A CONCEPTUAL FRAMEWORK FOR THE COMPARISON OF SMALL MODULAR REACTORS BASED ON PASSIVE SAFETY FEATURES, PROLIFERATION RESISTANCE AND ECONOMIC POTENTIAL

by

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"An academic reactor or reactor plant almost always has the following basic characteristics: (1) It is simple. (2) It is small. (3) It is cheap. (4) It is light. (5) It can be built very quickly. (6) It is very flexible in purpose. (7) Very little development will be required. It will use off-the-shelf components. (8) The reactor is in the study phase. It is not being built now.

On the other hand a practical reactor can be distinguished by the following characteristics: (1) It is being built now. (2) It is behind schedule. (3) It requires an immense amount of development on apparently trivial items. (4) It is very expensive. (5) It takes a long time to build because of its engineering development problems. (6) It is large. (7) It is heavy. (8) It is complicated."

Admiral Hyman G. Rickover, U.S. Navy, 1953

1 Contents

2 Introduction

Traditional energy forecasting envisions a rapid increase in energy demand. According to the International Energy Agency (IEA), the world energy demand may increase up to 47 per cent by 2035 if we continue along the current line already formally adopted and implemented policies. If we assume the introduction of some new measures, on a relatively cautionary basis, the IEA predicts that the energy demand increase could be reduced to 36 per cent. The so-called 450 Scenario[1], in line with current modelling practices and taking into consideration expected changes in the energy mix, suggests a maximum increase of around 22 per cent [1]. In other words, according to the IEA, unless we introduce more potent new energy saving measures on top of the already announced policy commitments, chances are we will be falling short of our GHG mitigation goal despite the advances in energy efficiency and policy made in previous years.

Although, the Global Energy Assessment does not agree with the IEA in terms of scenario development, its report also conveys the message that if we are to have a chance of limiting global warming to 2°C, a rapid reduction in global CO2 emissions from the energy sector is required [2].

In order to close the gap governments are faced with multiple possibilities on how to restructure their energy policies. Some of the options mentioned by the GEA are: (1) expanding the renewables portfolio, (2) improving energy efficiency, (3) modernizing fossil fuels; for example by stimulating the commercial deployment of Carbon Capture and Storage (CCS) or (4) by revitalising or expanding the share of nuclear power in the energy mix.

At present, nuclear energy constitutes a significant share (about 5%) of the world's total primary energy supply [3]. Consequently, without nuclear energy our current CO_2 emissions would, most likely, have been an equivalent percentage higher, depending on the replaced fossil fuel source. The argument can therefore be made that increasing our reliance on nuclear energy can further the global ambition to reduce greenhouse gas (GHG) emissions. When limiting our view to electricity generation, only trace amounts of GHGs are emitted during nuclear power plant operation. When expanding our view to span over the entire lifecycle, only very small amounts of CO_2 are emitted as a result of secondary processes like transport and excavation of raw materials. A publication by Weisser (2006) [4] reported, after having combined numerous figures on the GHG emissions of various electricity generation technologies, that the cumulative lifecycle emissions of nuclear power generation lie between 2.8 and 24 gCO_2eq/kWh. This estimate rates nuclear power to have a relatively similar GHG emission level as hydroelectric power (1-34 gCO_2eq/kWh) and onshore wind turbines (8-30 g gCO_2eq/kWh) [4].

Significant investments in nuclear infrastructure, made in the second half of previous century, have resulted in a large installed electricity generating capacity. This makes nuclear power on par with all but the most established non-GHG emitting energy resource[2] and has resulted in a high technological familiarity in a number of countries.

Nuclear power can also contribute to increased national energy security; the addition of nuclear power to a country's energy portfolio diversifies its power infrastructure and thereby diminishes its dependency on fossil resource imports from foreign countries[3][4]. Additionally, undertaking a nuclear construction project might lead to economic growth by means of job creation. Also, depending on certain geographical

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[1] The 450 Scenario sets out an energy roadmap, which is consistent with the goal of limiting the global average temperature increase to 2 degrees Celsius.

[2] Hydopower is currently the largest non-GHG emitting energy with a global installed capacity of 952 GW. Nuclear power has an installed capacity of 371 GW; its expansion has stagnated from 1989 onwards [3]

[3] The degree of this import independence is dependent on the availability of national uranium or thorium resources within national boundaries.

[4] It should be noted, that following the Fukushima accident the energy security in Japan has actually decreased as a result of mandatory safety inspections of its nuclear reactor fleet, leaving parts of the country's electricity grid undersupplied.

considerations and assumptions, nuclear power could be more cost effective than alternative forms of low-carbon electricity generation [5].

Despite all this, the electricity supplied by means of nuclear power has dwindled since 2002, when the number of online reactors reached a maximum of 444 units. Although the total net electrical capacity kept increasing steadily [6], it is expected, following the events in the wake of the Fukushima Daiichi nuclear disaster that this trend will come to end (e.g. [7]). Furthermore, for several countries the share of nuclear energy in their electricity mix was already in regress. It is noted in the World Nuclear Industry Status Report 2010-2011 [8] that the three countries which have phased out their nuclear power completely (Italy, Kazakhstan and Lithuania), together with Armenia had their nuclear peak power production in the 1980s. Furthermore, several other countries peaked in the 1990s (e.g. Belgium, Canada, Japan and Great Britain) and between 2001 and 2005 an additional seven countries experienced a decrease in their nuclear electricity supply (Bulgaria, France, Germany, India, South Africa, Spain and Sweden) [8]. However, the validity of this statement can be drawn into question considering that these events have only recently occurred and several sources sketch a different picture for Bulgaria [9], India [10] and France [11].

What cannot be denied however is that the current reactor fleet is aging [12]. There are several major problems standing in the way of the renewal of nuclear energy, these include: a short-term manufacturing bottleneck, a shortage of specially trained workers and managers, a sceptical financial sector and public opinion, the latter being of particular importance in the aftermath of the Fukushima disaster. Furthermore, the Global Financial Crisis followed up by European Sovereign Debt Crisis has done nothing but aggravate the situation, in particular for the countries who are currently entertaining the option of starting a nuclear program. Nonetheless, in a recent report the International Atomic Energy Agency (IAEA) announced that no less than 65 had expressed an interest to introduce nuclear power into their energy mix [13]. Of these 65 countries, 21 are in Asia and Pacific region, 21 are located in Africa, 12 are European and 11 are situated in Latin America.

Conventional modelling practice, executed by IEA, forecasts that by 2035 the share of generation from non-hydro renewable energy sources (e.g. wind, biomass, solar, geothermal and marine) may have increased more than five-fold, from 3% in 2008 to 16% by 2035 [1]. Furthermore, primary energy intensity of the economy is decreasing as a result of, among other things, energy conservation programs [14]. Therefore, when approaching the issue from a system wide perspective, countries planning on implementing nuclear power infrastructure should consider whether a nuclear power approach is compatible with a system dominated by renewables and energy efficiency.

It is suggested in reference [8] that such co-dominance is not possible. The first argument given is that overcapacity kills efficiency incentives. Large, centralized power-generation units like nuclear power plants (NPPs), are inclined to produce structural overcapacities, which will usually lead to lower electricity prices. These will discourage energy efficiency, because consumers tend to over-consume in times of resource abundance, which is inefficient. In other words, it is viewed that the power output of a typical NPP is too large. Another disadvantage of their size is that in the case of a systems failure a billion watts are removed from the grid [8], occasionally for extended periods. This gives rise to doubt, concerning their added value when it comes to improving the overall energy security.

The second argument is that renewables need flexible complementary capacity; increasing integration of renewables into the energy mix requires the support of medium-load complementary facilities set up in load-following capacity, not the presence of large, inflexible NPPs. However, dr. Lambertz, CEO of RWE Power, is of the mind that the growing energy demand cannot be met by the mass grid integration of renewables at the moment and that conventional power plants will still have a role to play in the foreseeable future. Furthermore, he stipulates that the continued development of renewables will be reflected in greater price volatility following the increased variability in power plant dispatch [15]. Table 2.1 provides an example of one of the challenges facing power plants and system operators; the typically large differences in the availability of renewable energy sources, in this case wind energy.

