of Swedish used nuclear fuel would generate ~80 pellets. The Multi-Mission Radioisotope Thermoelectric Generator described previously required 8 General Purpose Heat Sources that hold 4 pellets each, by adding 2 GPHS the use of 2 Multi-Mission Radioisotope Thermoelectric Generators could be used. A satellite could be loaded with two Multi-Mission Radioisotope Thermoelectric Generators and the total weight for these two Multi-Mission Radioisotope Thermoelectric Generators would be around 90 kg. By sending up one satellite every 3rd year using this technique, an alternative approach to the burial of the spent fuel could be achieved. This would substantially reduce the Swedish environmental impact of nuclear power.

To protect people handling the Multi-Mission Radioisotope Thermoelectric Generators from different radiation such as α, β, γ and neutron radiation, an effective shielding material needs to encapsulate the Mission Radioisotope Thermoelectric Generators.

Exergy

Exergy is not to be compared with energy, which cannot be destroyed only transformed into different forms. Exergy is a measurement on how useful the energy in the current system is i.e. it considers the conservation of mass and energy together with the second law of thermodynamics. If the exergy always is preserved then any change would be done without losses and be fully reversible, which implies that time would lack direction and meaning. It is also important to consider the exergy of the total system.

Thus, so called “energy saving lamps” may consume less electrical energy to light up a room, but the total exergy is much worse because of the cost of manufacturing and disposal of this type of lamp. A very important question is thus: “Where is energy lost?” Potential energy has high exergy and is useful. It should gradually be transformed into other forms of energy while lost exergy eventually ends up as room temperature water (which is not very useful energy wise and thus has a low exergy).

It is important to consider the remaining exergy in the curium part of the intermediate stored nuclear spent fuel before deciding what should be done with it. Exergy could be compared to the efficiency of energy, such as Exergy = η * Energy, where η is a value between 0 and 1. Spent nuclear fuel has a η around 0.9, because of its high temperature and radioactivity.

From an environmental and efficiency point of view, it is important to understand the exergy when comparing different strategies for the spent fuel. Conversions of energy between the different steps in the process and therefore the exergy losses that are involved can be described by the following equation:

$$E_{in} = E_{out} + E_{lost} + E_{destroyed}$$

Here $\eta = E_{out} / E_{in}$

To build rockets, satellites, separate out curium, make RTGs to convert thermal energy into electricity with a conversion ratio of only >6 % generates a small E_{out}, a big amount of E_{in} and E_{lost} and no $E_{destroyed}$ (if we do not send the satellite into the sun). By burying the curium together with the rest of the spent fuel in a deep geological repository the process of building a repository, manufacturing capsules and bury it has to be considered and compared. Keeping the spent fuel in an intermediate storage facility will cost energy, E_{in}, but maybe the future use of spent fuel could generate a big E_{out} and compensate for this [55].

4.2 Shielding of curium

The most commonly used isotope for RTGs has been Pu 238 for space missions. Pu 238 decays mostly through alpha emission and the alpha particles of energy of 5 MeV in the radioisotope heat unit (RHU) generate the thermal energy. The production of Pu 238 is made in Russia by putting Np 237 in a high neutron flux where Np 237 will capture a neutron to make Np 238 (half-life 2.11days) and then by beta decay turn into Pu 238 [30].

There have been many propositions for alternative isotopes to use in the General Purpose Heat Source R.C. O’Brian et al., showed in their article “Safe radioisotope thermoelectric generators and heat sources for space applications” that different isotopes could be interesting if the right shielding was used. The fuels suggested were different dioxide isotopes, so called metal oxides. Metal oxides have the advantages of raising the melting point and not burning since they are already in the oxidized state. Since curium is a neutron emitter one could consider to enriching it with the ^{16}O isotope to reduce the total neutron yields. Reducing the neutron yields is central to

being able to handle the fuel in a safer way. The chemical processing technique alters the composition of natural oxygen (99.757% ^{16}O, 0.038% ^{17}O and 0.205% ^{18}O) ratios to even higher ratios of ^{16}O. This is because the alpha-neutron (α,n) reaction cross section for ^{17}O and ^{18}O is considerable higher for alpha particles in the range of 0-6 MeV [30].

