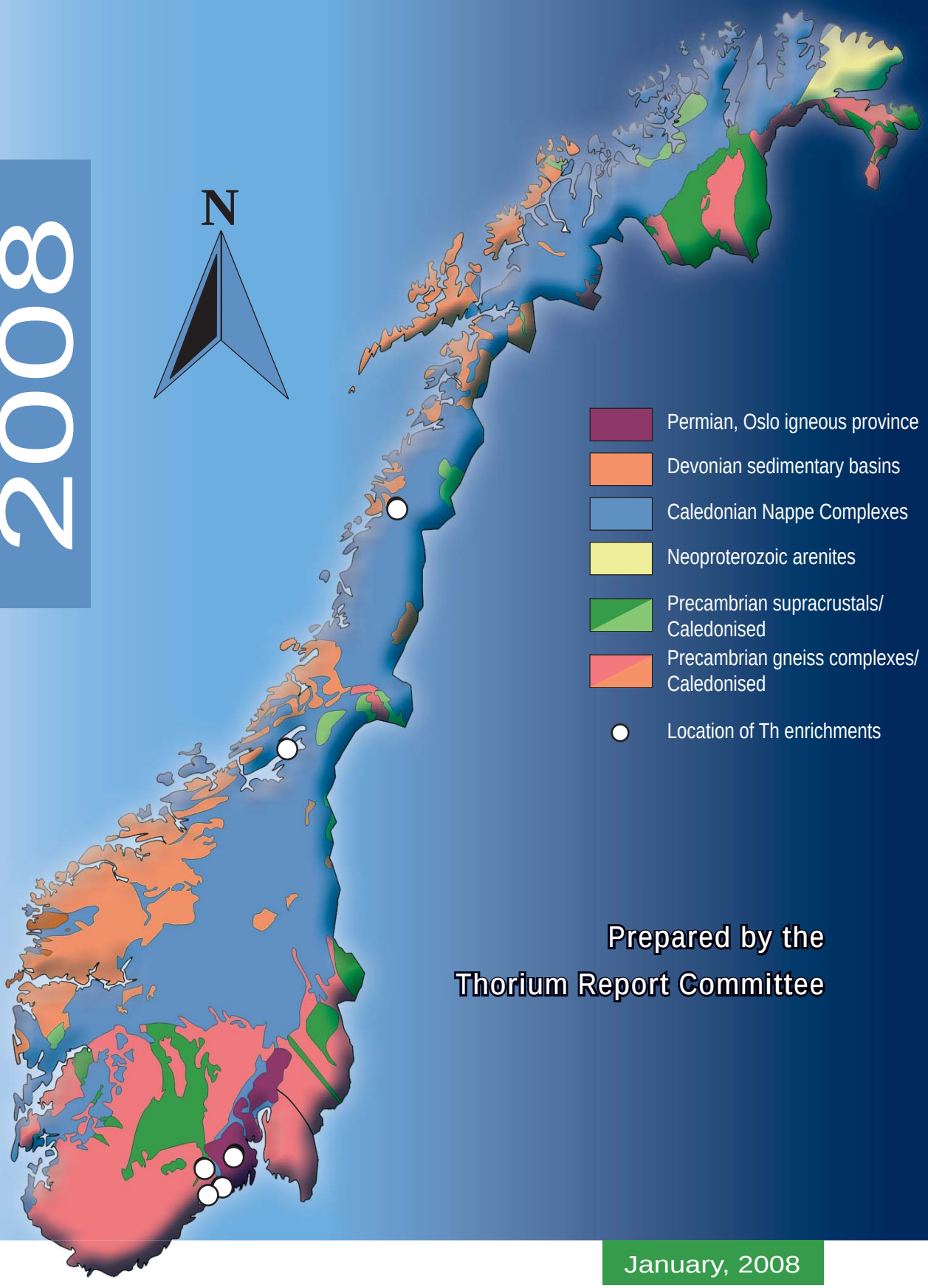
THORIUM AS AN ENERGY SOURCE
- Opportunities for Norway

2008



Prepared by the
Thorium Report Committee

January, 2008

Published by the *Thorium Report Committee, February 2008.*

The *Committee* was appointed by the *Research Council of Norway* (RCN),
on behalf of the *Ministry of Petroleum and Energy (OED).*

Secretariat: *Institute for Energy Technology (IFE)*
The *Research Council of Norway:* www.rcn.no
The *Ministry of Petroleum and Energy:* www.regjeringen.no\oed

ISBN 978-82-7017-692-2 (printed) ISBN 978-82-7017-693-9 (electronic)

Front Page Picture:
Source: Geological Survey of Norway (NGU)

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Foreword

The Norwegian Ministry of Petroleum and Energy decided that an investigation into the possibility of utilizing thorium as a source of energy should be carried out. The Research Council of Norway (RCN) was given the responsibility to organize this study and the *Thorium Report Committee* was established in March 2007.

The *Thorium Report Committee* had a broad international representation and included the following members:

The Chairman of the Thorium Report Committee:

- *Professor Mikko Kara*
  Director VAPO, Finland
  (Former Executive Vice President for the VTT Technical Research Centre of Finland)

Committee Members have been:

- *Professor Sven Kullander*
  Royal Swedish Academy of Science and Uppsala University, Sweden
  (also Chair of the Energy Committee at the Swedish Academy of Science)
- *Professor Dieter Röhrich*
  Department of Physics and Technology, University of Bergen (UiB), Norway
- *Dr. Thierry Dujardin*
  Deputy Director for Science and Development,
  OECD Nuclear Energy Agency (OECD/NEA), Paris
- *Dr. Yacine Kadi*
  Nuclear Engineer at the European Organization of Nuclear Research (CERN),
  Switzerland
- *Cand.Real. Sverre Hval*
  Principal Research Scientist, Institute for Energy Technology (IFE), Kjeller, Norway
- *Professor Brit Salbu*
  Isotope Laboratory, Norwegian University of Life Sciences (UMB), Ås, Norway
- *Professor Finn Ingebretsen*
  Department of Physics, University of Oslo (UiO), Norway

Committee Observers have been:

- *Dr.Scient. Bjørn Jacobsen*
  Adviser, Division for Science Physical Sciences and Technology,
  The Research Council of Norway (RCN)
- *Marius Knagenhjelm*
  Higher Executive Officer, The Norwegian Ministry of Petroleum and Energy (OED)

Secretariat:

- *Cand.Scient. Lise A. Moen*
  Senior Reactor Physicist, Institute for Energy Technology (IFE),
  OECD Halden Reactor Project, Norway

The Thorium Report Committee was given the following Mandate:

Terms of Reference (Mandate) for the Thorium Report Committee

"The Committee's work and the resulting Report shall establish a solid knowledge base concerning both opportunities and risks related to the use of thorium for long-term energy production. The work should be conducted as a study of the opportunities and possibilities (screening), based on a review of Norway's thorium resources and the status of key technologies.

The Report shall consider relevant technologies for utilizing thorium for energy production. This involves, for instance, various concepts for thorium based energy production, reactor technology, fuel cycles, challenges, risks, environmental problems including waste storage and an assessment of these aspects relative to other nuclear power production concepts.

The Report shall discuss significant challenges related to the key technologies, elucidate the international interest in developing thorium as an energy source, and assess Norway's possibilities for participating in potential future development of thorium as an energy source.

The Ministry emphasises that the Report shall be well balanced and knowledge based. The Committee is expected to consult international expertise when necessary, in order to ensure the quality of the Report. The Report shall be completed before the end of 2007, and the final Report shall be available in Norwegian."

This report has been made in order to fulfil the above Mandate.

Acknowledgements

The Thorium committee was established in the beginning of 2007 with well known experts in the nuclear area from Norway, OECD/NEA, Sweden and Switzerland as committee members. We had altogether 6 one day meetings in Oslo and some of the meetings continued into the following day. The hospitality of *The Research Council of Norway* was very well appreciated.

I would warmly thank all the committee members for their collaborative and active working attitude. Discussions at the meetings were lively and constructive. We did not avoid the necessary philosophical debate giving to all of us the useful background for this kind of long term energy matter. When discussing long term energy visions and trends they come close to our scientific and social consciousness leading to most interesting debates.

As external experts the Committee invited the following international nuclear area scientists and experts:

- *Dr. Hattangadi Suresh Kamath,*
  Director of Nuclear Fuels Group, Bhabha Atomic Research Centre (BARC), India.
- *Professor Kurt Kugeler,*
  Institute of reactor safety and -technology, RWTH Aachen University, Germany.
- *Professor Ulrich Ratzinger,*
  Institute for Applied Physics, J.W. Goethe-University, Frankfurt, Germany.
- *Dr. Claude Renault,*
  Chairman of the Generation IV Molten Salt Reactor Steering Committee, CEA France.
- *Dr. Gunnar Saxebøl,*
  Director of the Department for Radiation Protection and Safety,
  The Norwegian Radiation Protection Authority (NRPA).
- *Dr. Per Strand,*
  Director of the Department for Emergency Preparedness and Environmental Radioactivity,
  The Norwegian Radiation Protection Authority (NRPA).

External Reports have also been provided by:

- *K. Kugeler, N. Pöppe, S. Jühe, O. Schitthelm*: Use of Thorium in the nuclear energy technology - experiences in Germany. Institute of reactor safety and - technology, RWTH Aachen University, September 2007.
- *Ingvar Lindahl*: Thorium resources in Norway, The Geological Survey of Norway (NGU), October 2007.
- *Mette Seyersted*: Note to the Thorium Report Committee, The Norwegian Radiation Protection Authority (NRPA), November 2007.

They all gave us interesting and high quality lectures and reports on various aspects of thorium as a nuclear fuel and energy source. We are all grateful to these experts for their enthusiastic and knowledgeable input to the Committees work. Much of the information provided has been directly used in this report.

I would also like to thank *The Norwegian Ministry of Petroleum and Energy* for this very challenging task. This work has been at the same time most relevant from the global greenhouse gas point of view as well as from the point of view of the ongoing development within the nuclear

energy area. The price volatility of fossil fuels gave an extra flavour to this kind of long term thinking.

Last but definitely not least I thank our secretary Lise, who with her positive character carried out much more of the reporting work than we could reasonably expect.

Mikko Kara

Chairman of the Committee

1. EXECUTIVE SUMMARY

The introduction (Chapter 2) summarizes the energy situation worldwide, and summarised below are the major findings of the *Thorium Report Committee* with regard to the Mandate.

Thorium Resources in Norway (Chapter 3)

According to the US Geological Survey (*USGS* 2007), Norway has one of the major thorium resources in the world. The listed resources, i.e. 170 000 tonnes, have a potential energy content which is about 100 times larger than all the oil extracted to date by Norway, plus that of the remaining reserves.

Most of the thorium enriched minerals are situated within 3 main regions: the *Fen Complex* in Telemark County, the *Permian Oslo Province*, and on the Southeast coast of Norway, in the Kragerø and Langesund area. A series of thorium bearing minerals has been identified at these sites, while the *Fen Complex* is considered to be the most promising resource, with thorium amounting to about 0.1 – 0.4 wt% (weight percent).

Knowledge of Norwegian thorium enriched minerals and their grades is mainly based on results from uranium exploration carried out during two periods, from after the war to 1965 and from 1975 to 1985. The thorium levels were estimated from analyses of uranium in mineral samples, the correlation between uranium and thorium, and from helicopter and ground gamma surveys. Recent helicopter gamma surveys covering the *Permian Oslo Province* as well as the *Fen Complex* in Telemark County have confirmed previous gamma survey data.

Norway has *potential thorium resources,* but exploration specifically for thorium has never been undertaken. Knowledge of the grade and associated volumes is scarce. The quoted USGS 2007 weight estimates of the thorium resources date from the 1950s - 1960s and are uncertain. Thus, investigations of the resources, not only in the *Fen Complex*, but also at other sites in Norway, as well as mineralogical and mineral separation studies should be performed before it can be decided whether the thorium resources in Norway can be defined as an economical asset for the benefit of future generations.

The Front End of the Thorium Fuel Cycle (Chapter 4)

Production of thorium has been limited due to a lack of demand (it is used mainly in special glasses and alloys); it is a by-product of the separation of rare earth elements. The production of thorium is presently some hundred tonnes per year. The production reached about 1000 tonnes in the 1970s, and has decreased thereafter due to lack of demand.

Owing to its chemical toxicity, radiotoxicity and pyrophoricity, adequate precautions are required in the mining and processing of thorium. However, as a result of the very long half-life of thorium, limited quantities of pure thorium-232 can easily be handled, while some shielding is required for large amounts. Preparation of thorium fuel is somewhat more complex and more expensive than for uranium.

Thorium as a nuclear fuel is technically well established and behaves remarkably well in *Light Water Reactors* and *High Temperature Reactors*. It has demonstrated a very good neutron damage resistance due to its excellent chemical and metallographic stability.

Nuclear Reactors for Thorium (Chapter 5)

In the 1960s and 1970s, the development of thorium fuel for nuclear energy was of great interest worldwide. It was shown that thorium could be used practically in any type of existing reactors. A large amount of work was carried out and resulted in many interesting developments, including prototype *High Temperature Reactors*, *Light Water Reactors* and *Molten Salt Reactors*.

Most projects using thorium in their fuel cycles had been terminated by the 1980s. The reason for this seems to be threefold: (1) the thorium fuel cycle could not compete economically with the more well-known uranium cycle, (2) in many countries there was a lack of political support for the development of nuclear technology in the aftermath of the Chernobyl accident and (3) an increased worldwide concern regarding the proliferation risk associated with the reprocessing of spent fuel.

Nowadays, the use of thorium as a nuclear fuel is considered in a more or less active manner in three programs:

- In the long term the Indian program aims at burning uranium-233 and plutonium along with thorium in Advanced Heavy Water Reactors. These reactors will obtain about 75 % of their power from thorium; the uranium-233 and plutonium being previously produced in Fast Breeder Reactors.
- The Molten Salt Reactor, which is one of the six concepts developed by the Generation IV International Forum (GIF). In the GIF program the world's leading nuclear nations are gathering together technology to meet the world's future energy needs. Currently thorium is not a prioritized topic in GIF.
- The Accelerator Driven System (ADS) concept, which couples an accelerator, a spallation source and a sub-critical reactor, is being developed within the *European Atomic Energy Community (Euratom)* research Framework Program. At present, this concept focuses more on high-level waste transmutation than on energy production.

Thorium-232, used as fertile material for breeding uranium-233, has the main advantage, over uranium-238, that virtually no plutonium or other transuranic elements are produced, and the waste products are therefore free of long-lived α-emitters. Thorium used in ADS can facilitate plutonium or transuranium burning without the use of uranium-238. Some advantages of these systems, apart from the low radiotoxicity of the waste, are the increased safety features and a higher flexibility for breeding.

The Back End of the Thorium Fuel Cycle (Chapter 6)

Reprocessing thorium fuels will require more development than previously anticipated. Even if the familiar principles of solvent extraction are to be adopted (THOREX, thorium extraction), existing methods cannot be applied without modification. This would require a very substantial amount of development work.

Waste management will in principal follow known procedures and methods. Although fluoride and aluminium may have some effect on the conditioning of the high-level waste form, the key concern is the radiotoxicity of protactinium-231. This long-lived radionuclide is a candidate for partitioning into other actinides, but the present method appears not to be applicable to protactinium. Intermediate-level wastes would generally be the same as in the commonly used plutonium-uranium extraction process (PUREX) and their treatment needs no special investigation.

Radiation Protection of Man and the Environment (Chapter 7)

Radiation protection requirements for the thorium cycle will be lower than those of the uranium cycle. The enhanced levels of thorium and uranium and their daughters in the *Fen Complex* contribute to the highest outdoor and indoor gamma exposures to man ever reported in Norway, and are among the highest in Europe. However, natural background radiation is in principle not regulated. Doses to man and the environment from future potential exposures associated with the thorium fuel cycle will be regulated by the *Radiation Protection Act and associated Regulations*. However, authorisation requirements for mining and milling thorium are not included in the current radiation protection regulatory system and revision of the *Act* will be needed.

Regulation (Chapter 8)

Requirements for the establishment of thorium based industries in Norway are set by a series of acts and regulations, in particular the *Nuclear Energy Act* and the *Radiation Protection Act*. The *Act Concerning Nuclear Energy Activities* from 1972 regulates activities associated with the existing Norwegian research reactors. A conventional thorium-uranium based nuclear installation will most probably be covered by the current licensing requirements, whereas a pure thorium based system, such as an Accelerator Driven System (ADS) will not, and the *Nuclear Energy Act* would, therefore, probably have to be revised or amended.

Non-proliferation (Chapter 9)

Uranium based fuel cycles require enrichment and reprocessing facilities which use technology originally developed for military purposes and which leave large amounts of fissile plutonium isotopes in spent fuel. The thorium-uranium (Th-232/U-233) fuel cycles do not produce plutonium. Technically, one of the best ways to dispose of a plutonium stock pile is to burn it in a thorium-plutonium MOX fuel.

The proliferation resistance of uranium-233 depends on the reactor and reprocessing technologies. In the development of a reactor technology and its fuel cycle for civil purposes, the thorium fuel cycle should have an advantage in proliferation resistance that could be exploited. However, due to the lack of experience with industrial-scale thorium fuel cycle facilities, similar safeguard measures as for plutonium are considered mandatory until otherwise documented.

Economical Aspects (Chapter 10)

Due to a lack of data, it seems impractical to develop meaningful cost projections for any nuclear energy system using thorium. It seems obvious that the contribution of the raw material to the cost structure of the electricity generated will be small, comparable to that of the uranium cycle or even lower. The main economical challenges to the development of a thorium based energy production will be the acquisition of funding necessary to carry out the required research and development. As a comparison, in the 1970s Germany spent around 500 million euros in current money to develop a thorium fuel cycle and 2.5 billions euros for the *High Temperature Reactor* itself.

Research, Development, Education and Training (Chapter 11)

Several studies (e.g. EU, OECD/NEA) have identified the problem that an insufficient number of scientists are being trained to meet the needs of the current and future European nuclear industries. Norway also lost most of its specialists in nuclear sciences after the nuclear moratorium more than 25 years ago. The European higher education knowledge base has become

fragmented to a point where universities in most countries lack sufficient staff and equipment to provide education in all but a few nuclear areas.

Today, *Institute for Energy Technology (Institutt for energiteknikk, IFE)* operates two research reactors in Norway. IFE maintains a national competence in reactor technology and safety, based on the international *OECD Halden Reactor Project*. A broad spectrum of fuel- and material examinations is performed at the Halden Reactor.

The *University of Oslo* has Norway's only research accelerator. The main nuclear research, conducted in international collaborations, focuses on methods and equipment to measure nuclear structure properties.

The *Universities of Bergen* and *Oslo* are involved in basic research projects within nuclear physics, and nuclear methods relevant for reactor technology are used in many branches of science at the Norwegian universities. The Norwegian universities, in close cooperation with *Institute for Energy Technology*, are able to organize and provide courses in relevant nuclear physics, reactor theory, reactor operation, material science, radiation risk and radiation protection. In addition, there is an EU supported master program in radioecology at the *University of Life Sciences*.

Final Recommendations of the Thorium Report Committee:

1. *No technology should be idolized or demonized. All carbon-dioxide (CO_2) emission-free energy production technologies should be considered. The potential contribution of nuclear energy to a sustainable energy future should be recognized.*
2. *An investigation into the resources in the Fen Complex and other sites in Norway should be performed. It is essential to assess whether thorium in Norwegian rocks can be defined as an economical asset for the benefit of future generations. Furthermore, the application of new technologies for the extraction of thorium from the available mineral sources should be studied.*
3. *Testing of thorium fuel in the Halden Reactor should be encouraged, taking benefit of the well recognized nuclear fuel competence in Halden.*
4. *Norway should strengthen its participation in international collaborations by joining the Euratom fission program and the GIF program on Generation IV reactors suitable for the use of thorium.*
5. *The development of an Accelerator Driven System (ADS) using thorium is not within the capability of Norway working alone. Joining the European effort in this field should be considered. Norwegian research groups should be encouraged to participate in relevant international projects, although these are currently focused on waste management.*
6. *Norway should bring its competence in waste management up to an international standard and collaboration with Sweden and Finland could be beneficial.*
7. *Norway should bring its competence with respect to dose assessment related to the thorium cycle up to an international standard.*
8. *Since the proliferation resistance of uranium-233 depends on the reactor and reprocessing technologies, this aspect will be of key concern should any thorium reactor be built in Norway.*
9. *Any new nuclear activities in Norway, e.g. thorium fuel cycles, would need strong international pooling of human resources, and in the case of thorium, a strong long-term commitment in university education and basic science. All these should be included in the country level strategy aiming to develop new sustainable energy sources. However, to meet the challenge related to the new nuclear era in Europe, Norway should secure its competence within nuclear sciences and*

nuclear engineering fields. This includes additional permanent staff at the universities and research institutes and appropriate funding for new research and development as well as a high quality research-based Master and PhD education.

Concluding Remarks: The *Thorium Report Committee* finds that the current knowledge of thorium based energy generation and the geology is not solid enough to provide a final assessment regarding the potential value for Norway of a thorium based system for long term energy production. The *Committee* recommends that the thorium option be kept open in so far it represents an interesting complement to the uranium option to strengthen the sustainability of nuclear energy.

2. INTRODUCTION

Electricity has been generated for the purpose of powering human technologies since 1881. Electricity consumption and demand has increased continuously as a result of population growth and industrial development as well as increasing wealth. Decoupling of Gross Domestic Product (GDP) and energy growth has happened partially in most OECD countries, but not yet in the vast developing countries, i.e. China, India and Russia. Growth of energy consumption and demand will be the major challenge facing mankind in this century.

As an introduction to this report the present situation concerning the world energy and electricity balances, greenhouse gas emissions, fossil and nuclear fuel resources as well as a reference scenario describing projections of energy demand and carbon-dioxide (CO_2) emissions in the next 25 years are presented.

2.1 The World Energy Situation

There is a prevailing belief that the current model for the world's energy policy is not sustainable. The major reasons are well known: rising greenhouse gas emissions and their negative impact on climate, as well as concerns regarding the security of energy supply at affordable prices in the context of increasing needs of energy, notably in developing countries that are enjoying rapid economic growth.

There is no magical solution to avoid what *Claude Mandil*, the former Executive Director of the International Energy Agency (IEA), referred to as the "dirty, insecure and expensive" energy future that current trends are preparing for us. If better energy efficiency and energy savings are the first part of the solution, then both the development of cleaner fossil fuel technologies (with carbon capture and sequestration) and increased uses of renewable and nuclear energy are the second part.

According to the IEA reference scenario [1], which assumes that governments around the world maintain current policies, the world's primary energy demand needs are projected to grow by 55 % between 2005 and 2030 at an average rate of 1.8 % per year. Fossil fuels remain the dominant source of primary energy, accounting for 84 % of the overall increase in demand. Oil remains the single largest fuel though its share in global demand falls from 35 % to 32 %. Oil demand reaches 116 million barrels per day (bpd) in 2030, to compare to 84 million bpd in 2006. In line with the spectacular growth of the past few years, coal sees the biggest increase in demand in absolute terms, jumping by 73 % between 2005 and 2030, and pushing its share of total energy demand up from 25 % to 28 %. Most of the increase in coal use arises in China and India. The share of natural gas increases more modestly, from 21 % to 22 %. Some $22 trillion of investment in supply infrastructure is needed to meet projected global demand. Mobilising all this investment will be challenging.

Developing countries, whose economies and populations are growing fastest, contribute 74 % of the increase in global primary energy use in this scenario. China and India alone account for 45 % of this increase. OECD countries account for one-fifth and the transition economies the remaining 6 %. In aggregate, developing countries make up 47 % of the global energy market in 2015 and more than half in 2030, compared with only 41 % today.

In the context of such an ongoing fossil energy future up to 2030 and beyond, the world will continue to face two energy-related threats: a real and growing threat to the world's energy security and increased environmental harm caused by burning too much fossil fuel. The IEA reference scenario illustrates the former by noting that China's and India's combined oil imports will surge from 5.4 million bpd in 2006 to 19.1 million bpd in 2030, more than the combined imports of Japan and the United States today. Ensuring reliable and affordable supply will be a formidable challenge. As regards the latter, global energy-related CO_2 emissions jump by 57 % between 2005 and 2030, with the United States, China, Russia and India contributing two-thirds of this increase. China is by far the biggest contributor to incremental emissions, overtaking the United States as the world's biggest emitter in 2007. India becomes the third-largest emitter by around 2015. However, China's per-capita emissions in 2030 are only 40 % of those of the United States and about two thirds those of the OECD as a whole.

2.2 The World Electricity Situation

Historically the first power plants for electricity production were operated with hydro-power or coal. Today, although coal remains the largest contributor, electricity production relies on a more diverse mix, which includes natural gas, oil, nuclear, hydro and a small amount of other renewables such as biomass, solar, tidal harnesses, wind and geothermal sources. Figure 2.1 shows the world's electricity production by source in 2004 [2].

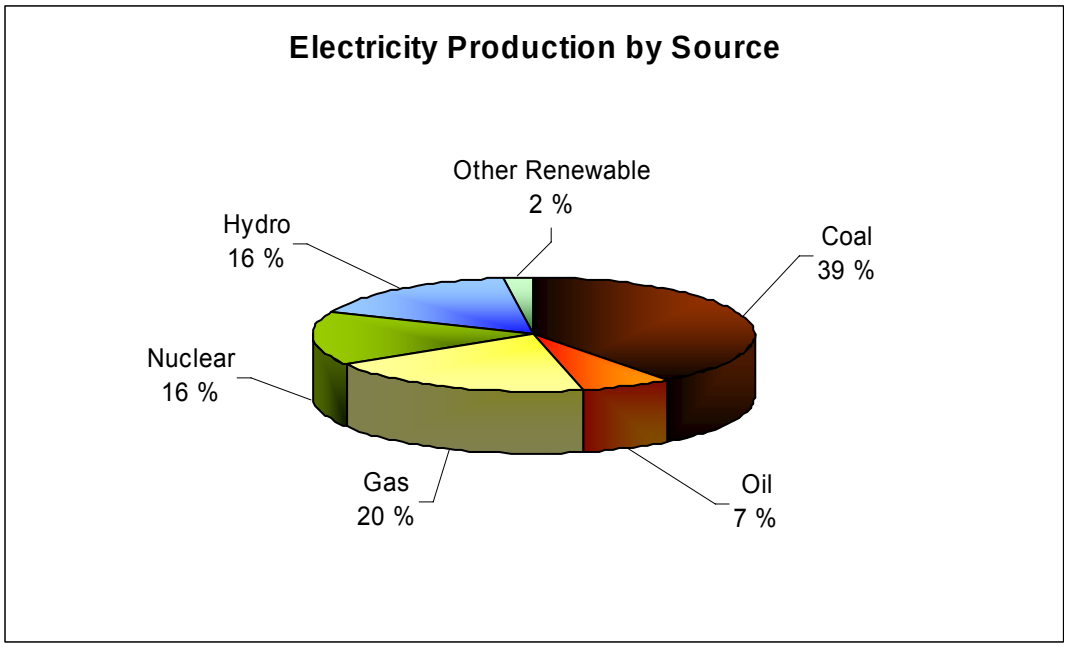

Figure 2.1: Electricity Production by Source in 2004.

(Source: World Energy Outlook 2006)

Since today's electricity production is mainly based on non-renewable sources, the availability of the required natural resources will become a growing concern. Coal is the most abundant fossil fuel and known global resources represent more than 250 years of current annual consumption, however it is also the biggest CO_2 emitter per kilo watt hour (kWh) produced (see Figure 2.2). The pace of depletion of the other fossil fuel resources (gas, oil) is more rapid although the numbers are questionable and strongly dependent upon technology improvements and prices.

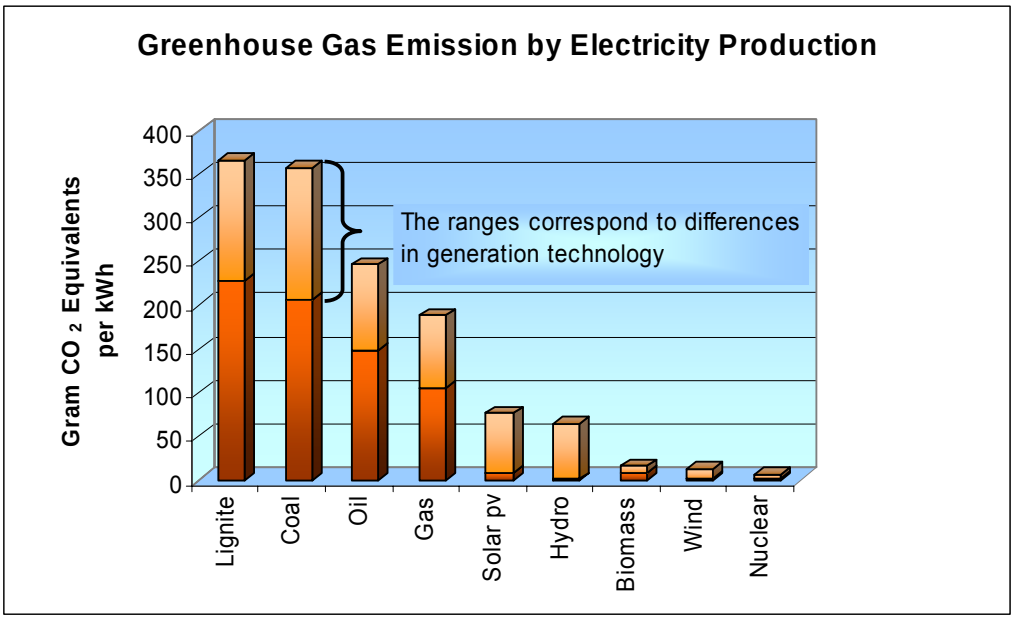

Figure 2.2: Greenhouse Gas Emission by Electricity Production Method.

(Source: OECD/NEA)

In the IEA reference scenario, global electricity use doubles between now and 2030 and even nearly triples in developing countries, its share of final energy consumption rising from 17 % to 22 %. Fossil fuels will continue to dominate the fuel mix for electricity generation, increasing their contribution from 66 % to 70 %. Therefore the global CO_2 emissions from the energy sector will see a two thirds increase, 75 % of which is due to the recourse to coal, especially in China and India. The historic and estimated future world energy consumption and carbon dioxide (CO_2) production are shown in Figure 2.3.

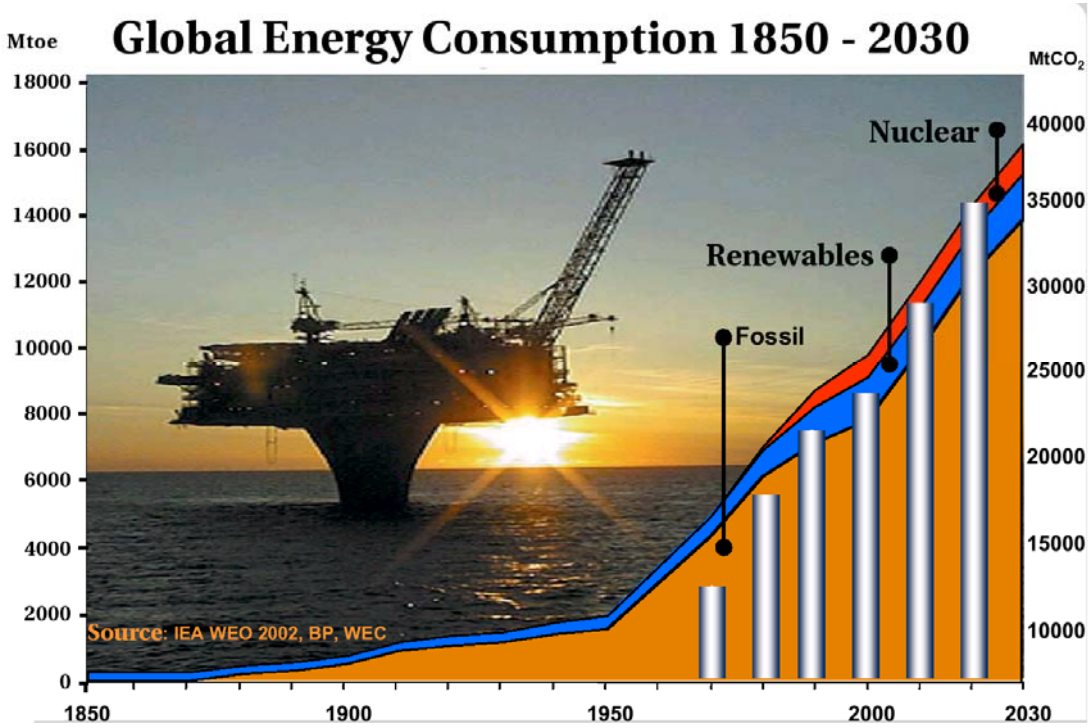

Figure 2.3: Historic and Estimated Future World Energy Consumption and Carbon Dioxide (CO_2) Production (IEA Reference Scenario).

(Source: IAEA bulletin n° 42/2, 2000)

2.3 The EU Situation

The current European energy policy was formally approved by the Brussels European Council on March 8 - 9, 2007. The Commission's Energy and Climate Change Package including an EU Strategic Energy Review was endorsed. The Council then underlined the vital importance of limiting the global average temperature increase to not more than 2°C above the pre-industrial level. It was stated that "an integrated climate and energy policy is needed" and that three objectives should be pursued:

1. *Increasing security of supply.*
2. *Ensuring competitiveness of European economies and availability of affordable energy.*
3. *Promoting environmental sustainability and combating climate change.*

The Council emphasised that the EU is committed to transforming Europe into a highly energy-efficient and low greenhouse gas emitting economy. It was stressed that in pursuing these objectives, the Member States' choice of energy mix and sovereignty over primary energy sources should be fully respected.

The EU Climate and Energy Package - Targets by 2020:

1. *Reduction of greenhouse gas emissions by 20 % compared with the 1990 level.*
2. *Reduction of energy consumption by 20 % compared with the 1990 level.*
3. *Increase the share of renewable sources in the EU energy mix to 20 %.*
4. *Increase the share of biofuels in transport petrol and diesel to 10 %.*

The Council emphasised that the EU makes a firm independent commitment to achieve at least a 20 % reduction in greenhouse gas emissions by 2020 compared with the 1990 level. The Council also stressed the importance of an internal market for gas and electricity and security of supply including the response to potential crises. A crucial issue is Carbon Capture and Sequestration which will be part of the European Strategic Energy Technology plan to be examined at the spring 2008 European Council meeting. The main remaining issue in creating an internal energy market is the unbundling of energy production and wholesale market from energy transmission and delivery. Also the increase of grid connections between EU countries is of vital importance.

As far as security of supply is concerned, it will be necessary to build emission-free coal fired power plants since coal and gas account for over 50 % of the EU electricity supply. The Commission is therefore working on designing a mechanism by which 12 large-scale demonstrations of sustainable fossil-fuel technologies can be in operation by 2015. The Commission believes that by 2020 all new coal-fired plants should include CO_2 Capture and Sequestration.

Saving 20 % of energy by 2020 is one of the EU's major goals. The Energy Efficiency Action Plan includes actions to make energy appliances, buildings, transportation and energy generation more efficient. If successful, the EU would by 2020 save 780 million tonnes of CO_2 release each year.

A Renewable Energy Roadmap paving the way towards a 20 % share of renewables in the EU energy mix by 2020 is part of the EU Climate and Energy Package. Member states will be required to establish National Action Plans for each of the renewable energy sectors – electricity, biofuels, heating and cooling. The roadmap also includes a coordinated development of biofuels within the EU so that by 2020 biofuels will reach in each country 10 % of transport petrol and diesel, compared with the current the average of 1 %.

Concerning nuclear energy, the Commission stated that it is one of the largest sources of CO_2 free energy in Europe but it is up to the member states to decide future developments. In 2004 the 152 reactors in the EU produced on average 6.7 TWh_e (Tera Watt Hours Electricity) per reactor corresponding to 15 % of the EU energy consumption or 31 % of the electricity consumption. To replace aging reactors, 50 new reactors need to be built before 2030. Safer operation and improved waste management will successively be implemented, an example being the new Finnish reactor. Although existing nuclear power plants are very competitive, high capital costs of any new nuclear investment greatly increase the business risk of nuclear energy for companies operating on purely market based conditions.

The 2004 EU consumption of renewable electricity is: hydro 304 TWh, wind 59 TWh, biomass 69 TWh and of nuclear electricity 986 TWh.

In the EU the role of the stock market in electricity trading can be predicted to increase. Currently, in the *Nord Pool* area more than 70 % of consumed electricity is traded via stock (*Nord Pool* became in 1996 the first international commodity exchange for trading electric power). This applies also to emission allowance trading, where the volume traded through stock is continuously increasing.

2.4 The Situation in Norway

The oil and gas industry is Norway's largest and most important industry. Since the start-up of activities on the Norwegian Continental Shelf (NCS) in the early 1970s, the industry has been characterised by growth and increased production. The situation today is different; oil production decreases every year and only a few discoveries are ready for development. At the same time, there are large expected resources on the NCS which remain to be proven. In spite of more than 30 years of production, only around one third of the total expected resources on the NCS have been extracted.

Norwegian oil production has remained at a plateau level of about 3 million barrels per day (bpd) since 1995, which is about 3.5 % of the world production, of which about 2.5 million bpd are exported. This makes Norway the world's tenth-largest oil producer and fifth-largest oil exporter. In a few years, oil production is expected to gradually decrease. From representing approximately 35 % of the total Norwegian petroleum production in 2006, gas production will probably increase its share to more than 50 % in 2013. The production of petroleum in Norway, which includes oil, gas, NGL (Natural Gas Liquids, a collective term for grades of liquid petroleum) and condensate (a type of light oil that contains a proportion of wet gas) from 1971 to 2006, is shown in Figure 2.4. In the longer term, the number and sizes of new discoveries will be a critical factor for the production level.

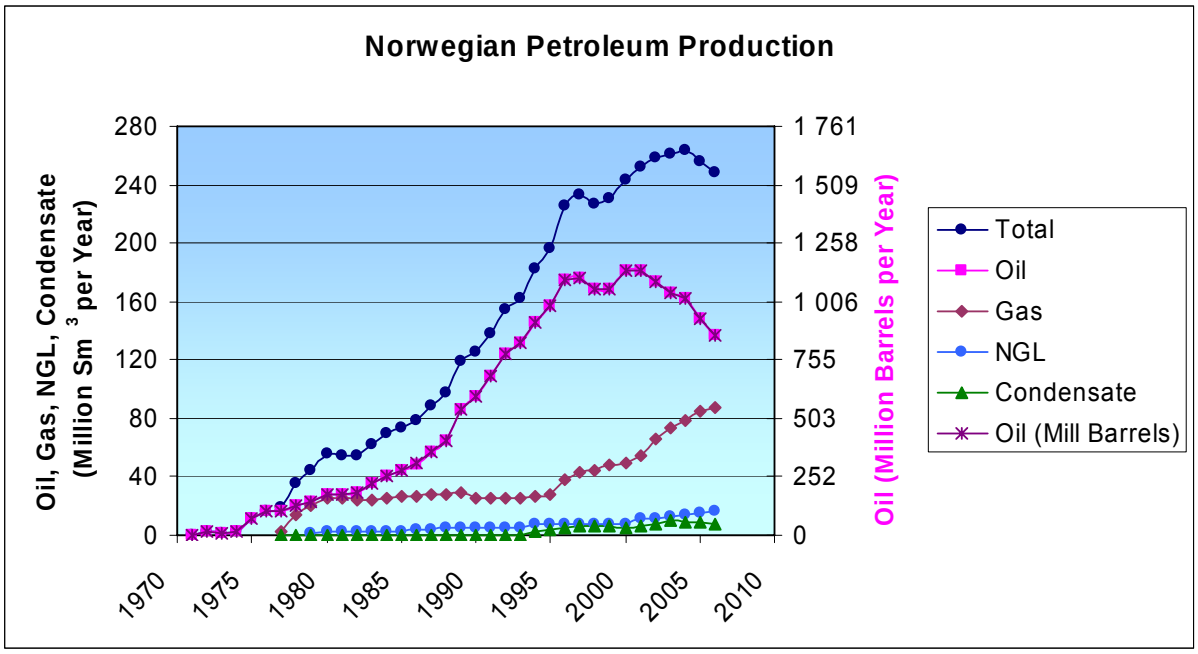

Figure 2.4: Production of Petroleum in Norway from 1971 to 2006.

(Source: Statistics Norway and Norwegian Petroleum Directorate)

Norway enjoys cheap and clean hydropower and has developed this resource extensively. Abundant offshore oil and gas resources and cheap hydropower have enabled Norway to enjoy a high level of security of electricity supply. Norway is the largest exporter of petroleum (oil, oil products and natural gas) in Europe, and is also contributing significantly to Europe's security of supply.

Norway deregulated its electricity market in 1991. *Statnett* was established as a transmission system operator from 1992, and a tariff system (access tariffs) was implemented - a prerequisite to customers' ability to choose their suppliers freely. *Statnett* established a power exchange, called *Statnett Marked*, in 1993; and initially this exchange covered only the Norwegian market. The exchange changed its name to *Nord Pool* when it became a common Norwegian-Swedish market, and *Svenska Kraftnät* stepped in as co-owner. During the 1990s, the Nordic countries (Denmark, Finland, Norway and Sweden) created a framework for a common electric power market based on open competition. The Nordic countries are the leaders in deregulating the electric power sector and, in particular, in organizing international trade in electricity. In 1996, *Nord Pool* became the first international commodity exchange for trading electric power. Approximately 200 utilities are competing to supply electricity to Norwegian customers in an open market, where customers can pick an energy provider at will and at no cost.

The electricity price development from 1996 to 2006 is shown in Figure 2.5.

A country's CO_2 Emission Allowance Units (EAU) can be bought and sold on the same market places as electricity.

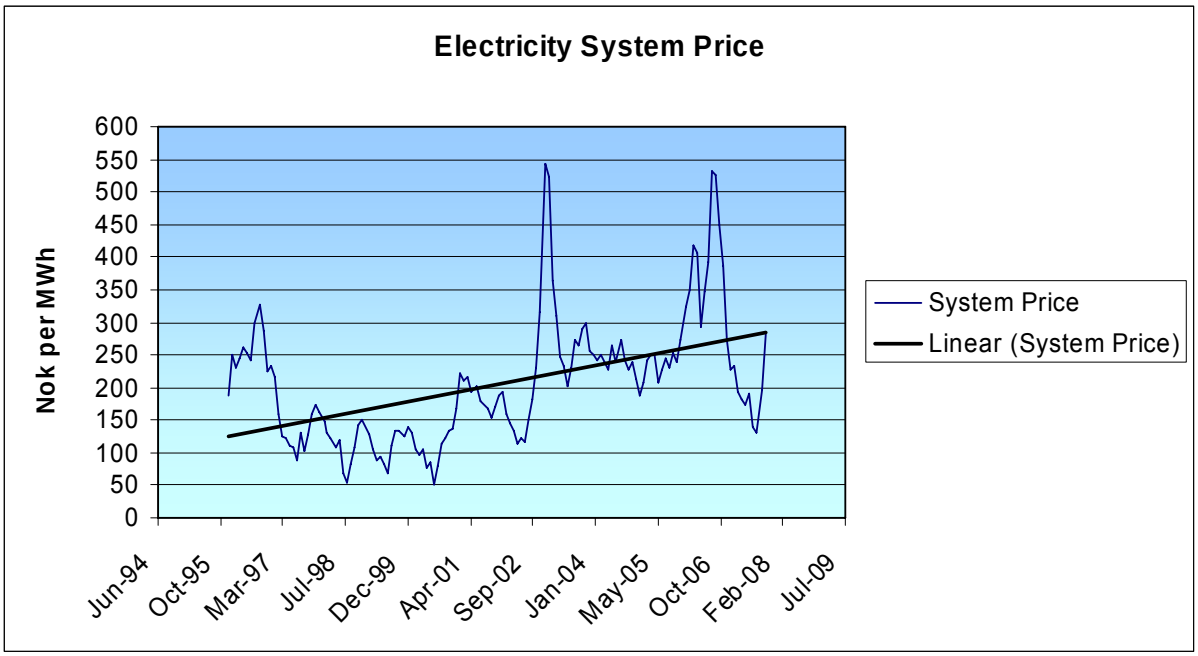

Figure 2.5: Electricity Price ("System Price") from 1996 to 2006.

(Source: Nord Pool)

Each of the four Nordic countries has a different mix of power generation facilities. Norway's power is almost 100 % from hydropower plants, Finland and Sweden have hydro, nuclear and other thermal plants, and Denmark has almost 100 % thermal production. About 25 % of the total Nordic power supply is generated by nuclear energy (30 % in Finland and 50 % in Sweden).

The low cost of electric power in Norway and a growing economy led to steadily increasing electric consumption throughout the 1990s. The electricity production in Norway has increased from around 60 TWh in 1970 to about 135 TWh in 2007. The variations from year to year are directly related to the weather variations since the production comes almost exclusively from hydro power (> 99 %). Figure 2.6 shows the electricity production and consumption in Norway from 1970 to 2007. The predictions toward 2020 are also indicated.

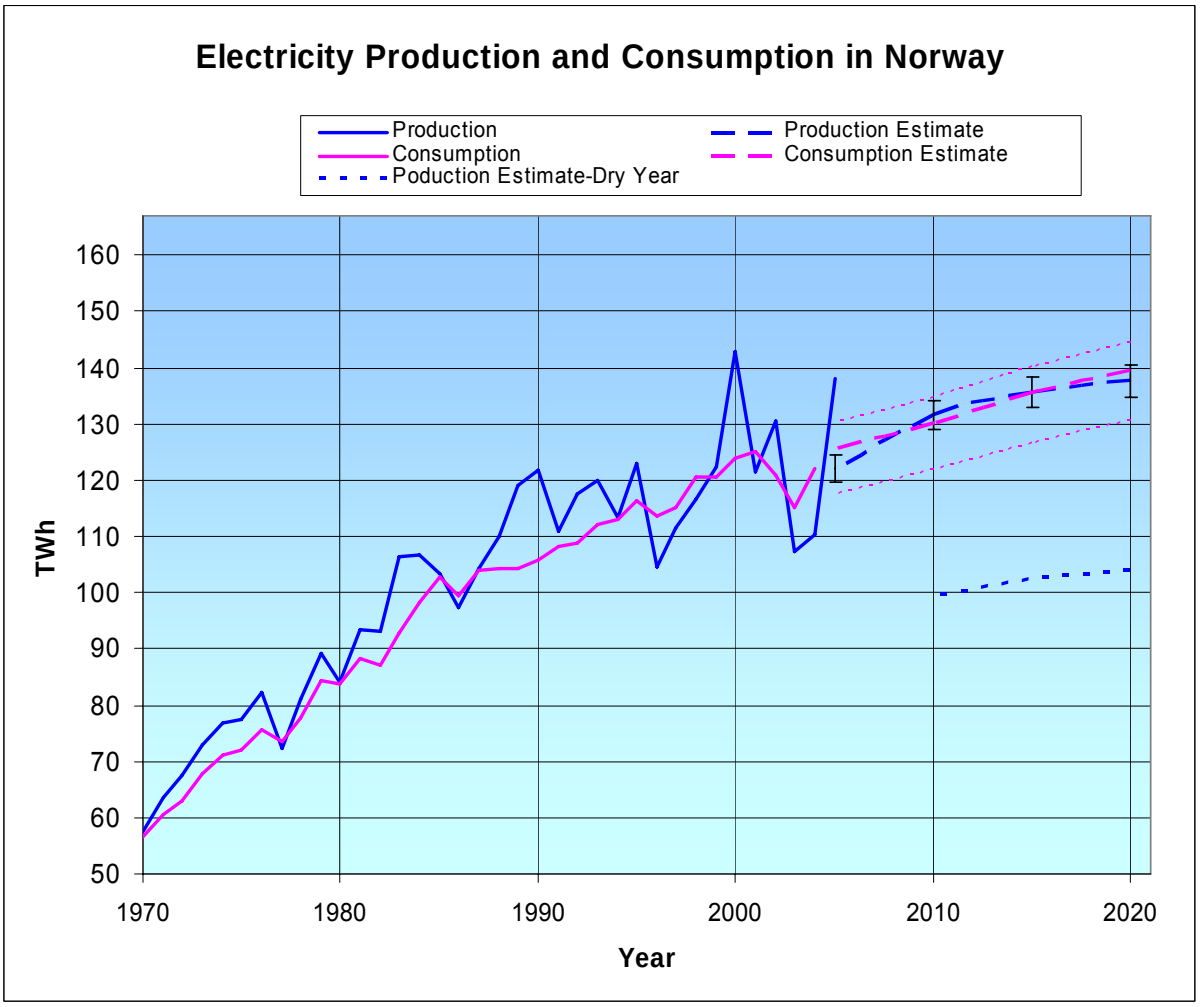

Figure 2.6: Electricity Production and Consumption in Norway from 1970 to 2007 (Estimated to 2020).

(Source: Norwegian Water and Energy Directorate, NVE)

The electricity production in 2020 is expected to be about 138 TWh. This production assumes a year with normal precipitation. In a dry year the production might be reduced to around 104 TWh. The consumption in 2020 is estimated to be about 140 TWh [3]. The role of the aluminum industry, which consumes about 8 TWh, may in the future change this consumption pattern and numbers. In 2020, with normal precipitations, the Norwegian electricity consumption is predicted to be slightly higher than the electricity production, while in a dry year a shortage in electricity production of about 36 TWh is expected.

Due to possible shortage in production, electrical power must be imported from other countries. Some changes will take place and give influence to the *Nord Pool* power capacity:

1. *New power cable between the Netherlands and Norway with a capacity of 6 TWh (finished 2010).*
2. *New nuclear power plant in Finland with a capacity of 12 TWh (finished 2011).*
3. *New power cable between Finland and Sweden.*

The total capacity for imported power to Norway will be 20 TWh in a dry year which is less than the expected shortage in the Norwegian electricity production. A dry year in Norway is normally correlated with a dry year in Sweden.