In Germany, for example, renewable system operators are statutorily entitled to a priority connection from the grid operator over fossil fuel powered energy stations (Renewable Energy Sources Act, Part 2, Chapter 1, Section 5(1)). Effective this means that when there is very little demand and an abundant production of renewable resources, conventional power sources, such as nuclear, will need to scale back their production in order avoid financial consequences. In the first weekend of October 2009 this resulted in NPP operators having to pay €500 per mega-watt hour for their electricity to be taken off. However, dr. Lambertz argues that the direct opposite situation can have equally devastating consequences. On the 6th of January 2010 only 300 MWe, out of an installed capacity of 25.800 MWe of wind turbines, was feeding electricity into the grid. In other words only 1.2% of the installed capacity could be used [15]. The same fluctuation profile can be applied to solar power generation. It has been reported that, in the period from February to July 2009, on average only 728 MWe was available per week, while the installed capacity at the time was roughly 9.5 GWe [15].

National Minimum and Maximum Wind Production 2010

	One hour	Denmark	Germany	Ireland	Great Britain
Minimum	MW	2	103	0	0
	Share of max	0.1%	0.5%	0.0%	0.0%
	Date and hour	9 Mar 20-21	29 Jun 09-10	13 Oct 11-12	4 May 02-03
Maximum	MW	3,333	21,204	1,214	2,058
	Date and hour	11 Dec 11-12	12 Nov 14-15	26 Dec 16-17	12 Nov 11-12

Table 2.1

Taken from Bach (2011) [16]

When taking the aforementioned into consideration, it can no longer be considered definitive that the network congestion is fully caused by the presence of conventional power plants. In 2009, Germany, a country with a relatively large share of renewables, had an installed wind power capacity of roughly 25.8 GWe and an installed PV capacity of 9.8 GWe [17]. Considering that the sun is often obscured by clouds (and of course not available at all during the night) and the winds does not always blow at a speed within the design specifications of a typical turbine, it is not unheard of that capacity availability can vary by 20 GWe or more on a daily basis. Taking into account that most countries plan to expand their renewable portfolio in the coming years, this generates a 'capacity-at-risk' or intermittency problem, which is only likely to increase in the near future. It is estimated that in order to support the intended growth of renewable energy, each MW solar or wind power added to the grid will require a 0,9 MW of base-load power back-up [15]. This point is illustrated in Figure 2.1, where the large spread in wind power output would stand to benefit from a base-load technology which is also able to operate in load-following mode.

Although considered a base-load generator, as described in reference [18], most second generation NPPs operating in the world today have been designed with strong power output manoeuvrability in mind. NPPs are not commonly known to possess such a load following capacity, due to the superfluousness of operating in any other capacity than base-load. This is a direct consequence of the relatively modest share nuclear power represents in a typical countries electricity mix. However, in France and Germany for example, NPPs typically operate in load following mode and follow variable load program which incorporates as much as two significant power changes per day [18]. Moreover, the minimum requirements for the manoeuvrability of modern reactors are enclosed in a series of documents known as the European Utilities Requirements (EUR). A NPP must be capable of daily load cycling operation between 50% and 100% of its rated power output, and must have a ramp rate of 3 to 5 per cent of its rated power per minute.

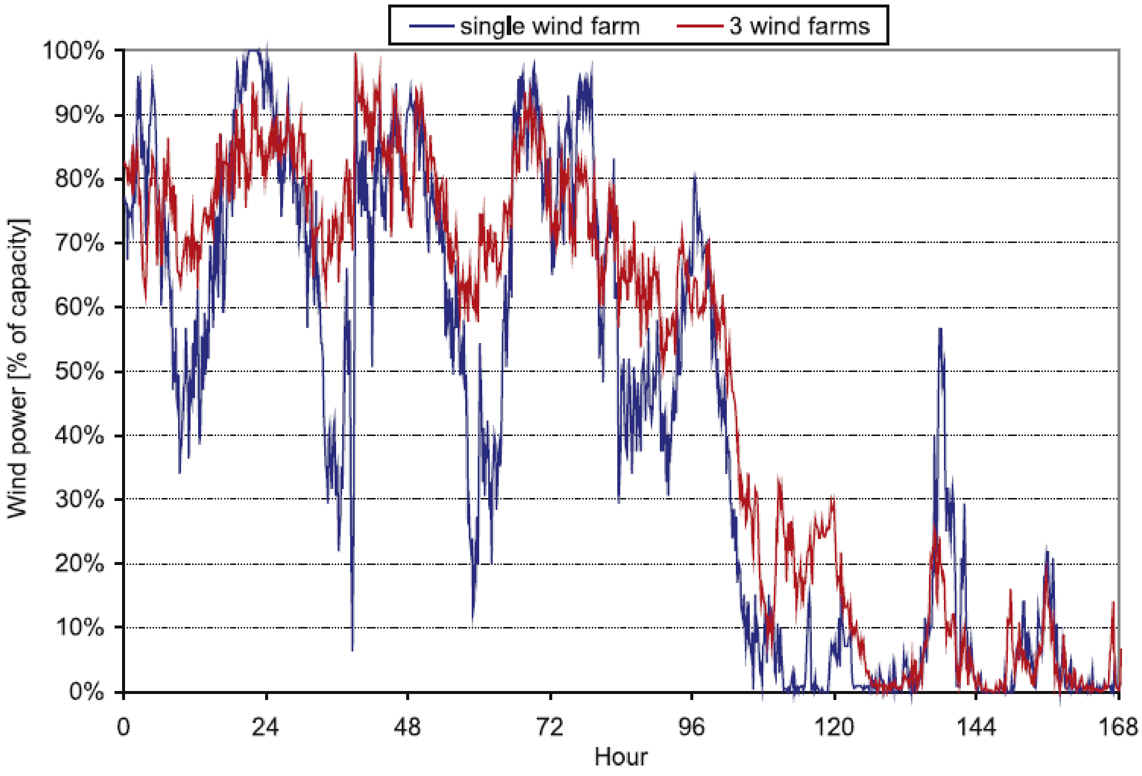

Figure 2.1| Typical wind load profile over a 7 day interval

Taken from Goić et al. (2010) [19]

The economic viability of nuclear power is also an issue. High capital costs and the technical complexity involved with NPP construction tend to increase the risk of construction delays and cost overruns and are therefore a major concern for the utilities sector, investors and lenders [20]. The same conclusion is reached by the IAEA, who published the following on the finance of new nuclear power plants:" *Economic viability is inescapable; no one is likely to invest in a financial black hole, nor build nuclear power plants for environmental reasons, unless they are demonstrably profitable and among the most cost efficient solutions*." Taylor (2011) [21] argues that although no intrinsic barrier exists to project financing[5] NPPs, banks will only agree to project finance a NPP after a successful construction track record has been established for that particular design. This, to some extent, requires the cheap capital provided by project finance.

Small Modular Reactors (SMRs) might be capable of breaking through the paradoxical nature of building commercially viable NPPs in high income economies. In recent years the IAEA has published several extensive reports on the status of SMR[6] development as well as several promising design features [22] [23]. SMRs are a subset of the Small Reactor group. Small refers to their equivalent electric power, which is 300 MW or less [24], and modular means that they can be constructed in an incremental fashion. SMRs have the potential to be small power output, flexible base-load power generators. This feature could allow them to perform the role of back-up system for the intermittent renewable capacity. In other words, SMRs could be supportive of the development of the renewable energy capacity, while upholding grid reliability and generating negligible amounts of GHGs. Therefore, SMRs might help to bridge our current fossil fuel based energy order with a future fully renewable based energy order.

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[5] Project finance is the most common form of finance for capital intensive projects. It has the advantage of efficiently allocating risk among project participants. This reduces the required return and allows for a higher debt ceiling
[6] Currently there are two definitions of such reactors which are in common use in the examined literature. Small and Medium-sized Reactors, which is the official definition maintained by the IAEA (e.g. [22]), and Small Modular Reactors

It should be noted that the development of SMRs and nuclear power in general, should be done within the context of furthering the ambition to transition into a sustainable society. As discussed in Turkenburg (2003) [25], debates on the advantages and disadvantages of nuclear power have culminated in the identification of the primary resistances to the adoption of nuclear power as a tool to reduce global GHG emissions. Among other things, included in this list of obstacles are: the safety aspects of NPPs, the management issues related to the storage of (high-level) nuclear waste, the dangers involved with the proliferation of nuclear material, the scarcity of fissile material and the cost of nuclear energy. The aforementioned (perceived) disadvantages of conventional nuclear power generation can utilized as restrictive boundary conditions in the evaluation of the various SMR designs. In this thesis the focus will be on three aspects of the list proposed by Turkenburg (2003) [25], these are: (1) the safety of the SMR designs, measured in terms of their passive safety; (2) the proliferation resistance of the underlying reactor technology and (3) the projected economic competitiveness of the design. The analysis will be restricted to the deployment potential of land-based SMRs, additionally as discussed in paragraph 4.1, the progression of the reactor design should be far enough for cost projection and certain technical specifications to be available.