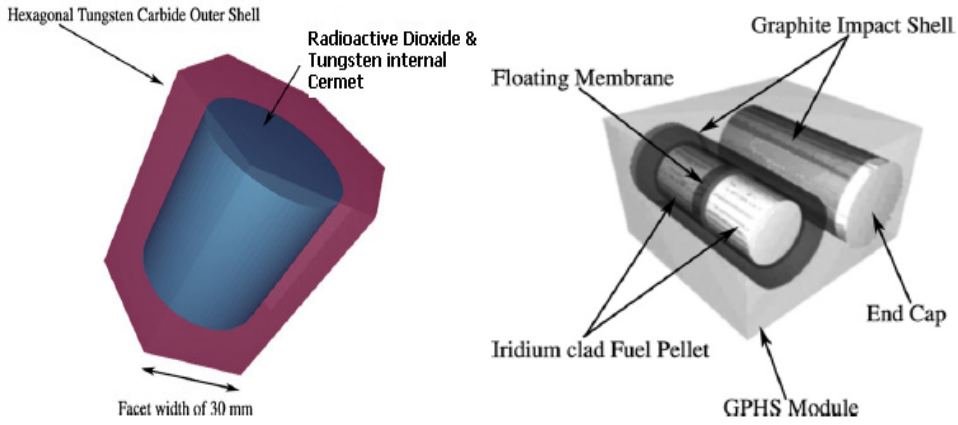

Figure 4.1 Graphic images over the encapsulated radioactive material in tungsten [30].

Figure 4.1 shows a hexagonal outer shell that should provide total encapsulation and a mechanical interface for the GPHS module. A benefit from encapsulating the fuel would be to reduce the exposure time of the dangerous radiation to the people handling the pellets and thus allow them to work additional hours with the radioactive material. The high melting point of tungsten is also beneficial if used to encapsulate the radioactive material in case of an accidental re-entry. To solve the problem with helium production from the alpha radiation within the tungsten-isotope, the porosity of the material must be adjusted via fabrication techniques [30]. Encapsulating the curium in tungsten would reduce the heat generated and because curium has such a high energy density, the heat loss would not be a problem as it would still have plenty left to generate electricity for the ion engine. More research into the fabrication of Cm pellets is needed, but this method clearly shows the potential to reduce the radiation and increase the safety of the pellets.

4.3 Transport

In chapter two it was suggested that the partitioning and separation facilities that would be used to create new fuel or targets for the future fast reactor were to be situated in close vicinity to the power plants. The best way to solve the multiple tasks for Sweden's future nuclear energy park would be to build all the plants and facilities needed to take care of the spent fuel from the LWRs on one site and to put it far away from the population to avoid unnecessary risks. This park could consist of one of the options discussed in Chapter 2 plus a facility to make curium based pellets. Whether this facility would need an automated or manual production line would require more research. An assembly place to put together the satellites would also be required.

The SMART-1 module was sent to space by piggybacking on with two other satellites an Ariane 5 rocket launched from Guiana space centre in French Guiana. The total cost of the SMART-1 satellite was €110 million, which is only one fifth of the normal price for a space mission that is put together by the ESA. Included in this cost was the building of the satellite, the launch and also the operations of the scientific instruments onboard. This low cost was mainly achieved by the small size of the satellite [34]. The price of producing curium today is unclear due to the lack of an existing curium fuel pellet factory, but there have been estimates of 2,000 USD/g for Cm 242 and 170 USD/g for Cm 244 [44]. These estimates are based on the presumption that curium would be used as a fuel in scientific RTGs and not in a program to separate out the curium from HLW to solve an environmental problem. There needs to be more research on partitioning, safety handling and separation techniques to get a more reliable price estimate for factory fabricated curium pellets.

4.4 Time and Trajectory Considerations for Deep Space Probes

This chapter gives a few comments on the duration and the specific trajectories of deep space probes. The starting point for this discussion will be the figure below showing the heliocentric velocity of Voyager 2 (launched in August 1977) as a function of the distance from the Sun.

It is interesting to note that the solar system escape velocity (to leave the solar system) decreases rapidly from 1 AU to 13-14 AU. The “gravity sling assist” of Jupiter in July 1979 and Saturn in August 1981 increased the velocity of the Voyager 2 space probe to significantly reach the needed solar system escape velocity. From this point on the speed of the probe is always much higher than the mentioned escape velocity. Uranus further added some push while the Neptune fly-by cost a bit.

The curve below (next page) is a suggestive trajectory for a careful planning to leave our solar system. The energy and time considerations, on one hand, and the scientific motivations, on the other. How much fuel is needed for “an extra fly-by” like Uranus in the case above? In any case, a Jupiter fly-by seems very rewarding considering speed.