2.5 Status of Nuclear Energy

Since 1954 when the first nuclear power plant was producing electricity for a power grid (Obninsk Nuclear Power Plant, USSR) the electricity production from nuclear has continuously been increased. Nuclear energy uses a controlled fission reaction to generate heat. In nuclear power reactors the heat produces steam that drives conventional turbines and generates electricity (see Appendix A: Introduction to Nuclear Energy). Except for the processes used to generate the steam, nuclear power plants are similar to conventional coal-fired generation plants. As of August 2007 there are 439 reactor units in operation in the world contributing to about 16 % of the world's electricity production. The majority of the reactors are located in the USA (104 units), France (59 units), Japan (55 units) and Russia (31 units). See also [4].

Figure 2.7 gives an overview of the countries having nuclear reactors in operation and the number of units in the respective countries. In addition to the existing reactors there are 31 units under construction, mainly located in Russia (7 units), India (6 units) and China (5 units).

Of the Nordic countries, only Finland and Sweden have nuclear power plants for electricity production. Sweden has 10 units producing yearly about 65 TWh which corresponds to about 50 % of the Swedish electricity production. There are 4 units in Finland producing about 22 TWh corresponding to about 27 % of the electricity production. In addition, one power plant is under construction in Finland and will come into commercial operation in 2011.

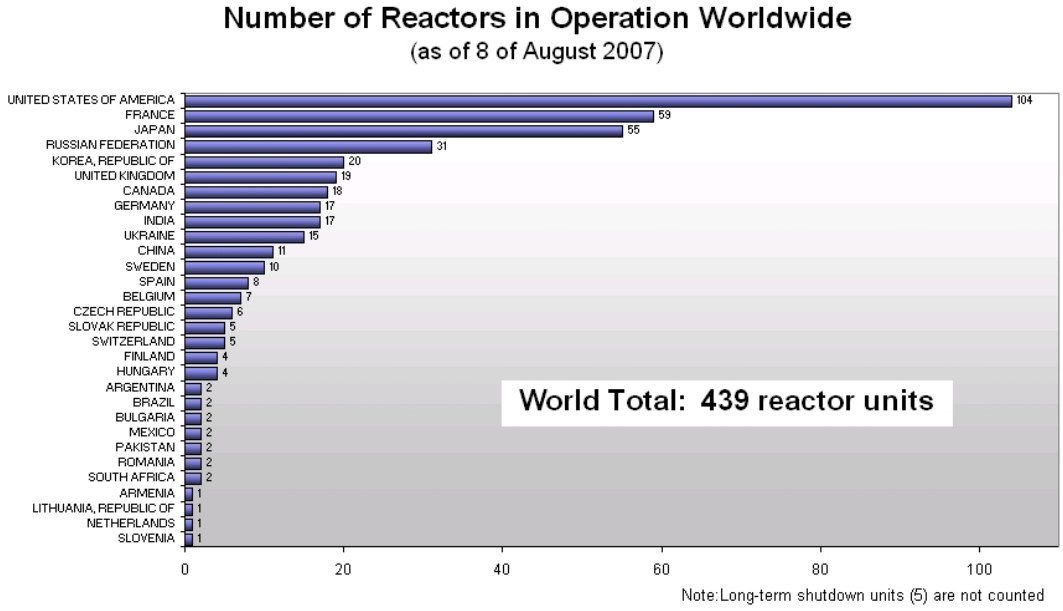

Figure 2.7: Numbers of Reactors in Operation Worldwide (as of August 8, 2007)

(Source: IAEA Power Reactor Information System (PRIS))

Today, most of the reactors in the world are more than 20 years old (~70 %), while as much as about 23 % are more than 30 years old. In 2030 more than 90 % of the reactors will be older than 30 years and about 80 % of the reactors will be older than 40 years (if no new units are built) and as a consequence of aging about 150 reactors will be shutdown at this time. The age distribution of operating reactors is shown in Figure 2.8.

The construction of a nuclear power plant including licensing and environmental assessments takes between 7 and 10 years. The planned lifetime of new reactors today is usually 60 years.

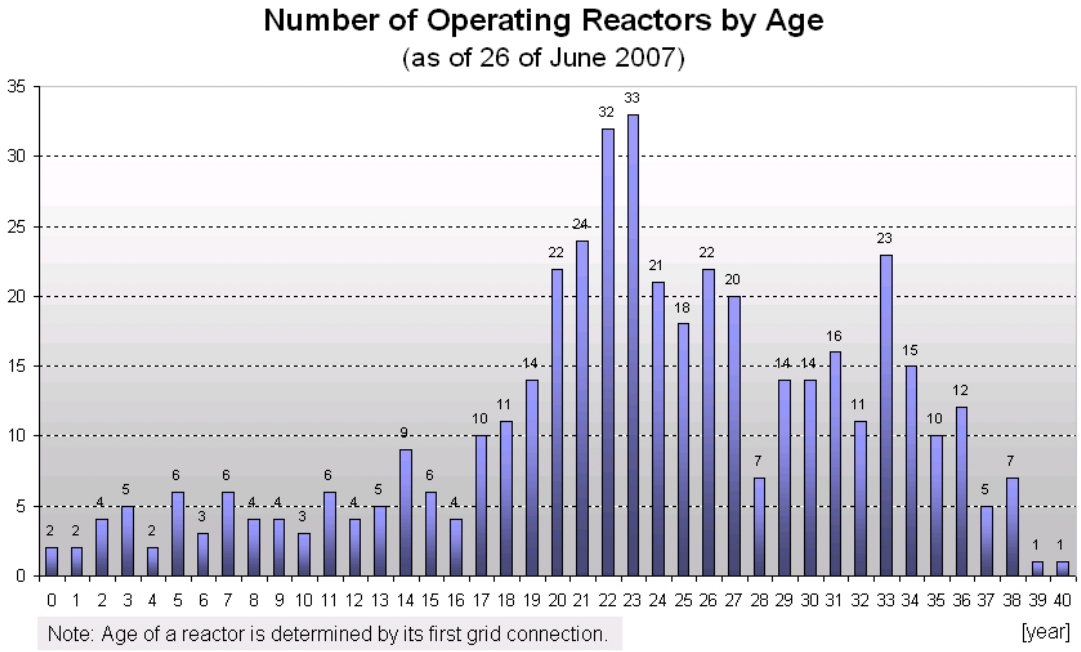

Figure 2.8: Number of Operating Reactors by Age (as of June 26, 2007).

(Source: IAEA Power Reactor Information System (PRIS))

The first reactors constructed in the 1950s and 1960s were early prototype reactors, so called *Generation I* reactors. The first commercial reactors were the so called *Generation II* reactors. Today, most of the reactors in operation are of the type *Generation II*. *Generation III* reactors are developments of any of the *Generation II* nuclear reactors incorporating evolutionary improvements in design which have been developed during the lifetime of the *Generation II* reactors, such as improved fuel technology, passive safety systems and standardized design. Some *Generation III* reactors are already in operation and some other *Generation III* or *III+* are in construction or planned. The next generation reactors, *Generation IV*, are a set of (theoretical) nuclear reactor designs currently under Research and Development (R&D). Figure 2.9 gives an overview of the evolution of nuclear power.

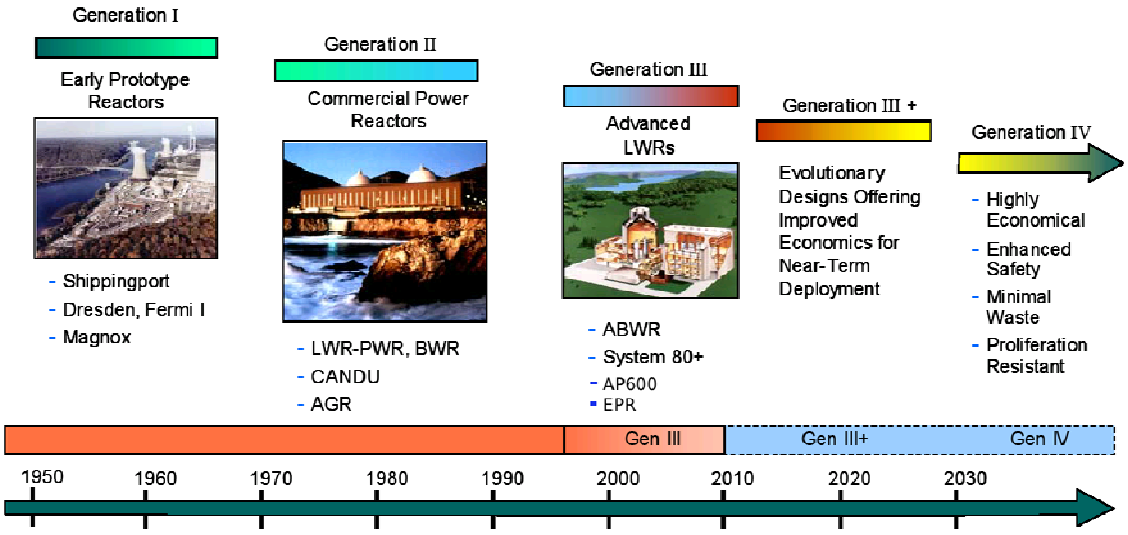

Figure 2.9: The Evolution of Nuclear Power.

Launched in 2000 by the US, the *Generation IV* initiative is aiming at fostering the collaborative efforts of the world's leading nuclear technology nations to develop next generation nuclear energy systems to meet the world's future energy needs. The Generation IV International Forum (GIF) defined eight goals for these systems in four key areas: economics; safety and reliability; sustainability; and proliferation resistance and physical protection. GIF also selected the six most promising systems that offer advantages in these four key areas and could be deployed commercially by 2030 [5].

Nowadays, GIF has the following 13 members: Argentina, Brazil, Canada, China, Euratom, France, Japan, Republic of Korea, the Russian Federation, Republic of South Africa, Switzerland, the United Kingdom, and the United States. Currently, eight Members are Parties to the Intergovernmental Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems (Canada, China, France, Japan, Republic of Korea, Switzerland, the United States and the European Atomic Energy Community (Euratom)) while the five others have not yet acceded to or ratified this Agreement

The European Atomic Energy Community (Euratom), established in March 1957, gives the ability to pursue Framework Programs for nuclear research and training activities. The main focus of Euratom is fusion energy covered by the ITER program (International Thermonuclear Experimental Reactor) which is an international collaboration to build and operate an experimental fusion facility. The objective of ITER is to demonstrate the scientific and technological feasibility of fusion energy. With the present know-how, it is hard to see the large scale commercialization of fusion based reactors before the middle of this century and not even very soon after that.

The role of thorium (Th) as fuel in nuclear reactors has been discussed since the early 1960s. The thorium isotope, Th-232, is not *fissile* which means that it cannot undergo fission if bombarded with neutrons. On the other hand, Th-232 is *fertile* which means that new fissile material uranium-233 (U-233) can be produced by irradiating thorium in a nuclear facility.

2.6 Uranium Resources

Total identified resources of uranium are divided into two groups: *Reasonably Assured Resources* (RAR) and *Inferred Resources* (IR). The Reasonably Assured Resources comprise the uranium that occurs in known mineral deposits of such size, grade and configuration that it could be recovered within given production cost ranges, with currently proven mining and processing technology. I.e. the *Reasonably Assured Resources* are very well known resources while *inferred resources* are less well known.

The categories are internally divided into various cost classes according to suggested extraction costs. The classes “below 40 US$/kgU”, “below 80 US$/kgU” and “below 130 US$/kg U” are the most widely used. Very often resources of type “RAR < 80 US$/kgU” are regarded as being equivalent to "proven resources”.

The global identified resources of uranium with mining costs up to 80 US$/kg amount to about 2.6 million tonnes. The uranium resources, when used in Light Water Reactors (LWRs), correspond to an energy equivalent of 28 billion tonnes hard coal. Deposits which contain large uranium resources and which can be mined in a cost-effective way are distributed to many countries, as shown in Figure 2.10 (US does not report resources in the IR category). These countries possess about 96 % of the global uranium resources.

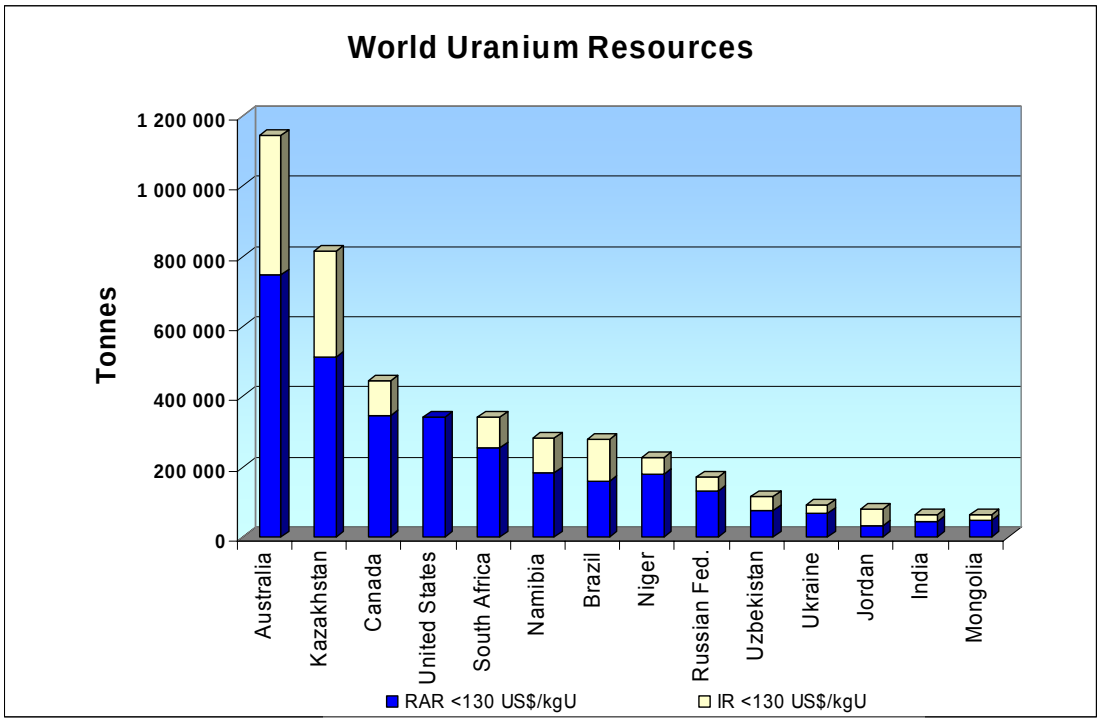

Figure 2.10: The World Uranium Resources (Mining Costs < 130 US$/kgU).

(Source: OECD/NEA and IAEA, Uranium 2005: Resources, Production and Demand)

If mining costs of up to 130 US$/kg are taken into consideration the global uranium resources are increased to 3.3 million tonnes [6]. The total uranium resources (independent of mining costs) are estimated at 15 to 20 million tonnes.

Many uncertainties must be taken into consideration when calculating the world's resources of fissile material for energy production.

In December 1997, nations in the United Nations Framework Convention on Climate Change (UNFCCC) agreed upon the Kyoto Protocol. In this context, The Nuclear Energy Agency (NEA) has investigated the effect of using nuclear energy to reduce greenhouse gas emissions. In this report [7] three different scenarios are considered:

1. *Variant I: "continued nuclear growth"*

   Assumes that nuclear power capacity would grow steadily, reaching 1120 GW_e in 2050.

2. *Variant II, "phase-out"*

   Assumes that nuclear power would be phased out completely by 2045.

3. *Variant III, "stagnation followed by revival"*

   Assumes early retirements of nuclear units in the short term (to 2015) followed by a revival of the nuclear option by 2020 leading to the same nuclear capacity in 2050 as in variant I.

Using *variant I* with no reprocessing from spent fuel, the known world uranium resources will be exhausted shortly after 2040, assuming total identified resources (RAR + IR) of 4.5 million tonnes (Figure 2.11). Reprocessing all Light Water Reactor (LWR) spent fuel and recycling the uranium and plutonium in mixed-oxide fuel (MOX) for Light Water Reactors (loaded with 30 % MOX and 70 % uranium oxide fuel) would lead to a cumulative saving of some 600 000 tonnes of natural uranium by 2050.

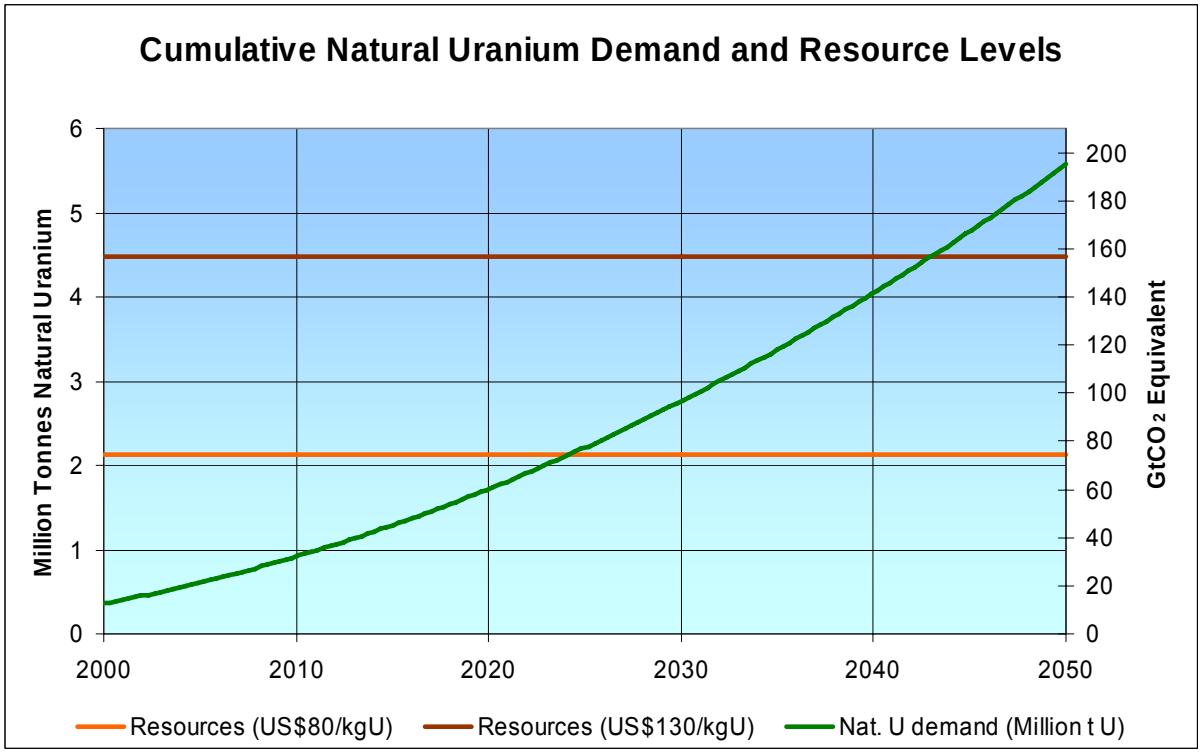

Figure 2.11: Cumulative Natural Uranium Demand and Resource Levels (Case: Continued Nuclear Growth).

(Source: Nuclear Energy Agency (NEA), Nuclear Power and Climate Change)

Under the assumption that nuclear power would substitute for a mix of fossil fuelled power plants emitting 800 gCO_2/kWh, which is an average value for the existing energy mix, the cumulated reduction of Greenhouse Gas Emissions to the atmosphere will be about 100 $GtCO_2$ in 2030 (see Figure 2.11).

The price of nuclear fuel for electricity production is influenced by many steps in the production process. The uranium mines produce U_3O_8 (*Yellow Cake*) that has to be *converted* to UF_6 which is the chemical form needed for the enrichment process of U-235. The conversion price is the processing cost to convert the *Yellow Cake* to UF_6 (given in US$/kg UF_6). After conversion to UF_6, the U-235 content must be increased in the *enrichment process*. This process has a unity cost (Separative Work Unit, SWU) depending on the required enrichment in the fuel and on the U-235 content in the tails (depleted uranium).

To produce 1 kg uranium with 4 weight percent (wt%) of U-235 requires 7.4 kg natural uranium and 8.4 Separative Work Units for the enrichment process. The spot price history of uranium is shown in Figure 2.12. Only a small part (maybe 5 – 10 %) of uranium consumed is traded at this spot price and the rest by bilateral contracts.

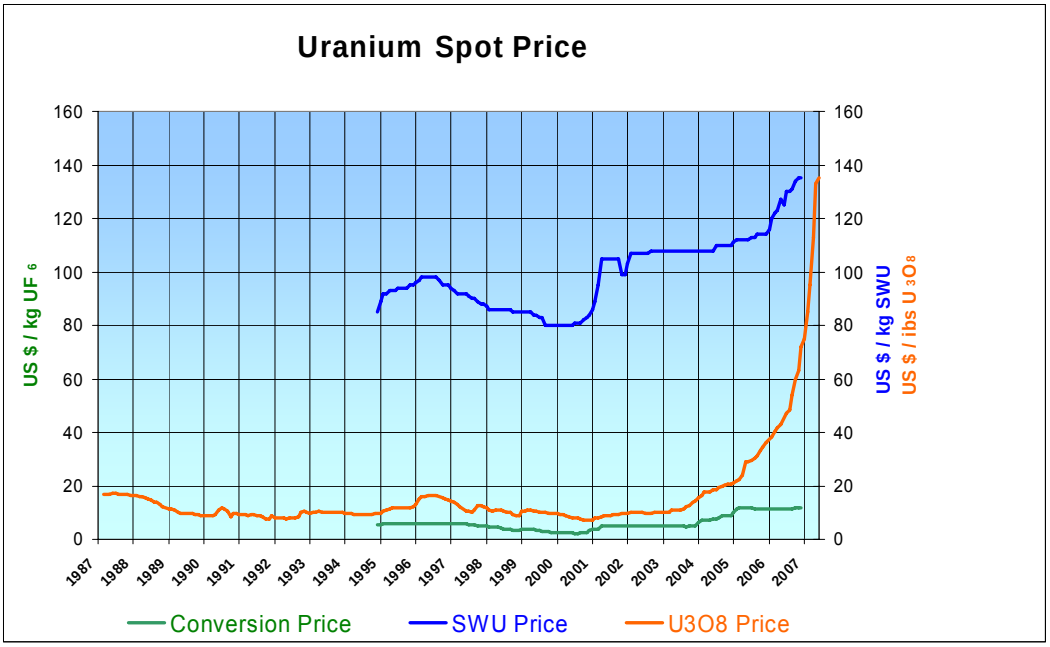

Figure 2.12: The World Uranium Spot Price History.

(Source: The Ux Consulting Company, LLC)

2.7 Thorium Resources

The primary source of the world's thorium is the rare-earth and thorium phosphate mineral, monazite. In the United States, thorium has been a byproduct of refining monazite for its rare-earth content. Monazite itself is recovered as a byproduct of processing heavy-mineral sands for titanium and zirconium minerals. Without demand for the rare earths, monazite would probably not be recovered for its thorium content. Other ore minerals with higher thorium contents, such as thorite, would be more likely sources.

Thorium deposits are found in several countries around the world as shown in Figure 2.13 [8]. The largest thorium reserves are expected to be found in Australia, India, USA, Norway, Canada, and in countries such as South Africa and Brazil. According to the US Geological Survey, the world thorium reserves and reserve base (resources) are shown in Figure 2.14.

According to Figure 2.14, Norway is known to have among the largest thorium reserves in the world, about 170 000 tonnes and about 150 000 tonnes as reserve base (resource). A *resource* refers to a situation where metals or minerals are enriched. The resources can be developed to a *reserve* (or *deposit*) when further investigations prove that the enrichment can be economically exploitable. This also implies that the metal or the mineral can be recovered using a viable process.

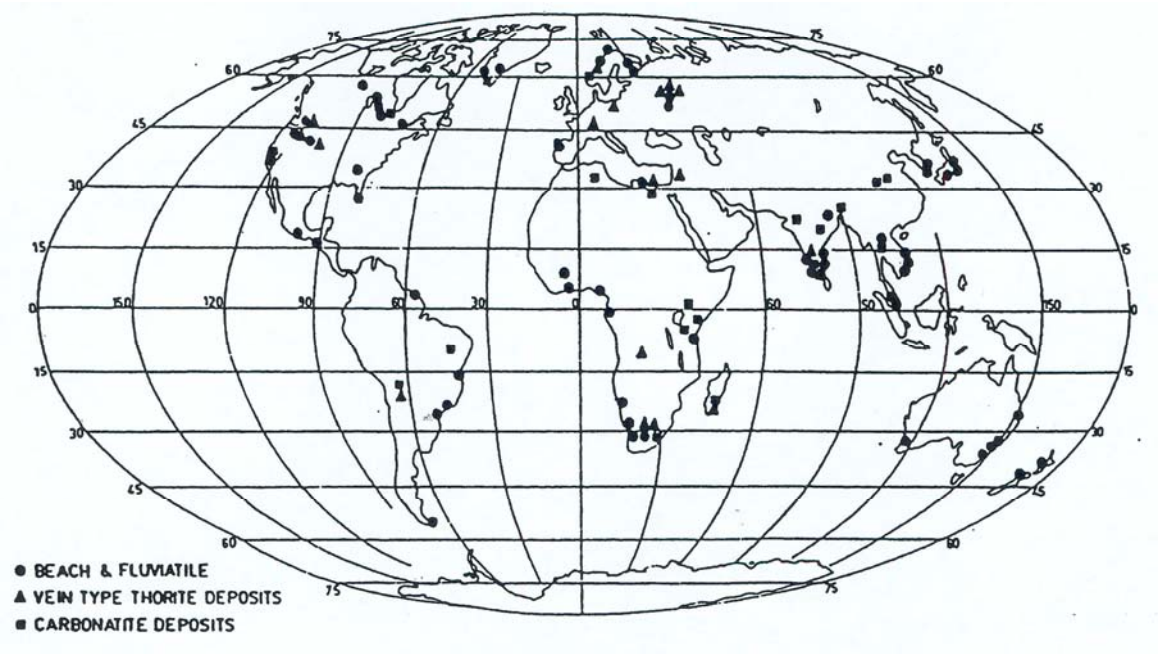

Figure 2.13: Thorium Deposits in the World.

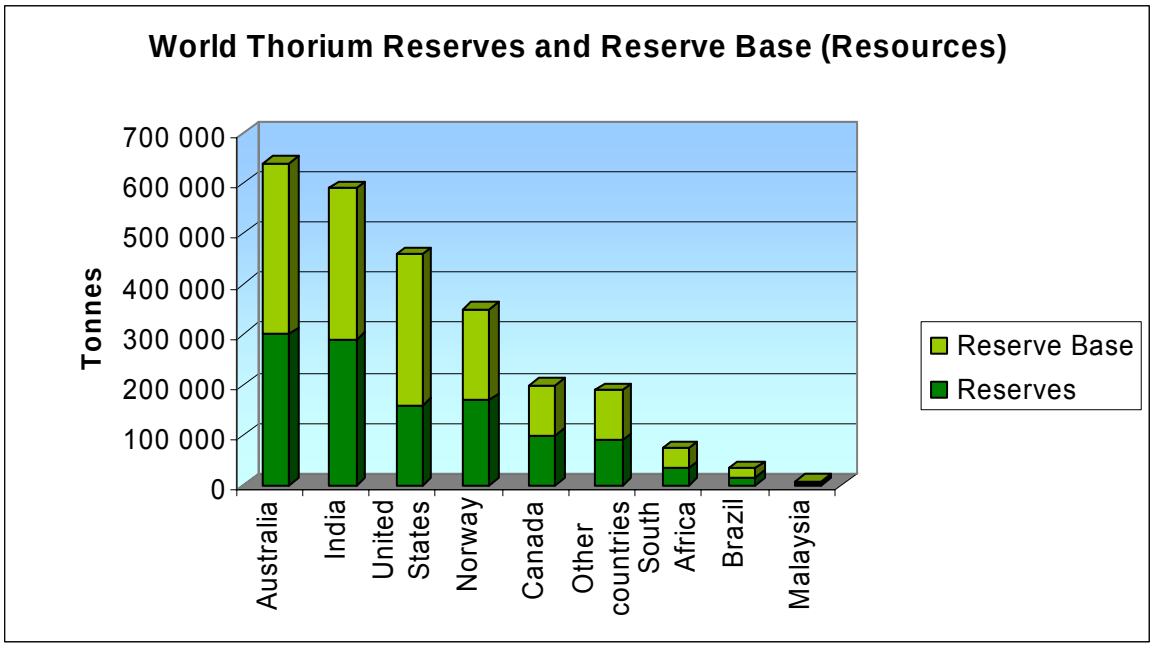

Figure 2.14: The World Thorium Reserves and Reserve Base (Resources).

(Source: US Geological Survey, Mineral Commodity Summaries, 2007)

2.8 Worldwide Activities on Thorium

Worldwide, the highest activity on thorium as a nuclear energy source is found in India where both Kakrapar-1 and -2 units are loaded with 500 kg of thorium blanket. Kakrapar-1 was the first nuclear reactor in the world to use thorium in the blanket, rather than depleted uranium, to achieve power flattening across the reactor core. In addition, the use of thorium based fuel is planned in 4 reactors, which are currently under construction.

Today, the average yearly electricity consumption per capita in India is only about 600 kWh per person while the global consumption is about 2500 kWh per person. With a population of more than 1 billion that is growing and one of the fastest growing economies in the world the Indian consumption continues to increase. Today, the total installed electricity capacity in India is about 130 GW (3.5 % of the world total) to which nuclear power contributes about 3.5 %.

India has about 1 % of the world's uranium resources while the thorium resources are one of the largest in the world with about 300 000 tonnes. With about six times more thorium than uranium, India has made utilization of thorium for large-scale energy production a major goal in its nuclear power program, utilizing a three-stage approach:

1. Pressurized Heavy Water Reactors (PHWRs), elsewhere known as CANDUs (CANada Deuterium Uranium) fuelled by natural uranium and Light Water Reactors (LWRs) of the Boiling Water Reactor (BWR) and VVER types. In this stage plutonium is produced.

2. Fast Breeder Reactors (FBRs) that use this plutonium-based fuel to breed U-233 from thorium. The blanket around the core will have uranium as well as thorium, so that further plutonium (ideally high-fissile plutonium) is produced as well as the U-233.

3. Advanced Heavy Water Reactors (AHWRs) that burn the U-233 and plutonium with thorium, getting about 75 % of their power from the thorium.

India's future program on thorium based nuclear power is important for India's long term energy security. Some research and development activities are also carried out on the Compact High Temperature Reactor (CHTR) and on the subcritical Accelerator Driven System (ADS) including the development of a high power proton accelerator.

Since the early 1990s, Russia has had a program to develop a thorium-uranium fuel, which more recently particularly emphasises the utilisation of weapons-grade plutonium in a thorium-plutonium fuel. The program is based at Moscow's Kurchatov Institute and involves the US company Thorium Power and US government funding to design fuel for Russian VVER-1000 reactors [9].

3. THORIUM RESOURCES IN NORWAY

The element thorium (Th) was first identified in 1828 by the Swedish chemist Jöns Jakob Berzelius (1779 - 1848) when he examined rock samples from Norway. He named the element thorium and the mineral thorite after Thor, the Norse god of thunder. In 1890, Professor Waldemar Christopher Brøgger (1851 - 1940) described thorium rich minerals, also containing niobium (Nb), yttrium (Y), zirconium (Zr) and rare earth elements (REE), from the *Permian Oslo Province* (the so called *Oslofeltet*) [10]. In 1895, thorium mining was performed at *Kragerø* on the South East coast of Norway, for production of thorium oxide for gas burners. About 2000 kg of thorite was produced. The production declined after one year, due to the low market prices of thorium associated with large monazite (a mineral containing thorium) resources identified in Brazil.

In Norway, thorium has never been specifically explored for. The knowledge of geological environments with elevated levels of thorium is mainly based on results from uranium (U) exploration carried out by the Geological Survey of Norway (NGU) during two periods. The first period was after the Second World War and up to the mid 1960s, while the second period was from 1975 to 1985. Based on analyses of uranium in a large number of mineral samples, the correlation between uranium and thorium in minerals and on helicopter and ground gamma surveys, the thorium levels were estimated. Recent helicopter gamma surveys covering the *Permian Oslo Province* as well as the *Fen Complex* in Telemark County confirmed in principle the old data, although the spatial resolution was improved. Additional information on thorium in Norwegian bedrocks is based on previous exploitation of rare earth elements (REEs) that are commonly positively correlated with thorium (Megon, 1973 [11]). In the MEGON Report it was shown that most of the minerals with elevated levels of thorium are situated within 3 main regions: the *Fen Complex* in Telemark County, the *Permian Oslo Province*, and on the Southeast coast of Norway, the Kragerø and Langesund areas. Investigations performed by the MEGON group confirmed previous data produced by *Norsk Bergverk A/S*, investigating the *Fen Complex* in 1955 - 1956 [11]. A series of thorium bearing minerals have been identified at the different sites in Norway, while the *Fen Complex* was considered as the most promising resource, with thorium amounting to about 0.1 – 0.4 wt%.

3.1 Resources - Geological Environments with Elevated Thorium Levels

The average concentration of thorium in the Earth's crust is about 7.2 ppm (parts per million), which is about 2 to 4 times more abundant than uranium, reflecting the difference in half-lives of thorium-232 ($1.4 \cdot 10^{10}$ years) and uranium-238 ($4.5\ 10^{9}$ years). Thorium (Th) and uranium (U) are enriched in magmatic processes, and the highest concentrations are found in silica (SiO_2)-rich rocks such as granites. The concentration of thorium and uranium in rocks such as gabbro and granite can vary by a factor of more than ten. Due to small variations in their chemical behaviours, thorium and uranium are enriched along different paths during weathering and near-surface geological processes: uranium is mobile under oxidizing conditions and is deposited under reducing conditions, while thorium is less mobile and is often concentrated in heavy mineral sands [12]. The most important geological environments in which thorium are enriched include:

1. *Alkaline complexes and their pegmatites*
2. *Granitic pegmatites*
3. *Carbonatites*
4. *Heavy mineral sand*

3.1.1 Thorium in Alkaline Complexes and their Pegmatites

Most of the global resources of thorium are found in alkaline rocks and associated pegmatites. In Norway the major alkaline rock is situated in the *Permian Oslo Province*. The thorium level is enhanced in the whole Province and mostly in the southern part. The geochemistry of the rocks, soil and stream sediments in the Province is well-known. During the last few years, the area has been investigated by means of airborne radiometric surveys confirming previous findings [12].

In the Langesundsfjord area, including *Stokkøy* and *Arøy*, several Th-containing minerals have been identified. The bedrocks, both intrusive and extrusive rocks, are markedly enriched in thorium. The concentrations of thorium from the pegmatites are often high, but the volumes available are not known. *Sæteråsen*, an extrusive volcanic rock, has been studied, drilled and mapped, due to the high concentrations of zirconium (Zr), niobium (Nb), yttrium (Y) and REE. The rock also contains high levels of thorium (500 ppm). Mineral separation performed in 1980 showed, however, that the rock (trachyte) was difficult to exploit with respect to these metals as it is very fine-grained. A preliminary estimate of the thorium volumes amounts to 8 million tonnes. *Høgtuva* in Nordland County has been identified as a potential resource for beryllium (Be), Zr, REE, U and Th. The beryllium resource has been drilled, and contains 0.5 million tonnes rocks with thorium (500 ppm). A wide zone of wall rock contains also similar grades of thorium. Thus, *Høgtuva* represents a low-grade thorium resource.

3.1.2 Thorium in Granitic Pegmatites

Granites are generally enriched in uranium and thorium, and especially pegmatites are usually the most enriched phase. The great advantage in exploiting pegmatites is that they are coarse-grained rocks. However, the grade of thorium in coarse-grained granitic pegmatites is difficult to estimate.

In the County of Østfold and along the coast of southern Norway from *Bamble* in Telemark County to Rogaland, granitic pegmatites containing Th-minerals have been identified. Pegmatites with Th-minerals were also identified in other granitic terrains, e.g. Nordland County, but these have not been studied in detail. In the old *Ødegården Mine* in *Bamble,* a flat-lying sill containing 0.1 wt% thorium at 100m depth has been reported. Already in 1906, 500 tonnes of the sill were taken out and thorium enrichment was carried out with some success. However, the information on this resource is scarce [12].

3.1.3 Thorium in Carbonatites

Carbonatites are exotic magmatic carbonate rocks, and these rocks are often exploited for phosphorus (P) and REEs. The carbonatites with high thorium levels are those enriched in iron (Fe) and/or often in magnesium (Mg) as observed in e.g. parts of the *Fen complex* in Telemark County. In addition to the *Fen complex*, known carbonatites in Norway include carbonatites on the coast of Finnmark (the islands *Stjernøy* and *Seiland*). The carbonatites from *Stjernøy* have been thoroughly investigated and contain low levels of Th, while the carbonatites at *Seiland* have not been examined in detail. On *Inderøy*, in the inner part of the Trondheimsfjord area, carbonate

veins with Th-minerals have been identified. The carbonates are thought to have been intruded from an underlying carbonatite. Locally these veins show high Th-grades [12].

3.1.4 Thorium in Heavy Mineral Sand

On a global scale heavy mineral sand, mostly beach sands, are an important host for several minerals such as zircon (zirconium silicate, $ZrSiO_4$), garnet, etc. and metals such as titanium (Ti), cerium (Ce), Y, etc. and Th. Such thorium resources having significant size and grade are not known in Norway, neither as sands nor as metamorphosed sands.

3.2 The Fen Complex in Telemark County

The Fen area is geologically unique with its magmatic carbonate rock, situated in the Precambrian gneisses of Telemark, about 120 km South West of Oslo. The central intrusive represents a cross-section of the feeder pipe of a volcano, active about 600 million years ago [13]. Around the central intrusive, with diameter of about 2 km, numerous satellite intrusions occur. Within the Fen area carbonatites and carbonatite-complex geology are dominant [10], [14], [13]. The rock types which occupy the largest fraction of the Fen surface area are søvite (calcite carbonatite), rauhaugite (dolomite carbonatite), rødberg (hematite–calcite–carbonatite) and fenite (alkali-metasomatised granitic gneiss) [15]. An overview is given in Figure 3.1.

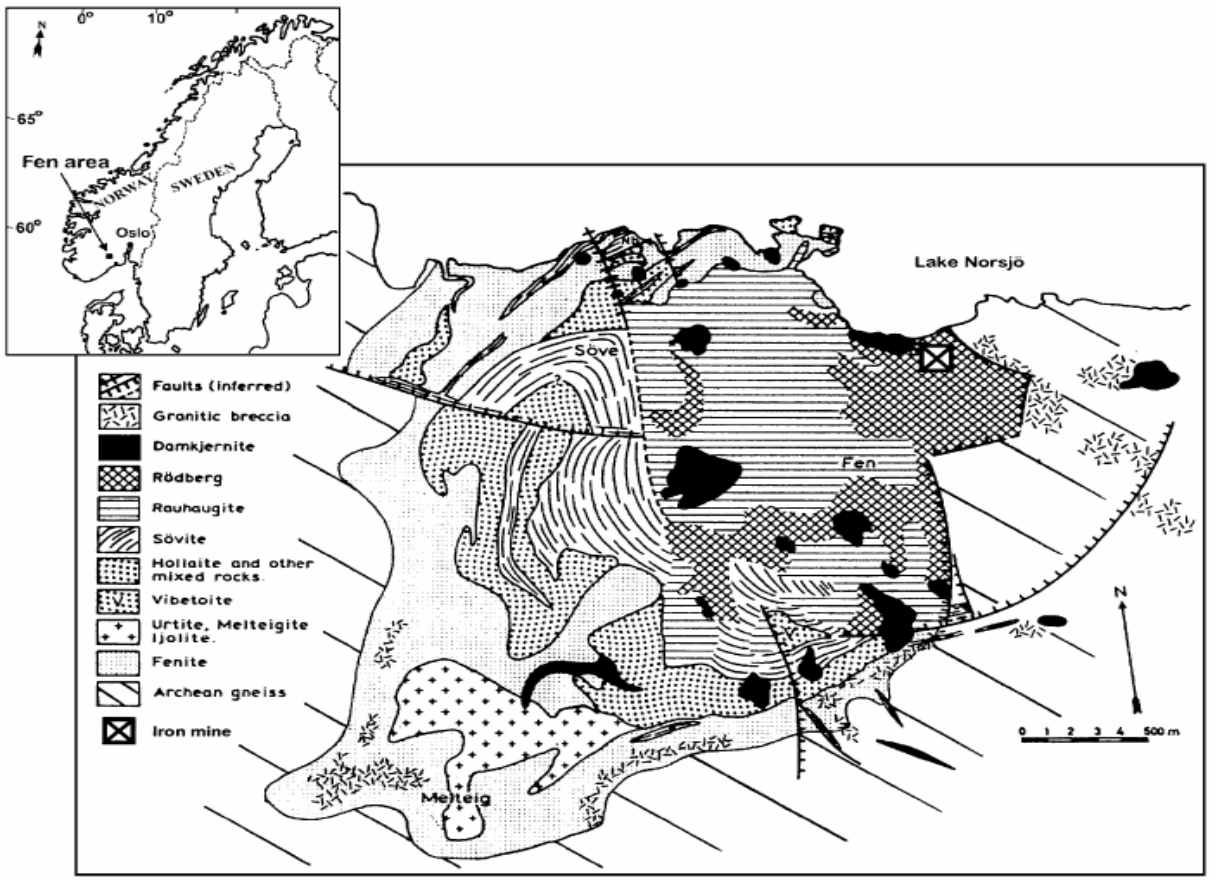

Figure 3.1: Bedrock Geology of the Fen Central Complex.

The thorium concentration in the Fen rocks is positively correlated with iron (Fe) [12]. Thus, the white carbonate part of the *Complex* has low levels of thorium. The summary report from previous investigations concluded that the thorium concentrations are:

- 0.4 wt% thorium in the iron ore.
- 0.2 - 0.3 wt% thorium in enriched zones in the iron-rich part of the ore.
- 0.13 - 0.17 wt% thorium in the iron-rich part in large volumes.

As shown in Table 3.1, the concentration of thorium (Th-232) is highest in rauhaugite and rødberg [15]. The iron-rich part of the carbonatite, rødberg, was mined for iron from 1655 until 1927, and søvite was mined for niobium (Nb) between 1953 and 1965. The carbonate rock has also been investigated for exploitation of phosphorus (P). Many pits and piles of tailings in the *Fen Complex* reflect the previous extensive mining activities. Today, about 350 dwellings are located within the *Fen Central Complex*.

Table 3.1: Activity Concentration of Thorium-232 (Th-232) in Rock Samples from the *Fen Complex*. 4 *Bq/kg (Becquerel [1]/kg) Th-232 is Equivalent to 1 ppm Th.*

Rock Type	Number of Samples	Th-232 (Bq/kg)
		Mean Range
Rödberg	9	3100 (390 - 5900)
Rauhaugite	9	600 (290 - 930)
Sövite	9	80 (20 - 190)
Fenite	8	130 (20 - 200)
Precambrian Gneiss	3	66 (68 - 63)

Based on recent helicopter surveys made by the Geological Survey of Norway [16] using a large NaI (Sodium Iodide) scintillator detector to detect the gamma radiation from the area, the distribution of thorium in the upper soils and bedrocks of the *Fen Complex* has been identified (Figure 3.2). Although thorium in deeper mineral deposits will not be included in the gamma measurements, the map shown in Figure 3.1 corresponds well with the distribution of rødberg in the bedrock of Fen. However, uranium is also present in a series of minerals present in the *Fen Complex*. Based on the helicopter survey where gamma radiation was determined, gamma energies from the Th-232 series and from the U-238 series could be differentiated. Thus, information of the distribution of uranium in the upper soils and bedrocks of the *Fen Complex* has also been obtained (Figure 3.3). Thus, traces of thorium-234 (Th-234), being a daughter of U-238 will also be present in the minerals. The distribution of thorium and uranium seems to correlate well, which indicates that uranium and its daughters will be present in the raw materials and the waste when mining and extraction of thorium is performed.

[1] *Radiation Emitted from a Source: Desintegration per Second = Becquerel (Bq)*

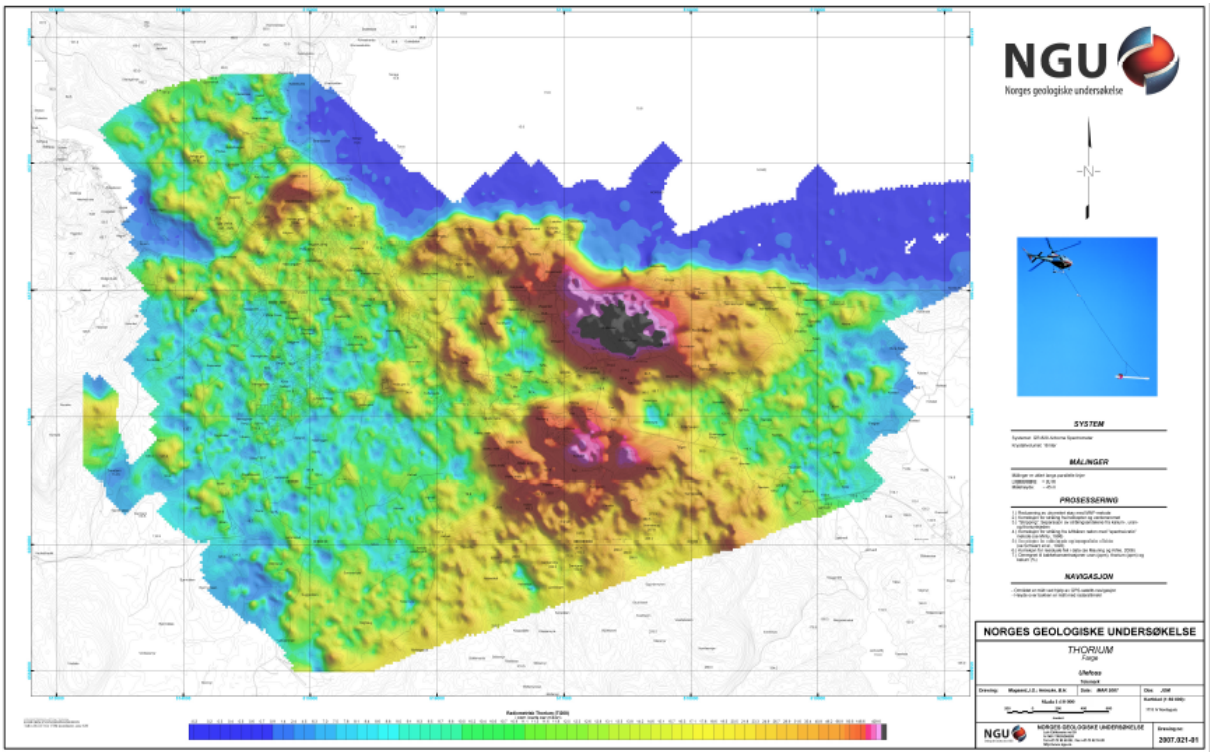

Figure 3.2: Thorium in the Upper Soil and Bedrocks of the Fen Central Complex.

(Source: Geological Survey of Norway, NGU)

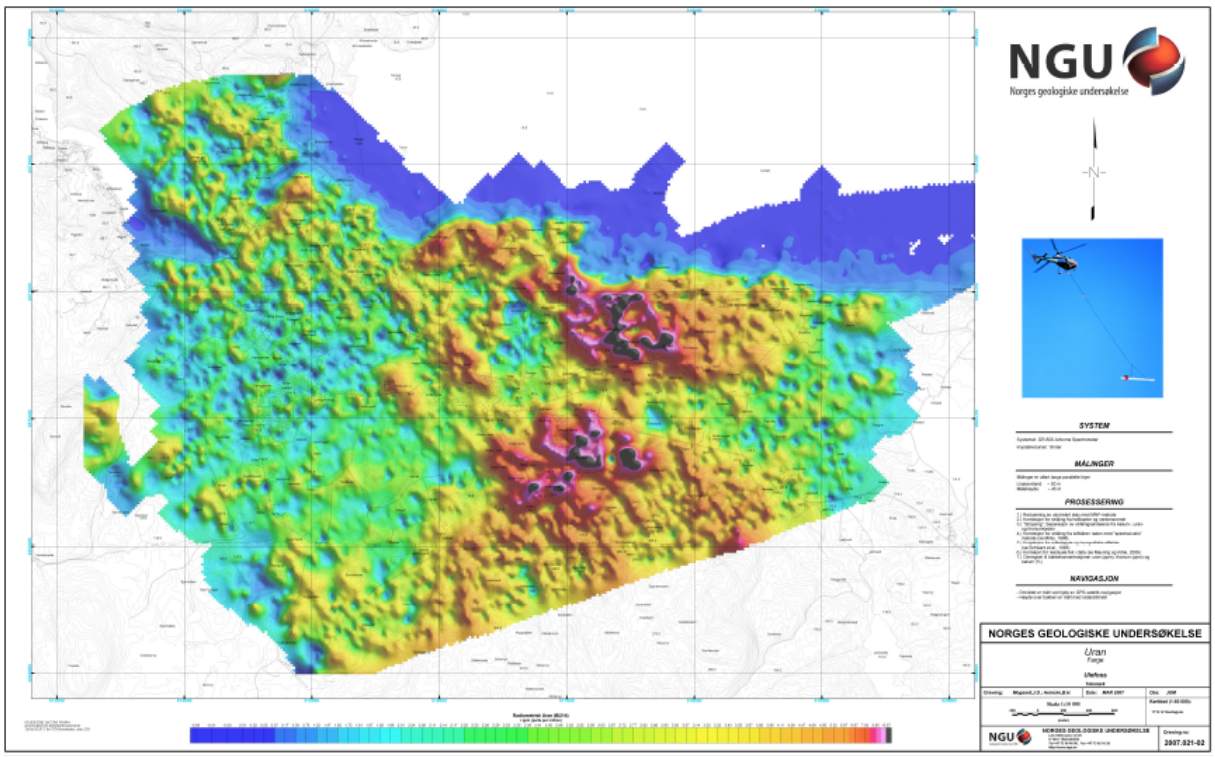

Figure 3.3: Uranium in the Upper Soil and Bedrocks of the Fen Central Complex.