Of particular interest are the economics of SMRs. Therefore, various known factors that influence the performance and competitiveness will be incorporated into the proposed mixed-modelling approach. Regarding several of the input values, although vendor data on several of the designs is available for comparison, the economics of nuclear power have been a source of great controversy in the past two decades [26] and major cost overruns haven given rise to mistrust in industrial estimates. Consequently, in this thesis an attempt will be made to construct independent estimates, either based on operational data or independent research.

In order for SMRs to be selected for use in the anticipated markets it must be demonstrated that they can be competitive. Therefore, the economics section will include quantitative estimates of the benefits of relying on novel design and deployment techniques while foregoing the advantages of economies of scale. As stated in reference [27], frequently encountered arguments for favouring SMRs over large reactors (LRs) include, but are not limited to:

1) Reduced design complexity as a result of the use of safety features which are more appropriate for the smaller reactor sizes.
2) The economies of mass production resulting from the use of multiple pre-fabricated modules.
3) The option of incremental capacity increase with the possibility of accelerated learning.
4) The sharing of common equipment and facilities, better known as co-siting.
5) Shorter construction periods and the option to apply ‘unit timing’, which is the option to spread out the investment over a longer timeframe if the investment climate becomes unfavourable.

However, a great deal of uncertainty still exists regarding if and how these advantages can be obtained as they also seem to be dependent on certain market variables (e.g. interest rate) and therefore need to be assessed on a per case basis.

In Chapter 3 of this thesis a detailed overview will be given of the background of SMRs, clarifying which studies have been carried out, what problems were encountered, what the current gaps in our knowledge are and which questions still require answering – based on a literature review. Chapter 4 aims to clarify the methodology and the modelling framework used in this research. It will define the boundaries of the research performed. Also it contains a concise statement of the research questions this thesis aims to tackle, including how these questions are to be answered and why it’s worthwhile to answer these questions. In chapter 5 an attempt will be made to provide an overview of the economic, safety and proliferation resistance features of the chosen SMR designs. The outcome of the analysis will be presented and discussed in chapter 6. Finally, in chapter 7 the most important findings and conclusions that can be drawn from this study will be summarized.

3 Literature Review

3.1 Competitiveness of SMRs

SMRs have the potential to contribute significantly to a decrease of GHG emissions due to our energy consumption and enhance energy security. Therefore, a great deal of research on SMRs has been devoted to their development and competitiveness in relation to larger NPPs and other conventional power generating facilities.

It has been suggested that SMRs could be viable base load alternatives to several older, smaller and less efficient coal fired power plants. These could then be retired on short notice or passed over for retrofitting with carbon capture and storage technology [28].

The Nuclear Energy Agency concludes that one of the main arguments in favour of SMRs is that: (1) they would also be suitable for deployment in areas with only small electrical grids or remote locations and (2) that due to their smaller upfront capital costs the financial risk of deploying a SMR is considerably smaller than for a LR [29]. Additionally, the added flexibility of incremental capacity increase could increase the attractiveness of building nuclear power plants for potential investors.

Regarding their role, it has been suggested that SMRs should be deployed primarily in regions where the potential for other forms of carbon-free electricity, such as solar or wind is too low to meet demand requirements [28]. Additionally, they could also be deployed in case of technical or market constraints, such as limitations on the transmission capacity or electricity demand growth projections that would render the construction of a GW-scale NPP a poor choice. In light of this, it is generally acknowledged that the SMRs which becoming available for commercial deployment in the next decade will be meant for different markets than LRs. Arguments like a desire to have more distributed electricity supply or a greater variety of electricity generating sources are often encountered and seem to be incompatible with large scale nuclear power deployment. Therefore, it is important to note that the markets for LRs and SMRs have diverging conditions and restrictions with respect to investments, siting, grid conditions, infrastructure, operation & maintenance and applications (e.g. [27], [29]).

Regarding the investment, it is a well-documented phenomenon that the overnight costs for LRs have been increasing in past decades. Even in France, the country which arguably underwent the most successful nuclear scale-up in an industrialized country, the construction costs were subject to substantial escalation over time [30]. It should be noted that this pattern of 'forgetting by doing' continued despite the French nuclear expansion program benefiting greatly from centralized decision making, regulatory stability and dedicated efforts for standardized reactor designs which allowed for a great deal of knowledge spill over. This might indicate that although the nuclear industry is often inclined to point out that public opposition and continuously changing regulations are the source of cost escalations, also the technology itself might be at fault. Contemporary conventional NPP designs are large in scale, cumbersome and require extensive complexity management capabilities during their construction and in-operation. It is noted by Grubler (2010) [30] that these technological characteristics limit essentially all classical mechanisms of cost improvement, such as standardisation, serial construction and large numbers of quasi-identical construction experiences, that might lead to technological learning. The only exception is economies of scale which result from the increase in unit size.

The opposite cost trend has been reported for the development, construction and operation of serially produced nuclear propulsion systems in the former Soviet Union. A study performed by the OKBM analysed the economics of serial production for hundreds of reactor systems which were constructed in-between 1963 and 1992 [31]. Some of the major findings of this study were that serial production could reduce the costs of propulsion reactor systems by approximately 30 to 35 per cent. The primary causes of this cost decrease were identified to be an increase in the productivity of labour, a decrease in R&D costs and a reduction of the costs of commercial equipment manufacturing, assembly and repair. Nuclear propulsion reactors exhibit similar features to small scale land based nuclear power systems and many companies that are developing SMRs have the construction of nuclear propulsion systems as one of their core competencies.

Rosner and Goldberg (2011) [28] state that in a number of countries the manufacturing facilities required to develop SMR technology can be developed domestically. A SMR industry could potentially be constructed on the foundations of an already existing domestic manufacturing expertise. Already present domestic industries that might have a beneficial effect on the beginning of a successful SMR program are naval shipbuilding facilities with experience with nuclear propulsion systems as well as nuclear component retailers and several types of equipment plants. For example, Babcock & Wilcox, the designing company of the mPower reactor (one of the SMRs investigated in this thesis), was also involved with the construction of the world's first nuclear submarine, the U.S.S. Nautilus.

Unsurprisingly, the U.S., which possesses a large under-utilized nuclear industry, is the first country in which letters of intent have been submitted to the Nuclear Regulatory Committee, requesting a SMR design licences.

It is reported that any nation possessing the required nuclear expertise could stand to benefit from the expected sustainable job growth within a wide range of professions, such as design, manufacturing, supply and construction activities [28]. Furthermore, it is indicated that some of the simpler SMR designs might even be beneficial to the national industries of countries with a lower established level of nuclear competence [27].

Although admittedly these effects are difficult to quantify, similar the macroeconomic and social effects have been known to affect the outcomes of the relative competitiveness assessments of alternatives of large investment decisions. An example is the Joint Strike Fighter Program, where many countries were persuaded to contribute to the development of a next generation multirole fighter aircraft, in return for the promise of increased turnover rates for national producers of specific component items such as engines and avionics [32]. A similar division of labour might be explored for certain regions, in which a high nuclear competence nation procures a nuclear commitment from its neighbours by pledging to place orders for (lower safety-grade) mechanical components in the neighbour's home-country industry.

An extension of this design philosophy is a multilateral approach in which the nuclear fuel cycle is set up in a so-called hub-and-spoke configuration. This concept envisions the creation of large regional/internal energy parks, referred to as 'hubs', that export a range of nuclear products such as fuel, hydrogen and even serially produced small sized sealed reactor cores, in the 20-100 MWe range, to the client, or 'spoke', states. At the end of their reactor lifetimes the modules would be sent back to their regional supplier to be decommissioned [33]. Apart from several significant advantages to the proliferation resistance, which will be discussed in paragraph 3.3, this multilateral approach might be used to address several of the issues which currently plague the nuclear fuel cycle. It is reported that a regional nuclear programs could function as a confidence-building activity due to its multinational character and would have a high follow through rate considering that project abandonment is politically more costly [34].

It should be noted that the traditional forerunners of nuclear power technology have not been very supportive of this idea as of yet. On the one hand this can be explained by the prerequisite that some control over any facility operated in this institutional framework has to be relinquished [34]. On the other hand it acknowledged that as of yet little economic analysis has been performed on these energy parks [35]. However any nations considering such a set up should at the very least have: (1) a fair amount of success in regional cooperation and economic integration, (2) a legal framework giving any spoke country guaranteed access to the services of energy park in exchange for foregoing an indigenous nuclear program and (3) a regional nuclear regulatory authority and institutions tasked with governing the nuclear energy infrastructure [35].