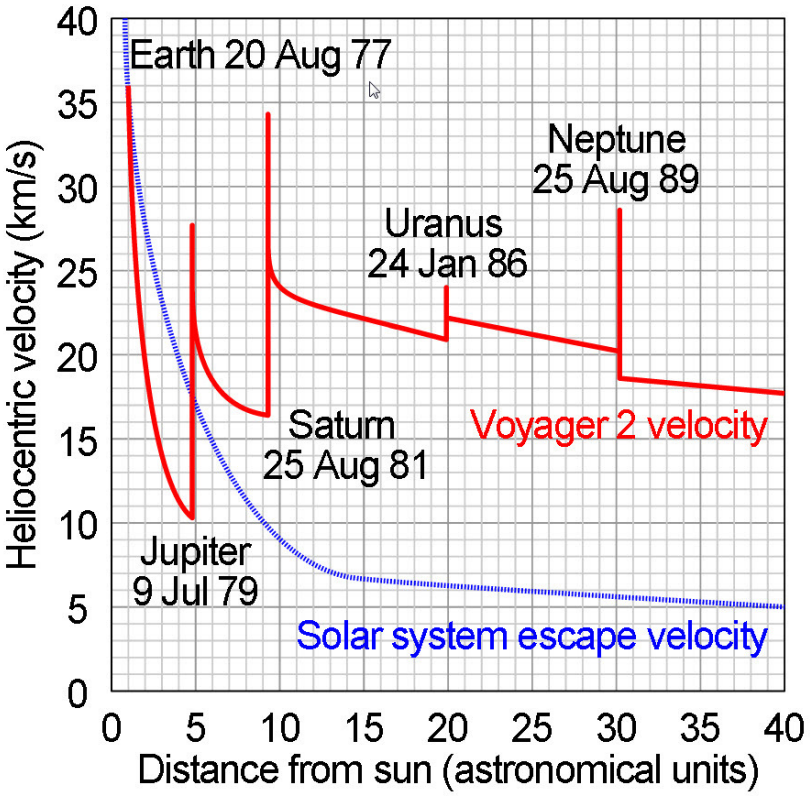

Figure 4.2 the heliocentric velocity of Voyager 2 (launched in August 1977) as a function of the distance from the Sun [61].

Because of the relatively short half-life of Cm, time and energy is important. The Voyager space probes using Pu-238 have been on their way for about 40 years, i.e. one half half-life. During this time the Voyager has spent about 100 kg of chemical fuel (hydrazine) corresponding to 536 kWh [56]. Hydrazine is a commonly used monopropellant for chemical thrusters[7].

Voyager was also built with three RTGs (each with 24 pressed Pu spheres), which provide the spacecraft with 470 W (at launch) and claimed to allow operations to continue until at least 2020 [58, 59]. The thrusters and the RTGs of the Voyager space craft are shown in the figure on the next page. (Note the placement of the RTGs)

Clearly the energy considerations depend strongly on the scientific mission and since leaving our solar system is a tedious task, it will always be tempting to study the outer planets "on the way out". Generally speaking it is therefore hard to specify the exact energy needed for a future deep space probe. It should also be pointed out here that the ion propulsion systems have developed

7 This fuel powered 16 thrusters for attitude control, etc. The hydrazine decomposition reaction produces ammonia, nitrogen and hydrogen. Cold gas thrusters use temperatures of 200 – 300 K, warm gas thrusters use temperatures up to 1600 K. The former has a specific impulse of 40 – 140 sec, while the latter reaches Isp = 245 sec.