(Source: Geological Survey of Norway, NGU)

The potential production of thorium concentrate from the *Fen Complex* will depend on the grain-size of thorium enriched minerals. Flotation is a well-established traditional method for mineral separation. In sulphide flotation the processes can utilise mineral grain sizes down to 20 micrometers (10^{-6} m) in the best cases [12]. In pegmatite flotation the mineral grain size has to be larger than 50 micrometers to get a satisfactory recovery. In the Megon report (1973) the possibility of exploiting the rare earth elements (REE) in the Fen carbonatite was evaluated. The conclusion stated that the REE minerals were so fine-grained that REE could not be enriched with satisfactory recovery using traditional techniques from the 1960s and 1970s. Therefore, other more advanced methods for mineral separation should be utilized and adapted to the specific deposits, but in general, separation of very fine-grained minerals is a great challenge.

Investigation on possible exploitation of thorium in minerals from the *Fen Complex* is currently proceeding. Based on archive iron-rich carbonatite samples, including drill cores, the Geological Survey of Norway has performed a preliminary study of the mineralogy [12]. Using electron microscopy, the grain size of the Th-containing minerals was found to be about 1 micrometer (Figure 3.4), indicating that the grain size of the Th-minerals is too small for enrichment by the traditional flotation processes. This is in accordance with previous mineralogical investigations, although a maximum grain size for REE minerals of 40 micrometers has been observed locally. Thorium occurs predominantly in the mineral monazite, but occurs also in other minerals, especially in REE minerals.

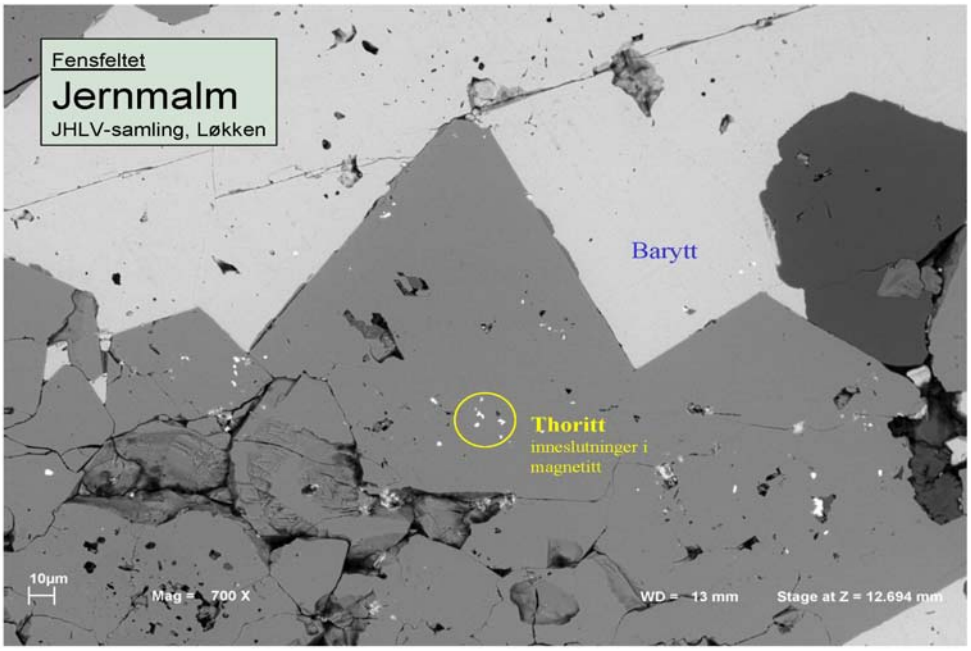

Figure 3.4: Photo of a Thin Section of Fen Iron-Rich Carbonatites Examined on SEM-Electron Microscope.

(Source: A. Korneliussen, Geological Survey of Norway, NGU)

3.3 Summary of the Thorium Resources in Norway

Figure 3.5 shows a geological map of Norway where the thorium rich areas are indicated. According to present knowledge most of the thorium enriched minerals are situated within the *Fen Complex* in Telemark County, the *Permian Oslo Province* covering Vestfold, in the Southeast coast of Norway covering the Kragerø and Langesund area, as well in Aust-Agder, Nord-Trøndelag, Nordland and Finmark.

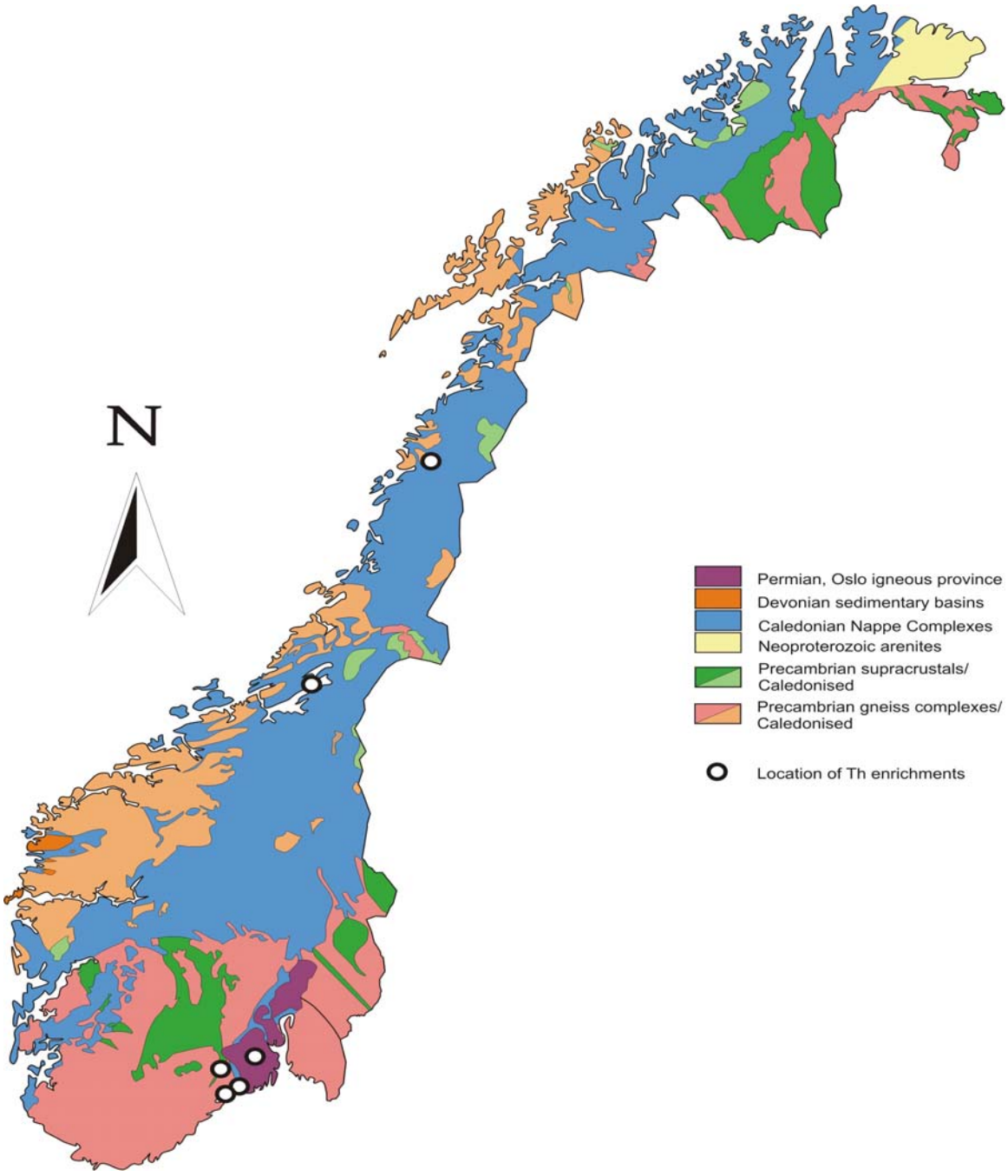

Figure 3.5: Map of Norway Showing the Location of Thorium-rich sites.

(Source: Geological Survey of Norway, NGU)

A series of thorium bearing minerals has been identified at different sites in Norway. Lindahl (2007) has collated present knowledge on the grades (wt%) and volumes of the identified thorium resources, as shown in Table 3.2. The grades vary from rather low (0.2 wt%) in Nordland to relatively high (0.4 wt%) in the *Fen Complex*. The volume estimates are, however, highly uncertain and in most cases not known. The *Fen Complex* has been considered as the most promising resource, with an estimated volume of about 170 000 tonnes referred to as reserve and 150 000 tonnes referred to as reserve base (US Geological Survey). The scientific basis for these old estimates of the Fen resources is somewhat unclear. These estimates have not been confirmed by the Geological Survey of Norway. Therefore, the volume given for the *Fen Complex* is referred to as "large", indicating that further investigations are needed to confirm the old estimates [12].

Table 3.2: Thorium (Th) Resources in Norway (Location, Concentration and Amount).

Location of the Resource	Grade of Thorium (wt%)	Volume (Million Tonnes)
Stokkøy/Arøy, Vestfold	0.2	?
Sæteråsen, Vestfold	0.05	8
Høgtuva, Nordland	0.02	0.5 (+)
Ødegården, Aust Agder	0.1	?
Ytterøya, Nord-Trøndelag	0.2	?
Fen (iron ore), Telemark	0.2 - 0.4	?
Fen (iron-rich), Telemark	0.1 - 0.2	Large

To conclude the discussion above, Norway has potential thorium *resources* that through more detailed investigations and the development in mineral processing can qualify to be a *reserve*. Already in 1895, about 2000 kg of thorite was produced during 1 year at Kragerø. The *Fen Complex* is considered as the most promising resource, and the thorium enriched carbonatite of the *Fen Complex* represents so far a "chemical reserve". However, most information available on the thorium resources in Norway is based on limited studies performed from the 1960s to the 1980s and knowledge of the grade and associated volumes is scarce and uncertain. Thus, thorough investigations of the resources, not only in the *Fen Complex*, but also at other sites in Norway, as well as mineralogical and mineral separation studies should be performed to decide whether thorium in Norwegian rocks can be exploited and thereby qualify to be a reserve.

4. THE FRONT END OF THE THORIUM FUEL CYCLE

Recently, the thorium cycle has been re-investigated by the EU [17] and the IAEA [18], [19], [20]. The present chapter is based on these sources and studies performed in the framework of the EU program "*Thorium Cycle as a Waste Management Option*".

Production of thorium has been limited due to the limited demand (it is used in special glass and alloys); it is a by-product of rare earths preparation. Its production is a few hundred tonnes per year only. Production of thorium reached about 1000 tonnes in the 1970s [21], to decrease thereafter due to a decrease in demand. During the period from 1960 to 1970, thorium was mentioned as both an alternative and a successor to uranium for nuclear power reactors. Therefore the possibility of geological exploration for new occurrences and methods to treat the various thorium-containing minerals became important. Already in 1895, about 2000 kg of thorite was produced during one year in Norway. In 1979 the current thorium production estimate was 200 tonnes per year [22].

The total amount of thorium which has been produced worldwide until 1988 is estimated at 37 500 tonnes [8] (some 17 000 tonnes in the US, 8 000 tonnes in Australia, 6 000 tonnes in China, 4 000 tonnes in India (to provide for the ongoing program) as well as about 2 000 tonnes in Germany). Some quantities are produced in other countries such as Canada, Malaysia, Brazil, Turkey and the former USSR. France has produced about 2 000 tonnes of purified thorium nitrate, sold mostly to the US.

It is interesting to note that in the USA alone, the ashes accumulated yearly from coal-fired power plants contain 700 tonnes uranium and 1700 tonnes thorium [23].

The half-life of thorium ($T_{1/2}$) is $1.4 \cdot 10^{10}$ years (U-238: $4.5 \cdot 10^{9}$ years). The element consists almost entirely of the isotope Th-232. Th-230, with a half-life of $7.5 \cdot 10^{4}$ years, is present in trace amounts. The fact that thorium has a longer half-life than uranium contributes to the fact that its natural occurrence in the earth's crust is higher.

Natural thorium has a density of 12 g/cm^3 (uranium: 18.7) and melts at 1750ºC (uranium at 1130ºC). Thus thorium is a highly refractory metal. Thorium dioxide melts at 3300ºC (UO_2 at 2800ºC).

The front end of the thorium fuel cycle comprises thorium ore processing, production of thorium metal or oxide and fuel fabrication.

4.1 Thorium Ore Processing

The processing of thorium ore for nuclear applications involves several activities:

- Mining.
- Extraction of the thorium-bearing minerals.
- Refining to remove impurities, especially neutron absorbing materials.
- Production of thorium metal or oxide.

These activities are addressed in the following sections.

4.1.1 Mining and Extraction

Thorium is found in a number of minerals. Therefore, there are several basic process flow sheets with modifications that can be considered for metal recovery [24]. The thorium is often associated with other minerals and may be present as a by-product of another process. Examples of this are the beach sands concentrates in which titanium and zirconium may be the metals of primary interest, or the processing of uranium ores, such as those found in the Elliot Lake area of Canada, where associated thorium can be recovered as a by-product. Hence there are several process alternatives.

Physical and Magnetic Separation: Individual minerals - ilmenite, rutile, monazite, zirconium, sillimanite and garnet - are separated by utilizing the differences in their physical properties, i.e. specific gravity, magnetic susceptibility, electric conductivity and surface properties. The wet concentrate is passed through rotary dryers at up to 150ºC, the dried feed then being electrostatically or electromechanically treated. The electrically conducting ilmenite and rutile constituents are separated first. The non-conducting monazite, being heavy and moderately magnetic, is isolated by high intensity magnetic separators. The resultant concentrate contains 98 % monazite. Usually, feeds containing 1 - 2 % of heavy minerals found in combination with monazite can be concentrated to 90 % heavy minerals with an overall recovery of 85 - 90 %.

Heavy Metal Chemical Extraction: The chemical properties of thorium and rare earth elements associated with it are similar, so their separation is difficult and time consuming. Concentrates of the rare earths must be chemically processed to separate them from the other components forming the mineral and from impurities. Monazite, the chief commercial ore from which thorium is extracted, is chemically inert and any chemical treatment for extracting thorium must initially be very severe to achieve the complete dissolution necessary for the separation of the rare earth elements, uranium and phosphates. The most common dissolution processes are: (1) acidic – using highly concentrated sulphuric acid [25], and (2) alkaline – using highly concentrated sodium hydroxide [22].

4.1.2 Refining

For nuclear applications thorium has to meet stringent requirements of purity, particularly concerning neutron absorbing elements. This purity is obtained industrially by solvent extraction, by ion exchange, or by direct chemical precipitation.

Solvent Extraction: This process makes use of the different solubilities of some heavy metal compounds in organic solvents which are immiscible with water. Successive transfer operations between the aqueous and organic phases lead to a relative concentration of one heavy metal in the aqueous phase and another in the organic phase. In purifying thorium by this process, the crude thorium produced is converted to a nitrate which exists in an aqueous phase. Contact between the aqueous carrier and the organic solvent (tributyl phosphate (TBP) in kerosene or hexane) transfers thorium to the solvent from which it is stripped by scrubbing with dilute nitric acid. Nuclear-grade thorium nitrate is obtained with more than 99 % thorium purity.

Ion Exchange: This is an effective way of separating individual lanthanides in a pure state.

Direct Precipitation: Either foreign ions are added to precipitate insoluble salts, or variations in solubility with pH of complex salts are exploited.

If there is uranium present with thorium and rare earth elements, a more sophisticated extraction flow sheet, with two TBP concentrations and appropriate scrubbing and stripping

stages in liquid-liquid contactors, would separate uranium from thorium nitrates. The purity obtained is 10 ppm uranium in Th, and about 3 ppm Rare Earth Elements in thorium.

4.1.3 Reduction to Thorium Metal or Thorium Oxide

Purified thorium nitrate is used as feed material for producing thorium. The reduction of thorium compounds to the pure metal thorium is not easy because at its high melting point of about 1700ºC thorium reacts readily with hydrogen, oxygen, nitrogen, carbon and many oxides. The metal is usually produced as a sponge or a powder by one of the following methods:

- Thorium tetrafluoride is prepared by heating thorium to 325ºC and exposing it to anhydrous hydrofluoric acid gas. The thorium tetrafluoride is then reduced with calcium at 800ºC in the presence of a zinc chloride booster. The booster reacts with calcium in an exothermic reaction thereby aiding in liquefying the thorium compound and fluxing the slag. The zinc is subsequently removed by pyrovacuum treatment at 1360ºC.
- Thorium oxide is obtained by precipitating thorium oxalate from a thorium nitrate solution with oxalic acid and calcinating the thorium oxalate at 650ºC or above. The thorium oxide is then reduced to metal with calcium at 1000 - 1100ºC, using calcium chloride as a flux.
- Electrolysis of tetrachloride or tetrafluoride: thorium tetrachloride may be obtained by chlorinating a mixture of thorium oxide and carbon at 600ºC and purifying the first distillate by redistillation. Alternatively, thorium tetrafluoride may be obtained as described above. Molten salt electrolysis of thorium chloride or fluoride in graphite crucibles, which act as the anode, with molybdenum as the cathode, results in thorium metal being deposited on the cathode.

4.2 Status of Thorium Fuel Fabrication Technology

The thorium fuel fabrication technology covers thorium metal fuel, thorium oxide fuel and thorium in mixed-oxide fuel.

4.2.1 Thorium Metal Fuel

Thorium metal pellets have been prepared to fabricate special fuel rods to be inserted into the CIRUS Thermal Indian Research Reactor for irradiation and U-233 extraction [26]. Thorium metal powder is obtained by reduction of ThO_2 by calciothermy (burning with calcium).

Thorium metal powder is quite ductile and can easily be consolidated by conventional powder metallurgical techniques. Thorium pellets of sintered density of about 98 % of theoretical density (TD) were produced by compacting at a pressure of 300 MPa followed by sintering in vacuum at 1300ºC for 1 hour.

Thorium has been converted into various shapes such as strips, blocks, rods, tubes, wires and foils, etc. by using conventional fabrication techniques. Extensive experimental investigations were carried out on the compatibility of thorium with various materials such as stainless steel, zirconium, chromium, vanadium, etc. It has been found that iron and nickel from stainless steel diffuse into thorium at 500ºC forming brittle phases such as $ThNi_x$. Thorium diffuses into zirconium at about 800ºC. Chromium and vanadium are both compatible with thorium up to 1000ºC.

Owing to its toxicity, radioactivity and pyrophoricity, adequate precautions are required in handling and processing of thorium. Pure or fresh thorium is a weak α-emitter but old thorium

with accumulated decay products also emits β-particles and penetrating γ-rays. However, as a result of the very long half-life of thorium, these emissions do not present a real danger for direct handling of limited amounts of thorium. Some shielding is required for large amounts. Thorium is a very active metal and in finely divided form it can be pyrophoric and in dust form it may be explosive. The presence of moisture and hydride as well as residual calcium may render thorium metal pyrophoric. The pyrophoricity can be minimized by giving thorium metal powder a protective treatment. This treatment usually consists in immersing the powder in an aqueous solution of a salt of a metal that is less electro-positive than thorium.

4.2.2 Thorium Oxide Fuel

The use of thorium in reactors has been mainly envisaged in the form of thorium dioxide (thoria) in view of the greater experience and proven performance of the oxide fuels. However, unlike the sintering of UO_x, where the sintering atmosphere and O/U ratio play an important role, ThO_2 being the most stable oxide known, can be sintered in any atmosphere such as air, hydrogen or vacuum.

Pelletizing ThO_2-based fuel pellets out of calcinated kernels seems to present the most attractive fuel fabrication process at present. It has been under development since the late seventies [27], [28], [29], [30], [31], [32], [33].

For fabrication of thoria kernels the sol-gel process has been developed. On the basis of this technology a procedure for production of pelletized fuel for water-cooled reactors has been developed. External gelation of thorium sols is used in order to produce gel kernels which could be calcinated and suitable for pelletization and sintering. The pelletizing of ex-gel-kernels avoids dust generation, is easy to remotize, and produces high density pellets with the desired microstructure at compaction pressures and sintering temperatures which may be even lower than for the conventional UO_2 manufacturing.

ThO_2-pellets can also be manufactured by the classical pelletizing process extruded from adequately pre-treated powder. The relevant experience is described in detail in the open literature:

- All ThO_2-based fuel in the US-programs from the early fifties until 1978 was produced by pelletizing from different powders. Other sophisticated approaches, such as duplex pellet manufacturing, were investigated on the laboratory scale. The necessary powders have been produced by nearly all known and applicable processes.
- The experience from India in fabrication of thoria-pellets using ex-oxalate powder was concentrated on powder granulation and on processing parameters as well as on sintering by the use of Mg-dopant as a sintering aid. The objective of this work was to improve the fabrication reliability for fast breeder blanket elements for the fast reactor at Kalpakkam, India.

The extrusion processes were also investigated. However, it seems that this approach is not very promising.

4.2.3 Mixed Oxide Fuel

Since thorium is a purely fertile material, a reactor containing thorium must also contain fissile material. The “driver” or “seed” material can be separated from the fertile fuel elements, but it is also frequently a mixed fuel: $(Th,U)O_2$ or $(Th,Pu)O_2$; where uranium could be U-235 or U-233. A

mixed fuel configuration permits to approach as closely as possible the homogeneous reactor state, the optimum for neutron conservation and breeding with thermal or epithermal neutrons.

The oxides are usually in shape of pellets/rods as sintered microspheres, or coated with pyrocarbon and included in a graphite matrix, or shaped into fuel "balls" as is the case for the pebble-bed reactors.

One of the major problems of the Th-232/U-233 fuel cycle is the penetrating γ-radiation of certain daughter products of U-232, which is always associated with U-233. Hence, handling of U-233 bearing materials requires remotization and automation in shielded glove boxes or hot cells for keeping the personnel exposure to radiation within permissible limits. Likewise, plutonium-239 (Pu-239) obtained by reprocessing spent uranium fuel from power reactors contains significant quantities of Pu-240, Pu-241 and Pu-238, which give rise to neutron and gamma radiation and hence also requires glove box containment and remote automated handling.

Indian Experience: Three fabrication methods have been explored for the $(Th,U)O_2$ and $(Th,Pu)O_2$ fuel pellets:

- Conventional "powder-pellet" route for both $(Th,U)O_2$ and $(Th,Pu)O_2$. This method involves simultaneous mixing-grinding of ThO_2 powder with UO_2 or PuO_2 powders, granulation, cold-pelletization of granules at 350 MPa and high temperature (1700ºC) sintering of pellets in an 15 % argon + 85 % H_2 atmosphere. With MgO doped ThO_2 powder, it was possible to achieve high pellet density in case of $(Th,U)O_2$ and $(Th,4\%Pu)O_2$ pellets at relatively low sintering temperature (1500ºC). Nb_2O_5 was found to have a more pronounced effect on densification of these ThO_2 based mixed oxides, achieving nearly 98 % of TD (Theoretical Density) at low sintering temperature (1200ºC).
- Pellet-impregnation technique for $(Th,U)O_2$ only. This concept consists of the preparation of relatively less radioactive ThO_2 pellets of low density by the conventional "powder-pellet" route in an unshielded area, impregnating these pellets in uranyl nitrate solution (U-233) in a shielded facility, followed by high temperature (> 1700ºC) sintering for densification. Fine U-233 bearing powders are avoided and handling of U-233 is restricted to specific sections of the fuel fabrication plants. $(Th,U)O_2$ pellets of high density (> 96 % of TD) and homogeneous uranium distribution could be reproducibly fabricated by this technique. However the limitation of the pellet-impregnation technique is that a maximum of 2 % U-233 could be introduced in the ThO_2 pellet.
- Sol-gel microsphere pelletization (SGMP) process for $(Th,U)O_2$ and $(Th,Pu)O_2$. This process was developed in collaboration with the Institute of Chemical Technology (ICT), KFA, Jülich, Germany. KFA's standard process, the external gelation of thorium for preparation of spherical fuel particles for high temperature gas-cooled reactor, was significantly modified to tailor-make gel microspheres. The sol-gel microsphere pelletization process minimises the radiotoxicity dust hazard as it is dust-free. It ensures excellent microhomogenity, microstructure and high density, and is amenable to automated and remote fabrication of highly radiotoxic $(Th,Pu)O_2$ and $(Th,U)O_2$ fuel pellets. Many other laboratories have tested the sol-gel process, among them KFA-Jülich and NUKEM in Germany.

4.2.4 Industrial Prototype Fabrication Plants for Fuels Containing Thorium

Among the more representative fabrication lines should be mentioned the semi-shielded KILOROD facility at *Oak Ridge National Laboratory (ORNL)*, and the development of the remote thorium-based fuel fabrication plant *TURF (Thorium-Uranium Recycle Facility)* by the ORNL team [34], [35].

Based in part on R&D pioneered by ORNL and other laboratories, private companies, especially *Du Pont* at Savannah River and *Battelle Pacific Northwest National Laboratory* at Hanford, Babcock and Wilcox, and others, under USAEC contracts, have installed pilot-to-industrial size facilities to produce the different types of fuel used in the prototype reactors, BWR, PWR and HTR. Fuel-based products could be obtained from *National Lead*, *Mallinckrodt*, *Nuclear Fuels Services*, among others.

Complete fuel element fabrication on the 10 kg scale was available at Babcock and Wilcox [36], and Allis-Chalmers had built a remotized fuel refabrication pilot facility at Rotondella, Italy, for Elk-River BWR fuel [37].

The KILOROD Fuel Fabrication at Oak Ridge National Laboratory (ORNL) was operated in the 1960s on a 10 kg/day scale. The process scheme was based on the sol-gel process and vibratory compaction of the powders into the pin cladding. 1280 kg of thoria were processed in an 8 months campaign. The average radiation exposures in the KILOROD program were 0.19 mSv (milli Sievert[2]) per man per week for the sol-gel process and the rod fabrication. Considering the accepted practical dose rates at the time (1968), which were of 20 - 50 mSv per year per operator, it was concluded that direct fabrication is feasible with (Th,3%U-233)O_2 fuels that contain less than 20 ppm U-232. With shadow shielding, this limit can be increased to 200 ppm. With shadow shielding and frequent cleaning and recycling, the limit is 600 ppm. Plants with larger capacities will require shielding when operating permanently.

Similar results were obtained at a *pilot fabrication plant of about 10 kg HM/batch operated* from 1965 *at Babcock and Wilcox* on the same ORNL process. Its cost was 1 million dollars in 1965. It consisted of a sol-gel preparation plant and a fuel fabrication plant. Both plants, like KILOROD, were partly shielded only. In 1968, the total production at the plant was about 1150 kg. Up to 1968, 119 rods (232 kg of (Th,3%U-233)O_2) were fabricated. The U-233 contained 42 ppm U-232.

The TURF ORNL Remote Fuel Fabrication Prototype Plant: As early as 1965, it was realised that industrial-scale fabrication plants would have to be totally remotized. ORNL engineers designed and built the TURF (Thorium-Uranium Recycle Facility) plant with remotely operated, manually maintained equipment having production capacities from 60 to 3700 kg/day of heavy metal. The TURF facility was intended to fabricate oxide (Th,U-233)O_2 fuel for water-cooled reactors and carbide fuel for HTGRs. The personnel radiation exposure is limited to 0.4 mSv/week. A shielding of 10 cm steel was chosen as a practical limit for semi-remote fabrication, because of the difficulty of working through a greater distance with gloved hands. A cost analysis based on these designs showed that remote fabrication of oxide and carbide fuels would cost almost 1.5 to 2 times more, respectively. The assumptions taken at the time do not now seem conservative, and heavier shielding would be necessary (~ 1.2 m) and hence possibly a higher cost.

Although such a plant could be built and operated, the fuel fabrication step has been the main roadblock to the development of the thorium fuel cycle to date. These technical complexities, especially penalizing 30 years ago, and the added costs, combined with other factors, have led to a provisory abandonment of this fuel cycle (except in India for strategic reasons).

The enormous technical progress accomplished since the 1960s, in mechanics, electrical motors, materials, electronics and computers, and practical examples of remotized fuel fabrication (MELOX in France) suggests that these fuel fabrication concepts be revisited so that their advantages can be realised in the not-too-distant future.

[2] *The Effect of the Absorbed Dose: Equivalent or Effective Dose = Sievert (Sv)*

The HTGR fuel element manufacture process was established in Germany, KFA Jülich and NUKEM, to make the reactor "pebbles" of about 60 mm diameter for the ATR and THTR reactors. About 650 000 such pebbles were manufactured. The coatings on the HTGR fuel particles operate at very nearly the same temperature as the fuel, 800 to 1300 ºC. These high temperatures intensify effects from radiation-induced dimensional changes, fission gas pressure build-up, and temperature-controlled diffusion processes. The effectiveness by which the fission products are retained in the different fuels is governed by a complex interrelation between many factors including radiation damage, swelling, chemical corrosion and composition changes at elevated temperatures. It can be deduced that, while HTGR carbide fuel can withstand very high temperatures for a long time and high burnup compared with LWR oxide fuel, it is much more complex and expensive to operate.

4.3 Fuel Properties and Performance

Considerable experience has accumulated in thorium fuel fabrication. For the Fort St. Vrain reactor 2448 hexagonal graphite fuel elements, 7.1 million fuel compacts and 26 000 kg of fissile and fertile material in TRISO-coated fuel particles were produced. This included almost 25 000 kg of thorium. It was irradiated at temperatures greater than 1300ºC to a maximum burnup in the fissile particles of 16 per cent fissions in initial metal atoms (FIMA) to a maximum fast neutron fluence of $4.5 \cdot 10^{25}$ n/m^2 with no evidence of significant coating failure. Over 50 tonnes of thorium fuel in ceramic form, clad in Zircaloy, was also manufactured for the Shippingport LWBR core.

4.3.1 LWR Fuel Performance

A number of irradiation tests was performed in water-moderated reactors to evaluate the in-pile behaviour of various (Th,U)O_2 containing fuel rods fabricated by either pelletizing, vipac or sphere-pac.

Fuel performance was extensively studied within the Oak Ridge National Laboratory irradiation program which started in 1961. The initial results on the performance characteristics were found to be favourable [38], [39]:

- In general, all thoria based fuels performed better than urania based fuels;
- Sol-gel derived (Th,U)O_2 fuels meet the basic performance requirements of a nuclear fuel;
- Vibration compacted (Th,U)O_2 fuel rods perform as well as those containing pressed and sintered pellets at moderate heat ratings up to 300 W/cm and burnup levels of 40 000 MWd/tHM (Mega Watt days per tonne Heavy Metal).

The work was continued by investigating the performance of (Th,U)O_2 fuel under different test conditions such as:

- High burnups and moderate heat ratings;
- High burnups and intermediate heat ratings;
- Moderate burnups and high heat ratings;
- Processing variables affecting the irradiation performance.

Additionally, special irradiation test programs had been performed under the *Light Water Breeder Reactor Program* and in the *Halden Boiling Heavy Water Reactor* [40] and in power reactors [41]. The behaviour of Th-based fuel in PWRs at heat loads up to 680 W/cm and to burnups of 80 000 MWd/tHM was studied. The Th-based fuel performed excellently compared to standard LWR fuel.

4.3.2 HTGR Fuel Performance

Thorium fuel performance was studied in the High Temperature Gas Cooled Reactor (HTGR) development program. These studies focused on fission product behaviour and irradiation performance of thorium or thorium-uranium mixed oxide in microsphere form.

Especially in the case of TRISO-coated pellets with SiC as a third layer, fuel particles showed excellent performance in irradiation tests under reactor operating conditions. This is true for all fuel kernels, whether UO_2, ThO_2, $(Th,U)O_2$, UC_2 or UCO. Under accident conditions, peak fuel temperatures for TRISO-particles are kept at or below 1600ºC. This is well below the 2000ºC limit at which significant fission product release has been measured.

Figure 4.1 shows typical fuel performance expressed as the release-to-production ratio of the fission gas Krypton-85 (R/B, where R = release rate; B = production rate) under reactor operating conditions and in irradiation experiments in material test reactors.

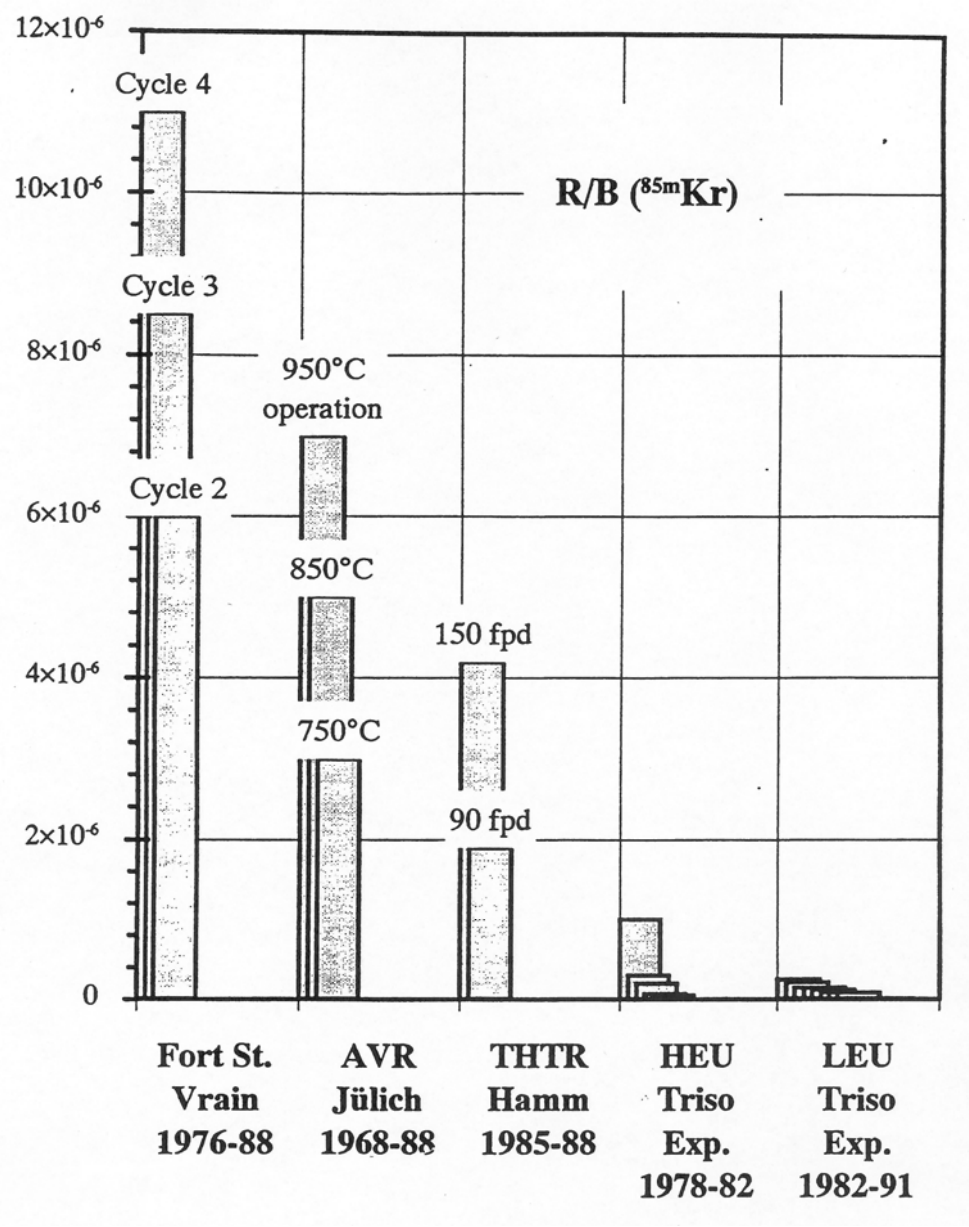

Figure 4.1: Fission Gas Release Rates with Modern, High Quality, TRISO Fuels.

5. NUCLEAR REACTORS FOR THORIUM

The thorium isotope, Th-232, is not *fissile* which means that it cannot undergo fission if bombarded with thermal neutrons. On the other hand, Th-232 is *fertile* which means that new fissile material uranium-233 (U-233) can be produced by irradiating thorium in a nuclear facility.

Like uranium-238 (U-238), the Th-232 isotope can be used in the nuclear fuel cycle as fertile material. Thorium-232 absorbs neutrons in the epithermal region and U-233 is produced via the decay of protactinium (Pa-233):

The Thorium-Uranium Fuel Cycle:

$^{232}Th + n \rightarrow {}^{233}Th$ (22 min) $\rightarrow$ ^{233}Pa (27 days) $\rightarrow$ ^{233}U (159 200 years)

Analogous to

The Uranium-Plutonium Fuel Cycle:

$^{238}U + n \rightarrow {}^{239}U$ (23.5 min) $\rightarrow$ ^{239}Np (2.4 days) $\rightarrow$ ^{239}Pu (24 110 years).Figure 5.1 shows the relevant isotopes in the thorium fuel cycle.

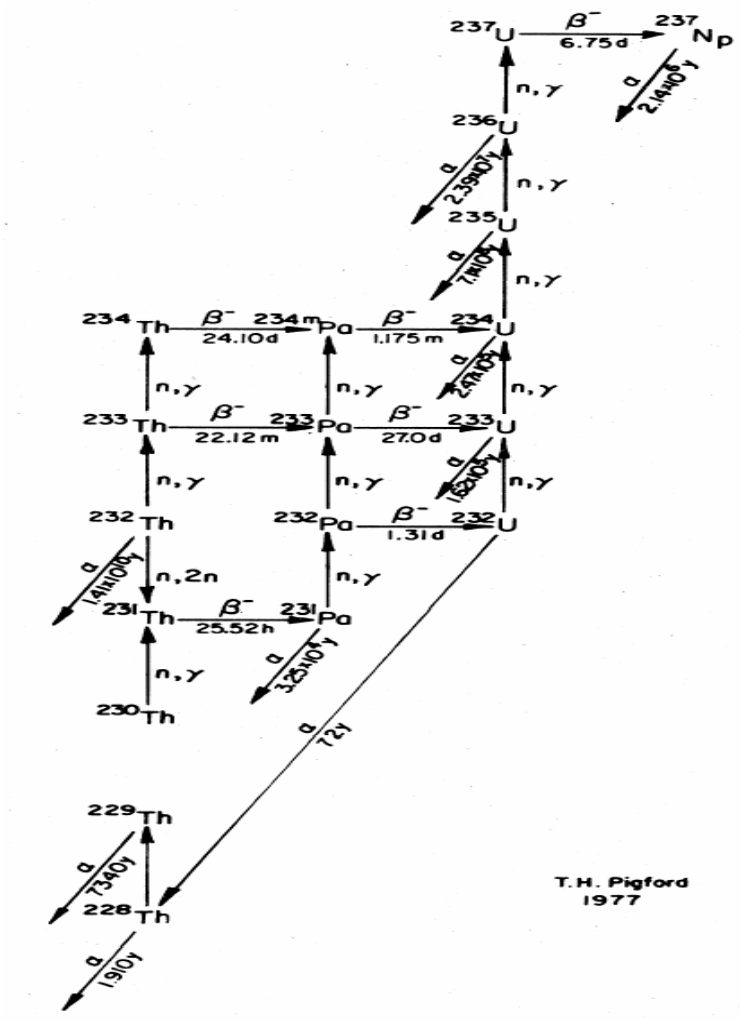

Figure 5.1: Relevant Isotopes in the Thorium Cycle.

5.1 Properties of the Fertile Material Thorium-232

Thorium-232 (Th-232) has many absorption resonances in the epithermal energy region (see Figure 5.2). These resonances cause the strong negative temperature coefficient of all reactors which contain this material and are designed in an appropriate manner. The absorption of neutrons in Th-232, followed by two successive beta decays, result in breeding of U-233. In the thermal energy region the absorption cross section of Th-232 ((n,γ)-reaction) is about 20 % larger than that of U-238. Above a threshold energy of 1.2 MeV (mega electronvolt) neutron capture will lead to direct fission of thorium however with a very small cross section (see Figure 5.3).

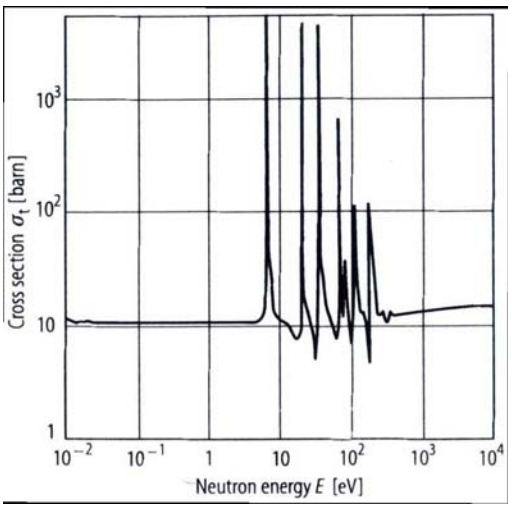

Figure 5.2: Absorption Cross Section of Thorium-232 (Th-232) in the Epithermal Neutron Energy Region.

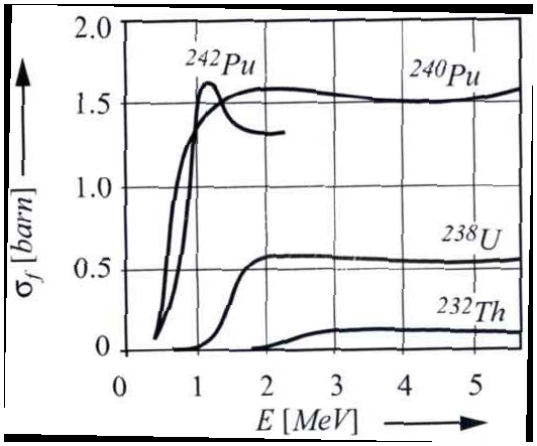

Figure 5.3: Fission Cross Sections at High Neutron Energies of the Isotopes Thorium-232 (Th-232), Uranium-238 (U-238), Plutonium-240 (Pu-240) and Plutonium-242 (Pu-242).

The formation of a high quantity of Pa-233 (half-life of 27 days) makes a management of Pa-233 necessary to optimize U-233 production in thermal reactors where the neutron balance is tight. This Pa isotope may not only decay to the fissile U-233, but may also be converted to Pa-234 via a (n, γ)-reaction with a high reaction cross section (110 barn). This leads to a significant loss with regard to the recovery of new fissile material. In addition, it causes a delayed reactivity increase after shutdown.

5.2 Properties of the Fissile Material Uranium-233

The fission cross section of U-233 is comparable to that of U-235 in the thermal neutron energy region. However, there are resonances above an energy of around 1eV (Figure 5.4). This has to be considered if shifts in the spectrum during massive transients in a thermal reactor are regarded. U-233 exhibits further fission and absorption resonances in the epithermal energy region which could influence the dynamic behaviour of a reactor loaded with large amounts of Th-232 and U-233. Like other fissile materials, U-233 has a small fission cross section at high energies.

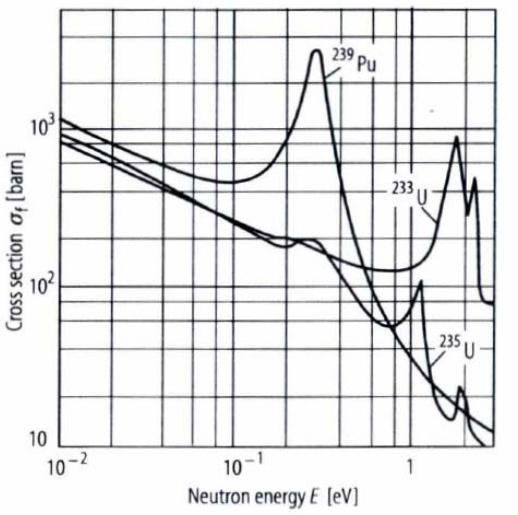

Figure 5.4: Fission Cross Sections of the Isotopes Uranium-233 (U-233), Uranium-235 (U-235) and Plutonium-239 (Pu-239) in the Thermal Energy Region.

The uranium isotope U-233 has very attractive features as a fissile material especially in the thermal energy region. For thermal neutron energies, U-233 has a higher *η-factor* than U-235 and Pu-239 (Figure 5.5). The *η-factor* is the number of neutrons produced per neutron captured by the fuel, defined by the expression:

$$\eta(E) = \nu(E) \cdot \frac{\sigma_f(E)}{\sigma_f(E) + \sigma_{n\gamma}(E)}$$

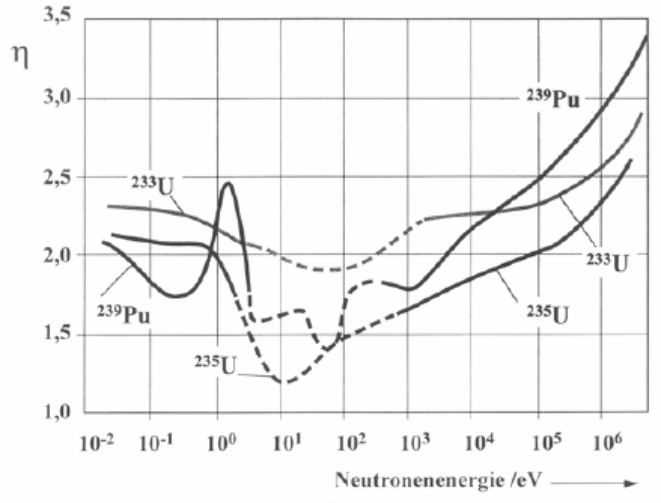

Figure 5.5: The η -Values of U-233, U-235 and Pu-239.

Table 5.1 shows the nuclear properties of the fissile isotopes U-233, U-235 and Pu-239 averaged over thermal energies. The large η-value allows the breeding of U-233 even at thermal energies. The effective fraction of delayed neutrons (β_{eff}) of U-233 is less than half of that of U-235, reducing the time constant of the feedback system for controlling a critical reactor.

Table 5.1: Comparison of the Nuclear Properties for the Fissile Isotopes U-233, U-235 and Pu-239.

isotopes	^{235}U		^{239}Pu		^{233}U	
spectrum	Thermal	Fast	Thermal	Fast	Thermal	Fast
σ_f (barn)	582	1.81	743	1.76	531	2.79
σ_c (barn)	101	0.52	270	0.46	46	0.33
$\alpha=\sigma_c/\sigma_f$	0.17	0.29	0.36	0.26	0.09	0.12
ν	2.42	2.43	2.87	2.94	2.49	2.53
$\eta=\nu\sigma_f/\sigma_a$	2.07	1.88	2.11	2.33	2.29	2.27
β_{eff} (%)	0.65		0.210		0.276	

Altogether, the favorable nuclear properties of U-233 are counter balanced by a loss of neutrons due to the residual Pa-233 and the lack of fast-neutron fission in Th-232 compared with U-238. As a consequence, it seems extremely difficult to design a thermal breeder reactor out of a Light Water Reactor fuelled with Th-232/U-233 under industrial conditions.

5.3 Past Experiences

In the 1960s and 70s, the development of thorium nuclear fuel was of great interest worldwide. It was shown that thorium could be used practically in any type of existing reactors. A large amount of work was carried out and resulted in many interesting developments, among these were prototype High Temperature Reactors (HTRs), Light Water Reactor (LWRs) and Molten Salt Reactors (MSRs).

In the mid 1970s the U.S. Electric Power Research Institute (EPRI) carried out a study of the prospects for improvements in the nuclear fuel cycle of modern LWRs by introducing thorium [42]. Several reactor types other than the LWRs have tried using thorium [43]. The most notable among these are the first gas cooled, graphite moderated reactor in the US (Peach Bottom, 40 MW_e, 1967 - 1969) and the first pebble bed reactor in Germany (AVR, 15 MW_e, 1966 - 1972) [44]. Table 5.2 gives an overview of the experimental reactors and the power reactors that have utilized thorium as nuclear fuel.

Table 5.2: Thorium Utilization in Different Experimental and Power Reactors (From IAEA-TECDOC 1450).

Name and Country	Type	Power	Fuel	Operation Period
AVR, Germany	HTGR Experimental (Pebble Bed Reactor)	15 MW_e	Th & U-235 Driver Fuel, Coated fuel particles, Oxide & dicarbides	1967 - 1988
THTR, Germany	HTGR Power (Pebble Type)	300 MW_e	Th & U-235 Driver Fuel, Coated fuel particles, Oxide & dicarbides	1985 - 1989
Lingen, Germany	BWR Irradiation-testing	60 MW_e	(Th, Pu)O_2 Test Fuel , Pellets	Terminated in 1973
Dragon, UK OECD-Euratom also Sweden, Norway & Switzerland	HTGR Experimental (Pin-in-Block Design)	20 MW_{th}	Th & U-235 Driver Fuel, Coated fuel particles, Dicarbides	1966 -1973
Peach Bottom, USA	HTGR Experimental (Prismatic Block)	40 MW_e	Th & U-235 Driver Fuel, Coated fuel particles, Oxide & dicarbides	1966 – 1972
Fort St Vrain, USA	HTGR Power (Prismatic Block)	330 MW_e	Th & U-235 Driver Fuel, Coated fuel particles, Dicarbides	1976 – 1989
MSRE ORNL, USA	MSBR	7.5 MW_{th}	U-233 Molten Fluorides	1964 – 1969
Borax IV & Elk River Reactors, USA	BWRs (Pin Assemblies)	2.4 MW_e 24 MW_e	Th & U-235 Driver Fuel, Oxide Pellets	1963 – 1968
Shippingport & Indian Point, USA	LWBR PWR (Pin Assemblies)	100 MW_e 285 MW_e	Th & U-233 Driver Fuel, Oxide Pellets	1977 – 1982 1962 – 1980
SUSPOP/KSTR KEMA, Netherlands	Aqueous Homogenous Suspension (Pin Assemblies)	1 MW_{th}	Th & HEU Oxide Pellets	1974 - 1977
NRU & NRX, Canada	MTR (Pin Assemblies)		Th & U-235 Test Fuel	Irradiation-testing of few fuel elements
KAMINI, CIRUS & DHRUVA, India	MTR Thermal	30 kW_{th} 40 MW_{th} 100 MW_{th}	Al & U-233 Drive Fuel, 'J' rod of Th & ThO_2 'J' rod of ThO_2	All three research reactors in operation
KAPS 1 & 2, KGS 1 & 2, RAPS 2, 3 & 4, India	PHWR (Pin Assemblies)	220 MW_e	ThO_2 Pellets For neutron flux flattening of initial core after start-up	Continuing in all new PHWRs
FBTR, India	LMFBR (Pin Assemblies)	40 MW_{th}	ThO_2 blanket	In operation

5.3.1 Light Water Reactor (LWR)

The Light Water Reactors (LWRs) utilize fuel based on uranium dioxide (UO_2), plutonium dioxide (PuO_2) and/or thorium dioxide (ThO_2) arranged in fuel rods. In the development of thorium-uranium oxide fuels in LWRs, two types of arrangements have been explored:

1. Homogeneous mixed oxides of thorium and highly enriched uranium in uniform lattices (the BORAX-IV, Indian Point I PWR, and Elk River BWR reactors).