In the foreseeable future, however, SMR designs will most likely remain a national enterprise. The state of affairs as of 2011 was that the mPower and NuScale reactors were expected to be approved within 6 years, with commercial deployment expected around 2020. Other SMR designs currently under development have an unknown approval time. One possible exception is the Toshiba 4S reactor [36] for which a letter of intent was submitted to the US Nuclear Regulatory Commission in early 2007. However, owing to its innovative sodium cooled design, a demonstration reactor will first need to build in order to be able to verify the design and its safety features. The construction of the proof of concept plant is expected to start before 2015 [27].

Typically, the designs which are furthest along are the integral PWRs. Next to the mPower and the NuScale reactor, the 335 MWe IRIS reactor, which was being developed by an international consortium led by Westinghouse, was reported to have reached an advanced design stage. However, the Nuclear Energy Agency reported that Westinghouse discontinued the development of the IRIS reactor in late 2010, in favour of an alternative integral design known as the Westinghouse SMR [29]. The World Nuclear News reported that several major US nuclear utilities have entered into an alliance with Westinghouse to support the licensing and deployment of the Westinghouse SMR in order to secure investment funds from the U.S. Department of Energy (DoE) [37]. This was done in response to an announcement made by the DoE in January 2012 that it wanted to facilitate SMR development by means of a 452 million US$ grant spread out over a five year period and two reactor designs [38]. NuScale Power has also indicated that it is interested in the DoE offer and has begun to form an alliance of its own with NuHub in attempt to secure one of the two available spots [39].

Other noteworthy designs are the Argentinian CAREM[7] reactor - of which a 200 MWe prototype reactor is scheduled to be built in the Formosa province following a decision by the Argentinian senate approving the construction of a fourth nuclear reactor - and the Chinese High Temperature Reactor Pebblebed Modular, or HTR-PM in short. The latter has a 250 MWth design generating approximately 100 MWe by means of a supercritical steam turbine loop. The construction of the first commercial NPP operating on HTR-PM technology has already been started and demonstration of a full scale power reactor module has been planned for 2013 [27].

As a result of a general lack of detailed SMR engineering data, the public availability of cost data is limited. This results in a great deal of uncertainty regarding the specific capital costs. Regardless, an IAEA Report [27] states that SMR designs would be less demanding of their potential customers. For example, the addition of capacity in smaller increments favours those with smaller upfront investment capabilities. Moreover, the lower power output is favourable for region or countries with smaller grids and the lesser degree of operational complexity and maintenance requirements is obliging to areas with a lower availability of supporting infrastructure and human resources. Furthermore, smaller safety regions may enable placement closer to the intended consumers. This means that population centres could potentially also benefit from non-electrical nuclear products such as desalination, district heating and hydrogen production.

An important side note made in the IAEA Report on SMRs [27] is that a small sized reactor does not necessary imply a small sized NPP and incremental capacity additions up to the point where the power output resembles that of a GW size NPP is a possibility. On the one hand this would be in violation with one of the conditions under which nuclear power and renewable energy sources can coexist; building large installation which could renders the addition of smaller renewable increments redundant due to structural overcapacity. On the other hand it could allow SMRs to capture the cost advantage that inherently eludes them as a result of economies of scale. Irrespective of this the IAEA seconds that SMRs possess the characteristics that Grubler (2010) [28] found to be lacking in LRs [27], namely their ability to be deployed in series, potentially leading to accelerated learning, as well as standardization allowing serial fabrication.

[7] Spanish: Central ARgentina de Elementos Modulares

Furthermore, most studies on the competitiveness of SMRs, including the recent Nuclear Energy Agency (NEA) report [29], limit their analysis to government funded projects. One of the limitations is that the validity of the assumption of lower than market conform interest rate assumptions does not necessarily apply to privately owned companies. For example, the importance of total costs, the height of the fixed costs and the capital-at-risk are known to matter more for privately owned companies, than the levelized electricity cost. This is in line with the IAEA, who also conclude that the competiveness of SMRs will most likely be dependent on more factors than cost assessments based on the economies of scale consideration [27]. This is based on the argument that when modular construction and serial deployment are involved, the financial assessment becomes less straight forward. Furthermore, privately owned companies often do not possess the financial resources to continue funding a project which is past its predetermined delivery date. Considering that nuclear reactors are notoriously known for construction delays, this could result in an increased construction project default, which considering the capital-at-risk, could result in an increase default rate of the company itself when the project drags out too long. Although SMR characteristics are typically considered to be more favourable under market conditions, uncertainty analysis should play a crucial role in the assessment procedure. In general, any cost assessment could stand to benefit from incorporating the effects of long-term macroeconomic and social risk [27].

3.2 Passive Safety Features

Traditional reactors largely depend on active safety features, which typically involve a command initiating an electrical of mechanical operation. Fully passive safety systems only dependent on physical phenomena (e.g. gravity, convection or resistance to high temperatures) and therefore do not require the proper functioning of engineered components. One such mechanism which finds broad usage in SMRs is a high surface to volume area relative to LRs. When properly implemented this feature largely makes the active heat removal systems, commonly found in LRs, redundant [40]. As a result, passive safety systems are the preferred method of decreasing the likelihood of accident scenarios.

In previous studies it has been concluded that in several reactor types, such as PWRs, advanced heavy water reactors, sodium cooled fast reactors and lead-bismuth cooled fast reactors it has become common practice to incorporate passive safety systems as safety grade back-up systems, in which safety grade is an indicator for the likelihood of failure of a certain component. The primary safety system, used during normal operation is usually still designed as an active system however [29]. It should be noted that any reactor design which has both passive and active heat removal systems, but is able to fully function on the use of only one, has an added layer of defence-in-depth. The reasoning behind the choice to make the primary heat removal system an active one, is that in many accident scenarios the primary system will remain functional and can therefore be used in conjunction with a dedicated passive safety system.

Various reactors of the same type utilize diverse plausible combinations of active and passive safety. It can therefore be assumed that no golden standard has been developed dictating the optimal cooperation between active and passive safety systems [29]. Although generally considered to be less reliable, one deliberation to incorporate a greater share of active safety systems into a design is prompted by the possibility of attaining more favourable plant economics. Active safety systems are usually better understood and require less space than passive safety systems. However, irrespective of the choice between active and passive safety systems, one common practice that does exist is: to design each independent system of being capable to handle the entire heat removal need of the reactor.

It has been reported that the current trend is to use natural convection in the primary cooling circuit when the rated power output is less than 150 MWe, but this is to be considered more as a guideline than as a rule [29].

According to the IAEA, only components which fall into the following categories may be designated as being ‘passive’ [41]:

A Physical barriers and static structures (e.g. pipe wall, concrete buildings
B Moving working fluids (e.g. cooling by free convection)
C Moving mechanical parts (e.g. check valves)
D External signals and stored energy (passive execution/active actuation, e.g. scram systems)

Passive systems are commonly attributed with higher reliability features, due to their lower unavailability as a result of hardware failure and human error compared to active systems. However, there is always a nonzero likelihood that physical phenomena might lead to a failure mode once the system is in operation. Deviation of the natural forces from their expected values may lead to system impairment. This especially applies to category B [42], which is the passive safety feature category of choice for most SMR designs under consideration in this thesis.

The IAEA has recognised the concerns that may arise as a result of the small amount of experience the industry has with similar systems. They summarized the most notable issues in a recent report [43]. These are:

- "Reliability of passive safety systems may not be understood as well as that of active safety system."
- "There may be a potential for undesired interaction between active and passive safety systems."
- "It may be more difficult to 'turn off' an activated passive safety system, if so desired, after it has been passively actuated."
- "Implications of the incorporation of passive safety features and systems into advanced reactor designs to achieve targeted safety goals needs to be proven, and the supporting regulatory requirements need to be worked out and put in place."

Several methodologies are being developed worldwide to quantify the reliability of passive safety systems. Several pronounced approaches for the assessment of thermal hydraulic passive safety systems are the European Union's RMPS [44], the Indian APSRA methodology [45] and the IAEAs Coordinated Research Project, which aims to develop a common approach for assessing passive safety systems [46]. However, all of these methodologies require detailed inputs, which at this stage of reactor development, are publicly unavailable. Specifically, a complete probability safety assessment would require (1) the collection of large volumes of data, (2) the identification of an event fault tree with corresponding damage states, (3) an assessment of how human actions shape events and (4) a database on the reliability of specific plant systems and components. Additionally, this process requires specialised software. The approach is relatively well documented and understood for actively forced circulation safety systems.