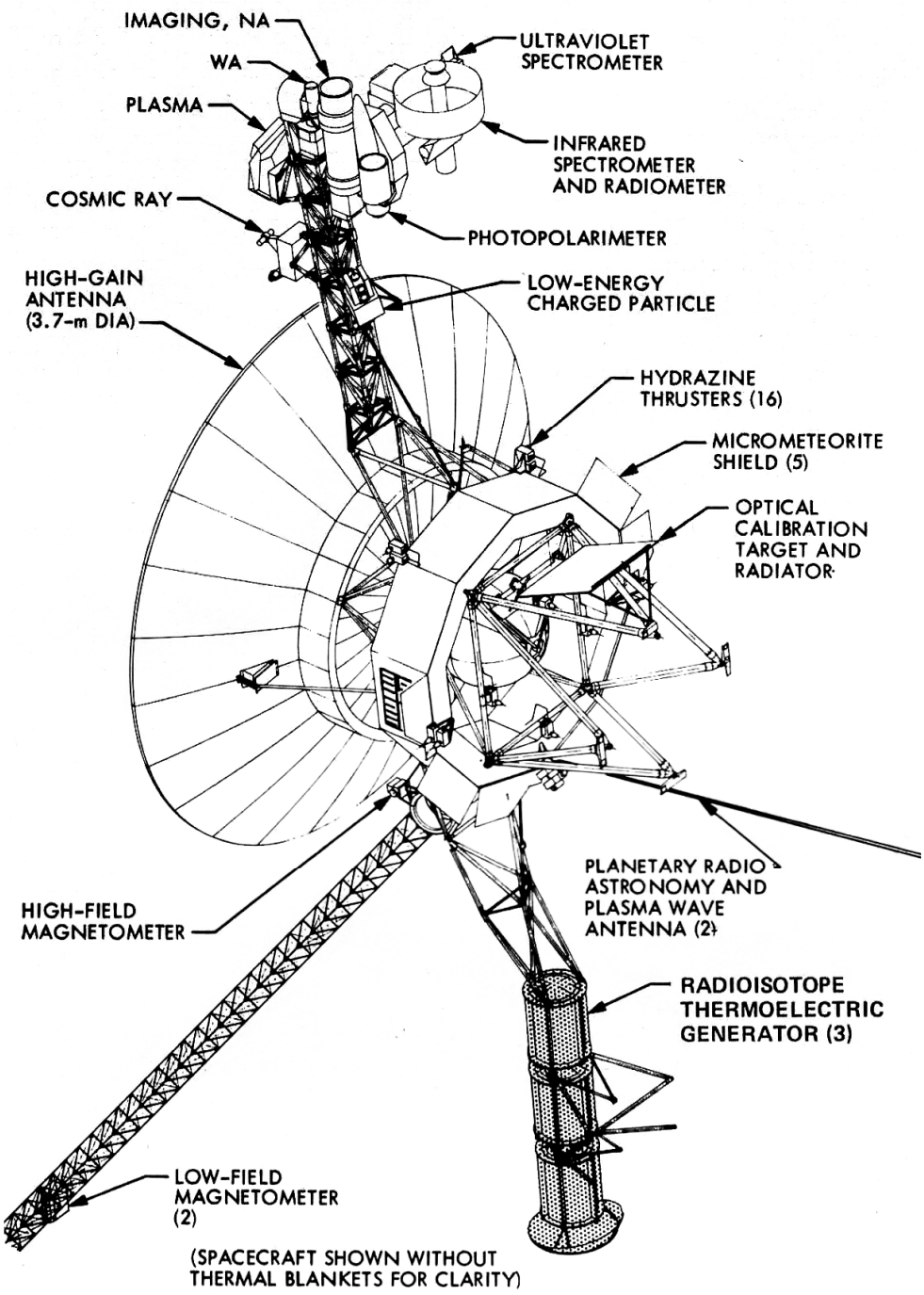

Figure 4.3 The Voyager 2 spacecraft showing the scientific instruments, the thrusters for attitude control and the three RTGs [59].

during the last few years. This is particularly true for the space program of Japan where development of an ion engine that has an exhaust velocity of 40 km/s and the propulsive efficiency of 50 % is being achieved [60]. With these new engines the performance of a deep space probe will significantly improve.

Although Cm has a relatively short half live, there should not be any significant problems in view of the energy and time considerations of a deep-space probe.

The use of chemical propulsion is still a superior choice when leaving LEO and also overcoming the solar system escape velocity, by using a spiral escape route from the gravitational pull of the Earth, ion engines combined with using “gravity sling assist” from planets is a very good option. The use of solar panels to generate electrical power works very well inside the perimeter of Jupiter, but when sending a probe further into space where the energy of the Sun diminish the use of RTGs are necessary.

Chapter 5

Conclusion

This thesis has not focused on anything but the possibility of using Cm in RTGs in deep space missions. Clearly, there are and will be other areas where Cm can and will be used. The major problem with nuclear power today is highly radioactive used nuclear fuel that is produced from fission of the uranium during the power production. This report has shown different combinations of reactors that could work together to resolve the problem without having to bury the radioactive material in a deep geological repository. These combinations rely particularly on using a fast reactor to burn most of the TRUs. The problem with a fast reactor is the difficulty in making targets or fuels with the curium elements. Since Curium is a neutron emitter and creator of higher actinides, such as californium, which is a strong neutron emitter, it could pollute the nuclear fuel cycle. The most recent approach to this problem is to store it for 100 years and let it decay into plutonium, which is a much easier material to burn in the fast reactors. This would require a storage facility that is highly protected and operated all hours of the day to secure the safety of the radioactive material.

There is instead the possibility to create new safe radioisotope heat units (RHU) with curium fuel. The need for more research into the fabrication methods, such as separation, partitioning and safety handling of such RHUs, is a must but the technology looks quite promising.