2. Heterogeneous arrangements where the blanket contains less uranium and is responsible for most of the in-core fissile generation (Shippingport PWR).

In the Shippingport Light Water Breeder Reactor (LWBR) program, the feasibility of net breeding of fissile isotopes in the core was investigated. The results confirmed that the ratio of the fissile content of the fuel at the end of operation to that at the beginning of operation was about 1.0139. The effort also identified some shortcomings of the LWBR technology relative to the LWR experience at that time, including a lower power density of the core (30 %), a need for high U-235 enrichments in the early stage of deployment, a more complicated design of a movable seed region as well as a more complicated recycling of uranium and thorium including extra shielding needed in the fabrication process compared to uranium and Pu recycling [45].

The main conclusions of the EPRI sponsored study on thorium cycle applications in the *Combustion Engineering System 80 PWRs* can be summarized as follows:

- Use of thorium with recycling[3] can in the long run increase energy output per mined tonne of uranium by about 85 % beyond the once-through[4] uranium cycle, and by 22 % beyond plutonium recycle.
- Even with the above, the thorium cycle may not be economically attractive relative to cycles with poor fuel conservation features because the early years' fuel demand is high, and savings occur in later years.
- Comparison of the characteristics of uranium and thorium based cores indicates that thorium fuelling is feasible, and modifications to a PWR designed to accommodate plutonium recycle do not appear to be required.
- The introduction of a totally new system of advanced converters into the US would probably require more effort and funding than can be justified.

In past attempts to improve LWR fuel cycle efficiency, it has been recognized [46], [47] that the advanced converter concepts attain superior fuel utilization by elimination of major parasitic reactivity control through one of three methods: adoption of on-line re-fuelling (CANDU, Molten Salt and Pebble Bed), special geometry to capture leakage neutrons (seed-blanket), or moderation control of neutron capture in fertile material (spectrum shift). At this stage only the CANDU reactor has been successfully deployed worldwide. All these cores can have improved fuel efficiencies when recycling the fissile material as well as with introduction of thorium in addition to uranium in the fuel.

5.3.2 High Temperature Gas Cooled Reactor (HTGR)

KFA Jülich, Germany, has four decades of experience with thorium and its use as a nuclear fuel. Thorium fuel with coated fuel particles (about 100 000) has been successfully tested in two High

[3] *Closed Fuel Cycle (the fuel is recycled): Reprocessing of spent fuel to extract the remining fissile material for further use.*

[4] *Open Fuel Cycle (once-through cycle): The fuel is wasted after use (some fissile material left).*

Temperature Reactors (HTR). The *Atom Versuchs Reaktor* (AVR, 50 MW_{th}) was in operation for more than 21 years. In the *Thorium High Temperature Reactor* (THTR, 750 MW_{th}), 675 000 graphite spheres containing coated fuel particles (~10 g ThO_2 / sphere) were used for three years after extensive approval procedures for the thorium fuel. Much of the work was dedicated to the question of high conversion and breeding with thorium cycles.

The THOREX (thorium extraction) process for the reprocessing of spent thorium fuel based on the PUREX (plutonium uranium extraction) process was also developed in Jülich. This development ended in the 1970s because breeders seemed superfluous and the US had virtually prohibited all use of highly enriched uranium. Ever since, all developments for modular High Temperature Reactors (HTRs) assume a low enrichment of the fuel (LEU; about 8wt% U-235).

Jülich has persistently studied the possibilities of the thorium cycle for e.g. the use in PWRs and in transmutation facilities. Jülich's know-how covers the production of thorium fuel, the operation of critical reactors and the reprocessing of thorium. The following is based on the report summarizing the German experience [48].

5.3.2.1 AVR (Atom Versuchs Reaktor)

The *Atom Versuchs Reaktor* (AVR) was the world's first high temperature gas cooled "pebble bed reactor", where the spherical fuel elements ("pebbles") were used for more than 21 years of operation. Thorium containing coated particle fuel was tested extensively with very good success. Different forms of spherical fuel elements and different types of coated particles were tested. Thorium containing fuel elements fulfilled all requirements.

The reactor used continuous loading and unloading of fuel elements, and therefore excess reactivity for the compensation of fuel burn-up was not necessary. This is very important for considerations on very severe reactivity accidents. The fuel elements were recycled several times before they reached their final burnup.

The AVR experiments demonstrated that the pebble bed reactor concept works and that such a reactor can be operated relatively easily. Many useful and new safety experiments have been carried out, especially the experiments to demonstrate the concept of self-acting decay heat removal, important for all new modular High Temperature Reactors (HTRs) worldwide. About 200 kg of thorium were inserted in the AVR, and the operation showed that there can be attractive features in the use of thorium in nuclear technology in the future [49], [50], [51], [52], [53].

5.3.2.2 THTR (Thorium High Temperature Reactor)

The High Temperature Reactor (HTR) can be operated with different fuel cycles. The fuel consists of the nuclides U-233, U-235, Pu-239, Pu-241 as fissile materials and Th-232, U-234, U-238 as fertile materials. In the pebble bed reactor, these materials can be used in mixed or separated form. Dependent on today's economic conditions and regarding the situation and perspectives of fuel supply for the next decades, the *open cycles* (without reprocessing) will have advantages for a time schedule of some decades. With rising costs of uranium the *closed cycles* (with reprocessing) will become interesting in the far future.

Cycles with different enrichments and different breeding materials have been developed and proved in mass tests in the AVR and the THTR. The following cycles are fully developed:

1. *Cycles with 93wt% enriched uranium and thorium (high enriched uranium (HEU) cycles).*

2. *Cycles with 20wt% enriched uranium and thorium (medium enriched uranium (MEU) cycles).*
3. *Cycles with 8wt% enriched uranium and U-238 as fertile material (low enriched uranium (LEU) cycles).*

Because of the International Nuclear Fuel Cycle Evaluation (INFCE) regulations, cycles with more than 20wt% enrichment are not allowed today because of non-proliferation requirements. Today, for all new HTR projects, LEU cycles are foreseen with around 8wt% enrichment and a burnup of 80 000 to 100 000 MWd/tHM.

Based on an HEU/Th concept similar to the THTR fuel elements (moderation ratio: N_C/N_{HM} = 325, heavy metal loading: 11.2 g/ball, burnup: 100 000 MWd/tHM, enrichment: ~10wt%) one gets characteristic data of the equilibrium core as given in Table 5.3 below.

Table 5.3: Characteristic Data of the Equilibrium Core.

Parameter	Value	Dimension
Average Enrichment (at start)	7.16	wt%
Burnup	100 000	MWd/tHM
Conversion Rate	0.59	-
Fissile Material Inventory	917	kg/GW_e
U_3O_8-demand (0.25 tail)	462	kg/Gwd_e
Loading U-235	1.81	kg/Gwd_e
Discharging U-233, U-235 and Pa-233	0.51	kg/Gwd_e
Discharging Np-239, Pu-239 and Pu-241	0.002	kg/Gwd_e
Pu(fiss)/Pu(tot) in Discharged Material	19	%
In-situ use of Pu-239 and Pu-241	98	%
Fast Fluence	5.1	10^{21} n/cm^2

These data show that production of plutonium is almost entirely avoided and that the power of the balls and the fast neutron dose stay within allowed limits.

The MEU cycle with medium enrichment (< 20wt%) would still be proliferation resistant (see Figure 5.6 for the balance of fissile material during the passage through the reactor). The pebble bed HTR allows major cycle flexibility. During normal operation of the AVR, fuel elements corresponding to 10 different options have been tested in a continuous change without any major changes of plant parameters [48].

The Thorium High Temperature Reactor (THTR) used spherical fuel elements loaded with U-235 (93wt% enriched) as fissile material and Th-232 as fertile material, both in the form of oxides.

The heavy metal was used in form of coated BISO particles (particles with two layers of pyrolytic carbon).

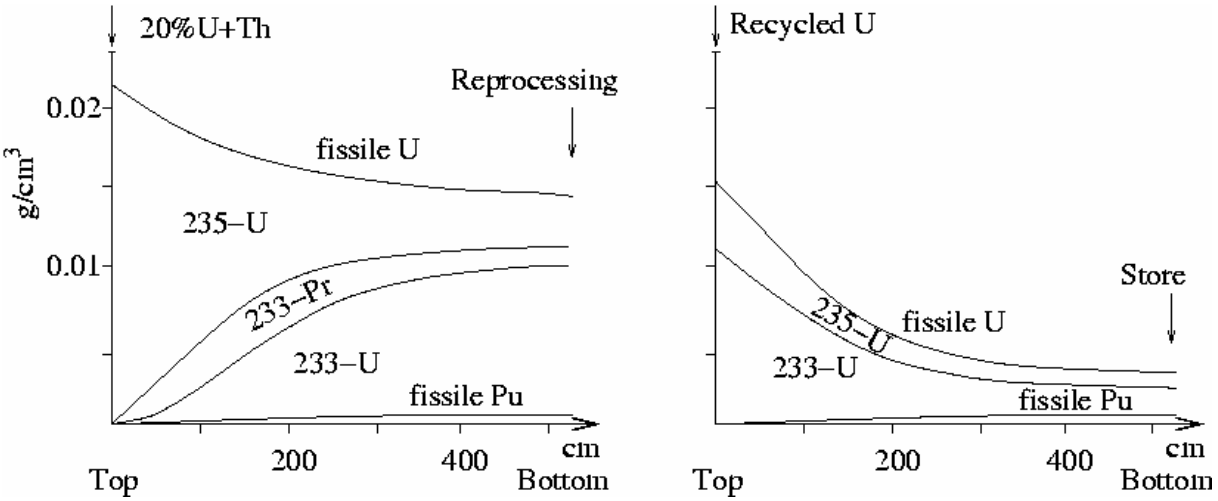

Figure 5.6: Balance of Fissile Material during the Passage through the Reactor in the Thorium/Denatured Uranium Cycle (MEU Cycle).

(Source: "Use of Thorium in the nuclear technology – experiences in Germany" [48])

It is of great importance in this context that the isotope Pa-233 not only may decay to the fissile material U-233, but may also be converted to Pa-234 by a (n, γ)-reaction with a high reaction cross section (110 barn). This leads to a significant loss with regard to the breeding of new fissile material. The neutron flux level and the fuel element design are additional parameters of great importance with regard to the utilisation of thorium in a HTR. The neutron flux should be as small as possible to realise high breeding gains.

During the open uranium cycle of the HTR, about 90 % of the plutonium bred during operation of the reactor is burned directly in–situ in the reactor. Therefore the plutonium content of the spent fuel elements is very low and additionally is denatured by higher isotopes.

Closed cycles, which can be operated with enriched uranium (20wt%) and thorium and which show a very good proliferation resistance, may show high conversion rates and therefore a reduced uranium demand. For the realisation of closed fuel cycles for HTRs as well as for other reactor concepts, economic reprocessing capacities have to be available [54], [55], [56], [57], [58] and [59].

5.3.2.3 HTR as Converter and Breeder Systems

Thorium allows in principle the realisation of breeding in helium cooled High Temperature Reactors (HTRs), in heavy water (D_2O) cooled systems and in Molten Salt Reactors (MSRs). These three options have been studied extensively in Germany in the past but only the HTR fuelled with thorium and U-233 is commented here. For high conversion or breeding in a HTR, the thorium cycle is favoured. However, a HTR Th-U breeder requires a low moderation ratio and a low fuel burnup. In the report summarizing the German experiences [48] a symbiotic HTR system consisting of two pebble bed HTRs are discussed, where one acts as a "pre-breeder", producing U-233 for the second HTR. Both reactors need reprocessing, and a breeding ratio of 1.026 (produces 2.6 % more fissile material than it consumes) was estimated. However, this design was never realized as a prototype.

5.3.2.4 Gas Turbine-Modular Helium Reactor (GT-MHR)

The Gas Turbine-Modular Helium Reactor (GT-MHR) was developed in the United States by General Atomics using a prismatic fuel. The use of helium as a coolant at high temperature and the relatively small power output per module (600 MW_{th}) permit direct coupling of the reactor to a gas turbine (a Brayton cycle). This feature results in power generation at 48 % thermal efficiency (50 % higher than in a conventional nuclear reactor in use today). The GT-MHR core can accommodate a wide range of fuels, including highly enriched uranium/thorium, U-233/Th, and Pu/Th. The use of highly enriched uranium/thorium fuel was demonstrated in General Atomics' Fort St. Vrain reactor in Colorado.

5.3.3 Molten Salt Reactor Experiment (MSRE)

This advanced breeder concept circulates the fuel in molten salt, without any external coolant in the core. The primary circuit runs through a heat exchanger which transfers the heat of the fission reaction to a secondary salt circuit for steam generation. It was studied in depth in the 1960s and is now being revived because of the availability of advanced technology for the materials and the components.

During the 1960s the USA developed the Molten Salt Breeder Reactor (MSBR) as the primary back-up option for the conventional fast breeder reactor and a small prototype was operated. This prototype, the Molten Salt Reactor Experiment (MSRE, 8 MW_{th}) was operated at ORNL between 1965 and 1969. Three fuel types were tested: uranium enriched to 30wt% with U-235, pure U-233 and Pu-239. The fuel salt consisted of 66 % LiF, 29 % BeF_2, 5 % ZrF_4 and 0.2%UF_4.In the USA, a project to produce a molten salt breeder was initiated in 1971. The electricity output was planned to be 1000 MW. The salt was composed of 71 % LiF, 16 % BeF_2, 12 % ThF_4 and 0.3 % UF_4. The reactor was graphite moderated and had two core zones: One fissile zone where most of the power was produced, and one fertile zone where the conversion from Th-232 to U-233 would take place. The breeding ratio of this reactor was estimated at 1.06, which means that it would produce 6 % more fissile material than it consumed. The salt reprocessing was quasi-continuous with a processing time of only ten days, which seems somewhat unrealistic. More recent calculations have concluded that the proposed reactor concept had slightly positive reactivity feedback coefficients, which is undesirable for safety reasons. The project was terminated in 1976.

5.3.4 Conclusion

All projects using thorium in their fuel cycles had been terminated by the 1980s (except in India). The reason for this seems to be threefold: (1) the thorium fuel cycle could not compete economically with the more well-known uranium cycle, (2) a lack of political support in many countries for the development of nuclear technology in the aftermath of the Chernobyl accident and (3) an increased concern worldwide regarding the proliferation risk associated with the reprocessing of spent fuel.

5.4 Future Nuclear Energy Systems

5.4.1 The Indian Advanced Heavy Water Reactor Design (AHWR)

India has made utilization of thorium for large-scale energy production a major goal in its nuclear power program, utilizing a three-stage concept:

1. Pressurized Heavy Water Reactors (PHWRs), elsewhere known as CANDUs fuelled by natural uranium and Light Water Reactors (LWRs) of the Boiling Water Reactor (BWR) and VVER types. This stage produces plutonium.
2. Fast Breeder Reactors (FBRs) use this plutonium-based fuel to breed U-233 from thorium. The blanket around the core will have uranium as well as thorium, so that further plutonium (ideally high-fissile Pu) is produced as well as U-233.
3. Advanced Heavy Water Reactors (AHWRs) burn the U-233 and plutonium with thorium, getting about 75 % of their power from thorium.

India has to fall back on its vast thorium resource (amounting to about 1/3 of the world's thorium reserves) for its long-term energy security. Accordingly, they have a road map for introducing ThO_2 in the blanket zone of Fast Breeder Reactors (based on (Pu-239,U)O_2 MOX fuel) at an appropriate growth level of the installed nuclear power capacity in the second stage of its nuclear energy program. In the third stage, another type of Fast Breeder Reactors based on (Th,U-233)O_2 MOX fuel will complete the program [60]. All the technologies involved at the *front end* as well as *back end* of (Th,U-233)O_2 MOX fuel at plant scale must be mastered. Their existing experience of the Th-fuel cycle at a pilot scale including experience of building the U-233 fuel based research reactor KAMINI will be valuable. The program to start construction of the thorium fuel based 300 MW_e Advanced Heavy Water Reactor (AHWR) within the next two years [61] is a move in that direction. In this way, adequate time will be available to overcome many technical problems involved in the large scale (Th,U-233)O_2 MOX fuel cycle, primarily arising from radiological hazards; presence of expected 1000 - 2000 ppm level of U-232 in the U-233 fuel and inertness of ThO_2.

The Advanced Heavy Water Reactor is a thorium fuel based vertical pressure tube type reactor (Figure 5.7) using heavy water (D_2O) as a moderator and boiling light water (H_2O) as a coolant in a mode of natural circulation at low pressure (~ 70 bar). This coolant replace the high pressure D_2O coolant (~ 100 bar) being circulated by a pump in existing horizontal pressure tube type PHWR. Designed for 100 years of plant life, the AHWR will utilize 65 % of the energy of the ThO_2 based fuel. AHWR is the first of its kind in the world not only because of its most attractive feature of heat removal from the reactor core by natural circulation under all conditions, but also due to the fact that it incorporates a host of other passive safety features. These are in line with the approach being developed worldwide for inherently safe reactor systems by incorporating features that do not call for any human intervention or any active control devices for reactor safety. The overall design parameters of the AHWR are shown in Table 5.4.

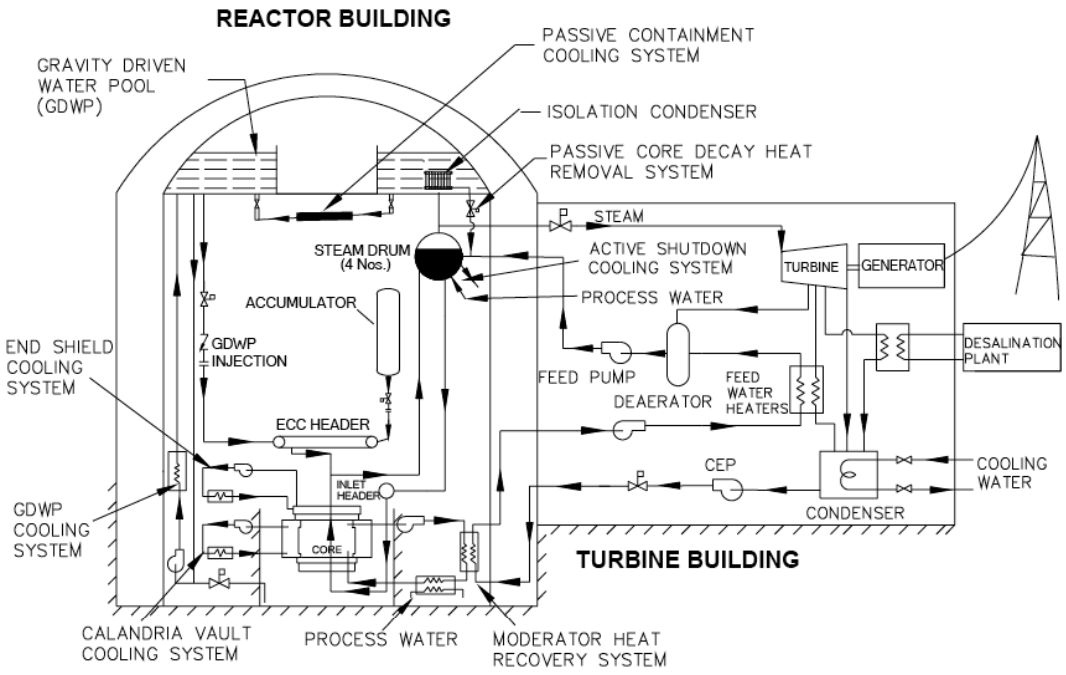

Figure 5.7: General Layout of the Advanced Heavy Water Reactor (AHWR).

Table 5.4: Design Parameters of the Advanced Heavy Water Reactor (AHWR).

Reactor power	920 MW_{th}, 300 MW_e
Core Configuration	Vertical, Pressure Tube type design
No. of pins in fuel cluster	54
	$(Th,Pu)O_2$ – 24 pins
	$(Th,U\text{-}233)O_2$ – 30 pins
Fuel burn up	24 000 MWd/tHM
Moderator	Heavy water
Reflector	Heavy water
Coolant	Boiling light water under natural
	Circulation
Steam pressure & temp.	70 bar & 285°C
MHT loop height	39 m
Number of coolant channels	452
Lattice pitch	245 mm (square pitch)
Primary shut down system	36 Shut off rods having B_4C
Secondary shut down system	Liquid poison injection in
	Moderator
No. of control rods	12 Nos. (4 each as regulation
	Rods, shim rods & absorber rods)

The AHWR fuel cycle will be implemented in stages – starting with the fuel cycle of "AHWR Critical Facility" which is being built at BARC followed by a closed fuel cycle of AHWR which will ultimately be self-sustaining with respect to U-233. The initial core will be made up entirely of (Th, Pu-239) MOX fuel assemblies, each being made of 54 fuel pins. Successively, the U-233 bred in the (Th, Pu-239) MOX fuel will be recovered and recycled as (Th, U-233) MOX. At equilibrium, the core of the AHWR will consist of composite fuel assemblies each having 24 (Th, Pu-239) MOX pins and 30 (Th, U-233) MOX pins arranged in three consecutive rings having fissile material compositions:

1st ring (inner ring): 12 pins consisting of (Th,U-233)O_2 with U-233 enrichment of 3.0wt%.

2nd ring (intermediate): 18 pins consisting of (Th,U-233)O_2 with U-233 enrichment of 3.75wt%.

3rd ring (outer ring): 24 pins consisting of (Th,Pu)O_2 with Pu enrichment of 3.25wt%.

The content of fissile plutonium isotopes will go down from an initial 75 % to 25 % at the equilibrium discharge burn-up level. However, to reduce the overall inventory of the waste, it is envisaged that thorium and U-233 will be recycled in the AHWR. Even though U-234 produced (along with U-235 and U-236) by neutron capture in U-233 has a negative influence on reactivity, it might be possible to recycle U-233 in AHWR with only a marginal penalty of less than 1000 MWd/tHM on discharge burn-up for each recycling.

The objective was to achieve a negative void coefficient of reactivity with light water as a coolant [62]. The core was optimized with respect to the fissile content, the maximum attainable reactivity, and low power peaking factors. Plutonium in the AHWR burns faster due to its large absorption cross-section that leads to a loss in reactivity. An option is available in the AHWR to reconstitute the fuel cluster after an averaged discharge burn-up of 24 000 MWd/tHM. Only plutonium pins in the outer rings are replaced by fresh fuel. The rest of the fuel cluster remains as it is. It is possible to obtain an additional burn-up of up to 20 000 MWd/tHM from the reconstituted cluster which improves U-233 production and reduces the reprocessing load. Reconstitution of the fuel cluster involves multiple enrichments for the (Th,Pu)O_2 pins, which will affect the fuel fabrication. However, the reconstitution improves the fuel conversion and hence the economy of the fuel cycle.

The plutonium composition has a strong bearing on the void coefficient. Although it is possible to achieve a negative void coefficient at the beginning of cycle, the void coefficient becomes positive as plutonium burns. With both plutonium and U-233 as fuel, the delayed neutron fraction (β_{eff}) is low. The β_{eff} for AHWR has been calculated as 2.8 mk[5]. Since a void collapse would introduce a positive reactivity, it will be one of the design basis accidents.

The fuel cycle time of the AHWR is 8 years of which 4 years are in-reactor residence, 2 years of cooling (to allow for >99.9 % conversion of Pa-233 to U-233), 1 year of reprocessing and 1 year for refabrication. For the initial few years, annual reload would consist of (Th,Pu)O_2 clusters only.

Part of the recovered thoria from the reprocessing plant will be recycled into the reactor immediately by using it for the fabrication of (Th,U-233) MOX pins. The rest will be stored for 17 - 20 years (~ 10 half-life of Th-228). Then it would be similar to fresh thorium and could be used to fabricate (Th,Pu)O_2 pins.

[5] *The reactivity unit mk is equal to 10^{-3}, where k refers to the value 1 of the multiplication factor in critical condition.*

5.4.2 Generation IV Reactors

Launched in 2000 by the US, the *Generation IV* initiative is a collaborative effort of the world's leading nuclear technology nations to develop the next generation nuclear energy systems to meet the world's future energy needs. The GIF members, gathered within the Generation IV International Forum (GIF) chartered in 2001, defined the eight following goals for these systems in four key areas; sustainability, economics, safety and reliability, proliferation resistance and physical protection:

1. ***Sustainability-1****: Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and provides long-term availability of systems and effective fuel utilization for worldwide energy production.*
2. ***Sustainability-2:*** *Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.*
3. ***Economics-1****: Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.*
4. ***Economics-2****: Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.*
5. ***Safety and Reliability-1****: Generation IV nuclear energy systems operations will excel in safety and reliability.*
6. ***Safety and Reliability-2****: Generation IV nuclear systems will have a very low likelihood and degree of reactor core damage.*
7. ***Safety and Reliability-3****: Generation IV nuclear energy systems will eliminate the need for offsite emergency response.*
8. ***Proliferation Resistance and Physical Protection****: Generation IV nuclear energy systems will increase the assurance that they are very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.*

In 2002, the GIF selected the six most promising systems that offer advantages in the four key areas and could be deployed commercially by 2030 [5].

Nowadays, the GIF has the following 13 members: Argentina, Brazil, Canada, China, Euratom, France, Japan, Republic of Korea, the Russian Federation, Republic of South Africa, Switzerland, the United Kingdom, and the United States. Currently eight Members are Parties to the Intergovernmental Framework Agreement for International Collaboration on Research and Development of *Generation IV* Nuclear Energy Systems (Canada, China, France, Japan, Republic of Korea, Switzerland, the United States and the European Atomic Energy Community (Euratom)) while the five others have not yet acceded to or ratified this Agreement which is the overarching legal basis for the GIF activities and includes provisions for the implementation of collaboration through a structure of System Arrangements (one per selected system) and Project Arrangements focusing on the key issues in each system.

The Table 5.5 gives the main characteristics of the six *Generation IV* systems in alphabetic order. It should be noted that, for sustainability reasons, most of the six systems use fast neutrons and a closed fuel cycle since it allows a better use of natural resources and minimizes the high-level waste to be sent to a repository. Moreover, the temperature ranges for the six systems are much higher than the operating temperature of the current light water reactors (below 330°C) which on

the one hand improves the thermodynamic efficiency and on the other hand in some cases allows the production of hydrogen with a thermo-chemical process to dissociate water.

Table 5.5: Overview of Generation IV Systems (Source: GIF 2007 annual report).

System	Neutron spectrum	Coolant	Temp. °C	Fuel cycle	Size (MW_e)
GFR (Gas-cooled Fast Reactor)	fast	helium	850	closed	1200
LFR (Lead-cooled Fast Reactor)	fast	lead	480 - 800	closed	20 - 180, 300 - 1200, 600 - 1000
MSR (Molten Salt Reactor)	epithermal	fluoride salts	700 - 800	closed	1000
SCWR (Super Critical Water-cooled Reactor)	thermal / fast	water	510 - 550	open / closed	300 - 700 1000 - 1500
SFR (Sodium-cooled Fast Reactor)	fast	sodium	550	closed	30 - 150, 300 - 1500, 1000 - 2000
VHTR (Very High Temperature gas Reactor)	thermal	helium	900 - 1000	open	250 - 300

GFR – The main characteristics of the *Gas-cooled Fast Reactor* are self-generating cores with fast neutron spectrum, robust refractory fuel, high operating temperature, high efficiency electricity production, energy conversion with a gas turbine, and full actinide recycling possibly associated with an integrated on-site fuel reprocessing facility. A technology demonstration reactor needed to qualify key technologies could be put into operation by 2020.

LFR – The *Lead-cooled Fast Reactor* system is characterized by a fast-neutron spectrum and a closed fuel cycle with full actinide recycling, possibly in central or regional fuel cycle facilities. The coolant could be either lead (most likely option), or lead/bismuth eutectic. The LFR can be operated as: a breeder; or a burner of actinides from spent fuel, using inert matrix fuel; or a burner/breeder using thorium matrices. Two size options are considered: a small transportable system of 50 to 150 MW_e with a very long core life; and a large system of 300 to 600 MW_e. In the long term, a very large system of 1200 MW_e could be envisaged. The LFR system could be deployable by 2025.

MSR – The *Molten Salt Reactor* systems present the very special feature of a liquid fuel. MSR concepts, which can be used as efficient burners of transuranic[6] elements (TRU) from spent LWR fuel, have also a breeding capability in any kind of neutron spectrum (from thermal to fast), when using the thorium or fast spectrum U-Pu fuel cycle. In both options, they have a very interesting potential for the minimization of radiotoxic nuclear waste.

SCWR – *Supercritical Water-Cooled Reactors* are a class of high temperature, high pressure water-cooled reactors operating with a direct cycle and above the thermodynamic critical point of water (374°C, 22.1 MPa). The higher thermodynamic efficiency and plant simplification opportunities afforded by a high-temperature, single-phase coolant translate into improved

[6] *Transuranic Elements (TRU) are elements with atomic numbers greater than uranium.*

economics. A wide variety of options are currently considered: both thermal-neutron and fast-neutron spectra are envisaged, and both pressure vessel and pressure tube are considered. The operation of a 30 to 150 MW_e prototype is targeted for 2022.

SFR – The *Sodium-cooled Fast Reactor* systems use liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. The reactor unit can be arranged in a pool layout or a compact loop layout. Plant size options under consideration range from small (50 to 300 MW_e) modular reactors to larger plants (up to 1500 MW_e). The two primary fuel recycle technology options are advanced aqueous and pyrometallurgical processing. A variety of fuel options are being considered for the SFR, with mixed oxide for advanced aqueous recycle and mixed metal alloy for pyrometallurgical processing. Owing to the significant past experience accumulated in several countries, the deployment of SFR systems is targeted for 2020.

VHTR – The *Very-High Temperature Reactor* is a next step in the evolutionary development of high-temperature reactors. The VHTR technology addresses advanced concepts for helium gas-cooled, graphite moderated, thermal neutron spectrum reactor with a core outlet temperature greater than 900°C, and a goal of 1000°C, specified to support production of hydrogen by thermo-chemical processes. The reference reactor thermal power is set at a level which allows completely passive decay heat removal, currently estimated to be about 600 MW_{th}. The VHTR is primarily dedicated to the cogeneration of electricity and hydrogen, as well as to other process heat applications. It can produce hydrogen from water by using thermo-chemical, electro-chemical or hybrid processes with reduced emission of CO_2 gases. At first, a once-through LEU (<20 wt% U-235) fuel cycle will be adopted, but a closed fuel cycle will be assessed, as well as potential symbiotic fuel cycles with other types of reactors (especially light-water reactors) for waste reduction.

The level of the GIF activities for the different systems is quite heterogeneous. For the time being, the SFR and the VHTR are the most active systems while the LFR and MSR are the less active. Moreover, at this stage, in the GIF framework, none of the *Generation IV* reactors is specifically designed or studied for the use of thorium; however the MSR explicitly considers the use of thorium.

5.4.3 Molten Salt Reactor (MSR)

While not strictly a fast neutron reactor, the uranium fuel in the Molten Salt Reactor (MSR) is dissolved in the fluoride salt coolant which circulates through graphite core channels to achieve some moderation and an epithermal neutron spectrum. Fission products are removed continuously and the actinides are fully recycled, while plutonium and other actinides can be added along with U-238. Coolant temperature is 700°C at very low pressure, with 800°C envisaged. A secondary coolant system is used for electricity generation, and thermochemical hydrogen production is also feasible.

Recent studies have focused on lithium and beryllium fluoride coolant with dissolved thorium and U-233 fuel. The attractive features of the MSR fuel cycle include: the high-level waste comprising fission products only, hence shorter-lived radioactivity; small inventory of weapons-fissile material (Pu-242 being the dominant Pu isotope); low fuel use (the French self-breeding variant claims 50kg of thorium and 50kg U-238 per billion kWh); and safety due to passive cooling up to any size.

The molten salt reactor was not completely mothballed after 1976. Some programs continued in Japan (from the 1980s), France (from the 1990s) and in Russia. In 2002 the Thorium Molten Salt

Reactor (TMSR) project was initiated in France (Laboratoire de Physique Subatomique et de Cosmologie (LPSC-lab) in Grenoble, Électricité de France (EdF) and Centre National de la Recherche Scientifique (CNRS)). The aim was to solve the problems that plagued the MSBR, for instance the bad reactivity coefficients, a positive void coefficient, unrealistic reprocessing, and the lifespan of the graphite moderator. One of the parameters studied was the radius of the salt channels in the graphite, and it was found that hardening the neutron spectrum by increasing the channel radius made the reactivity coefficients negative. On the other hand, hardening the spectrum requires a higher neutron flux, which in turn decreases the graphite lifespan. A spectrum sufficiently hard to give negative reactivity coefficients would lead to a lifespan of approximately 5 years for the graphite, which is problematic. On the positive side was the fact that the breeding ratio increased with the spectrum hardening.

As a result of the aforementioned studies, one decided to study the feasibility of a TMSR without graphite in the core. This reactor will have two zones: a homogeneous vessel with fuel salt where the fissions take place, and a surrounding blanket zone of fertile thorium. Calculations show that this concept has several good features: Very negative reactivity coefficients, high breeding ratio (~ 1.10), and of course no problem with the graphite lifespan. The price to be paid for this is that the initial fissile inventory of U-233 increases from 1 to 5 metric tonnes.

Many parameters of the non-moderated TMSR depend on the content of heavy nuclei in the fuel. For instance, the initial fissile inventory varies from 2.5 to 6.3 metric tonnes depending on the heavy nuclei proportion in the salt. The safety coefficients do not depend on the heavy nuclei content, but the breeding ratio does.

Chemical reprocessing of the fuel salt will consist of a batch process using fluorination to extract the uranium and neptunium. Also thorium, protactinium and the minor actinides will be extracted and returned to the fuel salt. What is left after the extraction process will be stored for three months to allow traces of Pa-233 to decay to U-233, and the uranium will then be extracted and returned to the fuel salt. Finally, the fission products that are dissolved in the salt will be extracted and the cleaned salt returned to the fuel salt. Some fission products (e.g. noble metals and gases) are not soluble in the salt and will be removed by helium bubbling and filtration in a separate online system.

The doubling time (the time required to produce enough U-233 to start another reactor) depends on the content of heavy nuclei in the fuel and the initial fuel composition, and may vary from 30 to 50 years.

Due to its hard spectrum, the TMSR is a good burner of plutonium and minor actinides. For example, LPSC-lab has calculated that a TMSR initially loaded with 11 metric tonnes of plutonium and minor actinides and 29 metric tonnes of thorium will burn 85 % of the plutonium and minor actinides in 45 years [63], [64], [65], [66], [67], [68], [69], [70], [71].

While Molten Salt Reactors offer many attractive features, there is much to be done before full-scale economic molten-salt breeders can be built. What is seen as the remaining primary research and development in different areas is summarized below [72], [73], [74], [75], [76], [77], [78], [79], [80], [81], [82], [83].

Physics: No crucial physics problems are seen in the development of molten-salt reactors, but in view of the relatively small breeding gain of an MSBR, particular care must be taken to ensure that the nuclear calculations are accurate.

- Need to reduce the uncertainty in the capture-to-fission cross section ratio of U-233.

- Additional measurements of the absorption cross sections of the salt constituent Li, Be, and F could reduce the uncertainty further.
- Verifying the temperature coefficients of reactivity, including Doppler coefficients.

Chemistry: An extensive program of molten-salt chemical research over the past 40 years has established a sound basis for understanding the chemistry of the pertinent fluoride salts. Additional work is desirable, however, in several areas:

- A better understanding is needed of the behaviour of the noble metal fission products in MSBR systems.
- Alternative processes for rare earth removal should be investigated, and chemical information on the protactinium removal process needs to be extended to a wider range of conditions.
- The phase behaviour of PuF_3 in molten salts ought to be studied more thoroughly to provide a firmer chemical basis for the use of plutonium.
- Further measurements are needed of the physical and chemical properties of molten salts, and particularly of the sodium fluoride/sodium fluoroborate salt mixtures which are the presently proposed secondary coolants for MSBRs.

Fuel Processing: Obtaining good breeding in a single-fluid reactor requires rapid on-site reprocessing to keep protactinium out of the high neutron flux as well as to remove fission products. The investigation of most of the steps needed to effect rapid reprocessing has not proceeded past preliminary laboratory tests.

Materials: Damage from high-energy neutrons eventually leads to deterioration of graphite at temperatures of interest for molten-salt reactors, and improved grades need to be developed and tested. The *Hastelloy-N* used in the past for containing molten salts is embrittled by helium produced in the metal when it is irradiated. Although the present material has been satisfactory for the Molten Salt Reactor Experiment (MSRE), an improved alloy is needed for power reactors. The resistance of the modified alloy to corrosion by fuel and coolants salts, particularly the fluoroborate, must be demonstrated at all reactor operating conditions.

Molten Salt Reactor Experiment (MSRE): MSRE operation has shown the compatibility of graphite with fissioning fuel at low power densities and low burnup, and a variety of in-pile capsules and loops have provided additional information. However, the experience to date has not included exposures at the power densities of breeder reactors. In-pile loop experiments thus need to be performed using appropriate materials at the peak power densities of high-performance breeders.

5.4.4 Accelerator Driven System (ADS)

An Accelerator Driven System (ADS) is a subcritical reactor, i.e. a reactor that cannot sustain a chain reaction on its own. The reactor can only run when neutrons are supplied from an external source. In the case of the ADS, the neutrons come from spallation of heavy nuclei like lead. In this context, spallation means that a high energy proton (500 - 1000 MeV) hits an atomic nucleus in a target (e.g. lead) and rips off a large number of neutrons (15 - 30), depending on the target material and the proton energy. The collision between the proton and the target nucleus may also result in fissioning of the nucleus, or the production of nuclei of light elements. Further, the neutrons that are produced will induce fission and nuclear transmutation reactions in the fuel. The high energy protons are supplied by a powerful particle accelerator which may be either a linear accelerator (LINAC) or a cyclotron.

5.4.4.1 History

ADS was first proposed by Nobel Prize laureate E.O. Lawrence in the 1950s, mainly as a method to produce fissile materials by neutron transmutation. Later, the concept was extended to "burn" nuclear waste. The idea behind "burning" of waste is to use the spallation neutrons to transform the waste into other isotopes with a much shorter half life, or, in the case of transuranics to cause fission. For example, the long-lived fission product technetium-99 (Tc-99) (half life = 211 000 years) changes to Tc-100 (half life = 16 seconds) when a neutron is absorbed in the nucleus. Thus, after only a few minutes the radioactive Tc-99 is transformed into the stable isotope ruthenium-100 (Ru-100). A full scale waste transmutation facility was never built, mainly due to the limitations in the accelerator technology.

By 1990, the accelerator technology had advanced significantly from the 1950s, and significant efforts leading to a subcritical molten salt system driven by a linear accelerator were extended in Los Alamos [84]. In 1993, another Nobel Prize laureate, Carlo Rubbia, revived the idea by proposing an ADS that could produce energy at the same time as it destroys both its own waste and waste from other reactors [85], [86], [87], [88], [89], [90], [91] (see Figure 5.8). Rubbia's concept was called the Energy Amplifier (EA). Several small experimental pilot studies have been done, but a full scale prototype of the Energy Amplifier has so far not been realized [92], [93], [94], [95] and [96]. The most active work on ADS in Europe is the MYRRHA project at Mol in Belgium, which started in 1997 and is planned to be in operation around year 2016 [97], [98], [99], [100], [101], [102] ,[103], [104] and [105]. MYRRHA is planned to have a subcritical core (k_{eff} ≈ 0.95)[7] with MOX fuel (35 wt% plutonium) and will be cooled by lead-bismuth eutectic. The accelerator will be a 1.5 MW LINAC that delivers protons with an energy of 600 MeV. The thermal power of the reactor is estimated at 60 MW.

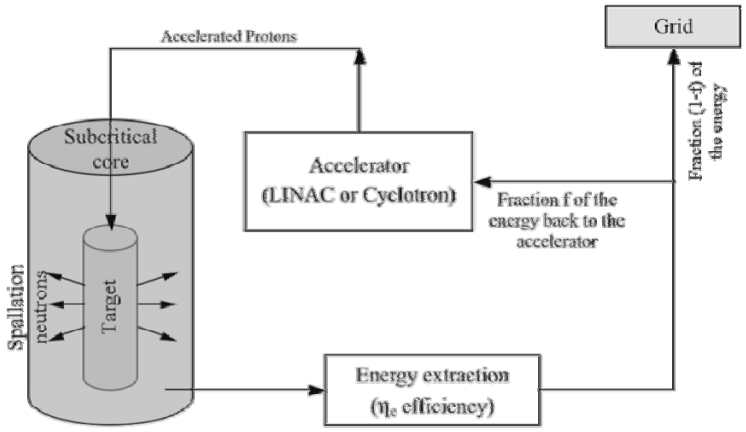

Figure 5.8: Scheme of an Accelerator Driven System (ADS).

5.4.4.2 Description of the ADS

The ADS consists of two main components:

1. The reactor core, whose fuel consists mainly of thorium. At the start-up of the ADS there will also be some U-235 or Pu or even transuranic waste in the fuel to increase the fission rate. The core is located close to the bottom of a double walled tank that is 25 meters tall and has a diameter of 6 meters (see Figure 5.9). The inner tank is filled with molten lead with a

[7] *The effective neutron multiplication factor (k_{eff}).*

temperature of 600 - 700ºC, or lead-bismuth eutectic with a temperature of 400 – 500ºC. At the upper part of the tank there are heat exchangers that transfer the heat from the molten metal to a secondary circuit (molten metal or water), eventually forming steam that drives the turbine. The lead-bismuth or lead is circulated by natural convection, that is, the heat supplied by the fissions in the core heats the molten metal such that it rises to the top of the tank where it is cooled by passing through the heat exchangers and returns towards the bottom. The outer tank is cooled by passive convection of air, which will remove the decay heat in case the cooling is interrupted or the inner tank ruptures.

2. The proton accelerator, which is located outside the containment building, supplies a high-powered (≈ 10 MW) beam of high-energy protons (500 - 1000 MeV) through a shielded beam guide to the spallation source inside the reactor core. The accelerator may be either a LINAC or a cyclotron.

The proton beam hits the target that is located close to the centre of the reactor core, and causes spallations that produce about 30 neutrons per incident proton. The neutrons enter the fuel in the core and cause two important processes:

1. Transmutation of the thorium into protactinium, which decays with a half-life of 27 days to U-233, which is fissile. Thus, fuel is produced from the non-fissile material thorium.
2. Fission in uranium/plutonium or even transuranic waste, present in the fuel, and in the U-233 that has been produced from thorium.

The reactor core produces 1500 MW of heat which is transformed into 675 MW of electricity. Of this electricity, 30 MW is used to drive the accelerator, and the remaining 645 MW is delivered to the electricity grid (See Figure 5.10). For a reactor with k = 0.98, the power of the proton beam must be around 10 MW to produce 1500 MW of heat. The power from the beam is thus amplified by a factor of 150, which is the reason for the name “Energy Amplifier”. The more subcritical the core is, the higher the power of the beam must be. For instance, if a core with k = 0.95 is used, the beam power must be around 25 MW. A beam power of 10 MW is more than the largest accelerators are capable of today, but it is considered to be realistic to develop accelerators of this size. The world's most powerful cyclotron is located at the *Paul Scherrer Institute (PSI)* in Switzerland, and was as an example operating in 1999 for 6000 hours with a beam power of 1 MW and an availability of 91 % of the scheduled time in 1999 [106].

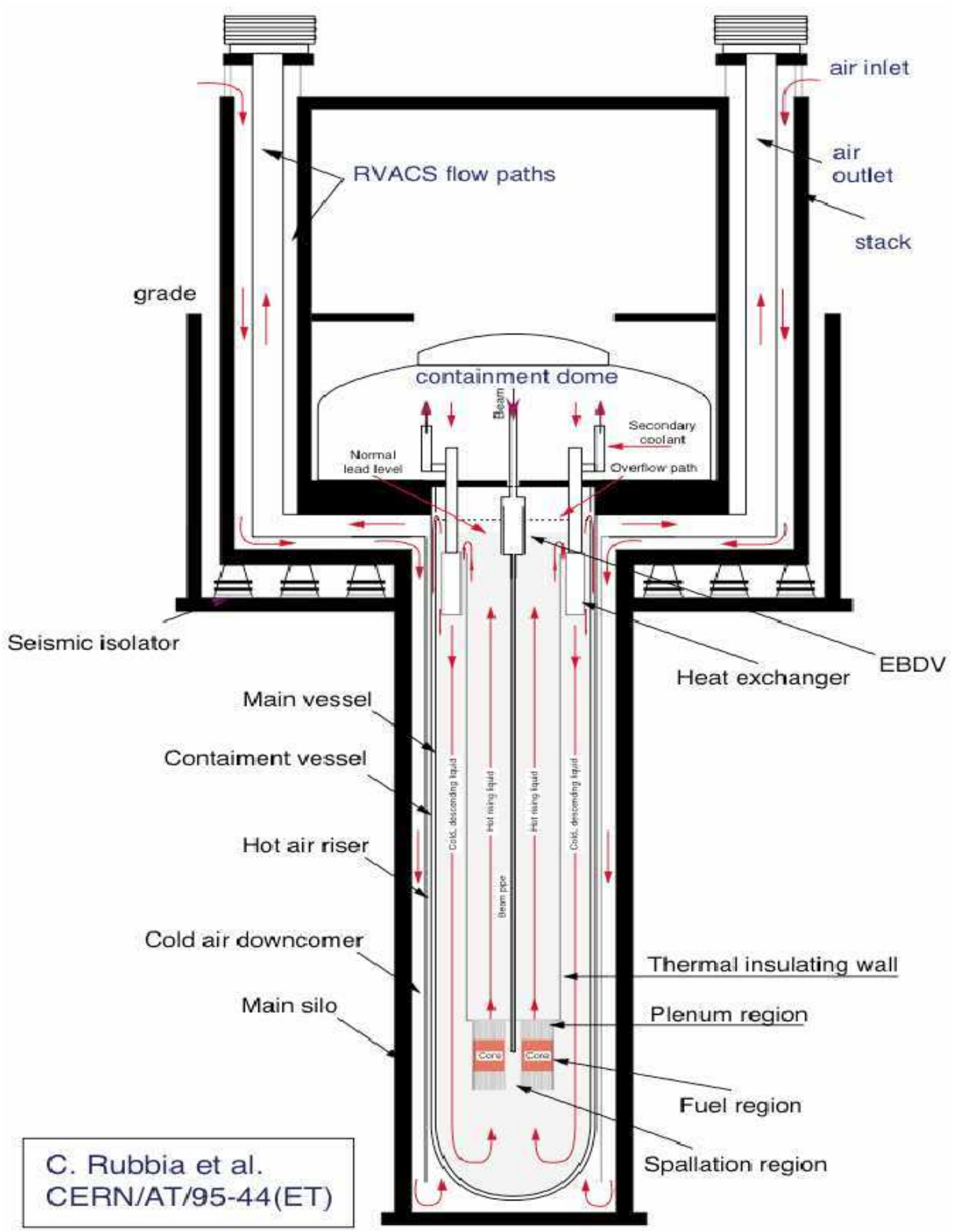

Figure 5.9: Carlo Rubbia's Energy Amplifier.

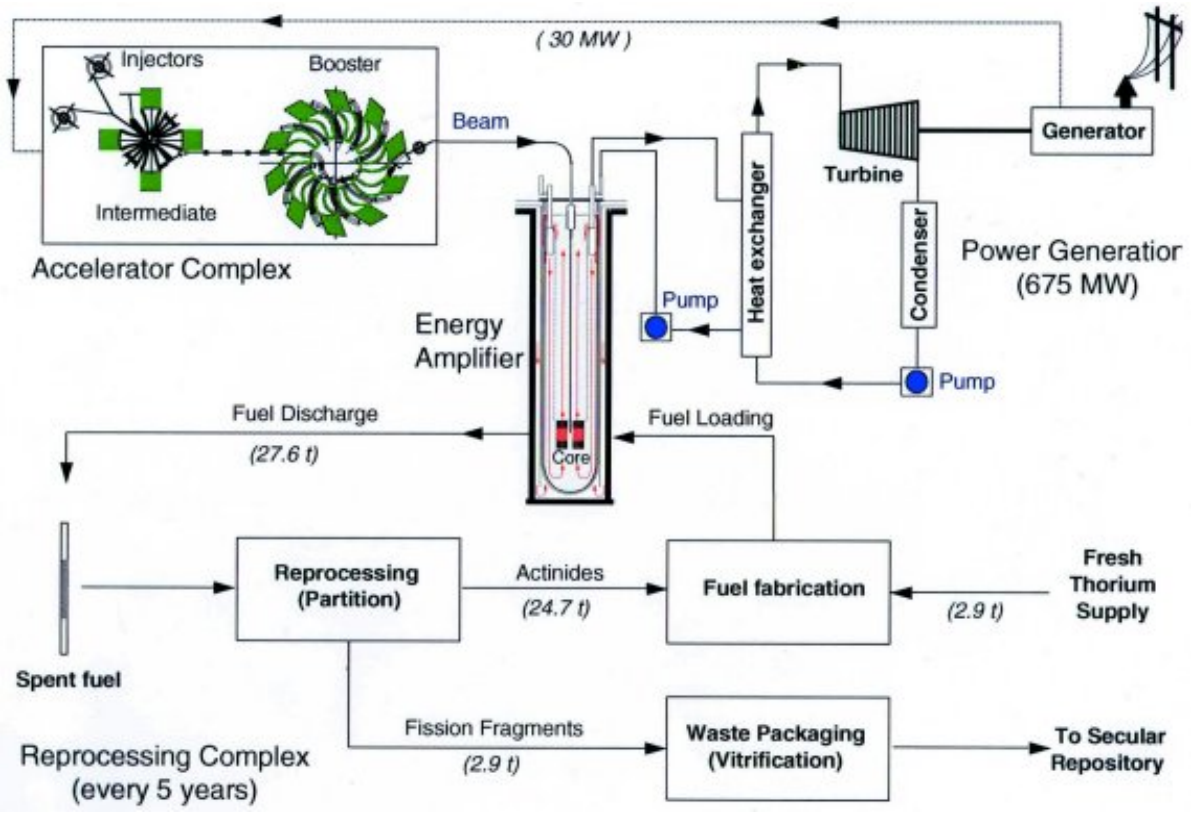

Figure 5.10: Schematic View of the Energy Amplifier System for Electricity Production.