3.3 Proliferation Resistance

The detonation of the first nuclear weapon in the course of warfare heralded the beginning of new era in global security relationships. The risk of nuclear weapons proliferation has been a central focus of international relations and was partially responsible for the founding of the IAEA in 1957. While nuclear power presents a viable solution to meet our energy demand while at the same time reducing our GHG emission, the associated risk of spreading nuclear weapons technology, also known as proliferation, poses a serious concern to the general public that needs to be addressed.

It is reported in Kiriyama et al. (2000) [47] that the exact definition of proliferation resistance tends to vary depending on which source is consulted. The risks of nuclear proliferation are typically clearer and defined along the line of the acquisition of weapons grade material by either a national or a subnational organisation.

A potential proliferator has many possibilities for acquiring nuclear material in the two predominant nuclear fuel cycles; once-through and recycling. The phases with the highest level of risk attached to them are the: enrichment of uranium, fuel fabrication, reactor operation and the back of the fuel cycle.

Furthermore, in the event of spent fuel recycling, the reprocessing phase can be added as a point in the fuel cycle with a heightened proliferation risk.

As explained by Close et al. (1995) [48] the major components affecting the proliferation resistance are (1) the material form, (2) the physical access, (3) the safeguards and security and (4) the conflicts that arise when trying to achieve the aforementioned points.

The material form of the nuclear material describes the radiological, chemical and physical characteristics that govern the complexity of acquisition and subsequent processing for use in a nuclear weapon. In this list the radiological component refers to radioactivity of the spent fuel, which determines whether it can be directly handled or not. This component is further subdivided in Becker et al. (2007) [49], where the proliferation resistance of a design is determined by: (1) the amount of plutonium discharged from the specific reactor design at the end of the fuel cycle, (2) the specific decay heat of the spent fuel and (3) the specific spontaneous neutron yield. A reactor design with a good proliferation resistance has a low plutonium discharge, a high level of specific decay heat and a high specific spontaneous neutron yield. The chemical component refers to the easy with which nuclear material can be processed for use in a nuclear weapon. Fuel material such as plutonium oxides requires very little in terms of processing in comparison to the low plutonium concentrations encountered in spent fuel. The physical component refers to the ease with which nuclear materials can be transported to a processing location. This component is especially influenced by size and weight. To extend the previous example, a container in which plutonium is stored as oxide powder is much easier to transport than plutonium in spent fuel form.

Physical access is described as the ease of access to nuclear materials and is highly dependent on the number and the types of barriers confining the material to a certain location and the ability of the barriers to cope with attempts to circumvent them [49]. Nuclear facilities that require little interaction between their personnel and the nuclear material can be imagined to possess a higher degree of proliferation resistance than facilities with a great deal of the nuclear material proceedings. One side note is that, although limited access typically improves the proliferation resistance with respect to physical access, the separation between the personnel and the nuclear material could also make the process of accounting for the nuclear material more precarious.

Safeguards and security comes in two varieties, domestic and international. Domestic safeguards are primarily concerned with material accounting and control and implementing measures for detecting and disrupting attempt to gain access to nuclear material. While domestic safeguards are typically involved with protection, international safeguards are concerned with verifying the accounting process, in addition to some containment and surveillance activities. There intended function is to confirm the assertions of the member states. International organisations such as the IAEA have developed their own criteria for the relative attractiveness of nuclear materials for use in nuclear weapons. These standards determine which types and quantities of nuclear material should have additional safeguards, considering that they are eligible for direct use in nuclear weapons. The approximate amounts of nuclear material for which the possibility of nuclear weapons fabrication cannot be excluded, also known as the ‘significant quantity’ (SQ) are given in Table 3.1. It should be noted that these quantities include the losses associated with conversion and manufacture. It should be noted that these numbers do not reflect the minimum material requirements for achieving critical mass.

Table 3.1gives an overview of the mass nuclear materials which constitutes an SQ for the chemical element plutonium and various isotopes of uranium. The table is divided into direct use and indirect use materials, of which only the direct use materials will be considered of interest in this thesis. This Indirect use materials is the generic term used to describe all nuclear materials that are not direct-use materials. These materials include low-enriched uranium which needs to be further enriched in order to become high-enriched uranium or irradiated in a reactor to produce Pu-239[8].

Material	SQ
Direct use nuclear material	
Pu[a]	8 kg Pu
^{233}U	8 kg ^{233}U
HEU ($^{235}U \geq 20\%$)	25 kg ^{235}U
Indirect use nuclear material	
U ($^{235}U < 20\%$)[b]	75 kg ^{235}U (or 10 t natural U or 20 t depleted U)
Th	20 t Th

[a] For Pu containing less than 80% ^{238}Pu.
[b] Including low enriched, natural and depleted uranium.

Table 3.1

Taken from IAEA Safeguards Glossary [50]

The accompanying values found in Table 3.1 are contested by Cochran and Paine (1995) [51] however, where it is claimed that the significant quantities displayed in Table 3.1 are not technically valid. Their argument hinges on three assertions. The first one is that if one took the *Fat Man* design deployed on Nagasaki in 1945 and replaced the 6.1 kilogram plutonium core with a 3 kilogram plutonium one, it would still have a respectable explosive force. This is reported to be a direct contradiction of the IAEA premise that the techniques required to field nuclear weapons utilizing a less than SQ of nuclear material, are only available to advanced nuclear weapon states. The second assertion is that several of the so-called 'highly sophisticated techniques' only available to advanced nuclear weapon states were designed and tested in the 1950s, more than half a century ago. For example, 'fractional crit' weapons; devices in which the core contains less than one critical mass of material, were first tested in 1951 [52]. Lastly, it is claimed that a well-designed safeguard program should recognise the fact that nuclear material can be obtained from multiple sources by setting the SQ to considerably lower levels than the minimum requirement for nuclear weapons manufacture. On a similar note it is stated that increasing the SQ to account for losses in conversion and manufacture is negligent due to the fact that significant portions of this waste stream can be recovered [51]. In summary, Cochran and Paine (1995) [51] state that in recognition of the fact that several key pieces of technical information are available in unclassified literature, it should be assumed that any proliferator has access to advanced weapons technology. Consequently they believe the SQs should be lowered to 1 kg of Plutonium and U-233 and 3 kg of U-235.

The reactor designs in this thesis are operated in two distinct fuel cycles, the once-through fuel cycle using uranium fuel and the mixed oxide (MOX) fuel cycle using uranium and plutonium fuel and applying reprocessing of spent fuel. In most cases the reactors are designed with fuel flexibility in mind, making the fuel cycle choice more of an economic consideration, as explained in paragraph 4.3.3.6.2, than a restriction

[8] The relevance of the boundary condition described in footnote (a), concerning effect of the isotopic composition of plutonium on its direct-use status, is discussed in paragraph 4.3.2.2

resulting from design limitations. Of these two, the proliferation resistance of the once-through cycle is best understood and is often also considered to possess the best proliferation resistance characteristics. This is because of the higher radioactivity of the spent fuel and the lack of a reprocessing step in which pure plutonium is separated from the heavy metal spent fuel mixture [53]. The conclusion of Pierpoint (2008) [53] is that the relatively weakest spots of the once-through cycle are the transportation steps located between fabrication and fuel-loading and between wet-cooling and dry (interim) storage. This makes intuitive sense, considering that mobile segments are more difficult to protect than stationary structures. However, considering that the fresh uranium fuel has a low enough enrichment to qualify as an indirect-use material, this does not directly aid the perpetrator in acquiring weapons-grade material.

Throughout most of the MOX fuel cycle, the proliferation resistance can be considered to be similar to the proliferation resistance of the once-through cycle. Several expected deviations might be that the proliferation resistance decreases in the front end of the fuel cycle due to the presence of plutonium in the MOX fuel rods and similarly a decrease in proliferation resistance in the back end of the fuel cycle because MOX reprocessing contains a pure plutonium stream [53].

One potential way to mitigate the risk of nuclear material proliferation is the hub-and-spoke concept, as introduced in paragraph 3.1. The reason for this is that all potentially proliferous activity such as fuel enrichment and spent fuel reprocessing would be confined to one region. If, additionally, the reactor design would require no on-site refuelling the reactor vessel could be sealed rendering access to the nuclear fuel assemblies even more difficult. Following the end of the core's lifetime, the reactor would be returned unopened to regional centre for the back end part of the fuel cycle. Systems that allow for continuous operation over an extended period without the need for intermittent refuelling are referred to in the industry as 'nuclear batteries' and enjoy the highest degree of proliferation resistance when combined with the hub-and-spoke concept. In addition, the client country would not require any fuel fabrication or management facilities, which would eliminate one rationale for developing nuclear research laboratories and educating a host of engineers and scientists whose experience could, at a later stage, be diverted to nuclear weapons manufacture [35].