This report shows an alternative way to use Cm 244 as a fuel for the Multi-Mission Radioisotope Thermoelectric Generators (MMRTGs). The Multi-Mission Radioisotope Thermoelectric Generator was chosen due to its rugged design. It does not have any moving parts that can malfunction or wear out. This is an important criterion for a spacecraft that must spiral its way for a long time out into deep space. The reliability of the satellite is vital because any repairs or parts replacements are not possible. Cm 244 is the only choice as fuel, because Cm 245 and Cm 246 do not generate enough heat output to drive the ion engine. Cm 242 and Cm 243 are not created in sufficient amounts from the used fuel from the reactor. The calculations have been made on Cm 244 to show the feasibility of this alternative solution, but of course all the different isotopes of Cm would have to be included in the satellite.

Energy production from the Swedish nuclear power plants generates a used nuclear fuel stream of 6-12 kg of curium each year. This amount of curium could be used to produce 80 General

Purpose Heat Sources (GPHSs), which would produce heat in a safe way inside 2 Multi-Mission Radioisotope Thermoelectric Generators. From Section 2.3, it is seen that the power production from alpha decay in Cm 244 would be enough to send the satellite far into space and not pose a risk for the Earth's environment.

Implementation of the concepts presented in this report lies a long time into the future. A decision must be made to select and build a particular combination of nuclear power systems as described in Chapter Two. If such a system could be realized and the separation techniques of the curium could be done on an industrial level, then the curium RHUs could be manufactured. Then sending curium out into space in a small satellite with an electric driven ion engine could be realized.

The cost of a project like this depends on many factors and to speculate in them now would be very hypothetical. However, it can be said that from the cost of €110 million for the SMART-1, a lot of money can be saved by mass production. Once the first prototype has been built and successfully launched, the price to build such spacecraft would drop significantly.

The cost of this project is highly dependent on assumptions about the future outcome of the nuclear fuel cycle, as well as on the availability of Pu RTGs. The cost also needs to be compared to storing and safekeeping the Cm for 100 years. Such a comparison may find that space disposal of Cm is not as crazy as it may first appear. In fact, it may be the most sensible way to proceed.

Bibliography

[1] Dagens Nyheter, “Förlorarna segrade - svensk kärnkraft genom tiderna,” published 26th of April 2010.

[2] Westlén D., “Why Faster is Better - On Minor Actinide Transmutation in Hard Neutron Spectra,” Doctoral Thesis, Fysik Ö11461 Trita-FYS, KTH, Stockholm, Sweden, 2007, pp.4-8.

[3] Wallenius J., ”Återanvändning av långlivat kärnavfall och sluten bränslecykel möjlig i nya reaktortyper,” Nucleus, No. 4, 2007, pp. 14-19.

[4] Organization for Economic Co-operation and Development. “Actinide and Fission Product Partitioning and Transmutation - Status and Assessment Report” Nuclear Energy Agency, Organization for Economic Co-operation and Development, OECD Publications, 1999, pp. 203-205.

[5] Svensk energi’s website,”www.svenskenergi.se/sv/Om-el/Karnkraft/Produktion/,” visited on 19 May 2010.

[6] *Actinide and Fission Product Partitioning and Transmutation. Status and Assessment Report*, OECD Publications, 1999, pp. 39-46.

[7] *Advanced Nuclear Fuel Cycles and Radioactive Waste Management*. OECD/NEA, 2006, pp. 16.

[8] Patricia. A. Baisden and Gregory R. Choppin, “Nuclear Waste Management and the Nuclear Fuel Cycle, in Radiochemistry and Nuclear Chemistry,” *Encyclopedia of Life Support Systems (EOLSS)*, Oxford ,UK, 2007.

[9] Svensk Kärnbränslehantering AB, “Verksamheten 2009,” 2010.

[10] Morra E.P.,”Transmutation of nuclear waste in a boiling water reactor with hafnium clad,” Graduation Thesis, Fysik Ö11461 Trita-FYS, KTH, Stockholm, Sweden 2008, pp. 3.

[11] OECD/NEA, Accelerator-driven Systems (ADS) and Fast Reactors (FR) in Advanced Nuclear Fuel Cycles - A Comparative Study, 2002, pp. 137.

[12] OECD/NEA, Accelerator-driven Systems (ADS) and Fast Reactors (FR) in Advanced Nuclear Fuel Cycles - A Comparative Study, 2002, pp. 118.