The reason for using a lead-bismuth eutectic as coolant instead of pure lead is to lower the melting point from that of lead (327°C) to 123°C, which reduces the probability of frozen slugs clogging the flow channels. It also reduces the heating requirements during reactor shutdowns. To reduce the induced radioactivity in the coolant, it would be desirable to use pure lead instead of the eutectic, but this is still under development.

In addition to its subcriticality, the EA is distinguished from most other reactor designs since it does not use control rods for the power control. The power of the reactor is proportional to the supplied beam power, which means that power control can be performed by adjusting the accelerator power. The same method is also used to compensate for reactivity changes caused by fuel burnup.

When compared with critical reactors, Accelerator Driven Systems have two specific characteristics:

1. If well designed, they prevent criticality accidents. It has been proposed [107] to take advantage of this sub-criticality in order to use certain types of fuel with poor neutronic properties: in particular, because of their small delayed neutron fractions, incineration of minor actinides appears to be feasible efficiently with ADS.
2. Spallation provides additional neutrons which can be used for increased breeding of U-233 or Pu-239. Another possible use of the additional neutrons is to transmute long-lived fission products and transuranics (TRUs).

In Norwegian media it has repeatedly been claimed that an ADS based on thorium cannot melt down like an ordinary reactor if the cooling is lost, because it is not critical. Unfortunately, this is not the case. A reactor without cooling will melt no matter how subcritical it is. When the reactor is shut down, the heat does not come from ongoing fissions in the fuel, it is the decay of the fission

products that produces the heat required to melt the reactor. The Energy Amplifier as shown in figure 5.9 will probably not melt if the heat removal from the lead-bismuth coolant stops, because it contains around 8000 metric tonnes of passively cooled metal that will absorb the heat. But, if the coolant leaks from the tank the core will certainly melt.

5.4.4.3 Sub-criticality and the Accelerator

In an ADS, the sustainability of the nuclear fission reactions is made possible because of the presence of an external source of neutrons provided by the proton beam; the working point of the core itself is far below criticality. Two alternatives, based on well established technologies, can be envisaged in order to fulfil the indicated requirements for the accelerator:

1. Super-conducting cavity LINACs, for which a vast experience exists in Europe, including on the industrial production of the cavities and of the associated RF sources, especially as a consequence of the realisation, now dismantled, of LEP200[8] at CERN and of the work at DESY (Germany) and elsewhere (Italy, France) in association with the TESLA[9] program. The most powerful LINAC is now being commissioned at the Spallation Neutron Source facility at ORNL.
2. Isochronous cyclotrons, for which the world reference with respect to the ADS question is PSI (Switzerland). There is also a strong experience accumulated at GANIL (France), CAL (France) and LNS (Italy) facilities. The guiding magnetic field can be produced either by ordinary magnets (PSI, GANIL, CAL) or super-conducting coils (LNS).

The system needed to drive an ADS represents only a reasonable extrapolation of what has already been achieved in current accelerator technology [108], [109], [110], [111], [112] and [113]. An important element for the accelerator is a high reliability in continuous operation and in particular the absence of "beam glitches", namely of short interruptions of the beam current. A beam interruption will induce very strong stresses in the window between the accelerator vacuum and the target and produce a fatigue of the reactor elements, reducing the lifetime of the system. So this topic is of first importance. The LEP200 super-conducting LINAC has demonstrated a remarkable level of continuity of operation, since a single short glitch would imply the immediate loss of the colliding beams. At PSI, however, some beam instabilities due to discharges in the RF-feed through to the cavities have been observed. It is believed that this problem can be overcome with a more advanced design of such components[10], which are known to operate without such inconveniences in other installations (CERN).

It is expected that both accelerator schemes (LINAC and cyclotron) will offer highly reliable and redundant methods to control the beam current and to switch it off extremely quickly (a few microseconds) and reliably if and when required by the sub-critical unit. However, it seems that the LINAC option is favoured by the international accelerator community [114].

5.4.4.4 Technology Issues of Medium-to-High Power Spallation Targets for ADS

The spallation target has to provide the highest possible neutron yield, be transparent to neutrons, and at the same time sustain a large beam power of 10 to 20 MW. In this respect, molten lead is almost an ideal candidate since it has also excellent thermodynamic properties and can contribute to cooling. The use of liquid targets is a tendency which is presently developing in

[8] *The LEP200 programme ultimately consists of about 3.6 GeV gradient for the continuous acceleration of relativistic electrons and positrons with a current of up to 7 mA.*

[9] *The TESLA programme aims at the realisation of advanced super conducting cavities, of high gradient and lower cost in order to open the way to the realisation of a Linear Collider in the 1/2 TeV range and eventually beyond.*

[10] New cavities are under development also at PSI.

the design of spallation neutron facilities. For instance, the *European Spallation Source (ESS)* [115] and the *Spallation Neutron Source (SNS)* [116] projects are developing liquid mercury targets, and the *Swiss spallation neutron source, SINQ* [117] is planning an upgrade to a liquid lead-Bismuth target. Tungsten, although acceptable from the point of view of spallation, is not favourable to neutron transport (neutron absorption and activation) and would clearly have to be used in solid form since its melting temperature is very high (3422˚C) with the additional difficulty that it can break (very brittle above 600 to 700˚C) or even explode if the proton source is pulsed.

From the neutronics point of view, both lead and eutectic lead-bismuth (Pb-Bi) mixtures are satisfactory. Lead-bismuth has the advantage of allowing operation at a lower temperature and might be chosen in a first stage for the design of an ADS demonstrator. The maximum temperature of the window (in a 6 mA, 600MeV beam Pb-Bi system) is about 500˚C, which can be handled with presently available materials such as ferritic, 9 % chromium steel. Going to pure lead would increase that maximum window temperature by about 200˚C, which requires developing new materials through technological R&D.

Because lead-bismuth targets produce significantly more radiotoxic elements (polonium-210) than pure lead, the long-term preferred solution is pure lead. We refer the reader to a discussion of these effects in the first item in [118] (pages 77 to 82). One assumes that through proper R&D, materials will be developed which can accommodate the high lead temperature including corrosion effects.

The target is presently an area where intense R&D is being carried out in Europe, within the 4th - 6th Framework Program of the European Commission (TESTRA, PDS-XADS and EUROTRANS projects). The Benchmark Working Group, a collaboration between 16 institutes, is particularly active in this domain [119]. All of this implies a careful design of the interface (window) between the accelerator and the effective target. The very low vapour pressure of lead makes it possible for liquid lead to be compatible with direct exposure to the accelerator beam pipe vacuum, which opens the possibility of a windowless solution for that interface [120] (Figure 5.11).

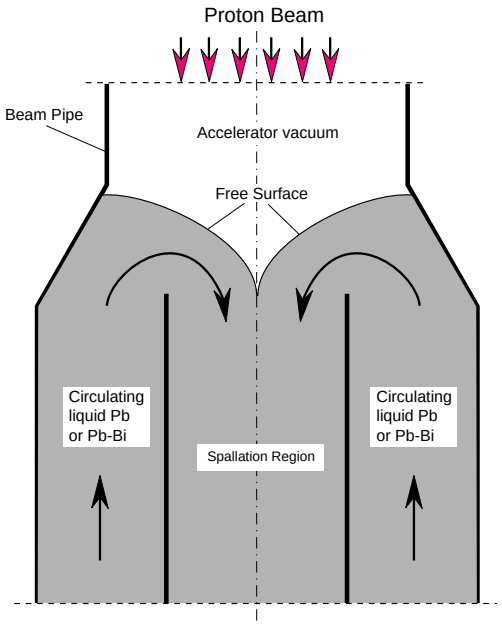

Figure 5.11: Sketch of a Windowless Interface between the Accelerator and the Spallation Region. (Because of the low vapour pressure of lead, the molten lead can be in direct contact with the beam vacuum. A cold trap type of device (not shown) can capture residual vapours.)

Following a first phase of R&D focused on the understanding of the basic principles of ADS, the programs have been streamlined and focused on practical demonstration of key issues. One particular demonstration covers spallation targets of high power (~ 1 MW class) and their effective coupling with a sub-critical core.

The MEGAPIE (MEGAwatt PIlot Experiment) project represents a key experiment in the ADS roadmap. The goal of this project was to design, build, operate and explore a 920 kg liquid bismuth eutectic (LBE) spallation target for 1MW of beam power, taking advantage of the existing spallation neutron facility SINQ at PSI, Switzerland (590 MeV) [121] see Figure 5.12. The MEGAPIE target was operated for 4 months between August and December, 2006. The results obtained show that there was an 80 % higher neutron flux as compared with a solid metal target, greatly exceeding the predictions. After a cooling off period, the examination of the now frozen target will continue for the next couple of years and will deliver invaluable information about the composition and behaviour of materials used in the experiment.

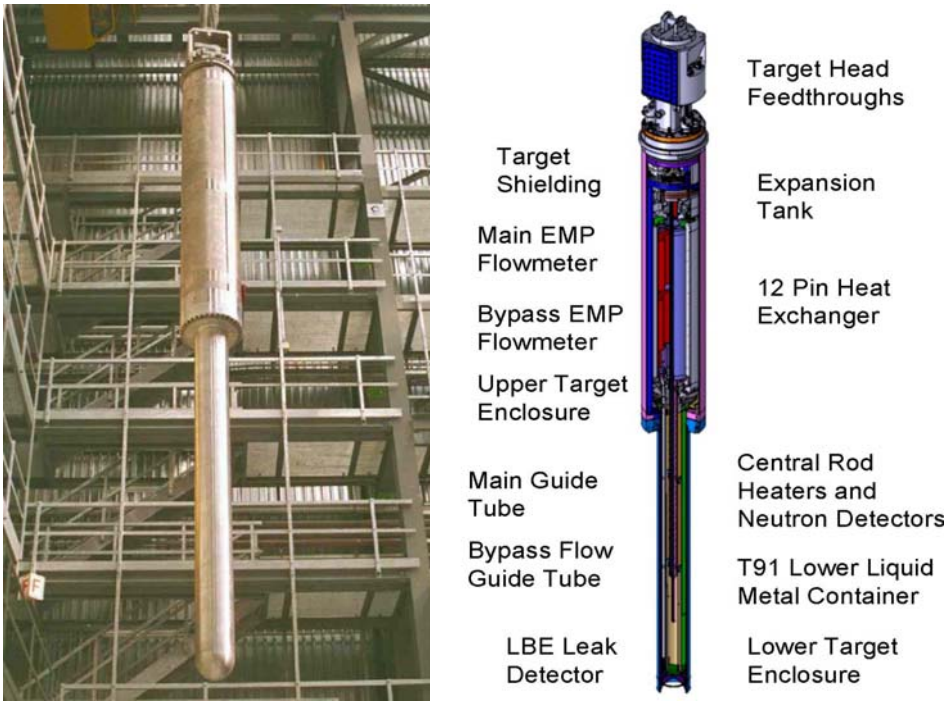

Figure 5.12: The Conceptual Design (Right) and the Five Metre Long Liquid Metal MEGAPIE Target. (The liquid metal Megapie target is installed in the hall at the spallation neutron source (SINQ). The target head: at the beginning of the experiment the target is filled with liquid metal through the large insulated tube in the centre of the picture.)

5.4.4.5 Safety Aspects of Accelerator Driven Systems

When a new nuclear system, such as the Accelerator Driven System, is proposed, the early investigation of potential severe accidents is important in order to point out the areas in which the design could be improved or whether passive devices could be introduced which would stop conceivable accident sequences.

For a heavy liquid metal cooled ADS, it was early recognized that a *complete Loss-Of-Flow (LOF)* would lead to problems if the accelerator was not switched off. Therefore, a first conceptual design [118] was based on natural circulation cooling and therefore required a rather tall vessel. A subsequent and more detailed design with a shorter vessel introduced the enhanced natural circulation cooling by gas bubble injection above the core [122]. This design still allows the

removal of the nominal power by pure natural circulation for a certain time. However, this would still require a complete heat removal by the secondary loops.

In a *Loss-Of-Heat-Sink (LOHS) Accident*, the core would slowly heat up (by about 200°C in 1000 seconds). Switching off the accelerator would get the power rapidly down to decay heat level. In case this shutting off is not done, one can rely on a simple passive means of interrupting the proton beam by using a melt-rupture disk at the side of the vacuum pipe through which the protons are streaming into the subcritical core [118], [123]. When this melt rupture disk is pushed inside the pipe by the coolant pressure, the pipe will be unconditionally filled up by the liquid metal coolant that will block off the proton beam.

An important initiator for *Loss-Of-Flow* and *Loss-Of-Heat-Sink* accidents is a *station blackout*. However, in an ADS this would also lead to the shutting off of the accelerator and the decrease of the power to decay heat level. The remaining problem would be the passive decay heat removal that can be achieved by natural air circulation cooling.

Loss-Of-Coolant (LOCA) Accidents, which are a concern in LWR safety, should pose no significant problem. First, a lead-bismuth (Pb-Bi) cooled system is at a low pressure; second, a pool design has been proposed in which all the primary coolant is in one vessel and there is no primary piping which might develop leaks; and third, a guard vessel surrounds the main vessel in case the latter would leak. Another important safety aspect of a lead-bismuth or lead coolant is that it is chemically rather inert and does not react strongly with air or water. This is an advantage relative to sodium-cooled fast systems.

Beam Power Accidents. Since the proton beam intensity can be increased to compensate for burn-up, one can also imagine that the beam power is increased accidentally. Figure 5.13 shows the power increase with an assumed beam power increase by a factor 2 in one second [118], [124]. According to accelerator specialists, this large increase is not credible. It can be seen that the power does not increase too much and levels off at about 1.5 times nominal. The elevated power leads to some pin failures after about 16.5 seconds. This would not have happened if the beam power was shut off actively or passively.

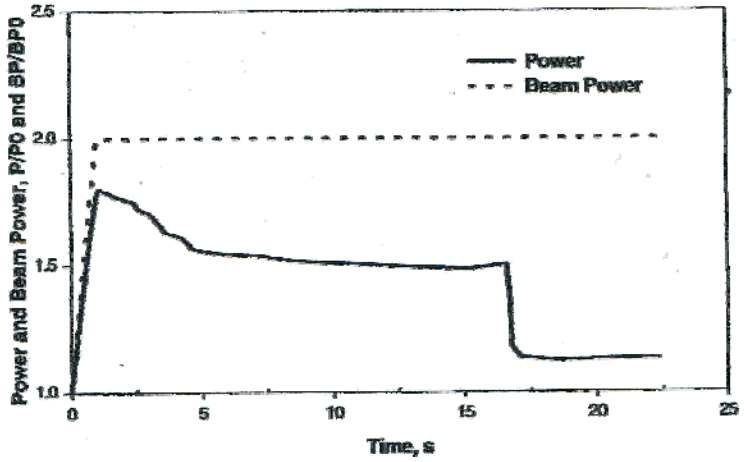

Figure 5.13: Assumed Beam Power Increase by a Factor of 2 in One Second.

Reactivity Accidents: It has also been realised that fast reactivity insertions do not lead to rapid power increases due to the subcriticality of an ADS which acts as a large delayed neutron group [124]. Figure 5.14 shows the power and reactivity histories due to a reactivity insertion of 170$/s,

for 15 ms (milli seconds), corresponding to a total insertion of 2.55$[11], in a lead-cooled ADS with an assumed subcriticality of -4$. The subcriticality is conservatively assumed to be relatively small and the ramp rate and total insertion to be rather high, particularly since an ADS will most probably have no reactivity control rods but only safety rods.

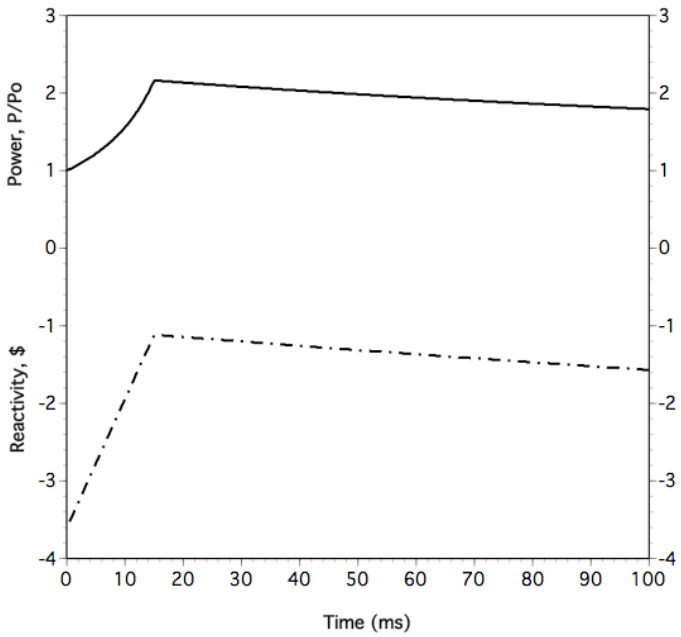

Figure 5.14: Power Excursion in a Lead-cooled Energy Amplifier (with k = 0.99) for a Slow Reactivity Ramp Insertion.
(The reactivity increases at a rate of 170 $/s for a period of 15 ms (this corresponds to a control rod withdrawal speed of 0.55 cm/ms in the case of a reactor). The accelerator is not shut-off [118].

In the case of the Energy Amplifier operated at $k = 0.99$, the power increases by a factor *2.5* after *15 ms,* and after *20 ms* the power decreases almost proportionally with the neutron source strength. If on the other hand the neutron source is maintained (the accelerator is not shut-off), the power remains almost constant in this time range. The average temperature of the fuel rises gradually, but at a much lower rate. Note that in this case the Doppler reactivity feedback is almost negligible and very much delayed (appears only after *23 ms*). The long time constant of the response implies that the heat loss from the fuel cannot be neglected anymore. In fact, there is sufficient time (of the order of a few seconds, as estimated by the convection studies described in [118]) for the natural convection mechanism to safely adapt itself to the new operating conditions without any fuel damage occurring.

The same reactivity accident was also calculated for the corresponding unscrammed critical sodium-cooled fast reactor [118], [124]. This led to a power pulse of more than 1000 times nominal and nearly complete core destruction. Two aspects explain the difference. First, the subcriticality of the ADS and second, the positive void coefficient that came into play in the accident in the critical reactor, but not in the ADS case. In the proposed ADS design, lead-bismuth is used which has a negative void coefficient.

To conclude, a lead or lead-bismuth cooled ADS with a good natural coolant circulation capability and a passive emergency decay heat removal system is very attractive to prevent or mitigate severe accidents. The critical *Loss-Of-Flow* and *Loss-Of-Heat-Sink* accidents in such a system will

[11] *One dollar (1$) is the reactivity insertion that will make a reactor prompt critical. It is equal to the effective delayed neutron fraction (β_{eff}), and therefore depends on the fuel.*

result in a slow coolant heat-up. This should normally lead to a manual shutdown. Fast and rather large reactivity insertions (which are of very low probability in a system without control rods) lead only to a limited power increase. This is due to the subcriticality of these systems. To stop such a limited overpower condition, a beam shut-off or the insertion of safety rods is necessary. The switching off of an accelerator is principally simpler than the insertion of safety rods.

5.4.4.6 Advantages of the ADS

Due to its special design, an ADS has some advantages when compared to more standard reactors.

1. The main advantage of an ADS is the subcriticality of the core. This gives an improved margin to prompt criticality leading to a runaway reactor like the Chernobyl reactor. However, this kind of reactivity-induced accident is highly improbable in modern nuclear power reactors, since they are constructed with negative power reactivity coefficients. This means that the reactivity decreases when the power increases, thus limiting power excursions. The Chernobyl reactor, on the other hand, had positive power reactivity coefficients, which lead to a rapidly accelerating power increase.
2. The largest benefit of the subcriticality is that it facilitates a more effective way of transmuting the minor actinides. The control of traditional reactors depends to a large extent on a small fraction of the neutrons emitted in the fission process that are delayed, varying from a few tenths of a second up to around 1 minute. For U-235 the delayed fraction of neutrons is 0.65 %, and this prevents rapid power increases for reactivity insertions less than 0.65 %. However, many of the minor actinides (Np, Am, Cm ...) as well as U-233 have lower delayed neutron fractions than U-235. If a substantial amount of minor actinides or U-233 is added to the reactor fuel, it will inevitably lead to a significantly reduced margin to prompt criticality. This problem does not exist for an ADS because the reactivity of the core can be reduced with an amount corresponding to the reduced margin to prompt criticality, at the cost of increasing the accelerator power to keep the power constant.
3. The production of long-lived actinides (Np, Pu, Am, Cm ...) is much smaller than for a conventional reactor fuelled with low-enriched uranium. The storage time for the waste may therefore be shortened from several tens of thousands of years to less than a thousand years. Here it should be mentioned that the storage time of waste from the fourth generation of reactors also will be significantly reduced compared with today's reactors, since actinide separation and recycling will be an integrated part of the fuel cycle.
4. The absence of control rods may simplify reactor control, since the power is proportional to the accelerator power. In any pressurized reactor system the penetrations of the control rods through the reactor vessel are possible leakage paths from the primary system. There is also a small possibility of control rod ejection, which would lead to a rapid reactivity increase.
5. Due to the high boiling point of the lead-bismuth eutectic (1670 °C), the pressure above the core can be kept at, or close to, atmospheric pressure. Low pressure reduces the risk of ruptures in the Cover Gas System (CGS). Of course, in the lower part of the reactor tank the pressure will be higher (≈ 28 bars) due to the 25 meters of lead. This is still much less than the 160 bar pressure used in the present generation of pressurized water reactors.

5.4.4.7 Disadvantages of ADS

There are also some drawbacks with the ADS:

1. The lack of operating experience due to the non-existence of a demonstrator or prototype.
2. The system is a combination of two complex machines, the reactor and the accelerator, each of which must be working for the system to produce power. The availability of the plant is therefore necessarily lower than for a conventional reactor. On the other hand, due to long burnup and small reactivity swings an ADS might not have to be shutdown as often as a critical reactor for changing or reshuffling the fuel.
3. Accelerators are large and highly advanced machines and will add a substantial amount to the overall cost of the power plant.
4. Due to the multitude of reactions (spallation, (p,xn), (α,xn), (n,γ) and fission) taking place in the lead-bismuth target, the coolant will become radioactive. Gases like xenon and krypton, and volatile elements like mercury, cesium, iodine, bromine and rubidium will find their way to the Cover Gas System (CGS) of the reactor. The activity in the CGS may be up to 100 000 times higher than in the CGS of "normal" reactors operating with lead-bismuth [106]. This will complicate maintenance and will also necessitate extra shielding of the CGS. Any reactor with bismuth in the coolant will produce the highly radiotoxic isotope Po-210 (half-life 138 days) by beta-decay of the activation product Bi-210. In an ADS there will also be production of other polonium-isotopes from (p,xn)- and (α,xn)-reactions. The most important isotopes produced by these reactions are Po-209 and Po-208, with half-lives of 102 and 2.9 years, respectively [106].
5. Since accelerators are large, they will be placed outside the containment building of the reactor. This may break the barrier between the primary system and the surroundings if there is a rupture in the beam tube or the window. The beam tube penetration of the containment is similar in nature to the steam line penetrations of for instance a boiling water reactor (BWR). However, in the case of a BWR there are fast acting isolation valves to close the steam lines in case of a leakage. This can also be fitted in the beam tube of an ADS, but a misaligned proton beam of several MW may burn through the beam tube and possibly also melt the isolation valve [106]. If a cyclotron is used, this problem may be avoided by placing it inside the containment building.
6. Lead at high temperature is highly corrosive, and special precautions must be taken to prevent degradation of the reactor vessel and the heat exchangers.
7. The window (the interface between the beam tube and the target) is exposed to an intense flux of both high energy protons and neutrons, and may need inspection or replacement at regular intervals.
8. A beam trip produces strong thermal stresses in the window and the supporting structure of the core, which over time may cause metal fatigue.

5.4.4.8 Conclusion

The ADS concept has been developed over several years, but has not yet reached a state allowing the building of a prototype power producing reactor. An ADS fuelled with thorium has some clear advantages over a conventional reactor:

1. *Much smaller production of long-lived actinides.*
2. *Minimal probability of a runaway reactor.*
3. *Efficient burning of minor actinides.*
4. *Low system pressure.*

The most important drawbacks of the ADS are:

1. *More complex than conventional reactors.*
2. *Less reliable power production due to accelerator downtime.*
3. *Large production of volatile radioactive isotopes in the spallation target.*
4. *The beam tube may break containment barriers.*

A power producing ADS using thorium may become a reality, but it is not reasonable to expect this to happen in the next 30 years. Much research and development remains to be done, especially in the fields of accelerator technology and material properties.

5.4.4.9 Status of ADS R&D Activities

The ADS has become a major R&D topic in Europe [120]. The resources presently allocated for the ADS in Europe are significant and are related to a large number of activities spanning from accelerator to materials and fuel technology. The total effort in the last decade is estimated to be of the order of 400 man-year/year. The organizations performing these activities range from national R&D bodies to universities with a significant participation of several major European nuclear industries. Important ADS activities are also going on or are planned outside Europe [125], e.g. in Japan (the *Japan Atomic Energy Agency (JAEA)* and the *High Energy Accelerator Research Organization (KEK)* joint project), in USA (the *Advanced Accelerator Applications Program (AAA)* recently completed a roadmap requested by the U.S. Department of Energy), in the Republic of Korea (the *Hybrid Power Extraction Reactor (HYPER)* Project at the *Korea Atomic Energy Research Institute's (KAERI)*), in India, and even in China where the government has approved its first R&D program.

An overview of selected Accelerator Driven System (ADS) projects is given in Table 5.6.

Table 5.6: Selected Accelerator Driven System (ADS) projects.

Project	Neutron Source	Core	Purpose
FEAT (CERN)	Proton (0.6 to 2.75 GeV) ($\sim 10^{10}$p/s)	Thermal (≈ 1 W)	Reactor physics of thermal subcritical system (k≈0.9) with spallation source - done
TARC (CERN)	Proton (0.6 to 2.75 GeV) ($\sim 10^{10}$p/s)	Fast (≈ 1 W)	Lead slowing down spectrometry and transmutation of LLFP - done
MUSE (France)	DT ($\sim 10^{10}$n/s)	Fast (< 1 kW)	Reactor physics of fast subcritical system - done
YALINA (Belorus)	DT ($\sim 10^{10}$n/s)	Fast (< 1 kW)	Reactor physics of thermal & fast subcritical system - done
MEGAPIE (Switzerland)	Proton (600 Me) + Pb-Bi (1MW)	-----	Demonstration of 1MW target for short period - done
TRADE (Italy)	Proton (140 MeV) + Ta (40 kW)	Thermal (200 kW)	Demonstration of ADS with thermal feedback - cancelled
TEF-P (Japan)	Proton (600 MeV) + Pb-Bi (10W, $\sim 10^{12}$n/s)	Fast (< 1 kW)	Coupling of fast subcritical system with spallation source including MA fuelled configuration - postponed
SAD (Russia)	Proton (660 MeV) + Pb-Bi (1 kW)	Fast (20 kW)	Coupling of fast subcritical system with spallation source - planned
TEF-T (Japan)	Proton (600 MeV) + Pb-Bi (200 kW)	-----	Dedicated facility for demonstration and accumulation of material data base for long term - postponed
MYRRHA (Belgium)	Proton (600 MeV) + Pb-Bi (1.5 MW)	Fast (60 MW)	Experimental ADS - under study FP6 EUROTRANS
XT-ADS (Europe)	Proton (600 MeV) + Pb-Bi or He (4-5 MW)	Fast (50-100 MW)	Prototype ADS - under study FP6 EUROTRANS
EFIT (Europe)	Proton (≈ 1 GeV) + Pb-Bi or He (≈ 10 MW)	Fast (200-300 MW)	Transmutation of MA and LLFP - under study FP6 EUROTRANS

The report "*Accelerator-driven Systems (ADS) and Fast Reactors (FR) in Advanced Nuclear Fuel Cycles*" [106], prepared by a panel of international experts under the auspices of the *OECD Nuclear Energy Agency*, clearly demonstrated the advantage of this positioning of ADS with respect to critical cores for concentrated management of waste. Hybrid systems perform as excellent dedicated minor actinide incinerators and offer the required flexibility for transition scenarios. This report also indicates that the meaningful reduction of the radiotoxicity of waste (at least a factor of 100) requires multi-recycling of fuels in which fuel losses to waste are to be very low.

The knowledge available today and at the end of the FP6 integrated projects (EUROTRANS and EUROPART) is and will be very important [126]. A better understanding and control over the different R&D topics required for ADS was and is still being gained, namely:

- Partitioning of high level waste (HLW) both via aqueous and pyro-reprocessing.
- Advanced fuels development heavily loaded with minor actinides (MAs).
- ADS Design addressing the key points of the ADS components:
  - ✓ accelerator development
  - ✓ feasibility of both window and windowless spallation targets

- ✓ material selection for the internal core structures, the spallation target and fuel cladding
- ✓ the sub-critical core design
- ✓ obtaining nuclear data for achieving a reliable design

The need for an experimental ADS in Europe and its objectives: In order to develop and test the technology for commercial deployment of accelerator driven systems, an experimental facility is required. This facility should in pilot scale prove the feasibility of operating a sub-critical reactor driven by a high intensity accelerator. It is needed to demonstrate the long-term applicability of corrosion control in lead-bismuth cooled reactors. It should further provide the fast neutron environment necessary for developing minor actinide based transmutation fuels. The availability of a fast neutron spectrum will be mandatory for development of more irradiation resistant structural materials that can increase the operational lifetime of commercial ADS or new generation critical reactors to be deployed in the future.

The experimental ADS can also be conceived with less demanding working condition in terms of facility rate availability (50 to 70 %) as compared to the one that would be envisaged for an industrial transmuter (70 to 90 %). Such a reasonable, but nevertheless acceptable, availability rate would allow accommodating the beam trips expected for the first-of-kind machine to be deployed. This will then allow improving the accelerator performance progressively towards the performances to be achieved for the industrial ADS.

The domain of interest of such experiment will be to show a reliable operation of the system, from start-up to nominal power level, up to shutdown, in presence of thermal reactor feedback effects. The presence of control rods in the system will allow verifying different modes of operation during fuel irradiation and the determination and monitoring of reactivity levels with "ad-hoc" techniques. The joint cooling of the target and of the sub-critical core will be demonstrated, together with the solution of some practical engineering problems of generic interest for an ADS, such as the configuration of the beam ingress into the core.

The possibility to run the experiment at different levels of sub-criticality and power ratings (realised e.g. with appropriate fuel loading patterns), will allow to explore experimentally the transition from an "external" source-dominated regime to a core thermal feedback-dominated regime. This transition is relevant, in particular to understand the dynamic behaviour of an ADS, which, in the future full scale demonstrations of transmutation, could have both a very low β_{eff} and very low Doppler reactivity effect.

One important project conducted at the Belgian Nuclear Research Centre *(SCK-CEN)* is the MYRRHA project. It aims to serve as a basis for the European experimental demonstration of transmutation in ADS and to provide protons and neutrons for various R&D applications. It consists of a proton accelerator delivering a 600 MeV, 2.5 mA (or 350 MeV, 5 mA) proton beam to a liquid Pb-Bi spallation target that in turn couples to a Pb-Bi cooled, subcritical fast nuclear core. The project started in 1997 and the aim is to put MYRRHA in service in 2016 - 2018. Additional information: [127], [128], [129], [130], [131], [132] and [133].

5.4.4.10 Roadmap for an Experimental ADS in Europe

The roadmap presented in April 2001 by the *Extended (actually European) Technical Working Group (ETWG)*: *"A European Roadmap for Developing Accelerator Driven Systems for Nuclear Waste Incineration"*, assumes that a major financial investment made as early as 2003, which was the scheduled year of the beginning of the FP6 (255 M€ over 4 years for R&D, engineering for the Detailed Preliminary-Design and the beginning of construction). This financial investment has

not yet been reached even when integrating the EC effort for P&T (31 M€) and the national efforts one can reach a total effort of 100 M€ including generic support R&D. This is certainly not enough as compared to the reasonable estimates of the ETWG. The reason for the delay as compared to the ETWG roadmap is that the option of high-level waste transmutation via ADS is not yet fully accepted by all European nuclear countries as the most appropriate method.

An updated roadmap [120] taking into account the outcome of the FP5 PDS-XADS project as well as the other FP5 ADOPT projects related to advanced fuel development, material and HLM technology and nuclear data that have led to start the FP6 integrated project EUROTRANS. There is a period of 4 years corresponding to the FP6 EUROTRANS period where potential show stoppers towards the deployment of an experimental ADS (labelled ADS (Phase1) in the ETWG roadmap or XADS in the PDS-XADS FP5 project) and later on the industrial ADS should be answered. At the end of this still generic development period one should enter a project dedicated structure for the construction of an experimental ADS that would last for 10 years for bringing the project to the full power operation of this facility. This will be including a first 3 years period for detailed engineering design of the facility, demonstration and testing of the reactor components and putting under beam the already established design of the spallation and finally preparing the construction site and the licensing file. The next period of 4 years would be dedicated to the construction of the components at the production site and the civil engineering at the facility site during 3 years and the fourth year would be dedicated to the assembly on site of the reactor as well as the accelerator components. In 2018 the facility will be then undergoing the commissioning tests for reaching the full operation stage somewhere at the beginning of 2020.

Alternatively, one could envisage coupling a commercially existing multipurpose reactor module of the type *SBVR-75/100* with an upgraded version of the PSI cyclotron (590 MeV, 3 mA) with a MEGAPIE like LBE spallation target (2 MW class). Indeed, the flexibility offered by this pool type reactor is eminently suited for the conversion into a subcritical configuration, which is achieved through: (1) the removal of the innermost fuel ring of the core and (2) the loading/unloading of the outermost fuel ring of the core to attain, by small reactivity steps, the desired sub-criticality level. Furthermore, this would have the advantage in principle of reducing the cost and the time needed to realise the experimental ADS.

In this configuration, a core power between 10 and 150 MW_{th} could be achieved in the subcriticality range between k = 0.9 and 0.98 for a maximum beam power of only 1.8 MW.

The multi-purpose reactor module SBVR-75/100 MW_e [126], [134] is a follow-up of the Russian program of lead-bismuth cooled fast neutron reactors for Alpha type submarines which has accumulated more than 80 reactor years of operational experience. It is an integral pool type reactor with the steam generators sitting in the same lead-bismuth pool at 400 - 480ºC.

The PSI ring cyclotron accelerator produces a proton beam with the highest power in the world. The protons are accelerated in the ring cyclotron to almost 80 % of the speed of light (about 236000 km/s), corresponding to an energy of 590 MeV. The proton current amounts presently to almost 2 mA, which results in a beam power of over 1 MW. The principal components of the ring cyclotron are eight sector magnets, with a total weight of 2000 tonnes, and four accelerator cavities (50 MHz frequency) each having a peak voltage of 730 kV.

6. THE BACK END OF THE THORIUM FUEL CYCLE

The back end of the thorium fuel cycle consists of fuel reprocessing and waste storage. It differs from the back end of a uranium cycle in some important aspects:

- Virtually no plutonium or other transuranic elements are produced.
- The waste products are free from the long-lived alpha-emitters.
- Due to the stability of thorium dioxide, more aggressive chemicals must be used in the process, thus increasing the corrosion problems in the reprocessing plant.

6.1 Thorium Cycle in Reactors

Breeding of U-233 in thermal reactors is difficult and requires excellent neutron economy and on-line fuelling-reprocessing. Breeding of U-233 in fast reactors is possible, but the doubling time of the fissile inventory is much higher (about 250 years) than in the case of Pu-239 breeding.

The reactor must be started with U-235 or Pu-239 (topping fuel), or with an accelerator-driven neutron generator. After a number of years, sufficient U-233 is formed to obtain equilibrium without further addition of topping fuel.

At present, the best thermal reactors for the thorium cycle are the Molten Salt Reactor (MSR), the Heavy Water Reactor (HWR, or CANDU) and the High Temperature Reactor (HTR). Thorium could also be used in Light Water Reactors (LWRs). Thorium can be used without any topping in conventional fast reactors.

6.1.1 Once-Through (Open) Thorium Cycle

In the case of the once-through cycle (open cycle) in LWRs, there is always a need for topping fuel. Maximum benefit from the produced U-233 requires as high as possible burnup, and U-233 is the best topping fuel for minimum actinide production. The introduction of thorium reduces the average annual consumption of uranium only by a few percent [135]. The radiotoxicity due to the actinides in the spent fuel is significantly reduced compared with that of spent UO_2 fuel. After 10 000 to 30 000 years of storage, the remaining U-233 dominates the radiotoxicity. However, in this last period of geological storage the actinides form only a minor risk, as fission products dominate the dose due to dispersion.

Reactors with higher fertile-to-fissile conversion ratios can benefit more from thorium fuelling. An example of what can be achieved with the once-through cycle in a CANDU heavy-water reactor is reported by the *Energy Research Centre of the Netherlands (ECN)* [136], [137]. In Figure 6.1 the curve labelled "*A*" represents the effect of using thorium with HEU topping. The thorium-cycle waste results in a relatively low waste toxicity immediately after the fission product period (up to about 600 years) because less plutonium and americium are formed. Later, the radiotoxicity becomes as high as that of unreprocessed LWR spent fuel because of Ra-226 from U-234. However, higher burnup could further reduce the radiotoxicity. In practice other topping fuels can be used (such as MEU), for economic reasons or to fulfil non-proliferation requirements, but this, unfortunately, leads to additional radiotoxicity contributions.

Similar results are obtained in the HTR, since it has the same conversion ratio and uses similar annual quantities of thorium and natural uranium fuel as the LWR. In a fast reactor, the open

cycle should not be considered to reduce radiotoxicity, because a fast reactor requires a much higher fissile inventory. The problem is the U-233 in the spent fuel.

6.1.2 Closed Thorium Cycle

In a closed cycle, the spent fuel is reprocessed to recycle U-233 and thorium and possibly also the other produced actinides, in particular protactinium. The ideal is a self-sustaining cycle, i.e. after a number of cycles there is no further need for topping fuel. Self-sustaining cycles are in principle possible for thermal as well as for fast reactors. In thermal reactors, this is only possible if there is an excellent neutron economy, on-line refuelling capability and, preferably, on-line removal of fission products. The best systems are molten-salt reactors (MSRs), followed by CANDU reactors and HTRs. However, in the latter two reactors, the maximum burnup that can be achieved is limited. Accelerators could also provide additional neutrons to facilitate a self-sustaining cycle (see Chapter 6.2).

Figure 6.1 shows some results for a CANDU reactor with HEU topping. The radiotoxicity of the waste is shown for the case if only uranium is recycled (curve B). Up to 1000 years, the remaining toxicity is caused mainly by Pu-238, and later by protactinium-231 (Pa-231). If plutonium and protactinium are recycled, it is demonstrated [137] that after two recyclings and after 1000 to 2000 years of storage, the radiotoxicity of the waste is already below the level of the uranium ore needed for fuel fabrication for the once-through PWR.

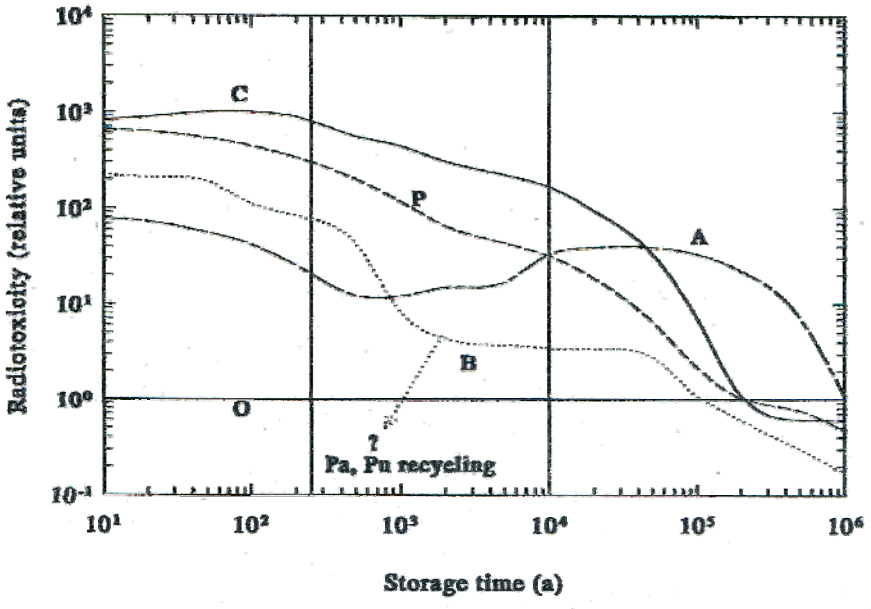

Figure 6.1: Actinide Radiotoxicity of the Waste if Thorium is used in a CANDU Reactor with Once-through Operation and HEU as Topping Material (curve A). Curve B shows the actinide radiotoxicity of a CANDU with HEU topping and recycling of U in equilibrium (curve B). Further reduction is possible if also Pu and Pa are recycled. References are the spent fuel of standard once-through CANDU (curve C) and PWR reactors (curve P) and the necessary U ore for a PWR (curve O).

The closed thorium cycle in a fast reactor breeds U-233. The radiotoxicity of the spent fuel could be further reduced if the uranium and the actinides are recycled. Due to the fast neutron spectrum, fissioning of produced minor actinides is more efficient than in a thermal reactor. A self-sustaining fast reactor based on the Th/U-233 cycle is possible, albeit that the doubling time of U-233 is quite large (about 250 years). Improved safety feed-back coefficients are obtained. The

Th/U-233 cycle waste, with the same reprocessing losses, should lead to a factor 10 lower radiotoxicity than that of the U/Pu cycle waste.

To conclude, there are advantages in using thorium in thermal reactors in a once-through mode, particularly if U-233 or U-235 (HEU) can be used as topping fuel. If the burnup of thorium fuel can be extended, the waste is less radiotoxic up to 20 000 years, and perhaps more importantly, has a smaller volume compared with that from (enriched) uranium. The closed Th-cycle offers the best performance with respect to long-lived waste radiotoxicity compared with U/Pu scenarios, in particular if fast reactors are used.

A recent study has shown that the radioactive waste from a thorium cycle will be significantly less than for the standard U/Pu cycle, in the same conditions. This is an advantage for thorium based fuels and has been confirmed in several studies; recently in the EU-supported study "Thorium as a waste management option" [138].One result of the study is illustrated in Figure 6.2 where the overall radiotoxic inventories (Sv/TWh_e) are compared for U/Pu and Th/U cycles as a function of elapsed time. The study is for a case where all actinides in a fast neutron reactor are recycled, assuming 0.1 % losses. It has been pointed out by *Dominique Greneche et al* [139] that there is a gain of a factor of 5 to 20 up to 10 000 years with thorium based fuels compared with U/Pu fuels. After about 20 000 years, the radiotoxic inventory of thorium based fuels becomes greater than that of U/Pu fuels. This is of less concern since in both cases the inventory is much lower than that of natural uranium itself. The comparison is based on the equivalent amounts of natural uranium that are needed to feed an open uranium cycle.

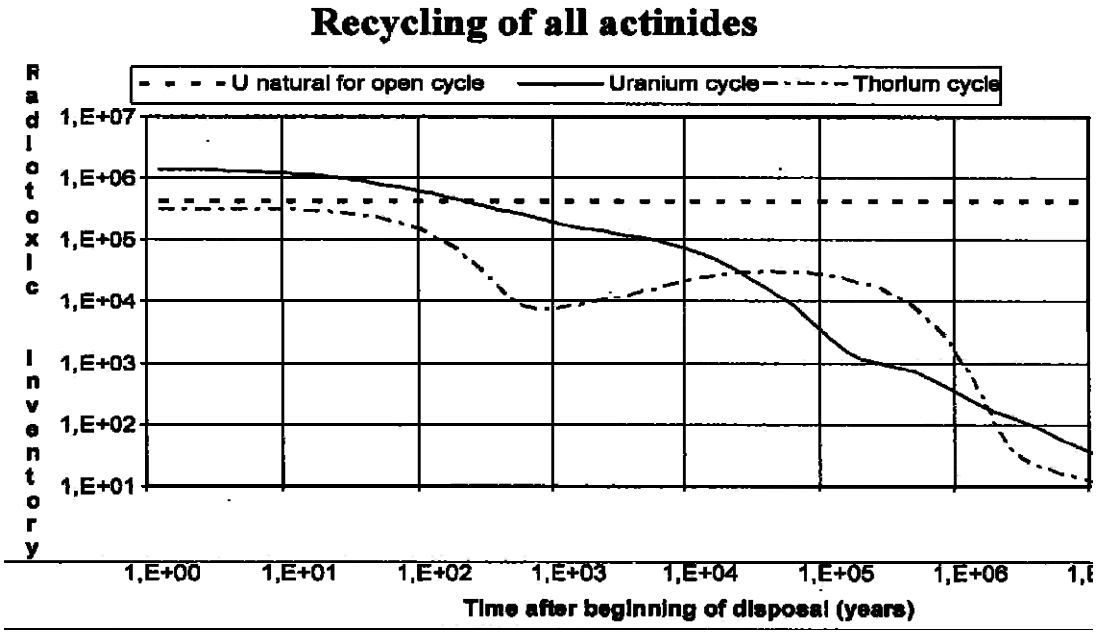

Figure 6.2 Reduction of Radiotoxicity as a Function of Time Elapsed after Disposal of the Waste in an Inventory for the Cases of Uranium and Thorium Cycles and with Recycling of all Actinides.

6.2 Thorium Cycle in Accelerator-Driven Systems

Critical reactors using thorium fuel have been operated in the past, motivated by the prospect of a high neutron yield per neutron absorbed, which U-233 offers over the whole neutron energy range, only slightly surpassed by Pu-239 for fast neutrons (see Table 5.1 in Chapter 5.2 for details). However, there is an associated disadvantage in breeding U-233: i.e. the production of Pa-233, which has a large neutron capture cross-section, and which must be compensated for by a

higher enrichment of fissile material. Also, U-233 fissions produce more Xe-135 (direct yield 1.4 % for U-233 versus 0.3 % for U-235) and samarium precursors (Nd-147, Pm-149) than U-235. These isotopes represent a significant fraction of the total neutron absorption by fission products. At mid-cycle they account for more than 50 % of the total fission product absorption.

The larger number of available neutrons is the second important advantage of ADS. This is relevant to:

- Burning of nuclear waste, in particular for transmutation of minor actinides and long-lived fission products;
- Breeding of fuel (in particular U-233) with short doubling time;
- Extending the fuel burnup.

A thermal ADS has advantages for long-lived fission product transmutation and for a thorium cycle without the necessity of topping fuel in the equilibrium situation. In the most advanced concepts, liquid fuel/coolant in the form of *molten salts* is proposed, based upon the thorium cycle (Los Alamos concept [140], [141], [142]), although many technical problems with regard to corrosion and inhomogeneity have to be solved before this option can become reality. The disadvantage of a thermal system is largely compensated for by the on-line fuelling and reprocessing option.

The CERN Energy Amplifier [143] is based upon a rather compact accelerator (cyclotron), coupled to a fast breeder lead-cooled reactor. The value of the k_{eff} is rather high to economise the energy production. A rather long cycle length of up to 5 years is proposed (see Figure 5.10 in Chapter 5.4.4.2: *Description of the ADS* for details). The closed thorium cycle is proposed with full recycling of all the actinides. This offers very low resource consumption and minimum waste production.

A study carried out for the Spanish government [144], based on a practical example, showed that a 1500 MW_{th} Energy Amplifier could destroy a net amount of *298 kg* of TRU per *GW × year* of thermal energy produced. In comparison, a PWR produces *123 kg* of TRU per *GW × year* of thermal energy produced (91 % Pu, 7.5 % Np, 1.4 % Am and 0.1 % Cm).

The radiotoxicity of the actinide waste should be similar to that of a fast reactor fuelled with thorium. However, slightly better performance is possible due to the increased burnup of the ADS. In the CERN Energy Amplifier, reprocessing losses of 0.01 % are assumed, yielding as a consequence a very low radiotoxicity, close to the level of radiotoxicity of coal burning. Further reduction is possible by transmuting the long-lived fission products, using the adiabatic resonance crossing technique [145].

It can be concluded that accelerator-driven systems have additional safety margins, which give operational flexibility to future systems for safe and clean energy production and/or waste transmutation. The accelerator offers the possibility of applying a closed thorium cycle, but also for an open, once-through cycle using thorium oxide with some topping fuel and a very high fuel burnup. The Energy Amplifier is one of the examples with high potential.

6.3 Reprocessing of Thorium-Base Irradiated Fuels and Waste Management

The true thorium cycle requires breeding and recycling of U-233. The reprocessing technology is proven on a commercial scale. Advanced dry processes are not proven for thorium, and for the time being, reprocessing options are limited to wet chemical operations in which the fuel

substance is dissolved in nitric acid and separated by solvent extraction. Thorium is considerably less amenable than uranium to such processing.