On a closing note, SMRs are not inherently more proliferation-resistant than LRs apart from that a SMR could contain less nuclear material when the rated power output of modular plant is less than would be the case for a LR. Most, if not all, of the reactor types that are currently being investigated for SMR designs could be upscaled to LR sizes and the reason that several SMR designs are further along in their development is that every commercial nuclear power program in history was built upon a foundation of small-scale research reactors. The foremost non-proliferation advantage of SMRs is more logistical in nature than technological and is the ability to significantly reduce access to nuclear material by means of centralized manufacturing and decommissioning as described above. Additionally, when designed to operate on a low enough power density, a weld-shut reactor vessel requiring no intermediate refuelling or maintenance would completely eliminate the need for a second party to access the reactor core, such as the 10 MWe 4S described in IAEA-TECDOC-1536 [23].

3.4 Nuclear Economics

As of August 2011 over 60 NPPs were under construction in 15 countries (including Taiwan), the majority of these countries are in Asia [54]. Despite extensive historical experience in NPP construction, the estimation of the construction costs has proven to be difficult. Many NPPs have been delivered behind schedule or at a higher specific total capital cost than computed in pre-determined amount. There are several factors that could explain this deviation, the first one being that the cost composition of NPPs is prone to unpredictability and cost overruns. For the current NPPs on offer the major components, such as the electricity generators, the steam turbine and the reactor, only account for a relatively small proportion of the total construction costs. The majority (approximately 60 per cent) of the costs are incurred during the on-site engineering phase [55]. Costs can easily pile-up in this phase following unforeseen consequences such as sudden changes in design or heightened safety requirements. A second explanation is that certain pre-construction estimates are deliberately made inaccurate as a result of conflicting interests. This could account for the numerous inaccurate reports and articles published over the years by vendors which were not the actual tenders, utilities not penalised for inaccuracy, governments, nuclear industry bodies, consultants and even academics on occasion [56]. Data that has been known to be more accurate in the past was acquired from turnkey completed plants, vendor tenders and the estimates of utility companies.

The majority of the NPP under construction are being made by either a single or a consortium of vendors, which could mean that very little reliable construction and plant operation data will be made publicly available afterwards. Subsequently, there will be no recent construction experience to interpret the estimates being put forward by the nuclear industry [57]. The EPR currently being constructed at the Olkiluoto site forms an exception as the contract Areva NP was awarded by TVO is on a full turnkey basis. This means that Areva NP is responsible for the plant construction and the finished product is sold to the intended owner for a fixed price level specified in the contract. Such an agreement is based on the notion that cost overruns are generally caused by factors within the control of the contractor. It should be noted though, that a cost increase as a result of causes outside of the control of the contractor, typically entitles the contractor to increase the target price by an amount equal to the additional costs incurred [58]. The sufficient room these definitions offer in legal terms combined with the dramatically escalating project costs and construction time has resulted in TVO demanding €2.4 billion from Areva NP for construction delays and Areva NP, in turn, demanding €1 billion from TVO [59].

Shrader-Frechette (2009) [56] concluded that the majority of 30 recent nuclear-cost studies did not take factors such as the nuclear-liability-costs (subsidized by taxpayers) into account when deriving the cost of electricity estimates. It points out that this may give off a flawed economic signal, leading to inefficient markets. Moreover, it is considered a sign of questionable research ethics and unequal treatment. This act of not including certain variables, that might exert an upward pressure on the cost of electricity, is dubbed ‘cost trimming’. Notable categories of cost trimming pointed out in Shrader-Frechette (2009) [56] are firstly, that most nuclear cost studies ignore the taxpayer subsidies that cover large portions of the costs, the largest of these being the cost for nuclear insurance. A recent report by the EC on environmentally harmful support measures in the EU member states concluded that if all liabilities were to be privately insured for the full risk of a severe accident, the upper damage estimate would result in a premium of 5 c$€_{2003}$/kWh [60]. The case-study was conducted in France and it was established that if this tariff were to be placed on top of the estimated generating cost of 2.5 c$€_{2003}$/kWh, the cost of electricity would effectively be tripled. However, the same study concluded that if the plant owner/operator would only cover all the currently agreed upon national and international liabilities by means of private insurance, the price of electricity (without support) would only increase by 0.8%, from 2.5 c$€_{2003}$/kWh to 2.52 c$€_{2003}$/kWh.

Secondly, most nuclear-cost studies, assume ‘overnight’ plant construction capital costs, which assume zero per cent cost of debt and an overnight NPP construction. A recent publication from MIT does incorporate a construction time of 5 years into their model [61]. This assumption is highly debatable however, considering that it is lower than the average worldwide construction time of roughly 6 years [8] and significantly lower than the U.S. average of more than 10 years [56]. Assuming an interest rate of zero per cent is considered to be misleading as well. All NPPs are susceptible to construction delays (subsequently resulting in cost overruns), equipment malfunctions[9] and energy-market competition, the real cost of capital is revealed to vary from country to country and utility to utility. Previous studies indicate that the cost of money is highly dependent on the organisational structure of the electricity sector. In a regulated monopoly, interest rates could be as low as 5 to 8 per cent, but competitive electricity market debt interest rates could be as high as 15 per cent [55]. These figures, although good indicators, are by no means definitive. According to the European Renewable Energies Federation, a consortium of banks consisting of

[9] Although in terms of immediate loss of human life resulting from severe accidents nuclear power is shown to have the best overall track record [235] in the period 1969-1996.

the Bayerische Landesbank, Handelsbanken, Nordea, BNP Paribas and J.P. Morgan granted a loan of €1.95 billion, approximately 60 per cent of the projected total costs, for the construction of the Olkiluoto reactor unit 3 at an interest rate of 2.6 per cent [26]. This is a remarkably low percentage taking into account that Finland is a part of the Nordic market covering Norway, Sweden, Finland and Denmark, which is generally considered to be one of the most competitive electricity markets in the world [62].

Thirdly, it is assumed in several heavily cited studies that lifetime load factors will be in 85-95% range [61] [63]. This is quite surprising as the lifetime average load factor of U.S. NPPs is roughly 70% [64]. It should be noted that SMRs could potentially have an even lower load factor when integrated in an energy system featuring a high penetration rate of renewable energy sources, although this downward tendency could also be compensated by adding possibilities for useful output besides electricity (e.g. district heating). It does merit some consideration that, from the 1980s onward, load factors have steadily improved and worldwide annual load factors now average approximately 80 per cent [65]. Several countries (including the U.S.) are currently achieving annual load factor averages in excess of 90 per cent; however, as will be discussed in paragraph 4.3.3.7, NPPs with a lifetime capacity of 90 per cent or more are still somewhat of a rarity.

Lastly, it is noted by Shrader-Frechette (2009) [56], that increasing the estimated reactor lifespan represents an effective way to diminish the estimated costs. The second generation of nuclear reactors were designed to last for 30 years and some licenses have been extended to cover a longer operating period. However, the average operational lifetime of the 130 NPPs which have already been retired is roughly 22 years [8]. It could be argued that the average operational lifetime of the first and second generation designs holds little meaning for estimating the estimated lifetime of contemporary designs given the advancements in technology and design experience. However, it does illustrate the point that unforeseen technical difficulties or economic considerations might cause certain plants to be retired ahead of their operational lifetime. Similarly, a NPP nearing the end of its planned operational lifetime could be refurbished to operate past its original design life if certain technical, economic and political issues can be resolved. However, this requires an additional investment.

Based on the abovementioned arguments, one could ask whether a nuclear analysis should use historical data or industrial projections to interpret the cost-characteristics of new NPP designs. What needs to be kept in mind is that data projections are, at least to some degree, hypothetical and untested and should be handled with care. Especially when provided by those who would stand to benefit from providing optimistic estimates. Shrader-Frechette (2009) [56] states that when the worst-case scenario for all the above mentioned cost engineering fallacies, the cost of electricity could be more than 6 times as high as projected. This figure was computed by assuming a change in (inter)national law obligating a nuclear operator to purchase a full-nuclear liability insurance could raise the price to three times it original value, as reported earlier. Furthermore, including a 15 per cent cost of money instead of 0 per cent, raising the construction time to 10 years instead of 0, and using historical averages for the load factor and operational lifetime could raise this figure by another 188, 150, 19-36 and 5 percent respectively.