[13] Masson H., Grenêche D., Chambrette P., “The Nuclear Waste Issue: Towards an Assessment of the Partitioning and transmutation of Actinides,” Proceedings of ICAPP 2007, May 13-18, Nice, France, Paper 7446, 2007, pp. 7,

[14] Westlén D., “Why Faster is Better - On Minor Actinide Transmutation in Hard Neutron Spectra,” Doctoral Thesis, Fysik Ö11461 Trita-FYS, KTH, Stockholm, Sweden, 2007, pp. 63-66.

[15] L.H. Baetslé, T. Wakabayashi, S. Sakurai., ”Status and Assessment Report on Actinides and Fission Product Partitioning and Transmutation,” OECD/NEA, November 1998 pp.15.

[16] *Actinide and Fission Product Partitioning and Transmutation. Status and Assessment Report*, OECD Publications, 1999, pp. 195-200.

[17] L.H. Baetslé, T. Wakabayashi, S. Sakurai, ”Status and Assessment Report on Actinides and Fission Product Partitioning and Transmutation,” OECD/NEA, November 1998, pp.16-21.

[18] Dr. R.G. WYMER et al.,”Actinide Separation Chemistry in Nuclear Waste Streams and Materials,” NEA Nuclear Science committee, OECD publication, 1997, pp. 47-48.

[19] C. Madic, M. Lecomte, P. Baron, B. Boullis. “Separation of long-lived radionuclides from high active nuclear waste,” C. R. Physique 3, 2002, pp. 797–811

[20] Organisation for Economic Co-operation and Development. “Actinide and Fission Product Partitioning and Transmutation - Status and Assessment Report” Nuclear Energy Agency, Organization for Economic Co-operation and Development, OECD Publications, 1999, pp. 19-20.

[21] Dr. R.G. WYMER et al.,”Actinide Separation Chemistry in Nuclear Waste Streams and Materials,” NEA Nuclear Science committee, OECD publication, 1997, pp. 69-70.

[22] Organization for Economic Co-operation and Development. “Actinide and Fission Product Partitioning and Transmutation - Status and Assessment Report” Nuclear Energy Agency, Organization for Economic Co-operation and Development, OECD Publications, 1999, pp. 203-206.

[23] Organization for Economic Co-operation and Development. “Actinide and Fission Product Partitioning and Transmutation - Status and Assessment Report” Nuclear Energy Agency, Organization for Economic Co-operation and Development, OECD Publications, 1999, pp. 196.

[24] J. Coopersmith, “Disposal of High-Level Nuclear Waste in Space” Annual Technical Symposium Texas A&M University, 1999.

[25] ESA, “By sunpower to the moon,” SMART-1, June 2003.

[26] Swedish space corporation's website on SMART-1, "www.ssc.se/?id=6217," visited on 16th of May 2010.

[27] Gary L. Bennett "Space Nuclear Power: Opening the Final Frontier," 4th International Energy Conversion Engineering Conference and Exhibit (IECEC) 26-29 June 2006, San Diego, California

[28] Gary L. Bennett et al. "Mission of Daring: The General-Purpose Heat Source Radioisotope Thermoelectric Generator," 4th International Energy Conversion Engineering Conference and Exhibit (IECEC), San Diego, California, 26-29 June 2006 pp 6-7.

[29] Radioisotope Power Systems Committee, National Research Council "Radioisotope Power Systems: An Imperative for Maintaining U.S. Leadership in Space Exploration," the national academies press, Washington DC, 2009, pp. 8.

[30] R.C. O'Brien et al., " Safe radioisotope thermoelectric generators and heat sources for space applications," Journal of Nuclear Materials 377, 2008, pp. 506–521

[31] Morse J.G., "Energy for remote areas," Science, New Series, Vol. 139, No. 3560, Mars 22, 1963, pp. 1175-1180.

[32] Robert D. Abelson et al., "Expanding Frontiers with Standard Radioisotope Power Systems," Prepared for the NASA Science Mission Directorate, January 12, 2005, pp. 11-18.

[33] Swedish space corporation's website on navigation services, "www.ssc.se/?id=13397," visited on 17th of May 2010.

[34] ESA website on SMART-1 FAQ, "www.esa.int," visited on 18th of May 2010.

[35] J. Wallenius, head of the Nuclear Reactor Department at the Royal Institute of Technology, Stockholm, 2010, personal communication.

[36] WWW Table of Radioactive Isotopes,"http://nucleardata.nuclear.lu.se/nucleardata/toi/nuclide .asp?iZA=960245," visit on 16 July 2010.