Reprocessing difficulties are especially severe with HTR-type fuels of whatever composition, since the particle coatings and graphite matrix are chemically resistant and troublesome to break down mechanically.

6.3.1 The "Head-End" Operations

The aim of these operations is to obtain, after preliminary operations leading to dissolution, a solution of U-Th nitrates together with a small quantity of minor actinides and fission products.

Extraction of U-233 from irradiated thorium rods was conducted extensively in the US from the 1950s until the 1970s. Almost 700 tonnes thorium were irradiated, delivering more than 1.5 tonnes of U-233. The separation operations were piloted at ORNL in the THOREX plant, and in the *Knolls Atomic Power Laboratory*. The process was later adapted to the Great "canyon" plants at Hanford and Savannah River (SRP), and the commercial direct maintenance plant of Nuclear Fuel Services at West Valley near Buffalo, now being dismantled, which reprocessed the first core of Indian Point 1 (95% ThO_2, 5% U-235O_2), but without recovering thorium (which was left in the waste).

The processes are explained in an interesting paper [146], which is one of the early documents giving details of the processes and of the adaptation of these large PUREX-type plants to the THOREX processes (Figure 6.3).

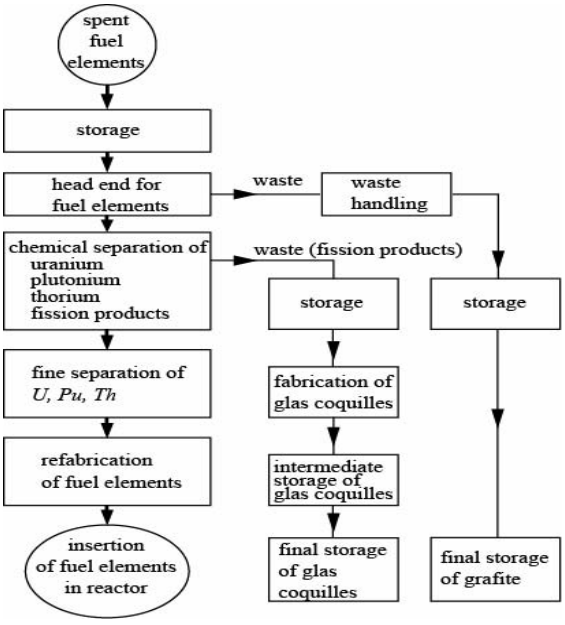

Figure 6.3: Outline of the THOREX Process (Thorium Extraction).

Head-end operations for fertile fuels to extract U-233: Due to its great stability, dissolution of thorium or ThO_2 is not as straightforward as that of U, and especially of UO_2. Strong nitric acid with HF is required, and the process takes a rather long time, especially for high temperature sintered compact ThO_2. Dissolution can take up to 35 hours. To prevent corrosion of the equipment by HF, aluminium nitrate was added as a buffer.

Similar processes have been used in India on a small scale for thorium and ThO_2 rods and in other countries (UK at Dounray, Canada at Whiteshell, Germany at Jülich and Karlsruhe). In

India, the engineering pilot scale laboratory has been replaced by a 50 tonnes/year reprocessing plant at the *Indira Gandhi Centre for Atomic Research (IGCAR)*, Kalpakkam, India.

In summary, "traditional" reprocessing plants can process irradiated thoria fuels, even though the dissolution is more difficult and precautions are required against HF corrosion.

Head-end operations for oxide LWR fuels: The pilot tests made at Oak Ridge were confirmed by "commercial runs" of the PWR Indian Point 1 $(U,Th)O_2$ core in the West Valley plant, for sizeable irradiation rates (16 000 MWd/tHM). The PWR fuel bundles clad in Zircaloy are sheared and the fuel is dissolved. It was found that, although irradiated $(U,Th)O_2$ dissolves quicker than unirradiated material, the dissolution is much slower than for UO_2 alone, as the thorium oxide is more compact and sintered. Concentrated nitric acid is needed with the resulting corrosion problems. The dissolver offgases will contain somewhat more Kr-85, depending on the proportion of U-233, as the yield of Kr-85 by fission is about double that of U-235. Iodine yield is about the same as for U-235.

Head-end operations for oxide or carbide HTR fuels are quite different and pose problems which have not been totally solved on the industrial scale, although many engineering tests have been performed, mainly at Oak Ridge, at General Dynamics in the USA and in Jülich (Germany), for their HTR programs. Limited tests have been performed in other laboratories, especially at the *National Committee for Nuclear Energy (CNEN)*, Italy. Coated oxide and carbide fuel is very difficult to process. This is mainly due to its PyC and impervious SiC coatings, and in part to the unavoidable carbon residue. The same is true for the TRISO-coated particles, for which different tests have been made in Jülich on the famous "pebbles" of the AVR and THTR reactors. Most of the irradiated pebbles have not been further processed. Direct dissolution of the coated particles has been tested at CNEN/ENEA and at IGCAR (India), using electrolytic dissolution. The laboratory tests are promising, although the dissolution times are longer and the equipment more complex than usual. It would be important to test such devices on a large scale to assess whether they work satisfactorily.

The complexity of these pre-treatments, and the technological problems associated with them, are one of the reasons why the High Temperature Reactor family has not been further developed. Broadly speaking, the coated kernel fuel is so resistant that it cannot be reprocessed by conventional reprocessing methods. The coated kernels, today, are seemingly the typical "once-through" fuel. They could be reserved for high temperature "burners" working on long once-through cycles (lasting many years), delivering high temperature gas. On the other hand, the "breeders", thermal or fast, should have fuel elements directly amenable to a simple reprocessing process, i.e. of the traditional oxide-in-clad type, water or liquid metal cooled.

6.3.2 Separation and Purification Operations: the THOREX Process

The reprocessing of thorium containing fuels may be carried out using the THOREX (thorium extraction) process. This was originally developed in the US at the *Knolls Atomic Power Laboratory (KAPL)* and the *Oak Ridge National Laboratory (ORNL)* in order to reprocess large amounts of irradiated thorium from light water reactors. Development continued in Germany for the reprocessing of High Temperature Reactor (HTR) fuel, which is described in the report from Kugeler [48].

The heavy metal and matrix graphite are first separated in the head-end of the reprocessing plant by burning of the graphite. For this step, the fuel elements can be crushed to pieces smaller than 5 mm. The burning of the coatings, including SiC-coatings, can be carried out in fluidised

bed ovens. The heavy metal ash is dissolved in the THOREX solution in the next step. Afterwards, separation into uranium, thorium and fission products is carried out in a solvent–extraction process in the chemical process part of the reprocessing plant. Additional equipment for the conditioning of high active waste, and for filtering gaseous fission products and other pollutants are necessary for the reprocessing of HTR fuel. The experiences gained with the reprocessing of LWR fuels may be used to a high degree in the THOREX process.

Extended research and development of the reprocessing of thorium containing fuel has been carried out in the last decades; and the feasibility of the process has been proven. The *JUPITER* facility, which includes the THOREX process and head end facilities, has been built and operated in Germany.

The PUREX (plutonium uranium extraction) process has become the reprocessing procedure generally used for all fuel types containing natural, slightly or highly enriched uranium together with lower or higher contents of plutonium. The THOREX process, on the other hand, has been developed for reprocessing uranium and thorium from thorium-based fuel. When the fuel contains appreciable amounts of U-238, plutonium will be produced and a combination of the THOREX and PUREX processes must be applied.

The THOREX process is technically less advanced and has the drawback that thorium nitrate exhibits a much lower distribution coefficient than uranium and plutonium. To drive thorium into the organic tributyl phosphate (TBP) phase, a strong salting agent is required. Aluminium nitrate, which has been recommended previously, has now been replaced by nitric acid in order to reduce the amount of radioactive waste. However, high acid concentrations are counter-effective in achieving high fission product decontamination. Therefore, several flow-sheet variants, with acid and acid deficient feed solutions, have been investigated in the past. In order to achieve high decontamination factors, a dual cycle THOREX process was developed. This process uses an acid feed solution in the first cycle and an acid deficient one in the second cycle.

According to recent investigations, a single cycle process with acid feed solution should provide the necessary decontamination factors. An immediate separation of thorium and uranium appears advisable in view of both fuel cycle strategy and process feasibility. Pulse columns should be preferentially used as extraction apparatus, at least for the extraction step.

Reprocessing of HTR-LEU fuel in existing PUREX plants has associated problems of criticality prevention due to the residual total fissile isotope content in the feed of greater than 2 wt%. Special precautions must therefore be taken in several processing units. The addition of soluble poison to the feed proved to be an unfavourable measure. A more suitable method to overcome criticality difficulties may be the application of a flow-sheet with lower tributyl phosphate (TBP) concentrations, less than 10 %.

6.3.3 Waste Treatment

The waste treatment from a THOREX reprocessing plant will not differ much from that of a modern PUREX reprocessing plant. However, two remarks seem appropriate:

- The fact that the thorium fuel cycle waste will contain less minor actinides and hence will be less radiotoxic for the first 10 000 years or so.
- Dissolution of thoria-based fuels requires a small proportion of HF, buffered by aluminium nitrate. The effect of the fluoride on the vitrification of the waste must be ascertained.

6.3.4 The Dry Processes

Dry processes have been envisaged for either the processing of carbide kernels or for the in-line processing of the molten salts of the molten salt breeder reactors.

For the kernels, a heat treatment to up to 2800 – 3000ºC followed by chloridation has been envisaged. For the molten salt fluorides, the fluoride volatilization process has been extensively investigated in a few countries (USA, Argone National Laboratory, France, UK, etc.), generally with success. Advantages of the process include its relative compactness and absence of aqeous effluents. The drawbacks are the problems of working with fluorine and the corrosion of the equipment.

In the case of the ADS (Energy Amplifier Project), pyroprocessing is regarded as a key technology in many aspects. In comparison with acqueous reprocessing, it promises:

- Compactness and simplicity;
- Less secondary wastes;
- Proliferation resistance (no separation of the TRUs);
- Fuel fabrication and reprocessing at the reactor site.

The spent fuel rods are chopped into small sections, which allow separation of the spent fuel from the cladding. Fission product gases released at this stage are collected through the ventilation systems on filters and sent for storage. The spent oxide fuel is then converted to metal. In this process calcium reacts with the oxide fuel to produce calcium oxide and heavy metals (U, Np, Pu, Am, Cm). The reaction takes place in a high temperature molten calcium chloride salt bath. Again, fission product gases released are collected and sent for storage. Metals such as Cs, Sr, and Ba are partitioned from the molten salt, which is periodically removed for storage. The resulting heavy metal is then sent for electro-refining. This is an electro-chemical process in which the thorium is separated from the actinide and fission product mixture. A NaCl-KCl molten salt at 1000 K is the transport medium. The thorium is collected at the cathode and removed periodically for further processing. Noble metal fission products (Zr, Mo, Ru, etc.) remain at the anode heel in the cell. The actinides and rare earth fission products remain in the molten salt. This salt is then sent for further treatment in the electro-winning process. This is also an electro-chemical process and is used to deposit the actinides (present in the form of chlorides) from the NaCl-KCl molten salt at the cathode of the cell.

However, development of the technology has not yet reached the stage where it can be demonstrated on an industrial scale.

7. RADIATION PROTECTION OF MAN AND THE ENVIRONMENT

Radiation protection refers to the protection of man and the environment from negative effects induced by ionizing radiation emitted from radionuclides. The Th-232 and U-238 decay series include a number of α-, β- and γ-emitting radionuclides. During the production of energy, other α-, β- and γ-emitting radionuclides are produced, and some of these will also be present in the waste. The α-radiation is short-ranged in air and external exposure to α-radiation is easily shielded. Internal exposure, when α-emitting radionuclides are included in cells due to inhalation or digestion, can produce detrimental effects, and these nuclides are referred to as highly radiotoxic. For β-radiation, internal exposure is also of importance, while high energy β-radiation also contributes to external radiation that must be shielded. The γ-radiation is long-ranged and the external exposure is of key relevance; lead shielding is required.

When living organisms are exposed to ionising radiation, free radicals are produced, which lead to oxidative stress and DNA damage, and which may result in effects such as reproduction and immune system failure, mutation, morbidity and mortality. At very high doses (several Sievert (Sv)[12]), deterministic effects such as acute radiation sickness can occur, while at low levels, stochastic effects due to chronic exposures, such as cancer, may arise in man. High dose-effect relationships for man and most organisms are well known, while at low doses (less than 50 mSv) the effects are still not fully understood. Information on low dose-effect relationships for non-human organisms is scarce, and is currently being studied in international research programs.

Natural radioactivity in the environment is not regulated, although the Norwegian Radiation Protection Authority (NRPA) provides advice if indoor exposure to radon is high. To protect man from "man-made" ionising radiation, conservative dose limits have been set by NRPA: 1 mSv/year for the public; and 20 mSv/year for occupational exposed workers in the nuclear industry., For the latter, permission can be given for a limit of 100 mSv over a continuous five-year period, on condition that the effective dose does not exceed 50 mSv in any single year (*Radiation Protection Act*). These limits are in accordance with international recommendations (ICRP). No dose limits are set for the environment. Furthermore, the ALARA principle, according to which doses should be kept "As Low As Reasonable Achievable" taking social and economic aspects into consideration, is also valid in Norway.

The use of thorium (Th) as an energy source includes:

Front End of the Fuel Cycle		Back End of the Fuel Cycle		
Mining and Milling	Extraction / Fuel Fabrication	Reactor Operation	Spent Fuel Storage and Reprocessing	Disposal of Waste and Spent Fuel

The dose contributions from the thorium cycle can be separated into exposures associated with the front end (mining, milling, extraction and fuel fabrication) and exposures associated with the back end of the cycle (operation, waste storage, reprocessing, and disposal of waste).

[12] *Units:*
Radiation Emitted from a Source: Desintegration per Second = Becquerel (Bq)
Radiation Absorbed by Living Organisms: Absorbed Dose = Gray (Gy)
The Effect of the Absorbed Dose: Equivalent or Effective Dose = Sievert (Sv)

7.1 Doses Associated with the Front end of the Thorium Cycle

Dose contributors in the front end (mining, milling, extraction and fuel fabrication) of the thorium cycle are naturally occurring radionuclides only. Due to the long half-life ($T_{1/2}$) of Th-232 (and U-238), very low doses are received from pure Th-232 (and U-238), while thorium minerals will also contain the α-emitting Th-228 ($T_{1/2}$ = 1.9 years). As for uranium, key dose contributors are α-emitting daughters such as radium (Ra) isotopes, as well as polonium (Po-216, Po-212) isotopes produced from radon (Rn). Radium isotopes, which are soluble in water, are often concentrated in ground water. Radon is a noble gas which can be released from minerals, rocks or ground water, but does not interact with matter. Thus, radium, radon, polonium and lead (Pb) isotopes can be displaced from their original deposits and can contribute to the dose to man and the environment. Due to the short half-life of Rn-220 ($T_{1/2}$ = 55.6 seconds) from Th-232, the migration of radon will be short before disintegration to particulate, reactive Po-216 occurs. In contrast, the potential migration of Rn-222 ($T_{1/2}$ = 3.825 days) from U-238 will proceed much further before its decay. Furthermore, the half-lives of the Rn-220 daughters are shorter, and hence their concentrations lower, than those of the Rn-222 daughters. Due to its short half-life, only traces of the high energy gamma-emitting daughter thallium (Tl-208) will be present per kg thorium. Thus, the impact of the radon daughters from thorium is lower and easier to handle than those from uranium. Doses received at a distance from a natural thorium source should thus be much less than those from a natural uranium source and the protection of crushed rocks and tailings can be obtained by a relative thin clay layer. Close to the thorium source (mining), the dose received can be comparable or somewhat higher than for uranium mining. It should, however, be noted that thorium resources also may contain uranium (levels of ppm).

Technologies to protect the offsite environment from thorium mining, milling, and refining effluents were reviewed in the US in the 1970s [147]. The waste arises from mining, milling, and refining of thorium (Figure 7.1). As the tailings pile containing the bulk of the thorium daughters will be a major source of potential release to the environment, a clay cover is usually needed. Some of the general differences between the front end cycle of thorium and uranium are outlined in Table 7.1. Information on Rn-220 is however, currently unavailable.

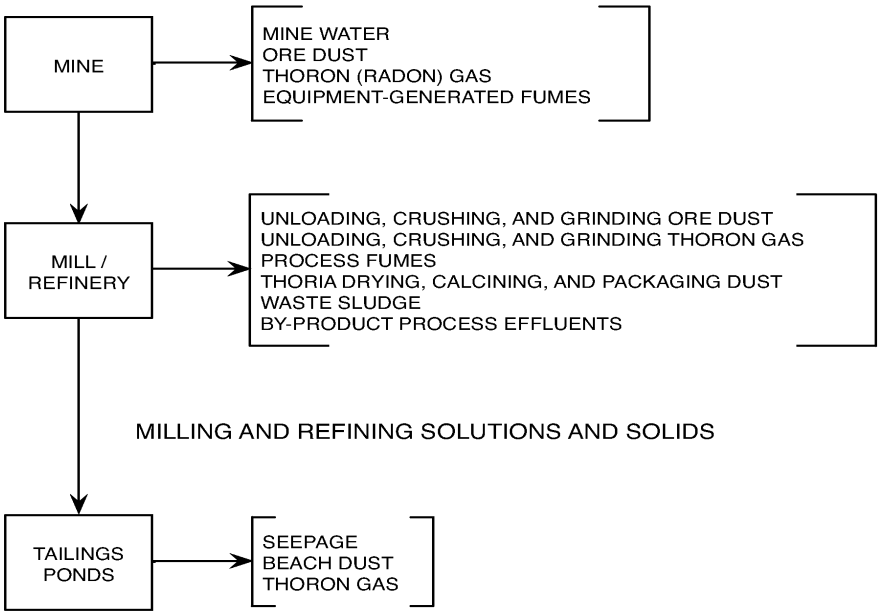

Figure 7.1: Typical Types of Wastes from Mining, Milling and Refining Thorium.

The radioactivity in the tailings originates from the ore, including a small percentage of thorium that was not recovered and the thorium daughters that were originally in equilibrium with the thorium. Uranium and its daughters may also be included at some of the sites. The daughters of Th-232 have all much shorter half-lives than the daughters of U-238. Thus, it is a great deal easier to stabilize thorium mill wastes than it is to stabilize uranium mill wastes [147].

Radioactivity from Ra-228 and its daughters will be almost down to zero within 50 years after the pile is stabilized. At that point, the bulk of the radioactivity will come from the residual thorium and uranium (if present) and their daughters. The shorter half-life of Rn-220 ($T_{1/2}$ = 54.5 seconds) from Th-232, versus 3.825 days for Rn-222 from U-238 will allow the throium tailings covers to be much thinner than those for uranium (by a factor of 6000). A 0.5 m earth cover would barely reduce the radon exposure from the uranium tailings, while the exposure from thorium tailings should be reduced significantly [147].

Table 7.1: General Differences between Thorium and Uranium Ores in Mining, Milling and Refining.

	Thorium Ore	Uranium Ore
1	Contained in igneous rock	Contained in sedimentary rock (sandstone)
2	Form: nodules peppered in a rock matrix	Form: fines coating sand kernels
3	High degree of grinding needed for feed material	Moderate crushing and grinding needed for feed material
4	High silica	Low silica
5	Gravity separations difficult	Gravity separations easier
6	High slime content in mill tailings effluent	High sand content in mill tailings effluent
7	Mineralogy, climatology, and topography are different than current uranium ores	Current mines mostly in arid regions
8	High pyrite and sulfides can affect mine water pH	pH usually about 7 (neutral)
9	Behaviour of thoron (radon-220) not well established	Radon-222 is better understood
10	Radium-228 and radium-224 daughters have short half-lives (6.7 years and 3.69 days) and require short-term control and containment	Thorium-230 and radium-226 daughters have long half-lives (1622 years and 75 200 years) and require long-term control and containment
11	Thoron (radon-220) has a short half-life (54.5 seconds). Requires less time to filter and less cover to contain	Radon-222 has a longer half-life (3.825 days). Requires more time to filter and greater cover to contain
12	Probably a greater hazard to miners	Probably not as great a hazard to miners

Fuel fabrication: Oxides of thorium, uranium and plutonium have very similar thermodynamical and thermophysical properties. Hence radiation protection associated with the manufacturing of oxide fuels for Th-232 and U-233 should be similar to that needed for the well known manufacturing of uranium oxide and mixed oxides fuels.

7.2 The Radiological Situation in the Fen Complex

Minerals with high levels of thorium have been identified in different counties of Norway; in the Permian Oslo Province, Vestfold, in Aust-Agder, Nord-Trøndelag, in Nordland and the Fen area

in Telemark (Chapter 3: Thorium Resources in Norway). The *Fen Complex* has been considered as the most promising resource, with thorium amounting to about 0.1 – 0.4 wt%.

Doses to man and the environment in *Fen* are attributed to the high Th-232 levels (Table 7.2, [15]) and the presence of uranium in the rock types rödberg (Th-232 average 3100 Bq/kg, can reach 5000 Bq/kg, Ra-226 from U-238 can reach 120 Bq/kg) and rauhaugite (Ra-226 up to 300 Bq/kg). The thorium levels in the Nordic rock types typically range from 0.5 to 350 Bq/kg and the thorium levels measured in the rödberg are the highest ever recorded in Norwegian bedrock [15]. The Ra-226 levels from U-238 in rödberg and rauhaugite are also high, concentrations above 100 Bq/kg are generally only found in black shales and radium-rich granites. The total effective dose from naturally occurring radiation received by the Norwegian population is currently estimated to be 2.9 mSv per year, while critical groups in *Fen* can easily receive more than 10 mSv per year [15]. Based on a recent investigation, the enhanced levels of thorium and uranium and their daughters in the *Fen Complex* contributed to the highest outdoor and indoor gamma exposures to man ever reported in Norway, and are among the highest in Europe. Thorium and associated radionuclides and metals are expected to represent an environmental problem in the *Fen area*, due to the presence of thorium bearing minerals in the rock types rödberg and rauhaugite. It is expected that the radionuclides can mobilise from surface minerals especially during flooding and acid rain episodes. Mobilisation processes may produce bioavailable radionuclide/metal species, which may accumulate in biota and contribute to internal exposure. The external and internal radionuclide exposures can induce biological responses of relevance for reproduction and growth, especially if synergism with associated metals takes place. However, doses to man and the environment from natural radiation are in principle not regulated. When man-made activities such as mining and milling are introduced, a revised version of the *Radiation Protection Act* will most probably be developed.

Table 7.2: Activity Concentrations of Th-232, Ra-226 (from U-238) and the Potassium Isotope K-40 in Rock Samples from the Fen Central Complex and Adjacent Areas.

Rock Type	Number of Samples	Th-232 (Bq/kg) [a)]	Ra-226 (Bq/kg)[b)]	K-40 (Bq/kg)[c)]
		Mean Range	Mean Range	Mean Range
Rödberg	9	3100 (390 - 5900)	70 (20 - 110)	310 (60 - 430)
Rauhaugite	9	600 (290 - 930)	120 (40 - 300)	60 (40 - 70)
Sövite	9	80 (20 - 190)	20 (10 - 60)	30 (20 - 40)
Fenite	8	130 (20 - 200)	50 (40 - 80)	1060 (750 - 1500)
Precambrian Gneiss [d)]	3	66 (68 - 63)	45 (43 - 46)	-

a*) 4 Bq/kg Th-232 is equivalent to 1 ppm Th, at radioactive equilibrium.*
b) 12.3 Bq/kg Ra-226 is equivalent to 1 ppm U, at radioactive equilibrium.
c) 310 Bq/kg K-40 is equivalent to 1 % K.
d) Outside the Fen Central Complex.

7.3 Doses Associated with the Back end of the Thorium Cycle

Radiation associated with the back end of the thorium cycle and associated safety aspects are discussed in Chapter 6. Doses to man and the environment will vary according to the fuel composition, reactor system and operations. There are advantages in using thorium in thermal reactors in a once-through mode, in particular if U-233 or U-235 (HEU) is used as topping material. If the burnup of thorium fuel can be extended, the waste is less radiotoxic up to 20 000

years, and, perhaps more importantly, has a smaller volume compared with the use of enriched uranium. The closed Th-cycle offers better performance with respect to long-lived waste radiotoxicity compared with U/Pu fuels, in particular if fast reactors are used.

Thorium based fuel cycles (Th/U) produce much less plutonium and associated minor actinides than uranium based fuels. The key dose contributions of concern are the generation of Pa-231, Th-229 and U-233 and their daughters. The radioactivity of the waste from a thorium cycle appears, however, significantly less than for the standard U/Pu cycle, in the same conditions. This is an advantage of thorium based fuels and has been confirmed in several studies, recently in the EU supported study "*Thorium as a waste management option*" [138].

7.4 Safety and Accidents

Following the installation and operation of the first nuclear reactor in USA, there has been a series of events and some major accidents (International Nuclear Event Scale (INES) 6 or 7) associated with uranium fuelled reactors. Furthermore, a number of criticality accidents have occurred off-line, in association with fuel production or spent fuel handling. IAEA has registered all events associated with nuclear reactor accidents, while UNSCEAR has also summarized information on criticality accidents and the number of fatalities and injuries associated with such accidents [149].

Most of the criticality accidents occurred during the early years from 1940 to the 1960s. The latest accident occurred in Japan in the 1990s due to human failure. Similarly, most of the accidents associated with nuclear reactors occurred during the 1950s and 1960s in prototype reactors, military reactors, *Generation I* reactors and early *Generation II* reactors. Among all these cases, three serious reactor accidents have occurred. In 1957 a graphite fire in an early designed air-cooled graphite moderated metal uranium reactor at Windscale, UK. Two such reactors operated from 1951 to 1957, when both were shut down due to the fire. A serious accident occurred in 1979 at Harrisburg in one of the Three Mile Island reactors due to human failure. The reactor was a pressurized *Generation II* reactor. Due to the safety precautions such as containment, however, only traces of gases, mostly noble gases, were released to the environment. In 1986, the worst ever accident in a nuclear reactor occurred in Chernobyl (a graphite moderated reactor) due to human failure and the construction features of the reactor (positive void coefficient). The accident resulted in the release of tonnes of uranium fuel and fission products since this reactor type does not have a containment building. A few such reactors are still operating in Russia. Based on the report from the Chernobyl Forum (IAEA, WHO, FAO), summing up the consequences 20 years after the Chernobyl accident [150], $14 \cdot 10^{18}$ Bq was released, 340 000 people were evacuated, in total 62 people died as a result of the accident, either due to the fire or due to radiation induced diseases such as thyroid cancer (about 6000 children were sick, 15 died, and the remainder were cured), and the social and economic consequences were large. Radiation effects (dead forest) were observed in the environment during the first year following the accident. According to the Forum scientists, the biodiversity 20 years after the accident is richer than before the accident. They attribute this circumstance to the fact that 340 000 people were evacuated so that the environment is no longer affected by human activities. Although the long-term consequences still are to be seen, the health and environmental consequences directly linked to radiation exposures are so far significantly less severe than claimed by many organisations.

Following these accidents, significant improvement in reactor safety and handling of nuclear materials have been made, both technically and culturally, in particular for the *Generation III* reactors and in the planning of *Generation IV* reactors. The largest nuclear reactor in the world,

presently under construction in Finland, is a *Generation III+*. The containment is built to withstand a crash from a Jumbo Jet and a core melt down.

The ADS is a subcritical system where the effective neutron multiplication factor is smaller than one by construction. Therefore, the resulting safety aspect is a deterministic one. The system is and remains subcritical at all times and Chernobyl type accidents are impossible.

7.5 Radiation Protection Act

As thorium and its daughters are radioactive elements, and as radioactive elements are produced in a nuclear reactor or an ADS, the key regulation authority with respect to radiation protection of workers (occupational exposed group), the public and the environment is the Norwegian Radiation Protection Authority (NRPA). The NRPA is the national competent authority for radiation protection, nuclear safety and security as well as for the corresponding legislation on radiation protection and nuclear energy. The NRPA is a Directorate under the *Norwegian Ministry of Health and Care Services* and also a competent authority for the *Ministry of the Environment* and the *Ministry of Foreign Affairs*. Furthermore, the NRPA assists other Norwegian ministries concerning questions about radiation protection, safety and security. NRPA has therefore been asked to summarize the requirements set by today's legislation and public management with respect to radiation protection of man and the environment (Appendix D1).

The key Norwegian legislation that applies to radiation protection of man and the environment in Norway includes:

1. *Regulation relating to work with ionising radiation (14 June 1985),*
2. *Act on Radiation Protection and use of Radiation (Radiation Protection Act) (12 May 2000 No. 36),*
3. *Regulations on Radiation Protection and use of Radiation (Radiation Protection Regulations) (21 November 2003 No. 1362).*

Additional legislation which also will be highly relevant is described in Chapter 8.

The Radiation Protection Act and Use of Radiation and the Radiation Protection Regulations include:

1. *General requirements (justification and basic principles for use of radiation)*
2. *Qualifications and training*
3. *Risk assessment and emergency preparedness*
4. *Requirements of authorisation (Radiation Protection Regulations Section 5)*
5. *Occupational exposure to ionising radiation*
6. *Special provisions on discharges to the environment and on waste treatment of substances which emit ionising radiation*
7. *Inspection powers of the Norwegian Radiation Protection Authority*

Some of these key areas are summarized below.

General Requirements: Basic radiation protection principles, i.e. the ALARA (As Low As Reasonably Achievable) principle, which includes justification, optimization principles and regulated dose limits, are described. The principles are in accordance with international recommendations (ICRP).

Import of thorium is subject to the principle of justification (*Radiation Protection Act, Section 5*). The NRPA may under certain conditions prohibit such import according to Section 20 *Prohibition of import and sale*: *The Norwegian Radiation Protection Authority may refuse the import or sale of any product or substance and any item that may involve a risk to health or environment due to radiation, provided that this is not in conflict with international agreements to which Norway has acceded.*

Requirements of Authorisation

Activities which have the intention to procure, use or handle substances that emit ionising radiation need authorisation from the Norwegian Radiation Protection Authority, for instance activities related to:

- Discharges of radioactive substances
- Facilities for the treatment, storage or disposal of radioactive waste
- Import and export of radioactive waste

However, authorisation requirements for mining and milling thorium are not included in the current radiation protection regulatory system. Such a requirement will be considered in future revisions of the regulations.

Occupational Exposure to Ionising Radiation

The occupational exposures to ionizing radiation under Section 21: Dose Limits

All radiation exposure shall be kept as low as reasonably achievable, and the following dose limits shall not be exceeded:

a) The dose limit for workers over the age of 18 is 20 mSv per calendar year. The Norwegian Radiation Protection Authority may grant dispensation for individuals where the nature of the work makes it impracticable to set an annual limit of 20 mSv. In such cases permission may be given for a limit of 100 mSv over a continuous five-year period, on condition that the effective dose does not exceed 50 mSv in any single year.

The dose limits for occupational exposure are in accordance with international recommendations.

Regulations Regarding the Environment

Activities involving treatment, storage, waste and releases are regulated by the authorization under the *Radiation Protection Act*, Chapter V. Regulations, Section 23 provides requirements for discharges where the best available technology should be applied to avoid discharges to the environment or to keep the discharges to the lowest possible level. Section 24 regulates the use of countermeasures to counteract any damage or inconvenience resulting from contamination and the management of waste, where best available technology should be applied to generate minimal waste, based on an overall assessment of current and future uses of the environment.

Safety Measures and Emergency Preparedness

The role of the National Radiation Protection Authority with regard to safety measures and emergency preparedness is described in several sections of the *Act Concerning Nuclear Energy Activities*.

Norway has established an emergency organization ("*Kriseutvalget for atomulykker*") in case of a nuclear event in Norway or if Norwegian territories will be affected by such an event abroad.

Norway has also acceded to all the IAEA radiation protection, radiation safety and non-proliferation conventions. Any potential future nuclear power in Norway would, however, require additional legal regulation. Such regulations could be adapted from existing international guidelines and regulations, such as IAEA safety standards for:

- Radiation protection
- Nuclear safety
- Activities that can produce radiation producing substances (e.g., mining)
- Waste handling and storage/depository

In summary, when the whole thorium cycle, including mining and milling of thorium, is taken into account, inadequacies are seen in the current regulatory framework. The potential establishment of Norwegian nuclear energy will require a revision of the *Radiation Protection Act* and/or the *Radiation Protection Regulations* to ensure that the legislation fully complements the *Act Concerning Nuclear Energy Activities*.

8. REGULATION

The use of thorium as an energy source can be divided into several scenarios e.g.:

1. *Exploitation of thorium minerals in Norway, i.e. mining and milling.*
2. *Import of thorium to Norway, or export of thorium from Norway.*
3. *Establishment of a thorium based nuclear reactor in Norway.*
4. *Establishment of a thorium based Accelerator Driven System (ADS) in Norway.*

These scenarios will call for different national regulation regimes managed by different regulating bodies. As the Norwegian Radiation Protection Authority (NRPA) is the national competent authority for radiation protection, nuclear safety and security as well as for the corresponding legislation on radiation protection and nuclear energy, NRPA has been asked to summarize the requirements set by today's legislation for using thorium as an energy source in Norway (see Appendix D1). In addition to the legislation associated with radiation protection (Chapter 7), the key Norwegian legislation that applies to such nuclear activities in Norway includes:

- Act relating to mining (30. June 1972),
- Act relating to acquisition of waterfalls, mines and other real estate (Industrial Licensing Act) (14. December 1917),
- Planning and Building Act (14 June 1985 No. 77),
- Regulations on environmental impact assessment (1 April 2005 No. 276),
- Act concerning nuclear energy activities (12 May 1972),
- Regulations on the Physical Protection of Nuclear Material (2 November 1984),
- Regulations on Possession, Transfer and Transportation of Nuclear Material and Dual-use Equipment (12 May 2000),
- Act relating to the generation, conversion, transmission, trading, distribution and use of energy etc. (Energy Act),
- Act relating to protection against pollution and relating to waste (The Pollution Control Act, 13 March 1981 No. 6) with regulations,
- Act relating to working environment, working hours and employment protection, etc. (Working Environment Act, 17 June 2005 No. 62),
- Regulations relating to systematic health, environmental and safety activities in enterprises (Internal control regulations, 6 December 1996 No. 1127).

Furthermore, Norway has acceded to a series of relevant Conventions and International Agreements where the Convention on Nuclear Safety (1994), the Treaty on the Non-proliferation of Nuclear Weapons (1968), Joint Convention on the Safe Management of Spent Fuel and the Safe Management of Radioactive Waste (1997), Convention on the Physical Protection of Nuclear Materials and Nuclear Facilities (1980, amended 2005) and the Paris Convention on Nuclear Liability (1960) are most relevant (see Appendix D1).

8.1 Exploitation of Thorium Minerals in Norway: Mining and Milling of Thorium

Mining and milling of thorium will have to be considered under legislation on mining, such as the:

- Act relating to mining (relevant authority: Ministry of Industry and Trade).
- Industrial Licensing Act (relevant authority: Ministry of Petroleum and Energy).
- Planning and Building Act (relevant authority: Ministry of the Environment).

Furthermore, contamination of the environment may occur during mining and milling of thorium making the following act and regulation applicable:

- Pollution Control Act (relevant authority: Ministry of the Environment).
- Regulations on environmental impact assessment (relevant authority: Ministry of the Environment).

As thorium and its daughters are radioactive, the Radiation Protection Act, the Radiation Protection Regulations, the Regulation relating to work with ionising radiation and the Act relating to working environment, working hours and employment protection will be essential if a thorium based industry is established. However, authorisation requirements for mining and milling thorium are not included in the current radiation protection regulatory system.

8.2 Import of Thorium to Norway and Export of Thorium from Norway

Import and export of thorium will require authorisation according to the:

- Regulations on Possession, Transfer and Transportation of Nuclear Material and Dual-use Equipment (import: Section 3 and export: sections 11 and 12).

The import of thorium is also subject to the principle of justification (Radiation Protection Act Section 5), and according to the Act *"The Norwegian Radiation Protection Authority may refuse the import or sale of any product or substance and any item that may involve a risk to health or environment due to radiation, provided that this is not in conflict with international agreements to which Norway has acceded"*.

In the case where export authorization is given, the export is subject to notification to NRPA and the Ministry of Foreign Affairs.

8.3 Enrichment, Fuel Production, Operation and Waste Disposal

The *Norwegian Ministry of Petroleum and Energy* is responsible for nuclear power plants and the *Norwegian Ministry of Health and Care Services* is responsible for other nuclear installations. The relevant legislation for these activities will be:

- Planning and Building Act.
- Regulations on environmental impact assessment.
- Act concerning nuclear energy activities.
- Radiation Protection Act.
- Radiation Protection Regulations.
- Energy Act.
- Pollution Control Act.

The Act concerning Nuclear Energy Activities and associated regulations include:

- Licence for constructing, owning and operating nuclear installations, as well as permits
- Supervision by the National Radiation Protection Authority

- Safety, security and emergency preparedness
- Fees and dues
- Nuclear liability (compensation and insurance)
- Control of the peaceful utilisation of nuclear energy

The Act concerning Nuclear Energy Activities applies to the following:

" (a) nuclear fuel: fissile material in the form of uranium or plutonium in metallic form, alloy or chemical compound, and such other fissile material as the Ministry may determine;

(b) radioactive product: other radioactive material (including wastes) which is made or has become radioactive by irradiation accidental to the production or utilisation of nuclear fuel;

(c) nuclear substance: nuclear fuel, other than natural uranium and depleted uranium, as well as radioactive products, except radioisotopes used for industrial, commercial, agricultural, medical or scientific purposes or which are intended for, and are directly usable for such a purpose;

(d) nuclear reactor: a structure containing nuclear fuel in such an arrangement that a self-sustaining chain process of nuclear fission can occur therein without the addition of neutrons from another source;

(e) nuclear installation: nuclear reactor installation; factory for production or processing of nuclear substances, factory for the separation of isotopes of nuclear fuel, factory for reprocessing irradiated nuclear fuel, facility for the storage of nuclear substances other than facilities intended exclusively for use as temporary storage accidental to the transport of such substances, and such other facilities, in which there are nuclear fuel or radioactive products, as the Ministry may determine"

Furthermore, the requirements for licensing a nuclear reactor will include reactors using:

- uranium fuel and
- thorium based fuel, where uranium or plutonium is used in conjunction with thorium to maintain the nuclear chain reaction

Therefore, according to NRPA, a reactor based on using an accelerator and pure thorium fuel does not fall under the present legal term "nuclear reactor" as such a system would be dependent on an external source of accelerated protons or neutrons to maintain the nuclear chain reaction (see Appendix D1). The Act concerning Nuclear Energy Activities would therefore need to be amended in order to incorporate a thorium-based Accelerator Driven System (ADS).

The Energy Act includes requirements on licensing for installations for the generation, conversion, transmission and distribution of high voltage electrical energy. The process for applications for licences is described in Appendix D1. There is also a fee for licence application and a due for supervision of nuclear installations as specified in The Act concerning Nuclear Energy Activities, section 57.

8.4 The Establishment of a Thorium Based Nuclear Reactor in Norway

Licences and permits for constructing, owning and operating nuclear installations are regulated by the Act concerning Nuclear Energy Activities and associated regulations.

Activities related to Norwegian thorium-based nuclear energy will inter alia be regulated by the following legislation: The Planning and Building Act with regulations, the Act concerning nuclear energy activities with regulations, the Radiation Protection Act, The Energy Act and the Pollution

Control Act with regulations. A series of related regulations will also be of relevance, such as the Act relating to the generation, conversion, transmission, trading, distribution and use of energy etc. (Energy Act), the Act relating to working environment, working hours and employment protection, etc.

According to NRPA, if Norwegian nuclear energy is to be established, the Act concerning nuclear energy activities will have to be reviewed and there will be a need for more detailed regulations on inter alia nuclear security and internal control (See Appendix D1). Such regulations might be based on international requirements and recommendations. The Radiation Protection Act and/or regulations might also be subject to amendment, for example in order to establish a requirement that mining and milling of thorium is subject to authorization from NRPA (Appendix D1).

8.5 The Establishment of a Thorium Based ADS in Norway

According to NRPA, the *Act Concerning Nuclear Energy Activities* is directed towards nuclear reactors using uranium fuel and thorium based fuel, where uranium or plutonium is used in conjunction with thorium to maintain the nuclear chain reaction. A reactor based on using a mass accelerator and pure thorium fuel does not fall under the present legal term "nuclear reactor" as such a system would be dependent on an external source of accelerated protons or neutrons to maintain the nuclear chain reaction. Consequently, the definitions of the *Act Concerning Nuclear Energy Activities* would therefore have to be amended in order to incorporate a thorium-based Accelerator Driven System.

8.6 Safety Measures and Emergency Preparedness

Norway has established an emergency organisation ("*Kriseutvalget for atomulykker"*) in case nuclear events occur in Norway or events in other countries affect Norwegian territory. "*Kriseutvalget for atomulykker"* is chaired by NRPA, and a series of advisors from different ministries, institutions and organisations or institutions are associated with the organisation. Norway has also acceded to all the IAEA radiation protection, radiation safety and non-proliferation conventions. Any potential future nuclear energy in Norway would, however, require additional legal regulation. Such regulations could be adapted from existing international guidelines and regulations, such as IAEA safety standards[13] for:

- Radiation protection
- Nuclear safety
- Activities that can produce radiation emitting substances (e.g., mining)
- Waste handling and storage/depository

8.7 Nuclear Liability (Compensation and Insurance)

The Paris Convention with additional protocols is implemented in Chapter 3 of the *Act of Nuclear Energy Activities*.

8.8 Control of the Peaceful Utilisation of Nuclear Energy

Norway has entered an agreement on international safeguards:

[13] *EURATOM-directives will also be considered in this work, as well as Swedish and Finnish legislation on nuclear power*

- The Agreement between Norway and the Agency for the Application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons (1 March 1972), and
- Protocol additional to the Agreement between the Kingdom of Norway and the International Atomic Energy Agency for the Application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons (29 September 1999).

Section 51, Control of the peaceful utilisation of nuclear energy from the *Act of Nuclear Energy Power Activities*, is detailed in Appendix D1.

8.9 Conclusions

To conclude, activities related to Norwegian thorium-based nuclear energy will inter alia be regulated by the following legislation: the *Planning and Building Act*, the *Act Concerning Nuclear Energy Activities*, the *Radiation Protection Act*, the *Energy Act* and the *Pollution Control Act*, all with regulations. In addition, mining and milling of thorium will have to be considered under the *Act Relating to Mining* and the *Industrial Licensing Act*. The import of thorium to Norway and the export of thorium from Norway are regulated by *Regulations on Possession, Transfer and Transportation of Nuclear Material* and *Dual-use Equipment*. The main rule is that import or export of thorium requires an authorization.

However, there are needs for amendments or more detailed legislation with regard to nuclear energy based on thorium. A conventional thorium-based nuclear installation will most probably be covered by the current licensing requirement laid down in the *Act Concerning Nuclear Energy Activities*, whereas a thorium-based Accelerator Driven System (ADS) will not. Consequently, the definitions of the *Act Concerning Nuclear Energy Activities* will have to be amended in order to cover any form of thorium-based nuclear installation. If Norwegian nuclear energy is established, the whole *Act Concerning Nuclear Energy Activities* will have to be reviewed and there will be a need for more detailed regulations on inter alia nuclear security and internal control. Such regulations might be based on international requirements and recommendations. The *Radiation Protection Act and/or regulations* might also be subject to amendment, for example in order to establish a requirement that mining and milling of thorium is subject to authorization from the NRPA

9. NON-PROLIFERATION

There are currently 9 countries that are known to have nuclear weapons. These are USA, Russia (previously USSR), United Kingdom, France, China, India, Israel, Pakistan and North Korea. Two of these, Israel and North Korea, have no civil nuclear energy. In the other 7 countries the decisions to build weapons were made prior to the development of civil nuclear energy. One exception is Pakistan, where civil nuclear energy was started in 1972, before the weapons program of the 1980s. However, the fissile materials for the weapons were obtained from a separate enrichment industry that was unconnected with the civil nuclear energy program.

The prevention of nuclear war is of utmost importance. The security threat posed by the proliferation of nuclear weapons has lead to the establishment of the international nuclear non-proliferation regime which comprises a network of treaties, institutions and the safeguards inspections regime. The cornerstone is the *Treaty on the Non-proliferation of Nuclear Weapons (NPT)*, supported by International Atomic Energy Agency (IAEA) safeguards. The NPT aims at preventing the spread of nuclear weapons, advancing and eventually achieving nuclear disarmament and facilitating the peaceful use of nuclear energy. A total of 189 countries have joined the NPT; India, Pakistan and Israel have never joined and North-Korea claims to have withdrawn. The NPT has been and still is an important pillar of Norwegian foreign policy [151].

Today, civil nuclear energy is employed in 31 countries with 439 reactors all together and there are plans for another 200 reactors within the next 20 years. Nearly all of the plans are for countries which already have nuclear weapons or where the non-proliferation measures are well developed.

Nuclear fission bombs are manufactured either from uranium-235 (U-235) or from plutonium-239 (Pu-239). U-235 is produced by enriching natural uranium to over 90 wt% of U-235 either by diffusion or by centrifuge methods. Ordinary light-water reactors need enrichment up to 3 - 5wt% of U-235 for the fuel to be useable. The spent fuel reprocessing and recycling needed for civil nuclear energy may also lead to diversion of fissile material. Thus it is obviously necessary to have strict international control of the fuel cycle under the auspices of the IAEA in order to prevent any future transfer of fissile material to the weapons sector.

One project initiated by IAEA is the *International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO)*. The objectives of INPRO are to help to ensure that sustainable nuclear energy is available in the 21st century, bring together all interested IAEA member states to achieve innovations in reactors and fuel cycles with inherent safety, minimizing risks of proliferation, and create a process that involves relevant stakeholders as well as ongoing initiatives at the national and international level.

Nowadays, civil nuclear energy is developed very openly, often in international collaborations. One such collaboration is the *Generation IV International Forum (GIF)* where the ambition is to develop reactors without risks for diversion of fissile materials to the weapons sector. The breeder reactors considered in the GIF collaboration do not require enrichment facilities and the planned reprocessing is planned to take place on site and include several unseparated trans-uranium elements not suitable for bomb manufacture.

Several initiatives are currently being discussed for a stricter control of the fuel cycle. One of them is the *Global Nuclear Energy Partnership (GNEP)* initiative proposed by the United States.

Nations with secure nuclear capabilities (fuel cycle states) would provide fresh fuel and recover used fuel to nations that agree to use nuclear energy for peaceful purposes. If fast breeder reactors with a closed fuel cycle are developed and with natural uranium as a fuel, many of the perceived problems with present Light-Water Reactors can be avoided as far as proliferation is concerned. There has not been a single example of proliferation with Light Water Reactors. The elimination of uranium enrichment and recycled actinide (not pure Pu-239) fuel precludes unauthorized handling.

The military development of bombs has been based on fissile U-235 and Pu-239. With an emerging civil nuclear energy technology based on thorium it could not be excluded that also uranium-233 (U-233) produced from thorium could be used for the weapons sector. The fissile weapons quality is evaluated in terms of:

1. *The critical mass of an isotope (or different isotopic composition),*
2. *the weapon yield degradation due to the pre-initiation caused by spontaneous fission neutrons and*
3. *the weapon stability degradation caused by heat emission.*

U-233 has been determined to be at least as efficient as U-235 as a weapon material, e.g. the critical mass of U-233 is approximately 5 - 8 kg (with neutron reflector) [48], [148] and U-233 has to be diluted by uranium-238 (U-238) below a level of 12 wt% in order to reach the proliferation limit [148].

Reprocessing of thorium-based fuel yields almost pure U-233 and therefore weapon-grade material. E.g. for the equilibrium fuel cycle of a uranium-thorium fuelled light-water reactor the recycled uranium typically contains about 55 wt% U-233 and 10 wt% U-235, which is a fissile content sufficient for nuclear explosives [152]. However, traces of U-232 are always present in fissile U-233. In the reactor, thorium is irradiated by neutrons leading not only to the isotope U-233 but also to a minute fraction of U-232. The appearance of gamma emitting nuclides (e.g. Tl-208) in the decay chain after U-232 (life time 68.9 years) would make an extracted material highly radioactive. The contact dose rate of 30 kg of U-233, with the small content of U-232, would after a few years be about 72 Sv/hour, which corresponds to a 50 % lethal dose after 5 minutes exposure to the bare mass [153]. The fractional quantity of U-232 depends on the neutron spectrum and on the burn-up and can reach 5000 ppm [148]. The presence of U-232 will thus create a radiation hazard which is sufficiently large to require remote handling within a short time after chemical separation. This problem rules out the military use of U-233 by the Nuclear Weapons States – they appear to have enough of the more conventional weapons material. On the other hand, terrorist groups or rogue states might want fissile material for immediate use rather than for stockpiling. U-233 could be decontaminated from decay products for a few days [154] or even weeks [155] allowing the fabrication of a weapon.

In conclusion, it seems that there is consensus amongst the experts that thorium based fuel cycles do not produce (weapon-grade) plutonium. However, the proliferation resistance of U-233 depends on the reactor and reprocessing technologies. The judgements range from that the risk of nuclear proliferation is “negligible” [156] to “very small (Radkowski LWR), medium (gas-cooled high-temperature reactor) or small (EA)” [157] and to statements that “U-233 requires the same level of safeguards oversight and physical protection as does plutonium” [154]. It is obvious that the thorium fuel cycle in general has advantages concerning the proliferation resistance, advantages that can be exploited in the design of a reactor technology and a fuel cycle merely for civil purposes and without connections to the military sector. However, due to the lack of experience with industrial-scale thorium fuel cycle facilities we adopt the view that similar safeguard measures as for plutonium are mandatory.