Summing these figures indicates that the total cost could increase by more than 600 per cent. However, due to the severity of several assumption made, this figure is not be interpreted as the difference between projections and actual costs. A more suitable designation would be the difference that might occur between projected and a worst case scenario study for a NPP with a tumultuous construction process in a highly competitive energy market and forced to buy a full-liability insurance, which is currently not required by (inter)national law. Shrader-Frechette (2009) [56] however, does persuasively argue that the cumulative load factors could potentially be a more suitable parameter than the higher end of the annual load factors. Furthermore, the assumption that a newly constructed NPP might actually have a shorter operational lifetime than intended by the designer is defendable. It has been established that under the specific circumstances, such as market deregulation, NPP operators will face increasing uncertainty regarding the revenues they receive from selling electricity to the grid and that the risk of early retirement is therefore ever present [66].

Within the context of nuclear economics, SMRs take up an ambiguous position, the specifics of which will be further explored in paragraph 4.3.3, which details the approach undertaken to model the economics of SMR. For now it can be established that the nature of the SMR construction process, which could potentially be industrial in scale, seems to protect it to a certain degree from the on-site complications. Furthermore, the reduced design complexity could prevent the bottleneck from shifting from NPP site to the factory floor. In other words, SMRs seem to reduce the probability of significant cost escalations occurring due to known unforeseen circumstances, which could give them an advantage for attracting investments at a lower rate. On the other hand, considering that they could be used in back-up capacity puts a downward pressure on their expected load factor which could make them expensive to operate in comparison to other alternatives.

3.5 Post-Fukushima Nuclear Landscape

On the 11th of March 2011 earthquakes of a magnitude 9.0 on the Richter scale hit the northeast of Japan, with a series of tsunami waves in its wake. The earthquake and the tsunami caused a loss of all AC power sources for units F1, F2 and F3, which were operational at the time, and F4 which was shut down for maintenance. Furthermore, there was a complete loss of the secondary sea-water cooling systems, the ultimate heat sink of the NPPs thermodynamic cycle. Emergency cooling systems had to run on energy battery power, because the tsunami had rendered all but one of the diesel generators (DGs) inoperable. These normally provide power to the emergency systems and at present it is believed that the remaining DG saved reactor units F5 and F6 [67]. The loss of core cooling in F1-F3 and maybe also the spent fuel cooling in F4 caused the water level to decrease due to evaporation, which eventually resulted in fuel dry-out in F1-F3. When the fuel reached 1200 °C the zirconium cladding started to oxidise and collapsed, which prompted the release of hydrogen and radioactive material into the pressure containment vessel (PCV). The release of steam and fission gasses from the reactor pressure vessel resulted in a pressure increase inside the PCV, which forced the operators to vent radioactive material into the atmosphere. Additionally, some of the hydrogen formed in the reactor pressure vessel made its out of the PCV, filled with inert nitrogen gas, and into the reactor building where it reacted explosively with oxygen. These two events combined resulted in the release of significant quantities of iodine, caesium, strontium and tellurium which have contaminated a large area surrounding the Fukushima site [67]. According to the latest estimates approximately 900,000 terabecquerels [68] of radioactive substance have been released and approximations for the cost of damages are in the order of 75 to 260 billion US$ [69].

The Fukushima Disaster ended a relatively uneventful period of nuclear power history. The relatively successful operation of roughly 440 reactors over the past 25 years combined with the absence of major catastrophe led to the popular belief that a large-scale global revival of nuclear power, the so-called ‘nuclear renaissance’ was a viable option for reducing the fossil fuel content of our energy portfolios. Irrespective of whether one supports or criticizes the concept of the ‘nuclear renaissance’, at this time it seems likely that Fukushima will have an impact on this ideology, although most sources deem it too soon after the event to allow for a sensible answer as to what this impact might be [34].

UBS, a Swiss global financing service, published an investment report only 3 weeks after the accident [70]. It concluded that the Fukushima disaster would affect safety standards, closures and investments in new nuclear plants. Regarding existing safety standards it expected that the focus would be on the lessons learned from Fukushima in particular those concerning the risk scenarios for seismic activity, flooding, back-up power systems and crisis management protocols. Furthermore, it envisioned increased resistance against the operation of aging reactors and a decreased willingness to grant lifetime extensions. Additionally, existing plants would most likely be subject to system upgrades bringing them in line with the highest safety standards. It was also predicted that even pro-nuclear countries such as France would not be able to get around closing some plants in order to demonstrate the political willingness to act and restore public trust in the installed nuclear power capacity. UBS reported to have identified 30 older reactors across the globe which had a higher risk of closure [70]. Moreover, the higher safety standards are expected to increase the already hefty nuclear capital costs even further, with overnight costs for new nuclear potentially increasing to 5000-6000 US$/kWe in Western-Europe and North-America and roughly 2000 US$/kWe in China. The corresponding cost of electricity of state-of-the-art plants was reported to be no less than 100 US$/MWh in high income economies and roughly 50 US$/MWh in emerging markets. HSBC, a multinational banking and financial service company, added that the expected loss of favour for the nuclear option could result in positive externalities for natural gas, energy efficiency and renewable energy systems [71]. The scenario in which countries massively adopt the natural gas (NG) option, also referred to as the 'Golden Age of Gas' Scenario by the International Energy Agency [72], is rationalized by NGs relative abundance and environmental benefits in comparison to other fossil fuels. The increases in energy efficiency and renewable energy are explained by: (1) an upward pressure on the expected CO_2 tax following the abandonment of nuclear option and (2) national re-evaluations of the energy policies.

Based on an analysis of the previous major accidents at Three Miles island and Chernobyl, the abovementioned studies conclude that the Fukushima disaster will have far reaching consequences. Both these accidents prompted a large public outrage which led to the formation of political movements questioning the safety of nuclear power generation and many countries adopting phase-out policies. Furthermore, the impact of Fukushima was expected to result in even larger implications than previous accidents, considering that: (1) Japan is an advanced economy utilizing advanced western reactor designs, not a totalitarian regime adopting substandard technology with a lacking safety culture, (2) the size and duration of accident were unparalleled with 3 cores having (partially) melted down and engineers being unable to stabilize the situation (which took until the 16th of December 2011 [73]) and (3) that the Fukushima disaster happened during a crucial stage in the energy and climate debate, as stated in Glaser (2011) [34].

In response to the accident, the IEA made some downward revisions concerning the projected growth of nuclear to New Policies Scenario in their 2011 World Energy Outlook, with the share of nuclear power being 5 per cent lower than projected in the previous edition [74]. Additionally, in view of the increased uncertainty regarding future nuclear deployment, the IEA has included a Low Nuclear Case, which examines the implications of a much smaller role for nuclear power. Its foremost premise is that the 393 GW nuclear power capacity installed at the end of 2010 drops to approximately 335 GW in 2035. This is just of half of what it would be in the New Policies Scenario (NPS). Such an event could have significant repercussions on energy security, diversity imports and CO_2 emissions, with the most important findings being that by 2035: (1) the coal demand will have increased by 290 mega tonnes a year and (2) gas demand will have increased by 130 billion cubic metres a year and on the upside (3) that renewable-based generation will have increased by roughly 550 TWh a year. The result of this is that an additional 90 billion US$ will be spent on the import of fossil fuel resources, which is 12 per cent higher than in the NPS. Additionally CO_2 emissions will rise by 2.6 per cent in comparison to the NPS. However, it pointed out that in countries which are currently highly dependent on nuclear power and traditionally limited in natural resources, such as: Belgium, France, Japan and Korea, the effects could be much graver [74]. The Low Nuclear 450 Case, which is the Low Nuclear Case variant of the traditional 450 Scenario, projects that in order to limit global warming to 2°C an additional 1.5 trillion US$ investment will be required from 2011 to 2035. Furthermore, following this path requires what the IEA refers to as a "*heroic achievements in the deployment of emerging low-carbon technologies, which have yet to be proven"*. In summary, mass abandonment of the nuclear option would most likely make the climate challenge more challenging than it already is and – according to the IEA - more costly as well.

Joskow and Parsons (2012) [75] report that although significant alterations have been suggested for the nuclear programs of Japan, Germany and Switzerland, most countries have not made major reforms to their nuclear energy programs nor have they implemented any new safety criteria that would severely alter the cost of the safety features. The large-scale nuclear growth scenario, as often encountered before Fukushima, can be considered increasingly unlikely, with certain countries re-assessing their nuclear commitment, but has not entirely been abandoned. It is reported in Goldston (2011) [76], for example, that economic modelling suggests that by 2100 nuclear power may provide roughly 30 per cent of the projected global electricity demand. The projected power production of 3600 GWe(y) implying that the installed capacity would have to increase by a factor of 12 from the 2010 benchmark of 300 GWe(y).