[37] M. Tondel et al.,"Increased incidence of malignancies in Sweden after the Chernobyl accident—a promoting effect?," American Journal of Industrial Medicine Volume 49, Issue 3, pages 159–168, March 2006

[38] Vattenfall AB, "Vattenfall Kärnbränslecykeln 2007," http://www.vattenfall.se /sv/radioaktivt-avfall.htm, visited on 9th of Mars 2011.

[39] Herman Cember and Thomas E. Johnson, "Introduction to Health Physics –fourth edition," Department of Environmental and Radiological Health Science, chapter 8. 2009.

[40] M. Konomura and M. Ichimiya, “Design Challenges for Sodium Cooled Fast Reactors,” Journal of Nuclear Materials 371, 15 September 2007, pp. 250-269.

[41] H. Takano et al.,” Transmutation of long-lived radioactive waste based on double-strata concept,” Progress in nuclear Energy, Vol. 37, No. 1-4, 2000, pp.371-376.

[42] G. Kessler, “Plutonium denaturing by Pu 238,” The first Int. Sci. Technol. Forum on Protected Plutonium Utilization for Peace and Sustainable Prosperity, Tokyo, Japan, Mar. 1-3, 2004

[43] E.H. Hall: "On a New Action of the Magnet on Electric Currents," *American Journal of Mathematics* vol 2, p.287-292, 1879.

[44] G.L. Kulcinski. “Basic elements of static RTGs,” NEEP 602 Course Notes (Spring 2000), Nuclear Power in Space, University of Wisconsin Fusion Technology Institute.

[45] Nyteknik “Fukushima ett år efter katastrofen,” http://www.nyteknik.se/nyheter/energi_miljo/karnkraft/article3423423.ece, visit on 12th of May 2012.

[46] H. Cember and T.E. Johnson, ”Introduction to Health Physics – fourth edition,” The McGraw-Hill Medical, p.69, 2009.

[47] K. V. Hecke and P. Goethals, “Research on advanced Aqueous reprocessing of spent nuclear fuel: literature study,” SCK-CEN-Belgium p.14, July 2007.

[48] Nyteknik “Wallenius drömreaktor får plats i en resväska,” http://www.nyteknik.se/nyheter/energi_miljo/karnkraft/article3423419.ece, visit on 12th of Mars 2012.

[49] D.M. Rowe, “ CRC Handbook of Thermoelectrics,” *University of Cardiff, Wales, UK,* July 14, 1995 by CRC Press - 701 Pages

[50] M S Lowell, “ The Lagrange Points” Royal Hospital School, Holbrook, UK, May 2007.

[51] The United Nations Office of Outer Space Affairs (UNOOSA), “United Nations Treaties and Principles on Space Law,” http://www.oosa.unvienna.org/oosa/SpaceLaw/treaties.html, visit on 17th of June 2012.

[52] The United Nations Office of Outer Space Affairs (UNOOSA), “Principles Relevant to the Use of Nuclear Power Sources In Outer Space,” http://www.oosa.unvienna.org/oosa/en/SpaceLaw/gares/html/gares_47_0068.html, visit on 17th of June 2012.

[53] James A. Van Allen, “The Geomagnetically Trapped Corpuscular Radiation,” Journal of Geophysical Reasearch VOLUME 64, NO. 11 NOVEMBER, 1 959

[54] Stephen C. Ko, Christopher S. Pugh, Richard L. Kiefer, Robert A. Orwoll, Williamsburg VA, Sheila A. Thibeault and Glen C. King (1998). Boron Containing Polyimides for Aerospace Radiation Shielding. MRS Proceedings, 551

[55] Adrien Gomez et al.," Exergy analysis for Generation IV nuclear plant optimization," INTERNATIONAL JOURNAL OF ENERGY RESEARCH Int. J. Energy Res. 2010; 34:609–625

[57] Dezsö Novák, " Testing of electronics and development of detectors for radiation environments," Akademisk avhandling, Fysik Ö11416 Trita-FYS, Kungliga Tekniska Högskolan i Stockholm, Nov 2001.

[56] E. Granot et al., "Hydrazine/air direct-liquid fuel cell based on nanostructured copper anodes," Journal of Power Sources, Volume 204, 15 April 2012, Pages 116–121.

[58] R. R. Furlong and E.J Wahlqvist, "US Space Missions using Radioisotope Power Systems", Nuclear News 42 (4), April 1999, pp. 26-34.