10. ECONOMICAL ASPECTS

Due to the lack of data, it seems impractical to develop meaningful cost projections for any nuclear energy systems using thorium. Historical examples give some ideas of the funding that may be required. For example, in the 1970s, Germany spent around 500 millions euros in current money to develop a thorium fuel cycle and 2.5 billions euros for the high temperature reactor itself. More recently, the GIF in its technology roadmap to develop *Generation IV* advanced nuclear energy systems estimated at around one billion dollars the funds needed to assess the viability and the performance of one system before any decision to develop and build a demonstrator, which of course would require large additional funding.

However some insights may be given regarding thorium as raw material and a comparison between conventional reactors and Accelerator Driven System (ADS).

Regarding the raw material, ores or concentrates of oxide of thorium, there is a small market supplying companies processing thorium for non-energy uses such as high-temperature ceramics, crucibles, catalysts, welding electrodes and some specific alloys. However, the use of thorium in most of these products has continuously decreased due to the burden associated with its natural radioactivity, encouraging processors to switch to non radioactive materials where possible.

A nearly constant commercial price in the range 30 - 35 US$/kg was published in the 1960s and 1970s, and due to the decrease of demand this publication was suspended. Fluctuations in the price of thorium have been minimized by its by-product status and a supply that far exceeds demand. In practice, today, many countries have fairly large stockpiles of thorium that they consider as waste.

If thorium was to be considered as raw material for energy production in the future, a more substantial market would develop. The raw material contribution to the cost of the electricity generated will remain very low comparable with or lower than that for the uranium cycle.

There is little information available regarding the capital cost of an ADS and nothing referring specifically to a thorium fuelled ADS. In 1997, the Euratom Scientific and Technical Committee (STC), the highest level advisory committee for nuclear research in the European institutions, was requested by the European Commission to evaluate the proposals for a nuclear energy amplifier system for electricity generation. The STC expressed a strong opinion (ref. Report EUR17616 EN): The STC not only questioned the complexity of an ADS ("*The STC does not consider it is realistic to pursue development of the whole system at once" and "sees significant technological and commercial risks in almost every aspect of the proposals*"), but was also convinced that for electricity production an ADS "*would not be economically competitive with the improved light water reactor systems now under development, such as the EPR*". The STC concluded that further work on ADS should be primarily aimed at waste management (actinides burning) rather than at energy production.

A comparative study between ADS and Fast Reactors (FRs) in advanced nuclear fuel cycles was issued in 2002 by OECD/NEA [106]. This study was carried out by a group of more than 35 experts from 15 OECD countries including all the major players in the nuclear field and 3 international organizations. Although this study was primarily focused on fuel cycle schemes to reduce the long term radiotoxic inventory of waste, some economic aspects were looked at. The expert group considered that a possible cost reduction of a sub-critical reactor compared to a

conventional reactor (for instance with the possible elimination of control rods) may be offset by cost increases related to complications in containment and other systems. The basic construction cost for an ADS was therefore set equal to that for a FR with an addition to cover capital costs of accelerator and target. The study also concluded that "*Fuel cycle schemes that involve the use of the more expensive ADS technology show an overall economic benefit by burning as much of the plutonium as possible in less expensive more conventional systems, i.e. MOX-LWRs and MOX-FRs*".

Since then, experts participating in different OECD/NEA standing technical committees or working groups have regularly concluded that energy production with an ADS cannot compete economically with critical reactor technology.

From an economic point of view, the complexity of ADS technology and the inescapable problems of technological development and definition of the safety features cast doubt on the particularly low future costs estimated by its promoters; about 30 % lower for a unit in the series already planned by the CERN group (Carlo Rubbia, 1996 [158]) compared with the costs (in USc/kWh) for a serial PWR in France announced in 1996 by Fernandès et al. Until successive units have actually been built and safety standards have been met, any estimate of this kind is bound to call to mind the extremely optimistic estimates of the investment cost of the nuclear kWh made in the 1960s (3 to 400 $/kW current value compared with 2500 $/kW effective cost) which widely underestimated technological problems. The extra costs involved in the addition of an accelerator compared with a present reactor should be compensated by the possible advantages of the potential simplicity of the subcritical reactor and a simpler fuel cycle. The present state of knowledge and technological know-how on the concepts to be explored is still insufficient. A certain number of projects are at the laboratory or the small pilot stage. None can provide a standard in terms of demonstration equipment. Complete industrial proficiency with respect to the new technological system will necessitate skills in numerous techniques and a long operational period will be necessary to demonstrate the feasibility of multi-recycling, something which has so far never been achieved for any nuclear reactor. At the same time, regulations to guaranty safety standards comparable to those of light water reactor technology by the protype stage must be gradually determined and will require often costly technological adaptation.

Altogether, the main challenges to develop a thorium based energy production might be the mobilization of the funding necessary to carry out the needed research and development for both the energy production technology and the associated fuel cycle. This research and development can not be done by Norway alone and should be seen in a European context. What concerns thorium-based energy production, one of the projects of the *Generation IV International Forum* would be well suited and for the back-end of the thorium fuel cycle, a European XT-ADS experiment leading to a demonstration of an ADS would be appropriate. The financial contributions to these projects would depend on the desired Norwegian involvements.

11. RESEARCH, DEVELOPMENT, EDUCATION AND TRAINING

Nuclear sciences cover all use of nuclear methods and ionizing radiation. Nuclear research and education provides knowledge as well as skilled workers for many important sectors in society. In 2000 the OECD Nuclear Energy Agency produced a report: *Nuclear Education and Training: Cause for Concern?* This document was compiled using information supplied by 200 organisations in 16 member countries. The agency demonstrated that it was possible that many nations were training too few scientists to meet the needs of their current and future nuclear industries. In addition, a number of studies over the past five years, by different European governments, have also identified the same lack of scientific personnel. This has been attributed to a decreased student interest, decreased course numbers, aging faculty members and aging facilities. Consequently, the European education skill base has become fragmented to a point where universities in most countries lack sufficient staff and equipment to provide education in all, but a few, nuclear areas.

Norway has also lost most of the specialists in nuclear energy technology after the nuclear moratorium more than 25 years ago.

Of particular concern appeared to be special skill-base deficits within nuclear radiological protection, radioecology and radiochemistry as well as technical reactor engineering fields at masters and doctorate levels (EURAC, 2007. ENEN-II). Skills in these areas are required not only to deal with currently installed nuclear capacity and decommissioned facilities, but also to meet the needs presented by likely new-build nuclear capacity. As recently stated by several EU politicians and experts, there are increasing pressures to build new nuclear power stations in many EU member nations. This pressure comes from the need to meet Kyoto greenhouse gas emission targets at a time when many currently installed, CO_2-clean, nuclear power stations are coming to the end of their useful lives. They also come from the decreasing stocks of domestic fossil fuels, with an increasing reliance upon politically unstable nations for the provision of oil and gas and from the increasing prices of domestic and imported fuels. Finally, the pressures are facilitated by new improved reactor systems that are being developed in Europe and the USA. Therefore, the need for nuclear competence is probably greater now than was earlier anticipated.

11.1 Master and PhD Education in Nuclear Sciences in Norway

Nuclear energy technology is an interdisciplinary field, including:

1. *Nuclear physics and chemistry, relevant for reactor physics and for transmutation.*
2. *Material science, materials under extreme radiation exposure, stress and temperatures.*
3. *Radiochemistry*
4. *Radiation protection, including dosimetry.*
5. *Thermodynamics – engineering approach.*
6. *Modelling and simulation – applied computer science.*
7. *Reactor theory, reactor stability and control.*
8. *Reactor safety – risk analysis.*
9. *Power engineering – modern turbine technology.*
10. *Waste management.*

11.Radioecology

12.Decommissioning

ADS requires in addition a very comprehensive field:

13.Accelerator technology.

The abbreviations below will be used in the following:

IFE = *Institute for Energy Technology (Kjeller and Halden)*
NTNU = *Norwegian university of Science and Technology (Trondheim)*
UiB = *University of Bergen*
UiO = *University of Oslo*
UMB = *Norwegian University of Life Sciences (Ås)*
UNIK = *University studies at Kjeller*

There are currently 4 universities in Norway providing full Master and PhD programs within nuclear sciences:

1. *UiO: Master in Nuclear Chemistry, Master in Nuclear Physics, Master in Material Science and PhD within the same areas.*
2. *NTNU: Master and PhD in Material Science.*
3. *UiB: Master and PhD in Nuclear Physics.*
4. *UMB: Master in Radiochemistry, EU Master in Radioecology and PhD within the same areas.*

Additional courses within nuclear sciences are given at universities and other institutions. There is already a basis for activities and competence in Norway for relevant research, education and training:

1. *Reactor used in operation training (IFE, UNIK).*
2. *Safety and reliability (man – machine) (IFE Halden).*
3. *Courses in multi component transport (from the oil industry) (NTNU, UiO).*
4. *Electrical power generation and turbine technology (NTNU).*
5. *Energy production and use (IFE, UiB, UiO, UMB).*
6. *Radiation protection (IFE, NTNU, UiB, UiO, UMB).*

11.2 Norwegian Competence in Nuclear Energy Technology

The first reactor in Norway was started at Kjeller in 1951, and made Norway a pioneering nation in nuclear technology. This is not so today, despite the high international status of the research reactor in Halden. Today *IFE* operates two old research reactors. Both are heavy water moderated and cooled. The thermal power of the Halden reactor is 20 MW, the Kjeller reactor thermal power is 2 MW.

The nuclear engineering development at *IFA* (now *IFE*) focused especially on nuclear reactors for ship propulsion and the development of computer codes for calculation of the fuel cycle and power distribution of power reactors. The computer codes were used in Germany, Switzerland, Sweden, Spain and USA.

This development and marketing was taken over in the 1970s by a spin off company, *Scandpower*. The nuclear part of *Scandpower* was bought by *Studsvik* in 1998, and the name was changed to

Studsvik-Scandpower. The company's software is still a world leader for reactor calculations. Customers include power companies and research institutes worldwide.

IFE is still the only institution that maintains the national competence in reactor technology. The mandate is:

"*IFE shall, on an idealistic and socially beneficially basis, do nuclear technology R&D and provide information on nuclear energy. IFE shall do basic research based on the Jeep II-reactor at Kjeller, and take care of important national duties within radiation pharmacy, the development and use of nuclear methods, radiation protection and radioactive waste.*

IFE shall also maintain a national competence in reactor technology and safety, based on the international Halden Reactor Project performed at the Halden Boiling Water Reactor (HBWR). A broad spectrum of fuel- and material examinations under realistic conditions are performed at the Halden Reactor. This provides an important data base for safety evaluations and the reliable operation of nuclear power stations and other complex industrial facilities."

Research, education and the maintenance of nuclear knowledge in Norway is important and somewhat independent of future national nuclear energy production. There is a need for more specialists today, and an introduction of nuclear energy will demand a considerable lager number of properly educated personnel.

11.3 Norwegian Supplier Industry

There is no industrial supplier of nuclear energy technology in Norway, but several companies have relevant competence.

Aker Kværner has expertise and experience in waste handling and storage, as well as in decommissioning of nuclear facilities. Those divisions are located in Great Britain.

The company *Thor Energy* is considering building nuclear power plants utilizing thorium. The plan is to buy standard power plants and to develop a new fuel cycle based on thorium.

11.4 Relevant Norwegian Research and Development

The only R&D environment for nuclear power in Norway is at *IFE*, primarily through the activity at the Halden Reactor. The activity there is highly esteemed internationally, both regarding the fuel research at the reactor and the comprehensive research on the interplay between humans, technology and organization. *IFE* has extensive experience in fuel testing, including thorium-containing fuel.

Studsvik-Scandpower and *Scandpower Risk Management* are capable of contributing to calculations of reactor core reactivity and power distribution as well as risk analysis.

Selskapet for INdustriell og TEknisk Forskning (SINTEF) may contribute with knowledge in the fields of advanced materials and high-temperature corrosion, which have been actualized because of the *Generation IV* reactor project.

The use of nuclear methods is used in many branches of science at the Norwegian universities. Several Norwegian university physicists and chemists do research and use methods relevant for reactor technology.

UiO has Norway's only research accelerator, a MC-35 Scanditronix cyclotron. The nuclear physics and -chemistry research at the cyclotron is organized in a centre for accelerator based research

and energy (SAFE). The cyclotron is used in various fields of research and in isotope production for medical use in collaboration with IFE and the national hospital (Rikshospitalet HF). The main nuclear research projects, including international collaborating groups, are methods and equipment to measure data that can be used to measure nuclear reaction cross sections for neutrons and for transmutation reactions. Recently a formal cooperation with IFE has been initiated, centred on these types of projects.

The research groups for material sciences at NTNU and UiO have both equipment and competence to perform relevant material studies for nuclear systems and safety.

In general, Norway today has few advantages in the field of nuclear energy, and is totally dependent on international cooperation. Cooperation within the *OECD Halden Reactor Project* is a possible starting point, but it is critically important to have other formal contacts with the international society.

Calculations and tests indicate that the Canadian heavy water reactor CANDU is close to producing as much U-233 from thorium as it consumes. Today somewhat more than 40 heavy water power reactors are operating in the world, of which about twenty are located in Canada. It is reasonable to assume that the competence within *IFE, Halden* could be used for relevant tests and model calculations in a possible further development in this area. Development of a reactor and a fuel cycle where the production of U-233 is at least as high as the consumption is a prerequisite to use thorium efficiently in a reactor. Regardless of which reactor type is chosen, it will be necessary to reprocess the spent fuel to achieve this goal.

Despite the fact that Norway has no commercial nuclear power plant, there is much competence in Norway concerning safety. This is partly due to the safety philosophy that has characterized the constructions in the North Sea, and the experience gained from accidents in the area. Furthermore, the *OECD Halden Reactor Project* is an important actor in the field of nuclear safety, both through the ongoing fuel research at the reactor, and through studies of the interplay between humans and technology in advanced laboratories where the effect of different systems and operator aids in simulated accident scenarios can be evaluated. The usefulness of the work within the Halden Reactor Project can be seen at Kola, where the safety of the Russian reactors has been significantly improved with the assistance of the Project.

Det Norske Veritas (DNV) and *Scandpower Risk Management* have relevant know-how in the fields of risk analysis, safety improvements and safety culture.

Research within nuclear chemistry, radiochemistry and radiation protection is performed by several groups at NTNU, UiO and UMB. This research includes measurements and analysis of background radiation from natural and from man made sources; the use of radiation in several fields; and basic research on the interaction of ionizing radiation with biological systems on the molecular level. The latter is also related to medical use of radiation and has an important impact on the advancement of dosimetry and safety regulations.

In addition, radioecology is a key research field at UMB, where environmental impact and risks associated with manmade and naturally occurring radionuclides are assessed.

11.5 Education in Nuclear Technology

The renewed interest in nuclear energy has promoted important efforts in nuclear education in many countries. This means that Norwegian students can obtain suitable education at several locations.

The Norwegian universities, in close cooperation with IFE, are able to organize and provide courses in relevant nuclear physics, reactor theory, material science, radiation risk / protection and in reactor operation. A national strategy project between IFE, UiB and UiO has recently been established, to secure and re-establish a national competence in nuclear technology.

EU and OECD have identified an urgent requirement for university trained candidates within nuclear sciences, including radioecology, in Europe and worldwide. Supported by EU's 6th FP ENEN-II project, a comprehensive joint international master program in radioecology is now offered at UMB.

International agreements and collaborations will be a necessity, and cooperation with several countries is possible. Language is a practical problem; here we have focused on countries and courses using the Swedish or English languages. Cooperation's, particularly with Finland and Sweden, should be established for the advanced and more specific nuclear reactor courses.

11.5.1 Finland

Finland's new fifth nuclear reactor has initiated renewed efforts in the education of nuclear personnel. The University of Oslo has already established collaboration with *Åbo Akademi*, where plans have been made for a series of relevant courses. See http://www.abo.fi/fak/mnf/fysik/teknfysik.htm for the information for students and the Appendix C: Education and Training for a list of courses.

11.5.2 Sweden

In Sweden academic activities take place at the following institutions

Chalmers University of Technology:

- Nuclear Engineering:
- Core Physics and Diagnostics
- Power Plant Safety and Technology
- Nuclear Measurement Techniques
- Nuclear Chemistry

Royal Institute of Technology:

- Reactor Physics
- Nuclear Power Safety
- Reactor Technology
- Nuclear Chemistry

Uppsala University Faculty of Science and Technology:

- Applied Nuclear Physics
- Neutron Physics
- Nuclear and Particle Physics
- Accelerator Physics

About the research of interest for the *Thorium Report Committee*, the Nuclear Engineering group at *Chalmers University of Technology* has lately studied in detail reactor physics aspects on using

thorium fuel in conventional BWR and heavy water cooled reactors. The work is planned to continue focused on a special design of thorium fuel for BWR. As part of the *Generation IV* research, a study on thorium driven Molten Salt Reactors is being initiated. Projects on thorium are planned also in the Nuclear Chemistry group.

The Reactor Physics group in Stockholm participates in the *Generation IV* activity and projects of interest are on ADS design, Monte Carlo Burn-up and Nitride Fuel.

In *Uppsala*, research is carried out on fuel and core diagnostics, back-end issues, new reactor concepts and safeguards and on geological environments for spent fuel. Measurements of neutron reaction cross sections carried out at the *Svedberg Laboratory* are of interest for any future ADS design.

The *Swedish Centre for Nuclear Technology (SKC)*, sponsored by the nuclear industry and the *Nuclear Power Inspectorate (SKI)* in Sweden, contributes towards maintaining and developing competence in nuclear technology in Sweden. The main activity of the SKC is the initiation and funding of PhD research projects at technical universities in Sweden. Another important activity is direct support to professors and lecturers at university departments for nuclear power technology and reactor physics.

Several universities in Sweden give courses in subjects related to nuclear science and technology. There are centres for Nuclear Technology both at the Royal Institute of Technology (CEKERT) and at Chalmers (CKTC).

SKC is also engaged in international cooperation on university education in nuclear technology through the World Nuclear University (WNU).

Besides the Academic research, The *Svensk Kärnbränslehantering AB* (SKB) runs a research laboratory in Oskarshamn, Äspö Hard Rock Laboratory, where research on long-term storage of radwaste is carried out. In Uppsala, the cyclotron of the Svedberg Laboratory provides neutrons and protons up to an energy of 200 MeV.

Studsvik, the previous centre for nuclear science and technology, now is a private company that provides qualified services on Waste Treatment, Decommissioning, Operating Efficiency and Service and Maintenance.

The Kärnkraftsäkerhet och Utbildning AB (KSU) is a private company owned by Vattenfall AB. It provides advice, development support and education for a safe running of the power reactors. KSU has currently 200 employees and has its head office in *Studsvik*.

More details are given in the Appendix C: Education and Training.

11.5.3 UK

NTEC is a newly founded consortium of universities and other institutions in the UK, providing postgraduate education in Nuclear Science and Technology.

NTEC offers part-time and full-time postgraduate courses in Nuclear Science and Technology. An extremely wide range of topics is available, from Reactor Physics to Nuclear Waste Disposal. This breadth of study is made possible by drawing upon the research and teaching expertise of the consortium partners, who together represent more than 90 % of the nuclear postgraduate teaching expertise that resides in the UK's universities and research institutes.

Individual subjects are presented in 'short course' modular format, providing excellent access to the program for engineers and managers in full-time employment who wish to advance their skill and knowledge base (Appendix C: Education and Training).

11.5.4 Canada

Canada has decided to refurbish four nuclear units. A newly founded university, the University of Ontario Institute of Technology (UOIT), will play an important role in nuclear research and education. (Prof. Rick Holt, Queen's University, private communication.)

The University Network of Excellence in Nuclear Engineering (UNENE) is an alliance of universities, nuclear power utilities, research and regulatory agencies for the support and development of post graduate nuclear education, research and development capability in Canadian universities. UNENE was established as a not-for-profit corporation by the Government of Canada.

With much encouragement from Ontario Power Generation (part of which used to be Ontario Hydro) and possibly some funding, UOIT has developed an undergraduate program in Nuclear Engineering, with 2007 as the first graduating year. Relevant course details are described in: https://connect.uoit.ca/uoit/program.do?from=subject&programID=47.

Professor Bill Garland has a very instructive website providing information and multimedia lectures for students and others interested in Nuclear Engineering as it relates to the program in the Department of Engineering Physics, McMaster University: http://www.nuceng.ca/index.htm

11.5.5 European collaboration

During recent years, international cooperation in nuclear education and training has contributed to revitalize the field, and new national and international organizations have been established. The "*Nuclear European Platform for Training and University Organizations (NEPTUNO)*" and the predecessor "*European Nuclear Education Network (ENEN)*" as well as the EU 6th FP projects *EURAC* and *ENEN-II*, in which Norway participates, have prepared future European nuclear education schemes, degrees and requirements.

11.5.5.1 The NEPTUNO project

The "*Nuclear European Platform for Training and University Organisations"* (NEPTUNO) builds on the achievements of the 5th European Framework Program which led to the establishment of the ENEN Association. The NEPTUNO project enhances further the harmonization of professional accreditation criteria and the associated training programs across the European Union.

The "International Seminar on the Nuclear Fuel Cycle" is a pilot training course planned for this purpose. The expected result is:

- An operational network of institutions for academic education at the Master, doctoral and postdoctoral level complemented with research organizations, regulatory bodies and industrial partners supporting research and development, bench-training and continual learning schemes.

The project is carried out under the coordination of the *French National Institute for Nuclear Sciences and Technology (INSTN)* by a consortium of 35 partners, including 25 universities and

10 research institutes or private companies from 19 countries. Twenty-six partners are also members of ENEN.

The aim of this project is to better integrate European education and training in nuclear engineering and safety to combat the decline in both student numbers and teaching establishments, thus providing the necessary competence and expertise for the continued safe use of nuclear energy and other uses of radiation in industry and medicine. The project focuses on a harmonized approach to education and training in nuclear engineering in Europe, and its implementation, including the better integration of national (governmental as well as industrial) resources and capabilities.

The expected result is an operational network for training and life-long learning schemes as well as on academic education at the master, doctoral and post-doctoral level, underpinning:

- Sustainability of Europe's excellence in nuclear technology, thereby contributing to the creation of a European Nuclear Knowledge Management Strategy
- Preservation of competence and expertise for the continued safe use of nuclear energy and other uses of radiation in industry and medicine
- Harmonized approaches to safety and best practices, both operational and regulatory, at European level within and across all Member States
- Harmonized approach for training and education in nuclear engineering.

In the network:

- The roadmap for nuclear education in Europe as developed and demonstrated in the Euratom FP5 project ENEN is implemented.
- The end-user relevance of the education at all levels by recruiting (part-time) professors out of industry and by providing (re-)training of nuclear industry personnel is warranted.
- Advanced courses, preferably at selected centres of excellence, are given:
- Bridging leading edge research and new knowledge generation with teaching and education.
- Creating nuclei of excellence for doctoral schools in nuclear engineering and sciences.
- Transnational access to research infrastructure, owned by governmental as well as industrial organizations, is facilitated.

11.5.5.2 The ENEN project

In the strategic goal for the European Union, the Lisbon EU 2000 summit meeting says: "*Although the number of nuclear scientists and technologists may appear to be sufficient today in some countries, there are indicators that future expertise is at risk. In most countries, there are now fewer comprehensive, high quality nuclear technology programs at universities than before. The ability of universities to attract top quality students, meet future staffing requirements of the nuclear industry, and conduct leading-edge research is becoming seriously compromised.*"

The "*European Nuclear Engineering Network*" (ENEN) project was launched under the 5th framework EC program in January 2002. It established the basis for conserving nuclear knowledge and expertise, created a European High Education Area for nuclear disciplines, and initiated the implementation of the Bologna declaration in nuclear disciplines.

ENEN was established afterwards on the basis of the European High Education Area by the partners of the ENEN and given a more permanent character and a legal status by the foundation of the ENEN Association, a non-profit international organization.

ENEN's mission is the preservation and the further development of higher nuclear education and expertise. The general goals of the ENEN Association are defined as follows:

With respect to the Academia:

- To develop a more harmonized approach for education in the nuclear sciences and nuclear engineering in Europe;
- To integrate European education and training in nuclear safety and radiation protection; and
- To achieve better co-operation and sharing of academic resources and capabilities at the national and international level.

With respect to the End-Users, such as nuclear industries, research centres, regulatory bodies and nuclear applications:

- To create a secure basis of skills and knowledge of value to the EU.
- To maintain an adequate supply of qualified human resources for design, construction, operation and maintenance of nuclear infrastructures, industries and power plants.
- To maintain the necessary competence and expertise for the continued safe use of nuclear energy and applications of radiation and nuclear techniques in agriculture, industry and medicine.

Within the framework of ENEN, a three week course is organized by four universities in the Central European Region (Bratislava, Budapest, Prague and Vienna), called “The Eugene Wigner course”. The Joint Research Centre Petten and the IAEA also support the course. The main organizer of the course is the Institute of Nuclear Techniques of the Budapest University of Technology and Economics.

The main emphasis of the course is to perform reactor physics experiments to enhance research reactor safety on three different research- and training reactors in three different cities (Vienna, Prague and Budapest). The experimental work is preceded by theoretical lectures aiming to prepare the students for the experiments (Bratislava). The students' work will be evaluated, and upon success the students will get a certificate.

11.5.5.3 The EURAC project

The Eu-funded EURAC project is a Coordinated Action of the 6th FP whose role is to strengthen in Europe the scientific academic competence and analytical skills within radioprotection, radiochemistry and radioecology and to secure the future recruitment of appropriately skilled post-graduates to meet the needs of European stakeholders. Recommendations from the EURAC project is currently followed up in collaboration with the ENEN association (EU funded ENEN-II project) to initiated Master education within nuclear sciences for the benefit of European students.

11.5.6 World Nuclear University (WNU)

“I am wholly in favour of the World Nuclear University. We are at the point where there is no sensible alternative to nuclear power if we are to sustain civilization. Obviously to replace the present use of fossil fuel with nuclear energy is a vast undertaking, and we will need a great number of trained engineers and scientists. The University would therefore have to come first."

James Lovelock

WNU was inaugurated in 2003 as a global partnership committed to enhancing international education and leadership in the applications of nuclear science and technology. The central elements of the WNU partnership are the global organizations of the nuclear industry WNA and WANO, the inter-governmental nuclear agencies IAEA and OECD-NEA and the leading institutions of nuclear learning in some thirty countries.

Within the UN system, the WNU is recognized as a "Partnership for Sustainable Development" by the UN Commission on Sustainable Development (CSD). WNU pursues its educational and leadership-building mission through programs organized by the WNU Coordinating Centre (WNUCC) in London.

The prospect of a steady worldwide growth in the use of nuclear technology – for power generation and in a diversity of sophisticated applications in medicine, agriculture, and industry – points to the need for a greatly expanded global cadre of nuclear professionals in the 21 century. The role of the WNU partnership is to support this growth.

After the summer 2007, USA, France, Russia, South Korea and Canada had seconded staff to the WNUCC. Secondment commitments have been received from India and the UK, and discussions are underway with governments and leading nuclear enterprises in Japan and China. An attractive concept, unfulfilled, is the placement on the WNUCC secretariat of regionally-supported representatives from Latin American and Africa.

12. CONCLUSIONS AND RECOMMENDATIONS

Energy Demand and Consumption: There is a prevailing belief that the current model for the world's energy policy is not sustainable. The major reasons are well known: rising greenhouse gas emissions and their negative impact on the climate, as well as concerns regarding the security of energy supply at affordable prices in the context of increasing needs of energy, notably in developing countries that are enjoying rapid economic growth.

Recommendation 1: No technology should be idolized or demonized. All carbon-dioxide (CO_2) emission free energy production technologies should be considered. The potential contribution of nuclear energy to a sustainable energy future should be recognized.

Resources: According to US Geological Survey (2007), Norway is known to have the third to sixth largest thorium resources in the world. These resources, i.e. 170 000 tonnes, have a potential energy content which is about 100 times larger than all the oil extracted by Norway to date as well as the remaining reserves, 4 250 million m^3. The information on thorium resources in Norway is, however, based on investigations carried out some 25 to 60 years ago, and no specific thorium exploitation has ever been carried out.

Recommendation 2: Investigation of the resources in the Fen Complex and other sites in Norway should be performed. It is essential to assess whether thorium in Norwegian rocks can be defined as an economical asset for the benefit of future generations. Furthermore, the application of new technologies for the extraction of thorium from the available mineral sources should be studied.

Thorium Fuel: In the 1960s and 70s, the development of thorium fuel for nuclear energy was of great interest worldwide. It was shown that thorium could be used practically in any type of existing reactor. Thorium fuel production and the technical feasibility of the use of thorium in conventional reactors were also demonstrated. Thorium fuel has been tested in the Halden Reactor on several occasions. Due to the worldwide focus on uranium, modern technologies such as automated fuel processing have not been tested on thorium.

Recommendation 3: Testing of thorium fuel in the Halden Reactor should be encouraged, taking benefit of the well recognized nuclear fuel competence in Halden.

Reactor Technology: Today, most of the reactors in operation are of the *Generation II* type, while new constructions will be based on *Generation III* and *III+*, which are significantly improved with respect to safety, security and economics. The next generation reactors, *Generation IV*, which are currently being developed, are expected to be commercially available in 25 – 30 years. Among the *Generation IV* reactors, the high temperature reactors, fast breeder reactors and molten salt reactors are most suitable for the use of thorium. Within the GIF (Generation IV International Forum), the use of thorium is only explicitly considered in Molten Salt Reactors. However, this concept is at present not prioritized.

Recommendation 4: Norway should strengthen its international collaboration by joining the *Euratom fission program* and the GIF program on *Generation IV* reactors suitable for the use of thorium.

Accelerator Driven System (ADS) Concept: The ADS concept has been generically developed since 1990, but the construction of a prototype has not yet been launched. An ADS fuelled with

thorium has some clear advantages compared with currently operating reactors; much smaller production of long-lived actinides, minimal probability of a runaway reactor and efficient burning of minor actinides. The lack of experience in operating such a complex system is a major drawback. However, the expected development within the on-going EUROTRANS project should provide information about the feasibility of the ADS concept. It is commonly agreed by OECD countries that energy production with an ADS cannot compete economically with current reactor technology.

Recommendation 5: The development of an ADS using thorium is out of the scope of the Norwegian capability alone. Joining the European effort in that field should be considered. Norwegian research groups should be encouraged to participate in relevant international projects, although these are for the time being focusing on waste management.

Radioactive Waste from the Front end of the Thorium Fuel Cycle: The dose burden of waste arising from mining and extraction of thorium is significantly smaller than that from uranium, due to the short half-life ($T_{1/2}$) of Rn-220 ($T_{1/2}$ = 56 sec) compared with the half-life of Rn-222 ($T_{1/2}$ = 3.8 days) from U-238 decay chain.

Radioactive Waste from the Back End of the Thorium Fuel Cycle: In contrast to the U-Pu fuel cycle, plutonium and other transuranics are not produced in a pure Th-232/U-233 cycle. The radiotoxic inventory of the waste from the Th-U cycle is significantly lower than that of a U-Pu cycle under the same conditions during the first 1000 years. Already after 100 years, the radiotoxic inventory of the Th-U cycle is significantly lower than that of natural uranium used in an open-cycle for the same amount of energy.

Recommendation 6: Norway should bring its competence with respect to waste management to an international standard, and collaboration with Sweden and Finland could be beneficial.

Radiation Protection of Man and the Environment: Compared to the uranium cycle the radiation protection associated with the thorium cycle is in general of less concern. This is especially so for the back end of the ADS. The competence to assess doses and impact to man and the environment from the thorium cycle in Norway is, however, limited. Already today, the high outdoor gamma doses for instance in the *Fen Complex* call for restrictions in use of the area. Doses to man and the environment from future potential exposures associated with the thorium fuel cycle will be regulated by the *Radiation Protection Act and associated Regulations*. However, authorisation requirements for mining and milling thorium are not included in the current radiation protection regulatory system and revision of the *Act* will be needed.

Recommendation 7: Norway should bring its competence with respect to dose assessment related to the thorium cycle to an international standard.

Regulation: The *Act Concerning Nuclear Energy Activities* from 1972 regulates activities associated with the existing Norwegian research reactors. A conventional thorium-uranium based nuclear installation will most probably be covered by the current licensing requirement, whereas a pure thorium-based system such as an Accelerator Driven System (ADS) will not. In such a case, the *Act Concerning Nuclear Energy Activities* will require revision.

Non-proliferation: The Th-232/U-233 fuel cycles do not produce plutonium. The proliferation resistance to U-233 depends on the reactor and reprocessing technologies. In the development of a reactor technology with its fuel cycle for civil purposes, the thorium fuel cycle should have an advantage concerning the proliferation resistance that can be exploited. However, due to the lack

of experience with industrial-scale thorium fuel cycle facilities we adopt the view that similar safeguard measures as for plutonium are mandatory until otherwise documented.

Recommendation 8: Since the proliferation resistance of uranium-233 (U-233) depends on the reactor and reprocessing technologies, this aspect should be of key concern if any thorium reactor is built in Norway.

Educational and Competence needs: Several studies (e.g. EU, OECD/NEA) have identified the problem that an insufficient number of scientists are being trained to meet the needs of the current and future European nuclear industries. This has been attributed to decreased student interest, decreased course numbers, aging faculty members and aging facilities. Norway has also lost most of its specialists in nuclear sciences after the nuclear moratorium more than 25 years ago. The European education skill base has become fragmented to a point where universities in most countries lack sufficient staff and equipment to provide education in all, but a few, nuclear areas. Of particular concern are special skill-base deficits within technical reactor engineering fields, basic and applied nuclear sciences.

Recommendation 9: Any new nuclear activities in Norway, e.g. thorium fuel cycles, would need strong international pooling of human resources, and in the case of thorium strong long-term commitment of the education and basic science side. All these should be included in the country level strategy aiming to develop new sustainable energy sources. However, to meet the challenge related to the new nuclear era in Europe, Norway should secure its competence within nuclear sciences and nuclear engineering fields. This includes additional permanent staff at the Universities and research institutes and appropriate funding for new research and development as well as high quality research-based Master and PhD education.

Concluding Remarks: The *Thorium Report Committee* finds that the current knowledge of thorium based energy generation and the geology is not solid enough to provide a final assessment regarding the potential value for Norway of a thorium based system for long term energy production. The *Committee* recommends that the thorium option be kept open in so far it represents an interesting complement to the uranium option to strengthen the sustainability of nuclear energy.

13. APPENDIX A: INTRODUCTION TO NUCLEAR ENERGY

The main elements in a thermal nuclear reactor core are fuel, moderator, coolant and control rods. The fuel is mainly uranium oxide (UO_2) or mixed oxide (MOX) which consists of depleted uranium mixed with fissile plutonium.

Natural uranium consists of 99.3wt% (weight percent) U-238 and 0.7wt% of the isotope U-235. Fissions mainly occur in the fissile isotope U-235; hence the fuel is enriched by increasing the concentration of fissile material. In the enrichment process the fissile isotope U-235 is extracted from natural uranium, which then becomes depleted uranium containing about 0.2wt% U-235. In commercial reactors the fuel enrichment is around 3 - 5wt% of U-235.

During irradiation in a nuclear reactor some of the U-238 in the uranium fuel will be converted to plutonium (Pu-239 and further to Pu-240, Pu-241 and Pu-242). Of these isotopes Pu-239 and Pu-241 are fissile and will also contribute to the power production in the reactor core. This can to some extent compensate the depletion of U-235 that occurs during a cycle. After irradiation the fissile plutonium isotopes in the spent fuel can be extracted in a reprocessing facility and utilized in so called MOX fuel where it is mixed with depleted uranium (tail from the enrichment process).

The heat production in a nuclear reactor appears in the fission process in the fuel. Fissions occur when a nucleus of fissile material (U-235, U-233, Pu-239 or Pu-241) present in the fuel is hit by a 'slow' neutron (thermal neutron) and divides into two smaller nuclei (fission products), gamma radiation and 2 - 3 new neutrons (fission neutrons). The heat production from the fission process is mainly kinetic energy from the fission products. An illustration of the fission process is given in Figure 13.1. The fuel pellets are assembled in fuel pins encapsulated in cladding Figure 13.2.

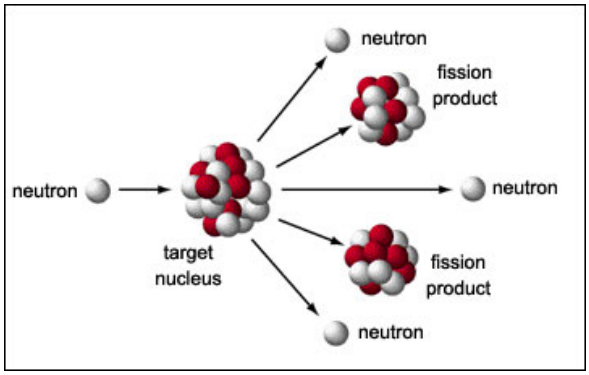

Figure 13.1: Illustration of the Fission Process (Target Nucleus can be U-235, U-233, Pu-239 or Pu-241)

Figure 13.2: Fuel Pellets and Fuel Pellets Arranged in a Fuel Pin.

(Window in cladding shows the fuel pellets inside)

The neutrons released in the fission process are so called fast neutrons and must be moderated to thermal neutrons before they can be absorbed by other fissile nuclei and produce fission. The moderator material will slow down and to some extend absorb neutrons in the reactor core. Moderator material can be water (H_2O), heavy water (D_2O) or graphite. If the production of neutrons in the core is in equilibrium with the loss of neutrons the chain reaction is self-sustaining and the reactor is said to be *critical*. The chain reaction (neutron balance) and the resulting reactor power level is precisely balanced by insertion of neutron absorbing materials such as control rods (cadmium, silver, boron) to the reactor core.

The heat from the fission process is transferred from the fuel pellets through the cladding to the coolant material. Often the coolant and the moderator is the same, but they may also be different materials. In a Boiling Water Reactor (BWR) the coolant boils and produces steam directly in the core. The steam is used to drive turbines for electricity production. In Pressurized Water Reactors (PWRs) the high pressure suppresses boiling in the core and steam generators outside the reactor core are used to produce steam for the turbines.

14. APPENDIX B: NUCLEAR REACTOR TECHNOLOGY

Worldwide, several nuclear reactor types are used for energy production. The different types are usually classified according to the main feature of the reactor i.e. the moderator method (material). The most widespread power reactor types are:

- *Light Water Reactors* where both the moderator and coolant are *light water* (H_2O). To this category belong the Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs).
- *Heavy Water Reactors* where both the moderator and coolant are heavy water (D_2O). To this category belongs the Pressurized Heavy Water Reactors (PHWRs) or the co called CANDU reactors.
- *Graphite Moderated Reactors* where the moderator consists of Graphite. The coolant material is either gas or light water. In this category there are Gas Cooled Reactors (GCRs) and Light Water cooled Gas Reactors (LWGR).
- *Fast Breeder Reactors (FBRs)* and other experimental installations.

The numbers of reactor units of the above mentioned types are illustrated in Figure 14.1. Each of these reactor types is briefly described below.

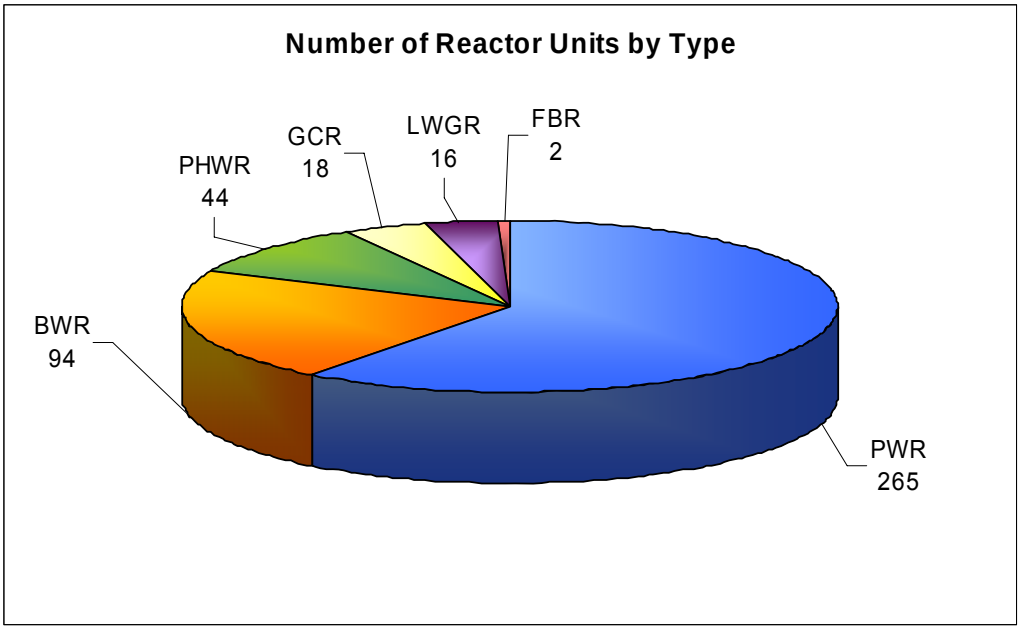

Figure 14.1: Reactor Units by Type as of December 2007.

(Source: Power Reactor Information System, PRIS)

14.1 Appendix B1: Pressurised Water Reactor (PWR)

The Pressurized Water Reactor (PWR) belongs to the light water reactor type. PWRs are the most common type of power producing reactors and are widely used all over the world. Today, there are 265 PWR units generating electric power accounting for about 61 % of the world's current power reactors. PWRs are generation II reactors that use ordinary light water (H_2O) as both coolant and moderator in the reactor core. The water is held at pressures around 160 bars to prevent boiling and is heated to 320 – 330˚C by the fission process as it passes through the core. It transfers energy to a secondary loop that produces steam (saturated at 275 °C) which drives the steam

turbine and, in turn, a generator to produce electricity. The steam cycle is typically 33 % efficient. A principal layout of a PWR is shown in Figure 14.2.

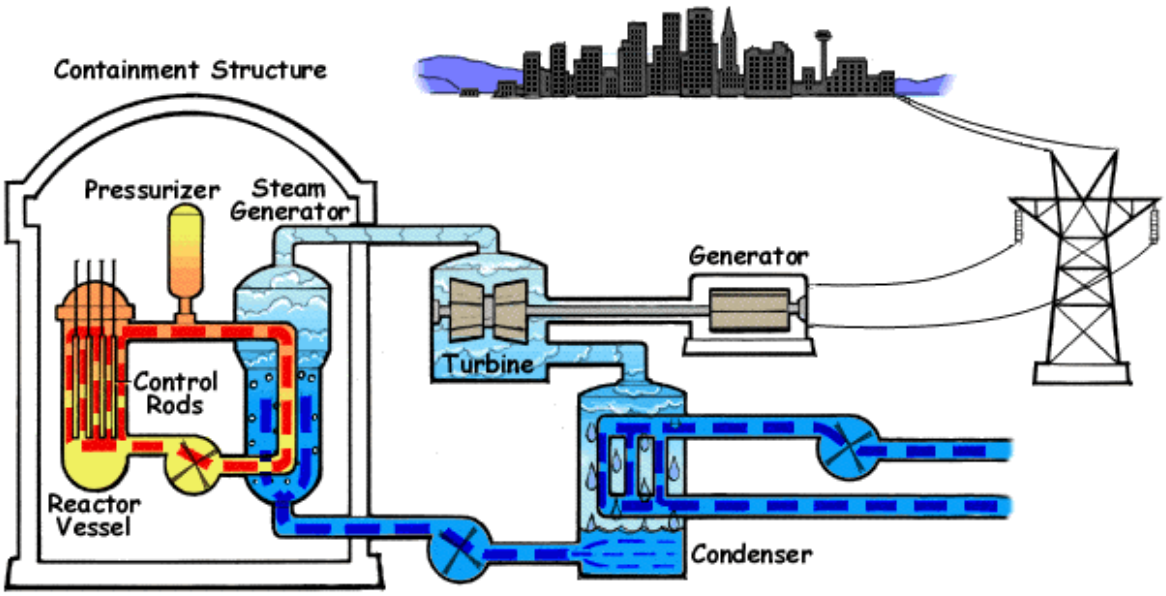

Figure 14.2: Pressurized Water Reactor (PWR).

The uranium used in PWR fuel is usually enriched several percent in U-235. After enrichment the uranium dioxide (UO_2) powder is sintered to create hard, ceramic pellets. The cylindrical pellets are then put into tubes of a corrosion-resistant zirconium metal alloy (Zircaloy). The fuel rods are grouped in fuel assemblies, called fuel bundles, which are positioned in the reactor core. A typical PWR has fuel assemblies consisting of 14x14 to 17x17 fuel rods, with a length of about 4 meters. A large reactor would have about 150 - 250 such assemblies with a total of 80 - 100 tonnes of uranium.

Re-fuelling for most commercial PWRs is after 18 - 24 months where approximately one third of the core is replaced.

Another design belonging to the light water reactor type is the Russian VVER. The reactor is of the PWR design. The fuel is low enriched (ca. 2.4 – 4.4wt% U-235) uranium dioxide (UO_2) pressed into pellets and assembled into fuel rods.

14.2 Appendix B2: Boiling Water Reactor (BWR)

The Boiling water reactor (BWR) also belongs to the light water reactor type and is the second most common nuclear reactor in commercial operation. Today, there are 94 BWR units generating electric power accounting for about 21 % of nuclear reactors installed. BWRs use ordinary light water (H_2O) both as coolant and moderator. In a BWR, water is constantly fed into the bottom of the primary vessel and then boils in the upper part of the reactor core. The steam generated, at a pressure of 70 bars and temperature around 290˚C, is routed directly to the turbine.

A modern BWR fuel assembly comprises 74 to 100 fuel rods, and there are up to approximately 800 assemblies in a reactor core, holding up to about 140 tonnes of uranium. Fuel load and efficiency are similar to the PWR. A schematic drawing of a BWR is shown in Figure 14.3.

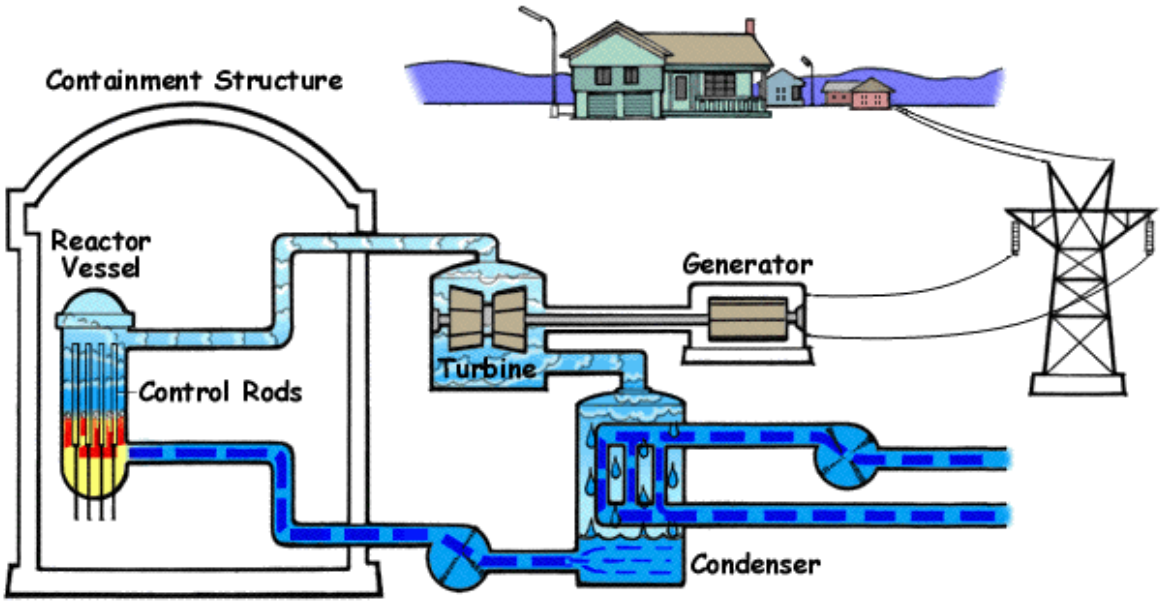

Figure 14.3: Boiling Water Reactor (BWR).

14.3 Appendix B3: Pressurised Heavy Water Reactor (PHWR or CANDU)

The Pressurised Heavy Water Reactor (PHWR) is a reactor fuelled with natural uranium and cooled and moderated with heavy water (deuterium, D_2O). This reactor design is often called CANDU (Canada Deuterium Uranium) since it was developed in Canada. There are 44 heavy water moderated reactors based on the CANDU design in operation worldwide accounting about 10 % of the nuclear reactors installed.

The PHWR/CANDU design is similar to the PWR in that fission reactions inside the reactor core heat coolant - heavy water in CANDU and normal (light) water in PWR - in the primary loop. This loop is pressurised to prevent boiling and steam formation. As in a PWR, steam is generated in a secondary coolant loop at reduced pressure to drive the turbine and generator. CANDU overall thermal efficiency is typically about 31 %. A major difference is that, whereas the core and moderator of a PWR are in a single large, thick-walled steel pressure vessel, the CANDU fuel bundles and coolant are contained in some hundreds of horizontal pressure tubes penetrating a large tank of heavy water moderator. A schematic drawing of a Pressurised Heavy Water Reactor (PHWR) or CANDU is shown in Figure 14.4.

Because of the good neutron economy the CANDU reactor design can utilize natural uranium dioxide (UO_2) containing 0.7wt% U-235 as fuel. A CANDU fuel assembly consists of a number of Zircaloy tubes containing ceramic fuel pellets arranged into a cylinder that fits within the horizontal fuel channels in the reactor. The assemblies have between 28 and 43 half-meter long fuel tubes lying end to end in a fuel channel.

One feature of the CANDU design is the online re-fuelling. New fuel assemblies are inserted in one end of the fuel channel while old assemblies are unloaded from the opposite side.