[59] NASA, "Voyager 2 Craft Details," NASA-NSSDC Spacecraft Details, http://nssdc.gsfc.nasa.gov/nmc/spacecraftDisplay.do?id=1977-076A, visited on 11th of April 2013

[60] Shunjiro Shinohara, "Research and Development of a Novel Electrodeless Plasma Rocket Engine Using a Helicon Source," Kyushu University, Interdisciplinary Graduate School of Engineering Sciences, http://www.jsps.go.jp/j-grantsinaid/12_kiban/ichiran_21/e-data/e67_shinohara.pdf, visited on 7th of April 2013

[61] NASA, "Basics of space flight Section I," Chapter 4. Interplanetary Trajectories, http://www2.jpl.nasa.gov/basics/bsf4-1.php, visited on 11th of April 2013

Appendix – Computer Codes

Matlab code for diagram 2.4

```
NCm244          = (6.022E23/244);                                   %atom/g

TCm244          = 18.1;                                             %half-life

lambdaCm244     = log(2)/TCm244;                                    %decay constant for Cm 244

QCm244          = 5902.0*1000*1.602E-19;                            %J/decay

t               = 0:10:350;                                         %350 years

ACm244          = lambdaCm244*NCm244*exp(-(lambdaCm244*t));         %Activity/year

PCm244          = ACm244*(QCm244/(365*24*3600));                    %Heat power/year

TPu240          = 6563;                                             %half-life

lambdaPu240     = log(2)/TPu240;                                    %decay constant for Pu 240

APu240          = -NCm244*((lambdaCm244*lambdaPu240)/(lambdaPu240-lambdaCm244))
                  *(exp(-lambdaPu240*t)-exp(-lambdaCm244*t));       %Activity/year

QPu240          = 5256.0*1000*1.602E-19;                            %J/decay

PPu240          = APu240*(QPu240/(365*24*3600));

Ptot            = PCm244+PPu240;

figure(3)
plot (t,log2(PCm244),t,log2(PPu240),t,log2(Ptot))                   %plot of total power
legend('Cm244','Pu240','Tot')
title('Heat Production from 1g of Cm244')
xlabel('Time(year)')
ylabel('Power(Watts)')
```

Matlab code for diagram 2.5

```
t           = 0:5:350; %from 0-350 years

NCm242      = 6.022E23/242 %atom/g
TCm242      = 0.446 %year
lambdaCm242 = log(2)/TCm242 %1/year
QCm242      = (6215.56*1000*1.602E-19)/(3600*24*365) %J/decay
ACm242      = lambdaCm242.*NCm242.*exp(-(lambdaCm242.*t)); %decay/year

PCm242      = QCm242*ACm242;
```

```
NCm243      = 6.022E23/243 %atom/g
TCm243      = 29.1 %year
lambdaCm243 = log(2)/TCm243 %1/year
QCm243      = (6168.8*1000*1.602E-19)/(3600*24*365) %J/decay
ACm243      = lambdaCm243.*NCm243.*exp(-(lambdaCm243.*t)); %decay/year

PCm243      = QCm243*ACm243;

NCm244      = 6.022E23/244 %atom/g
TCm244      = 18.1 %year
lambdaCm244 = log(2)/TCm244 %1/year
QCm244      = (5902*1000*1.602E-19)/(3600*24*365) %J/decay
ACm244      = lambdaCm244.*NCm244.*exp(-(lambdaCm244.*t)); %decay/year

PCm244      = QCm244*ACm244;

NCm245      = 6.022E23/245 %atom/g
TCm245      = 8500 %year
lambdaCm245 = log(2)/TCm245 %1/year
QCm245      = (5623*1000*1.602E-19)/(3600*24*365) %J/decay
ACm245      = lambdaCm245.*NCm245.*exp(-(lambdaCm245.*t)); %decay/year

PCm245      = QCm245*ACm245;

NCm246      = 6.022E23/246 %atom/g
TCm246      = 4730 %year
lambdaCm246 = log(2)/TCm246 %1/year
QCm246      = (5475*1000*1.602E-19)/(3600*24*365) %J/decay
ACm246      = lambdaCm246.*NCm246.*exp(-(lambdaCm246.*t)); %decay/year

PCm246      = QCm246*ACm246;

Ptot        = PCm242 + PCm243 + PCm244 + PCm245 + PCm246

%loglog(t,PCm243,t,PCm244,t,PCm245,t,PCm246,t,Ptot)
plot(t,log2(PCm242),t,log2(PCm243),t,log2(PCm244),t,log2(PCm245),t,log2(PCm246
),t,log2(Ptot))

legend('Cm242','Cm243','Cm244','Cm245','Cm246','Tot')
title('Heat from 1g of the different isotopes')
xlabel('time(year)')
ylabel('Power(Watts)')

%semilogx(t,PPu239)
```