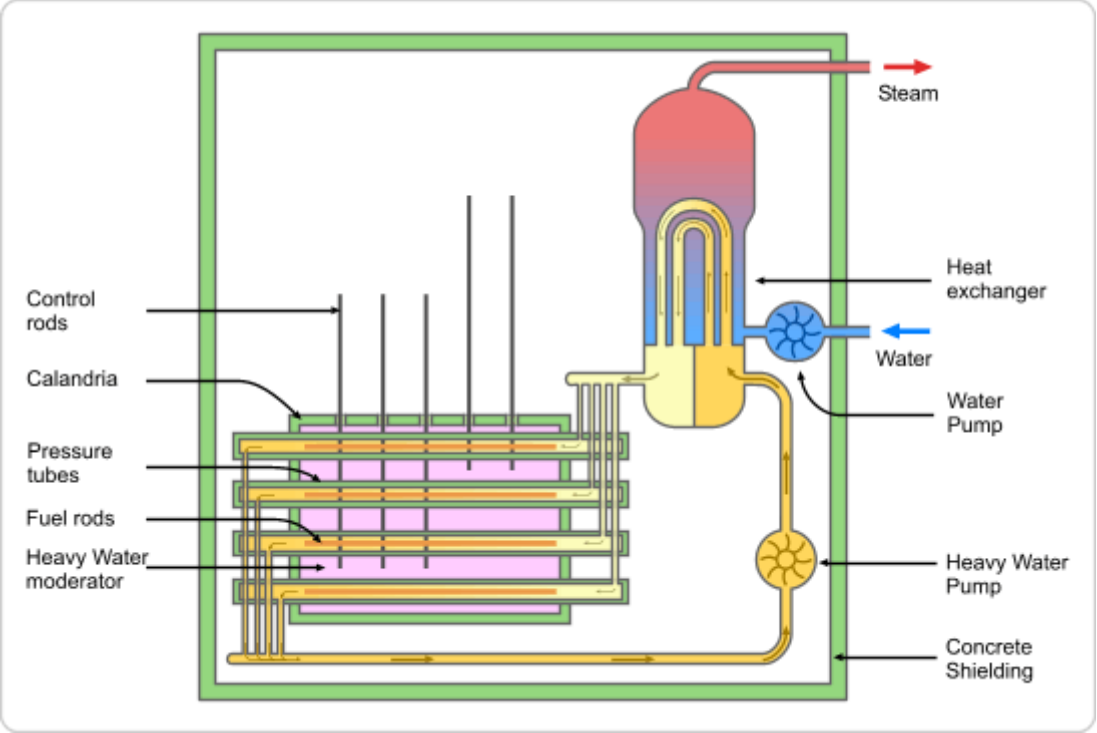

Figure 14.4: Schematic drawing of a Pressurised Heavy Water Reactor (PHWR) or the so called CANDU reactor.

14.4 Appendix B4: Gas Cooled Reactors (GCR)

The gas-cooled reactors (GCRs) belong to the Graphite Moderated Reactor types. These reactors use graphite as the neutron moderator and either carbon dioxide (CO_2) or helium (He) gas as coolant. The advantage of the design is that the coolant can be heated to higher temperatures than water. As a result, higher plant efficiency (40 % or more) can be obtained compared to the water cooled design (33 - 34 %).

The newest gas cooled reactor type is the HTGR (High Temperature Gas cooled Reactor), which is cooled by helium and moderated by graphite. In this reactor coolant temperatures as high as 950ºC can be achieved. A Schematic drawing of an Advanced Gas Cooled Reactor (AGR) is given in Figure 14.5.

Also belonging to the Graphite Moderated Reactor types are the Light Water cooled Gas Reactors (LWGR). These reactors are moderated by graphite and cooled by light water.

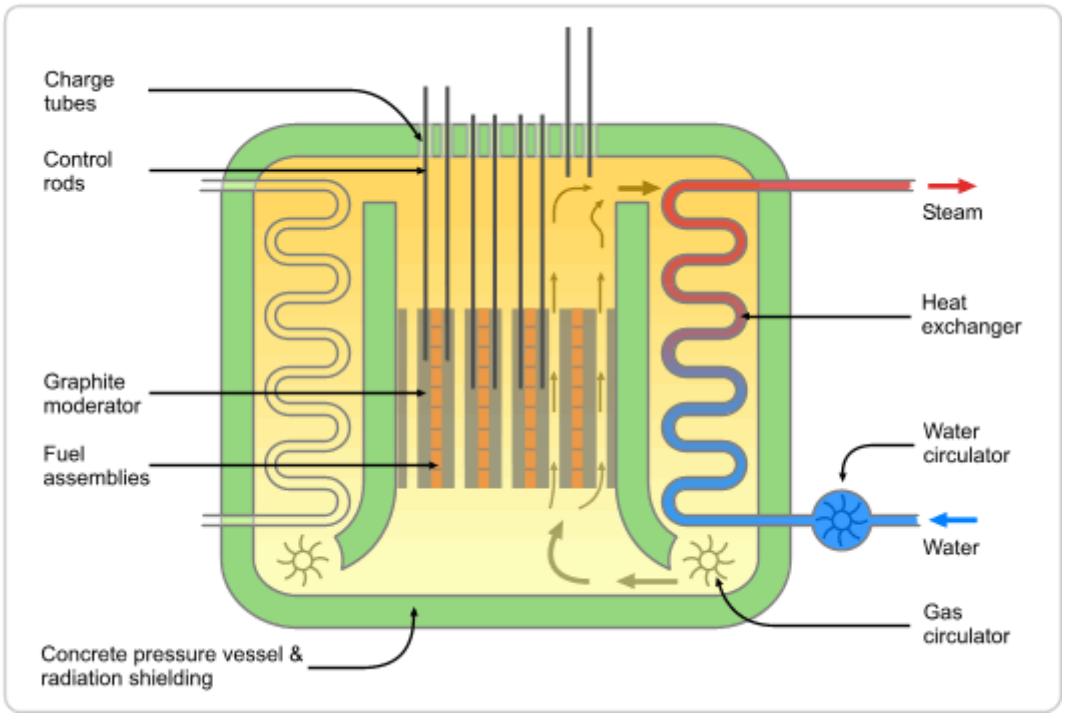

Figure 14.5: Schematic Drawing of an Advanced Gas Cooled Reactor (AGR)

14.5 Appendix B5: Fast Breeder Reactor (FBR)

The reactors described above have in common that they are all thermal reactors. In thermal reactors the fast neutrons released in the fission process must be "slowed down" (thermalized) by collisions with the moderator atoms. Fast reactors, on the other hand, utilize the fast neutrons directly for producing fissions. While fast neutrons are less likely to be absorbed by U-235 or plutonium-239 than thermal neutrons, the highly enriched fuel used in fast breeder reactors allows for a self-sustaining nuclear chain reaction. For this reason, no moderator is required to thermalize the fast neutrons.

All large-scale FBRs have been Liquid Metal Fast Breeder Reactors (LMFBRs) cooled by liquid sodium to transfer heat from the core to steam used to power the electricity generating turbines. FBRs usually use a mixed oxide (MOX) fuel core of up to 20 % plutonium dioxide (PuO_2) and at least 80 % uranium dioxide (UO_2).

The fast breeder reactor (FBR) is a fast neutron reactor designed to breed fuel by producing more fissile material than it consumes. In many FBR designs, the reactor core is surrounded in a blanket of tubes containing non-fissile uranium-238 (U-238) which, by capturing fast neutrons from the reaction in the core, is partially converted to fissile plutonium (Pu-239), which can then be reprocessed for use as nuclear fuel. Other FBR designs rely on the geometry of the fuel itself (which also contains U-238) to attain sufficient fast neutron capture.

15. APPENDIX C: EDUCATION AND TRAINING

Finland:

The following courses are initiated or will be started at Åbo Akademi:

- Fall 2006 Nuclear Structure
- Spring 2007 Emerging Nuclear Energy Systems I (ENES-I)
- Spring 2007 Radiation Safety
- Fall 2007 Radiation Safety Laboratory
- Fall 2007 Nuclear Physics
- Spring 2008 Radiation Safety (+ Laboratory if needed)
- Spring 2008 Not yet decided, may be ENES-I/II or Reactor Physics

The program has been discussed with the following parties and has received very positive response from the following institutions:

Olkiluoto Reactor Plant, Helsinki University of Technology, Advanced Energy Systems, Fortum Nuclear Services, STUK, the Finnish Radiation and Nuclear Safety Authority, Technical Faculty of Åbo Akademi, Neutron Physics, the Royal Institute of Technology (KTH), Stockholm.

Sweden:

Göteborg, Chalmers University of Technology:

Undergraduate courses in nuclear engineering are collected as a special "line" (inriktning in Swedish) in the master program in "Applied Physics". A volume of 30 ECTS in nuclear engineering courses, divided to 4 courses, each consisting of 7.5 ECTS. The courses are given jointly by the groups (formerly departments) of Nuclear Engineering and Nuclear Chemistry, Chalmers. All courses are given during the fall semester, whereas the spring is designated for the master thesis (diploma) work.

The courses contain about 2 laboratory exercises per package, with written reports. In the spring term, or incidentally in the summer, the students perform a one-week long laboratory course at reactors at a research institute. During a very long period, these were made in Studsvik. Since the closing of the R2 and R2-0 reactors in Studsvik, the laboratory exercises were performed at the Kyoto University Research Reactor Institute. The current course package is meant to represent a broad basis in nuclear engineering, and has, in an overview, the following content shown in the Table 15.1 below.

Table 15.1: The Current Course Package in Nuclear Engineering.

Nuclear Engineering I	Nuclear Engineering II	Nuclear Engineering III	Nuclear Engineering IV
Basic concepts in nuclear physics	Neutron sources	Criticality, the four factor formula	Heat transfer and heat removal
Cross sections	Kinetics in nuclear reactions	The transport equation	Single and two phase flows
Radiation interaction with matter	Neutron moderation	One and two group diffusion	Present nuclear reactors
Biological effects of radiation	Ficks law, diffusion	Reactor dynamics	Past and future nuclear reactors
Neutron interaction with matter	Neutron leakage	Changes in reactivity	Core stability
Neutron induced reactions, activation	The diffusion equation	Reactor calculation methodology	Nuclear reactor performance/safety
Limits and effects of radiation on humans	Chemistry of actinides	Composition of nuclear waste	Design basis accidents/transients
Detector principles and neutron detection	Production and chemistry of super heavy elements	The Swedish system	Core instrumentations
Fission	Radioactivity in nature	Transmutation	Separation techniques
Measurement statistics			Basic solvent extraction
Electronics in nuclear experiments			

Graduate courses given at Chalmers are:

- Advanced Reactor Theory
- Radiation detection and measurement
- Actinide Chemistry
- Solvent Extraction

Stockholm, Royal Institute of Technology:

Undergraduate courses in the master program are given regularly. For the year 2008 they are:

Spring 2008

- Nuclear reactor engineering
- Nuclear power safety
- Chemistry and physics of nuclear fuels
- Transmutation of nuclear waste
- Nuclear reactor dynamics and stability

Fall 2008

- Nuclear Physics
- Radiation protection, dosimetry and detectors
- Reactor Physics
- Light water reactor chemistry
- Thermal hydraulics in nuclear engineering
- Nuclear chemistry

- Fuel cycle and waste management
- Numerical methods in nuclear engineering
- Radiation damage
- Generation IV reactors

Uppsala University, Faculty of Science and Technology:

Undergraduate courses are given yearly in:

- Energy physics
- Nuclear energy – Technologies and Systems
- Nuclear physics
- Nuclear and particle physics
- Nuclear science with modern applications

Graduate courses:

- Nuclear Fuel Diagnostics and Safeguards
- Accelerator physics and technology
- Ionizing radiation and detectors

Contacts in Sweden:

- Assoc. Professor Janne Wallenius, Head of Reactor Physics, KTH,
- +46-8-55 378 200, +46-70-603 6306
- Professor Imre Pázsit, Director of CKTC, Chalmers University, +46-31-177 230 81
- Assoc. Professor Ane Håkansson, Research leader Nuclear Fuel Diagnostics and Safeguards, Uppsala University, +46-18-471 3251, +46-70-167 9095
- Nils-Olov Jonsson, Director of SKC, +46-8-7396861, +46-76-134 6861
- Professor Jan Blomgren, Head of Applied Nuclear Physics Uppsala University,
- +46-18-471 3788, +46-70-167 9003

UK:

A description of all course modules offered by NTEC is available at http://www.ntec.ac.uk:

- N01 Reactor Physics, Criticality & Design, The University of Birmingham
- N02 Nuclear Fuel Cycle, HMS Sultan
- N03 Radiation & Radiological Protection, The University of Manchester
- N04 Decommissioning / Waste / Environmental Management, The University of Liverpool / Westlakes
- N05 Water Reactor Performance and Safety, Imperial College London
- N06 Reactor Materials & Lifetime Behaviour, The University of Manchester
- N07 Safety Case Development, HMS Sultan
- N08 Particle & Colloid Engineering in the Nuclear Industry, University of Leeds
- N09 Policy, Regulation & Licensing, The University of Manchester
- N10 Processing, Storage & Disposal of Nuclear Waste, The University of Sheffield

- N12 Reactor Thermal Hydraulics, HMS Sultan
- N13 Criticality Safety Management, HMS Sultan
- N14 Risk Management, City University, London
- N21 Geotechnical Aspects of Radioactive Waste Disposal, UHI Millennium Institute
- N22 Public & Political Aspects of Nuclear Decommissioning, UHI Millennium Institute
- N23 Environmental Impact Assessment, The University of Liverpool/Westlakes
- N24 Environmental Decision Making Applied to Decommissioning, Lancaster University /Westlakes
- N29 Decommissioning Technology & Robotics, Lancaster University
- N30 Design of Safety Critical Systems, Lancaster University
- N31 Management of the Decommissioning Process , The University of Birmingham
- N32 Experimental Reactor Physics, The University of Manchester

The following eight courses will also be offered in distance learning multimedia format:

N01, N02, N03, N04, N10, N13, N29, N31.

Canada:

The undergraduate courses at UOIT are:

- Corrosion for Engineers
- Environmental Effects of Radiation
- Nuclear Fuel Cycles
- Nuclear Plant Design and Simulation
- Nuclear Plant Operation
- Nuclear Reactor Design
- Principles of Fusion Energy
- Radioactive Waste Management Design
- Risk Analysis Methods
- Strength of Materials

Graduate courses given by UNENE:

- UN 0600 / Industrial Research Project, University of Western Ontario / Staff /
- UN 0601 / Control, Instrumentation and Electrical Systems in CANDU based Nuclear Power Plants / J. Jiang /
- UN 0602 / Nuclear Fuel Waste Management / D. Shoesmith /
- UN 0603 / Project Management for Nuclear Engineers / M. Bennett /
- UN 0700 / Industrial Research Project, University of Waterloo / Staff /
- UN 0701 / Engineering Risk and Reliability / M. Pandey /
- UN 0702 / Power Plant Thermodynamics / R. Chaplin /
- UN 0800 / Industrial Research Project, McMaster University / Staff /
- UN 0801 / Nuclear Plant Systems and Operations / G. Bereznai /
- UN 0802 / Reactor Physics / E. Nichita /

- UN 0803 / Nuclear Reactor Safety Design / V. Snell /
- UN 0804 / Reactor Thermal hydraulics / N. Popov /
- UN 0805 / Radiation Health Risks and Benefits / D. Tucker /
- UN 0900 / Industrial Research Project, Queen's University
- UN 0901 / Nuclear Materials / R. Holt /
- UN 0902 Fuel Management / H. Bonin /
- UN 1000 / Industrial Research Project, University of Toronto
- UN 1001 / Reactor Chemistry and Corrosion Lister / D. Lister /

16. APPENDIX D: RADIATION PROTECTION

16.1 Appendix D1: Note from the Norwegian Radiation Protection Authorities

Note

Statens strålevern
Norwegian Radiation Protection Authority

To:	The thorium report committee	Att. Secretary Lise Moen	Our ref:
From	Norwegian Radiation Protection Authority	Att: Mette Seyersted	200700775/532/OJH

Date: 6 Nov. 2007

Introduction

The thorium committee has requested an input from the Norwegian Radiation Protection Authority (NRPA). This input is based on the question of whether today's legislation and public management would accommodate the establishment of thorium-based nuclear power in Norway. The main purpose of this input is to give an overview of the current Norwegian legislation, in particular the legislation on radiation protection and nuclear energy.

This is an unofficial translation of the Norwegian version of the document and in case of discrepancies between the two versions, the Norwegian version will prevail. The quoted legislations are also unofficial translations.

The NRPA is the national competent authority for radiation protection, nuclear safety and security as well as the corresponding legislation on radiation protection and nuclear energy. The NRPA is a Directorate under the Norwegian Ministry of Health and Care Services and also a competent authority for the Ministry of the Environment and the Ministry of Foreign Affairs. Furthermore, the NRPA assists other Norwegian ministries concerning questions about radiation protection, -safety and -security.

Limits and assumptions

Several assumptions must be assumed as fulfilled before discussing the regulatory management and legal adaptation required to establish Norwegian nuclear power, *inter alia* the Parliament's will to allow development of Norwegian nuclear power, the availability of necessary competence and resources of the parties involved.

Norwegian legislation that is or might be applicable (not exhaustive)

- Act relating to mining (30. June 1972)
- Act relating to acquisition of waterfalls, mines and other real estate [Industrial Licensing Act] (14. December 1917)
- Planning and Building Act (14 June 1985 No. 77)
- Regulations on environmental impact assessment (1 April 2005 No. 276)
- Act concerning nuclear energy activities (12 May 1972)
- Regulations on the Physical Protection of Nuclear Material (2 November 1984)
- Regulations on Possession, Transfer and Transportation of Nuclear Material and Dual-use Equipment (12 May 2000)
- Act relating to the generation, conversion, transmission, trading, distribution and use of energy etc. (Energy Act)
- Act on Radiation Protection and use of Radiation (Radiation Protection Act) (12 May 2000 No. 36)

- Regulations on Radiation Protection and use of Radiation (Radiation Protection Regulations) (21 November 2003 No. 1362)
- Act relating to protection against pollution and relating to waste [The Pollution Control Act] (13 March 1981 No. 6) with regulations
- Act relating to working environment, working hours and employment protection, etc. (Working Environment Act) (17 June 2005 No. 62)
- Regulation relating to work with ionising radiation (14 June 1985).
- Regulations relating to systematic health, environmental and safety activities in enterprises [Internal control regulations] (6 December 1996 No. 1127)

Norwegian legislation that should be considered regarding:

- **Mining and milling of thorium**

The Radiation Protection Act and the Radiation Protection Regulations will be applicable for activities relating to mining and milling of thorium. Furthermore, such activities will have to be considered under more general legislation on mining, such as the Act relating to mining (relevant authority: Ministry of Industry and Trade) and the Industrial Licensing Act (relevant authority: Ministry of Petroleum and Energy). Mining and milling of thorium will also have to be considered under the Planning and Building Act and the relating Regulations on environmental impact assessment, both under the responsibility of the Ministry of the Environment. Please see below for further elaboration of parts of this legislation.

- **Enrichment, fuel production, operation and waste disposal**

Applicable legislation for these activities will be inter alia the Planning and Building Act and the relating Regulations on environmental impact assessment, the Act concerning nuclear energy activities, the Radiation Protection Act and Radiation Protection Regulations, the Energy Act and the Pollution Control Act. Please see below for further elaboration of some of this legislation.

Relevant Conventions and International Agreements Norway has acceded:

- Convention on Nuclear Safety (20 september1994)
- Treaty on the non-proliferation of nuclear weapons (1 July1968)
- Joint Convention on the Safe Management of Spent Fuel and the Safe Management of Radioactive Waste (29 September 1997)
- Convention on the Physical Protection of Nuclear Materials and Nuclear Facilities (3 March 1980, amended 2005)
- The Paris Convention on Nuclear Liability (29 July1960) with additional protocols
- Convention on Early Notification of a Nuclear Accident (26 September 1986) and several bilateral agreements
- Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency (26 September 1986)
- Nordic Mutual Emergency Assistance Agreement in Connection with Radiation Accidents (17 October 1973)
- Convention on Environmental Impact Assessment in a Transboundary Context (25 February 1991)
- OSPAR Convention for the Protection of the Marine Environment of the North-East Atlantic (22 September 1992)

The Planning and Building Act (Ministry of the Environment) - Chapter VII-a Environmental Impact Assessment

Mining and milling of thorium, operation as well as waste disposal will have to be assessed according to Regulations on environmental impact assessment (Ministry of the Environment). Threat and risk assessments relating to these activities will have to be included in the environmental risk assessment.

Act concerning Nuclear Energy Activities and regulations include:

- Licence for constructing, owning and operating nuclear installations, as well as permits
- Supervision by the National Radiation Protection Authority
- Safety, security and emergency preparedness
- Fees and dues
- Nuclear liability (compensation and insurance)
- Control of the peaceful utilisation of nuclear energy

The Act concerning Nuclear Energy Activities and regulations is based on a set of definitions:

*"**Section 1** [1] (definitions)*

For the purposes of this Act, the following definitions apply:
(a) nuclear fuel:
fissile material in the form of uranium or plutonium in metallic form, alloy or chemical compound, and such other fissile material as the Ministry[2] may determine;
(b) radioactive product:
other radioactive material (including wastes) which is made or has become radioactive by irradiation accidental to the production or utilisation of nuclear fuel;
(c) nuclear substance:
nuclear fuel, other than natural uranium and depleted uranium, as well as radioactive products, except radioisotopes used for industrial, commercial, agricultural, medical or scientific purposes or which are intended for, and are directly usable for such a purpose;
(d) nuclear reactor:
a structure containing nuclear fuel in such an arrangement that a self-sustaining chain process of nuclear fission can occur therein without the addition of neutrons from another source;
(e) nuclear installation:
nuclear reactor installation;
factory for production or processing of nuclear substances,
factory for the separation of isotopes of nuclear fuel,
factory for reprocessing irradiated nuclear fuel,
facility for the storage of nuclear substances other than facilities intended exclusively for use as temporary storage accidental to the transport of such substances,
and such other facilities, in which there are nuclear fuel or radioactive products, as the Ministry may determine;
…..."

Licence
The Act concerning Nuclear Energy Activities regulates licences for constructing, owning and operating nuclear installations, as well as permits:

*”**Section 4.** (Licence for nuclear installation)*[1]

It shall be unlawful to construct, own or operate a nuclear installation[2] *without a licence granted by the Crown. The licence shall be valid for a specified place for operation. As a rule the duration of the licence should be limited to a specific period. The transfer of a nuclear installation*[2] *or the operation thereof to a new owner or operator*[2] *needs special licence..*

A licence for the construction of a nuclear power plant should not be granted before the Storting (translator's note: the Norwegian Parliament) has given its approval. The matter should be submitted to the Storting when proposals for the construction site of the nuclear power plant are presented and the question of the operator/ownership is clarified.”
“ [1] Cf. Sections 55. 56 and 58.
[2] See Section l.”

*”**Section 10.**[1] (The Norwegian Radiation Protection Authority)*

The Norwegian Radiation Protection Authority is the highest specialist agency as far as questions of safety are concerned. It functions as the institution making recommendations and giving advice to the Ministry concerned. The Authority shall prepare and submit recommendations on all applications concerning licences and permits, and shall on its own initiative put into effect all such means measures as it deems necessary for reasons of safety. It shall be the duty of the Authority to ensure that all rules and conditions pertaining to safety precautions are complied with and put into effect, as well as such orders that are given in pursuance of this Act.

[1] Amended by Act no. 142 of 18 December 1992. ”

The Ministry concerned with nuclear power plants is the Norwegian Ministry of Petroleum and Energy and the Ministry concerned with other nuclear installations is the Norwegian Ministry of Health and Care Services.

Requirements for licensing a nuclear reactor would include reactors using:
- uranium fuel and
- thorium based fuel, where uranium or plutonium is used in conjunction with thorium to maintain the nuclear chain reaction.

A reactor based on using a mass accelerator and pure thorium fuel does not fall under the present legal term "nuclear reactor" as such a system would be dependent on an external source of accelerated protons or neutrons to maintain the nuclear chain reaction. The Act concerning Nuclear Energy Activities would therefore need to be amended from its present form in order to incorporate a thorium-based Accelerator-Driven System (ADS).

The National Radiation Protection Authority
The role of the National Radiation Protection Authority is described in several sections of The Act concerning Nuclear Energy Activities, among others in Sections **10, 11, 13, 14.** Section 10 is quoted in the above paragraph, section 11 is quoted below:

*”**Section 11**. (Construction and commissioning of nuclear installation).*

\ The Norwegian Radiation Protection Authority[2] *shall exercise continuous supervision over the construction of nuclear installations. In particular it shall ensure compliance with the terms and*

provisions of the licence, as well as ensuring the implementation of all necessary measures required safety precautions, including such safety measures as described in the provisionally authorised safety reports. Measures described in the safety reports may be altered by the Norwegian Radiation Protection Authority[2] providing this does not conflict with safety considerations.

2. Before a nuclear installation[3] is put into operation, the operator[3] must have obtained[4] authorisation for this from the Norwegian Radiation Protection Authority[2]. Before granting such authorisation the Authority must be satisfied that:
a the technical standards of the installation, the operating regulations, safety measures and accident emergency plans are sound,
b. the management and personnel of the installation have the necessary qualifications and clearly defined areas of responsibility,
c security has been furnished in accordance with Sections 35 and 37 of this Act.
d. all the necessary authorisations have been obtained from the competent authorities in accordance with other legislative provisions.

3. In good time before the nuclear installation is put into operation the operator shall submit to the Norwegian Radiation Protection Authority[2] a complete safety report on the installation.

4. The Norwegian Radiation Protection Authority[2] may, if it believes this will assist it in its evaluation of the installation, give separate consent to a limited trial operation, subject to such conditions as may appear necessary.
[2] See Section H).
[3] See Section l
[4] Cf. Section 55."

Safety measures and emergency preparedness
Norway has acceded all the IAEA radiation protection, radiation safety and non-proliferation conventions. Any potential future nuclear power in Norway would, however, require additional legal regulation. Such regulations could be adapted from existing international guidelines and regulations, such as IAEA safety standards[1] for:

- Radiation protection
- Nuclear safety
- Activities that can produce radiation producing substances (e.g., mining)
- Waste handling and storage/depository

Fees and dues (The Act concerning Nuclear Energy Activities, section 57)
There is a fee for licence application and a due for supervision of nuclear installations as specified in section 57:

"Section 57 *(fees and dues)*

1. A handling fee shall be payable for the consideration by the authorities of a licence application. The fee shall accompany the licence application or be paid in instalments in accordance with Ministry regulations.

[1] *EURATOM-directives will also be considered in this work, as well as Swedish and Finnish legislation on nuclear power*

2. Dues shall be paid for the supervision undertaken by the Norwegian Radiation Protection Authority in connection with the construction and operation of nuclear installations.
3. Fees and dues shall be determined by the Crown."

Nuclear liability (compensation and insurance)
The Paris Convention with additional protocols is implemented in Chapter III of the Act of Nuclear Energy Activities.

Control of the peaceful utilisation of nuclear energy
Norway has entered an agreement on international safeguards:

- The Agreement between Norway and the Agency for the Application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons (1 March 1972), and
- Protocol additional to the Agreement between the Kingdom of Norway and the International Atomic Energy Agency for the Application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons (29 September 1999).

Section 51. *(Control of the peaceful utilisation of nuclear energy) from the Act of Nuclear Energy Power Activities*

"The Crown may issue any provisions necessary to ensure and ascertain by supervision that nuclear installations, nuclear fuel, radioactive products and other materials used for providing nuclear energy that fall under the scope of international safety supervision pursuant to an agreement to which Norway has acceded, are only used for peaceful and non-explosive purposes. Norwegian inspectors shall, for the purposes of supervision, have right of access to nuclear installations and to other places where the aforementioned materials and equipment are located or assumed located. The inspectors are entitled to obtain such information as is deemed necessary in order to ascertain whether such installations, materials and equipment are to be used for peaceful and non-explosive purposes. To the extent that an agreement on international safety supervision exists to which Norway has acceded, foreign inspectors shall also have the right of access to information and, accompanied by Norwegian inspectors or other authorised persons, nuclear installations, etc., in accordance with point two."

Export of thorium
Export of thorium is regulated by Regulations on Possession, Transfer and Transportation of Nuclear Material and Dual-use Equipment, sections 11 and 12. The main rule is that export of thorium requires an authorization and in the case such authorization is given, the export is subject to notification to NRPA and the Ministry of Foreign Affairs.

Import of thorium
Import of thorium is regulated by Regulations on Possession, Transfer and Transportation of Nuclear Material and Dual-use Equipment, Chapter II (section 3). Here the main rule is that import of thorium requires an authorization.

The Radiation Protection Act and the Radiation Protection Regulations include

- General requirements (justification and basic principles for use of radiation)
- Qualifications and training
- Risk assessment and emergency preparedness
- Requirements of authorisation (Radiation Protection Regulations Section 5)
- Occupational exposure to ionising radiation
- Special provisions on discharges to the environment and on waste treatment of substances which emit ionising radiation
- Inspection powers of the Norwegian Radiation Protection Authority

Basic principles for radiation protection
The Radiation Protection Act and Use of Radiation Section 5 stipulates the basic radiation protection principles (justification, optimization and dose limits)

Requirements of authorisation according to the Radiation Protection regulations (Section 5)
Activities intending to procure, use or handle substances that emit ionising radiation need authorisation from the Norwegian Radiation Protection Authority, for instance:

- Discharges of radioactive substances
- Facilities for the treatment, storage or disposal of radioactive waste
- Import and export of radioactive waste

Authorisation requirements for mining and milling thorium are not included in the current radiation protection regulatory system. Such a requirement will be considered in future revisions of the regulations.

Occupational exposure to ionising radiation
The Radiation Protection Regulations regulate occupational exposure to ionizing radiation under Section 21 (extract):

> ***Section 21 Dose limits etc.***
> *All radiation exposure shall be kept as low as reasonably achievable, and the following dose limits shall not be exceeded:*
> *a) The dose limit for workers over the age of 18 is 20 mSv per calendar year. The Norwegian Radiation Protection Authority may grant dispensation for individuals where the nature of the work makes it impracticable to set an annual limit of 20 mSv. In such cases permission may be given for a limit of 100 mSv over a continuous five-year period, on condition that the effective dose does not exceed 50 mSv in any single year.*

The dose limits for occupational exposure are in accordance with international recommendations.

Regulations regarding the environment
The Radiation Protection Act and Use of Radiation and the Radiation Protection Regulations regulate radiation exposures to the environment. Activities involving treatment, storage and final disposal of radioactive waste require authorization under the Radiation Protection Regulations, as described in Chapter V. Selected sections of Chapter V are quoted below:

> ***"Section 23 Regulation of discharges***
> *Undertakings which cause discharges of radioactive substances shall have approval to do so from the Norwegian Radiation Protection Authority cf. section 5 o). The undertakings shall*

use the best available technology such that discharges to the environment are avoided or kept to the lowest possible level.

Section 24 Order to investigate and carry out countermeasures
The Norwegian Radiation Protection Authority may order undertakings which cause or may cause radioactive contamination or radiation in the environment to carry out investigations and take measures which may reasonably be demanded in order to:
d) counteract any damage or inconvenience resulting from the contamination.

Radioactive waste shall be dealt with in such a way as to cause the least possible damage and inconvenience. The waste treatment shall:
a) generate minimal waste,
b) be carried out using the best available technology. To minimise waste problems, a basis shall be taken in technology which, based on an overall assessment of current and future use of the environment and of economic factors, gives the best results.

Import
Import of thorium is subject to the principle of justification (Radiation Protection Act Section 5). The NRPA may on certain conditions prohibit such import:

Section 20 Prohibition of import and sale
The Norwegian Radiation Protection Authority may refuse the import or sale of any product or substance and any item that may involve a risk to health or environment due to radiation, provided that this is not in conflict with international agreements to which Norway has acceded.

Inadequacies in the current regulatory framework
The potential establishment of Norwegian nuclear power will probably require an evaluation of the Radiation Protection Act and/or the Radiation Protection Regulations to ensure that the legislation fully complements the Act concerning nuclear energy activities. For instance, one may need to consider authorization requirements (from the NRPA) for mining and milling thorium.

The Pollution Control Act with regulations

- The purpose of the Pollution Control Act is to protect the outdoor environment against pollution and to reduce existing pollution, to reduce the quantity of waste and to promote better waste management.
- The pollution control authority may on application issue a permit for any activity that may lead to pollution (Section 11). The Pollution Control Act comprises radiation to the extent decided by the pollution control authority (Section 6). Today the Pollution Control Act does not regulate radioactive pollution from nuclear installations.

Internal Control Regulations
According to Regulations relating to Systematic Health, Environmental and Safety Activities in Enterprises (Internal Control Regulations), enterprises shall have systematic measures designed to ensure that the activities of the enterprise are planned, organised, performed and maintained in conformity with requirements laid down in or pursuant to the health, environmental and safety legislation. This legislation comprises legislation on radiation protection, working environment, pollution control and prevention of fire and explosion.

According to the Act concerning Nuclear Energy Activities Section 6 (quoted below), the Norwegian Government may issue detailed provisions and rules regarding nuclear installations and the treatment of nuclear substances, and through such provisions ensure that requirements concerning internal control and internal control systems are met. Such detailed provisions will have to be considered in regard to a revision of the nuclear energy legislation.

> **Section 6.** ***from*** *the Act concerning Nuclear Energy Activities (provisions)*[1]
>
> *"The Crown may issue detailed provisions regarding the construction and operation of nuclear installations*[2]*. The Crown may also issue rules regarding the manufacture, handling, packaging and marking, carriage, storage, sale, and other ways of holding nuclear substances or other kinds of nuclear fuel or radioactive products.*
> [1] *Cf. Sections 55 and 56.*
> [2] *See Section l."*

The Energy Act

The Energy Act includes requirements on:

- Licensing for installations for the generation, conversion, transmission and distribution of high voltage electrical energy
- Local area licensing for the construction and operation of such installations

Applications for licences etc.

The process for applications for licences might be as follows:

- A proposal for an assessment programme shall be sent from the proposer to the relevant competent authority (who will circulate it for comments)
- Following circulation for comments, the relevant competent authority will prescribe an assessment programme
- The prescribed assessment programme will form the basis for the environmental impact assessment
- Environmental impact assessment by the proposer
- Application for licence according to the Act concerning nuclear energy activities and the corresponding environmental impact assessment are sent jointly to the Ministry of Petroleum and Energy
- The application and environmental impact assessment are circulated for comments
- The application and environmental impact assessment is submitted to the Parliament?
- The NRPA considers the application etc. and makes a recommendation to the Ministry of Petroleum and Energy
- The Ministry of Petroleum and Energy coordinates the process with other authorities incl. other licences, finalizes the consideration of the whole application and submits the case to the Norwegian Government
- The case is submitted to the Parliament
- Decision by Royal Decree (Act concerning nuclear energy activities Section 4)

Summary:

Legislation that is or may be relevant for nuclear energy based on thorium
Activities related to Norwegian thorium-based nuclear energy will inter alia be regulated by the following legislation: The Planning and Building Act with regulations, the Act concerning nuclear energy activities with regulations, the Radiation Protection Act with regulations, The Energy Act and the Pollution Control Act. In addition, mining and milling of thorium will have to be considered under the Act relating to mining and the Industrial Licensing Act.

The needs for amendments or more detailed legislation in regard to nuclear energy based on thorium
A conventional thorium-base nuclear installation will most probably be comprised by the current licensing requirement laid down in the Act concerning nuclear energy activities, whereas a thorium-based Accelerator-Driven System (ADS) will not. Consequently, the definitions of the Act concerning nuclear energy activities will have to be amended in order to comprise any form of thorium-based nuclear installation. If Norwegian nuclear power is established, the whole Act concerning nuclear energy activities will have to be considered thoroughly and there will be a need for more detailed regulations on inter alia nuclear security and internal control. Such regulations might be based on international requirements and recommendations. The Radiation Protection Act and/or regulations might also be subject to some amendments, for example in order to establish a requirement that mining and milling of thorium is subject to authorization from the NRPA.

Export of thorium
Export of thorium is regulated by Regulations on Possession, Transfer and Transportation of Nuclear Material and Dual-use Equipment. The main rule is that export of thorium requires an authorization and in the case such authorization is given, the export is subject to notification to NRPA and the Ministry of Foreign Affairs.

Import of thorium
Import of thorium requires authorisation according to Regulations on Possession, Transfer and Transportation of Nuclear Material and Dual-use Equipment, Section 3. The Radiation Protection Act stipulates that import of thorium shall be justifiable and may on certain conditions prohibit such import.

16.2 Appendix D2: Decay Properties of the Th-232 and U-238 Decay Series

Decay properties of radionuclides included in the Th-232 and U-238 decay series.

Thorium series **Uranium series**

nuclide	decay mode	half life	α-energy MeV (γ-energy keV)	product of decay		nuclide	decay mode	half life	α-energy MeV (γ-energy keV)	product of decay
						^{238}U	α	4.468·10^{9} a	4.270 (50)	^{234}Th
^{232}Th	α	1.405·10^{10} a	4.081 (64)	^{228}Ra		^{234}Th	β$^-$	24.10 d	0.273 (63, 92)	^{234}Pa
^{228}Ra	β$^-$	5.75 a	0.046 (14)	^{228}Ac						
						^{234}Pa	β$^-$	6.70 h	2.197 (131, 881)	^{234}U
						^{234}U	α	245500 a	4.859 (53)	^{230}Th
^{228}Ac	β$^-$	6.25 h	2.124 (911, 969)	^{228}Th						
^{228}Th	α	1.9116 a	5.520 (84)	^{224}Ra		^{230}Th	α	75380 a	4.770 (68)	^{226}Ra
^{224}Ra	α	3.6319 d	5.789 (241)	^{220}Rn		^{226}Ra	α	1602 a	4.871 (186)	^{222}Rn
^{220}Rn	α	55.6 s	6.404 (550)	^{216}Po		^{222}Rn	α	3.8235 d	5.590 (510)	^{218}Po
						^{218}Po	α 99.98 % β$^-$ 0.02 %	3.10 min	6.115 0.265	^{214}Pb ^{218}At
						^{218}At	α 99.90 % β$^-$ 0.10 %	1.5 s	6.874 2.883	^{214}Bi ^{218}Rn

						^{218}Rn	α	35 ms	7.263 (609)	^{214}Po
						^{214}Pb	β^-	26.8 min	1.024 (352, 295)	^{214}Bi
						^{214}Bi	β^- 99.98 % α 0.02 %	19.9 min	3.272 (609, 1764, 1120) 5.617	^{214}Po ^{210}Tl
^{216}Po	α	0.145 s	6.906 (805)	^{212}Pb		^{214}Po/ ^{210}Tl	α / β^-	0.1643 ms / 1.30 min	7.883 / 5.484 (800, 298)	^{210}Pb
^{212}Pb	β^-	10.64 h	0.570 (239, 300)	^{212}Bi		^{210}Pb	β^-	22.3 a	0.064 (47)	^{210}Bi
^{212}Bi	β^- 64.06% α 35.94%	60.55 min	2.252 6.208 (727)	^{212}Po ^{208}Tl		^{210}Bi	β^- 99.99987% α 0.00013%	5.013 d	1.426 5.982 (266, 304)	^{210}Po ^{206}Tl
^{212}Po	α	299 ns	8.955 (2615, 728, 585)	^{208}Pb		^{210}Po	α	138.376 d	5.407 (803)	^{206}Pb
^{208}Tl	β^-	3.053 min	4.999 (2615, 583, 511, 860, 277)	^{208}Pb		^{206}Tl	β^-	4.199 min	1.533 (803)	^{206}Pb
^{208}Pb	.	stable	.	.		^{206}Pb	-	stable	-	-

17. LIST OF FIGURES

18. ACRONYMS AND ABBREVIATIONS

$(Th,U)O_2$	thorium uranium dioxide
AAA	Advanced Accelerator Applications Program, USA
ADS	Accelerator Driven System
AHWR	Advanced Heavy Water Reactor (Indian design)
ALARA	As Low As Reasonably Achievable
AVR	Atom Versuchs Reaktor
BARC	Bhabha Atomic Research Centre, India
Be	beryllium
bpd	barrels per day
Bq	Becquerel (amount of material that will produce 1 nuclear decay per second)
Bq/kg	Becqurel per kilogram
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium Reactor
Ce	cerium
CEA	The French Atomic Energy Commission
CGS	cover gas system
CHTR	Compact High Temperature Reactor
CKTC	Chalmers University, Sweden
CNEN	The National Committee for Nuclear Energy, Italy
CNRS	Centre National de la Recherche Scientifique
CO_2	carbon-dioxide
D_2O	heavy water
DNV	Det Norske Veritas
EA	Energy Amplifier
EAU	CO_2 Emission Allowance Units
ECN	Netherlands Energy Research Foundation, Petten, the Netherlands
EdF	Électricité de France
ENEN	European Nuclear Education Network
EPRI	U.S. Electric Power Research Institute
ESS	The European Spallation Source (project to design and construct a next generation facility for research with neutrons)
ETWG	Extended (actually European) Technical Working Group
EU	The European Union
Euratom	The European Atomic Energy Community
F	fluorine
FBR	Fast Breeder Reactor
Fe	iron
FIMA	per cent fissions in initial metal atoms (unit used for nuclear fuel burnup)
GDP	Gross Domestic Product
GFR	Gas-Cooled Fast Reactor (Generation IV concept)
GIF	The Generation IV International Forum
GNEP	Global Nuclear Energy Partnership
GT-MHR	Gas Turbine-Modular Helium Reactor
GW_e	Giga Watt Electricity (1 000 000 kW_e)
H_2O	water
HBWR	Halden Boiling Water Reactor
HEU	High Enriched Uranium (> 20wt% U-235)
HTGR	High Temperature Gas Cooled Reactor
HTR	High Temperature Reactor
HWR	Heavy Water Reactor
HYPER	Hybrid Power Extraction Reactor project (The Korea Atomic Energy Research Institute (KAERI) ADS system)
IAEA	International Atomic Energy Agency
ICT KFA	Institute of Chemical Technology, KFA, Jülich, Germany
IEA	The International Energy Agency
IFE	Institute for Energy Technology (Institutt for energiteknikk) at Kjeller and Halden, Norway
IGCAR	The Indira Gandhi Centre for Atomic Research, Kalpakkam, India
INES	International Nuclear Event Scale
INFCE	International Nuclear Fuel Cycle Evaluation

INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
INSTN	French National Institute for Nuclear Sciences and Technology
IR	Inferred Resources
ITER	International Thermonuclear Experimental Reactor (Fusion)
JAEA	The Japan Atomic Energy Agency (formerly Japan Atomic Energy Research Institute, JAERI and Japan Nuclear Cycle Development Institute, JNC)
KAPL	Knolls Atomic Power Laboratory, USA
k_{eff}	effective neutron multiplication factor
KEK	The High Energy Accelerator Research Organization, Japan
Kr-85	krypton-85 (^{85}Kr)
kWh	kilo Watt hour
LBE	Liquid Bismuth Eutectic
LEU	Low Enriched Uranium (< 20wt% U-235)
LFR	Lead-Cooled Fast Reactor (Generation IV concept)
Li	lithium
LINAC	linear accelerator
LOCA	Loss-Of-Coolant Accident
LOF	Loss-Of-Flow (possible accident of an ADS system)
LOHS	Loss-Of-Heat-Sink (possible accident of an ADS system)
LPSC-lab	Laboratoire de Physique Subatomique et de Cosmologie (in Grenoble)
LWBR	Light Water Breeder Reactor
LWR	Light Water Reactor
MEGAPIE	The Megawatt Pilot Experiment
MEU	Medium Enriched Uranium (<20wt% U-235)
MeV	Mega electronvolt (unit of energy)
Mg	magnesium
MgO	magnesium oxide
micrometer	1 millionth of a meter (10^{-6} m)
microseconds	1 millionth of a second (10^{-6} s)
MOX	Mixed-Oxide (uranium and plutonium in mixed-oxide fuel)
MPa	mega pascal (1000 000 Pa = 10^6 Pa)
MSBR	Molten Salt Breeder Reactor
MSR	Molten Salt Reactor (Generation IV concept)
MSRE	Molten Salt Reactor Experiment
mSv	milli Sievert (10^{-3} Sv or 1/1000 Sv)
MTR	Material Test Reactor
MWd/tHM	Mega Watt days per tonne Heavy Metal (unit used for nuclear fuel burnup)
MW_{th}	Mega Watt Thermal
n	neutron
NaI Detector	Sodium Iodide Detector (detector for measurement of uranium enrichment)
Nb	niobium
Nb_2O_5	niobium pentoxide
NCS	Norwegian Continental Shelf
Nd-147	neodymium-147 (^{147}Nd)
NEA	OECD Nuclear Energy Agency
NEPTUNO	Nuclear European Platform for Training and University Organizations
NGL	Natural Gas Liquids (a collective term for grades of liquid petroleum)
NGU	The Geological Survey of Norway
Np-239	neptunium-239 (^{239}Np)
NPT	Treaty on the Non-proliferation of Nuclear Weapons
NRPA	The Norwegian Radiation Protection Authority
NTNU	Norwegian university of Science and Technology
NVE	Norwegian Water and Energy Directorate (Norges Vassdrags- og energidirektorat)
OECD/NEA	OECD Nuclear Energy Agency, Paris
OED	The Norwegian Ministry of Petroleum and Energy
ORNL	Oak Ridge National Laboratory, USA
P	phosphorus
Pa-231	protactinium-231 (^{231}Pa)
Pa-233	protactinium-233 (^{233}Pa)
Pa-234	protactinium-234 (^{234}Pa)
Pb	lead
Pb-Bi	lead-bismuth
PHWR	Pressurized Heavy Water Reactor (known as CANDU)
Pm-149	promethium-149 (^{149}Pm)

Po	polonium
Po-208	polonium-208
Po-209	polonium-209
Po-210	polonium-210
Po-216	polonium-216
ppm	parts per million
PRIS	IAEA Power Reactor Information System
PSI	Paul Scherrer Institute, Switzerland
Pu-238	plutonium-238 (^{238}U)
Pu-239	plutonium-239 (^{239}Pu)
Pu-240	plutonium-240 (^{240}Pu)
Pu-241	plutonium-241 (^{241}Pu)
Pu-242	plutonium-242 (^{242}Pu)
PuF_3	plutonium trifluoride
PuO_2	plutonium dioxide
PUREX	plutonium uranium extraction prosess
PyC	pyrolytic carbon
R&D	Research and Development
Ra	radium
Ra-226	radium-226 (^{226}Ra)
RAR	Reasonably Assured Resources
RCN	The Research Council of Norway
REE	Rare Earth Elements
Rn	radon (Rn-222)
Rn-220	thoron (a radon isotope)
Ru-100	ruthenium-100 (^{100}Ru).
RWTH Aachen	Die Rheinisch-Westfälische Technische Hochschule Aachen
SCK-CEN	The Belgian Nuclear Research Centre
SCWR	Supercritical-Water-Cooled Reactor (Generation IV concept)
SFR	Sodium-Cooled Fast Reactor (Generation IV concept)
SGMP	Sol-Gel Microsphere Pelletization
SiC	silicon carbide
SINQ	Swiss spallation neutron source at Paul Scherrer Institut (PSI), Switzerland
SINTEF	Selskapet for INdustriell og TEknisk Forskning
SiO_2	Silica (Silicon Dioxide)
SKB	Svensk Kärnbränslehantering AB
SKC	The Swedish Centre for Nuclear Technology
SKI	The Nuclear Power Inspectorate in Sweden
SNS	The Spallation Neutron Source (accelerator-based neutron source being built in Oak Ridge, Tennessee, USA)
Sv	Sievert
SWU	Separative Work Unit (used in the enrichment process)
$T_{1/2}$	half-life (time to reduse the amount of radioactive material to the half)
TBP	tributyl phosphate
Tc-100	technetium-100 (^{100}Tc)
Tc-99	technetium-99 (^{99}Tc)
TD	theoretical density
Th	thorium
Th-230	thorium-230 (^{230}Th)
Th-232	thorium-232 (^{232}Th)
Th-234	thorium-234 (^{234}Th)
ThO_2	thorium dioxide (thoria)
THOREX	thorium extraction process
THTR	Thorium High Temperature Reactor
Ti	titanium
Tl-208	thallium-208
TMSR	Thorium Molten Salt Reactor
TRU	transuranic (elements with atomic numbers greater than uranium)
TURF	Thorium-Uranium Recycle Facility
TWh	Tera Watt Hours (1 000 000 000 kWh)
TWh_e	Tera Watt Hours Electrisity (1 000 000 000 kWh_e)
U	uranium
U-233	uranium-233 (^{233}U), fissile isotope of uranium
U-234	uranium-234 (^{234}U)

U-235	uranium-235 (^{235}U), fissile isotope of uranium (0.7 wt% in natural uranium)
U-236	uranium-236 (^{236}U)
U-238	uranium-238 (^{238}U)
U_3O_8	Triuranium octaoxide (Yellow cake) (directly from the uranium mines) (An oxide form of uranium that is the most common chemical form) found in nature.
UC_2	uranium carbide
UCO	uranium oxycarbide
UF_6	Uranium hexafluoride (a compound used in the uranium enrichment process)
UiB	University of Bergen, Norway
UiO	University of Oslo, Norway
UMB	Norwegian University of Life Sciences at Ås, Norway
UNENE	University Network of Excellence in Nuclear Engineering
UNFCCC	United Nations Framework Convention on Climate Change
UNIK	University studies at Kjeller
UO_2	uranium dioxide
UOIT	University of Ontario Institute of Technology, Canada
USGS	The US Geological Survey
VHTR	Very-High-Temperature Reactor (Generation IV concept)
VVER	Russian version of the Pressurized Water Reactor (PWR)
WNU	World Nuclear University
WNUCC	WNU Coordinating Centre in London
wt%	weight percent
Xe-135	xenon-135 (^{135}Xe)
Y	yttrium
zircon	zirconium silicate ($ZrSiO_4$)
Zr	zirconium
β_{eff}	delayed neutron fraction
$	reactivity unit, 1$ is the reactivity insertion that will make a critical reactor prompt critical

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