This view is most definitely not in line with the de facto policy consensus in Germany. Even before it decided to phase out its nuclear power by 2022, one major assumption in every published energy outlook was that the role of nuclear power would be decreasing over time [34]. Although the government and the various NGOs and industrial sources might disagree on best overall strategy to achieve the countries GHG reduction scenarios, the fundamental assumption which is made in all these studies is the impending end of the nuclear era. Among other things, the prophesizing of the end of the nuclear power has given the German energy projections a distinctive undertone in comparison to other forecasts.

The International Energy Agency forecasts a continuous increase in German electricity demand of 1.0 and 0.8 per cent for the business-as-usual case and the 450 scenario, respectively [74]. These estimates outline the expected development paths for European OECD countries in-between 2009 and 2035. Contemporary German energy concepts on the other hand establish the reduction of electricity demand as a forecasting guideline. A recent publication by the BMU unravels that the German electricity consumption should have decreased by 10 per cent in 2020 and 25 per cent by 2050 [77], 2008 acting as the point of reference.

As pointed out in Glaser (2011) [34], the necessity of this vigorous persecution of energy demand reductions is ultimately a consequence of removing a major low-carbon base load technology. This also explains why the German forecast deviate that much from the projected development paths presented by the westernized countries where the nuclear option remains intact. However, the conviction with which Germany dismissed its nuclear infrastructure, partially fuelled by a trust in renewable energy technology innovation and a resistance against nuclear power generation predating the events at Fukushima [78], has seen some mimicry. The following year resulted in several other countries announcing a re-evaluation of their nuclear energy policy, including Belgium, France and Japan.

Therefore, the events at Fukushima have made a rapid up-scaling of global nuclear capacity far less likely. However, the resilience which several nuclear power programs have bounced back since this disaster can be taken as an indicator that several countries will continue to rely on, or might event start to pursue nuclear power as a means to reduce their GHG emissions and enhance energy security. Worries about global climate change are likely to result in a fair amount of restraint with regard to excluding a low-carbon technology, which could hypothetically play a key role in addressing future electricity demand, from the energy mix. This might especially be the case in countries where the technology is already in use. Considering this, Glaser (2011) [34] suggests that the coming period, in which he expects relatively little nuclear capacity will be added, should be used to evaluate the '*technical and institutional approaches to nuclear power*'. This could result in several important lessons being learned, one of them being the approach towards lifetime extensions. Nuclear operators could come to realize that even with significant equipment upgrades, the technology they are operating at the present can no longer be thought of as being state-of-the-art, in particular with regard to the innovative safety features present in more modern designs. The other one lesson concerns responsible energy policy. Whether one agrees or disagrees with Germany's radical response to the Fukushima disaster, the circumstances have shown just how important it is to have alternatives energy strategies to fall back on in case an exogenous shock reveals a fundamental flaw in a certain energy technology. This could be of relevance again, if carbon capture and storage ever makes it to the mass deployment phase

Everything considering, there is currently no way to accurately determine what effect the Fukushima accident will have on the development of SMR technology. Although an obvious consequence could be that SMRs, falling under the denominator 'nuclear power systems', would be stigmatized in an equal fashion as LRs, several vendors have seized the opportunity to dissociate their designs from the light water reactor technology used at the Fukushima NPP, especially advertising their design's passive safety features.

Considering that the two SMRs that are expected to hit the market first will not do so in the coming years, as discussed in paragraph 3.1, this might play to the vendors advantage by providing them with ample time to let the current upheaval of anti-nuclear public sentiments subside a bit before their designs become commercially available.

3.6 Swedish Case Study

In the wake of the Fukushima disaster, in line with the global trend, the popularity of nuclear power generation in Sweden has diminished. As is to be expected, the figures generated by polls tend to vary. While the Swedish daily Dagens Nyheter reported that support for a nuclear phase-out has reached an all-time high with 36 per cent of the population favouring a nuclear phase out (March 20th 2011) [79], a similar poll conducted by NOVUS in May 2011 concluded that a mere 24 per cent of the population was in favour of a politically driven decision for nuclear phase-out [80]. Both sources are in reasonable agreement on the 2008 percentage of public opposition however, which is reported to be positioned at the 15 per cent mark.

Another similarity, is that the countries so called 'core support'; the portion of the population that prefers the role of nuclear power in the energy mix to remain the same, was reported to have remained largely the same in the NOVUS poll and even increased in the Dagens Nyheter poll. These outcomes stand in stark contrast to the aforementioned German case and the outcome of the 2011 Italian referendum in which more than 94 per cent of the voters opposed a restart of the Italian nuclear program [81]. The variations in response to the Fukushima accident illustrate the difficulty of linking public support for nuclear power to objective measures and that the nuclear power debate is as much driven by sentiment and cultural issues as it is by rational argumentation.

An analysis of the various factors influencing national nuclear support, conducted by the Ipsos Social Research Institute (ISRI), concluded that a strong inverse relationship exists between a country's support for nuclear power and its overall level of energy dependence [82]. At first, it might seem counter-intuitive that a lower share of dependency on foreign energy resources could lead to a higher support for nuclear power. However, after taking into consideration the social, cultural and political factors, it can be argued that people living in countries with a lower energy dependency are more inclined to support nuclear power in order to preserve their independence than the inhabitants of a country that is heavily dependent on energy imports. Public opinion in relatively dependant countries may be disposed towards the belief that since they already rely heavily on imports, there is little benefit to gain by switching to an energy source that is perceived to be more dangerous [82]. With the vast majority of its electricity being generated by hydroelectric and nuclear power plants, Sweden can be considered to be relatively autonomous with regard to its electrical power generation and is only dependent on the outside world for the coal, gas and oil import required for its industrial, transport and residential sectors. Considering that, regardless of which poll one adheres to, the majority of the Swedish public still supports its nuclear infrastructure, the theoretical framework presented by the ISRI appears to hold up quite decently.

Per January 1st 2011 the Swedish Parliament has overturned a decision made following a referendum 30 years ago to allow Swedish firms to replace the existing ten reactors [83] that provide roughly 40% of the country's electricity [84]. Although the implementation of the law heralded a new era for Swedish Nuclear, the new rules still contain restrictions. New reactors are only permitted at the existing three power plants and a new reactor may only begin operation as an older one is permanently shut down. This arrangement would limit the role of nuclear power to the traditional position it already holds, because it only encourages the construction of largest designs of 1600-1800 MW(e). This limits the attractiveness of implementing SMRs. Moreover, it might even discourage the future use of Generation-IV reactors in general, due to none of the current designs being envisioned to provide sufficiently large generating capacities [85]. Generation III light water reactors, which could be eligible to replace the current fleet of reactors are: AREVA's European Pressurized water Reactor (EPR) at 1650 MWe, the General Electric-Hitachi Advanced Boiling

Water Reactor (ABWR) at 1350-1460 MWe, and the Mitsubishi Nuclear Energy Systems, Inc. U.S. Advanced Pressurized Water Reactor (US-APWR) at 1700 MWe. However, no plans for replacing the current reactor fleet are known to exist at the time.

4 Scope, Research Questions and Methodology of this Thesis

4.1 Scope

The SMRs under investigation in this thesis are all advanced reactor designs, a designation which can be further subdivided into the so-called ‘evolutionary designs’ and ‘innovative designs’, which are predicates attributed to reactor designs in accordance with their degree of novelty. The definitions of these terms are given in line with IAEA-TECDOC-936 [86]:

Advanced design: “A design of current interest for which improvement over its predecessors and/or existing designs are expected. Advanced designs consist of evolutionary designs and designs requiring substantial development efforts.”

Evolutionary design: “An advanced design that achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining design proveness to minimize technological risks.”

Innovative design: “An innovative design is an advanced design which incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice.”

Furthermore, the development of advanced reactor projects moves through a number of subsequent design stages. These are the conceptual design stage, the preliminary design stage and the detailed design stage. Although no actual definitions are given, these stages are described in IAEA-TECDOC-88 [87] as follows:

The **conceptual design stage** is the stage in which the initial concept and the plant layout are developed. The **preliminary design stage** is the stage in which the essential R&D activities are completed (with the exception of some non-critical items). The **detailed design stage** is the stage which eventually results in a complete plant design with the possible exception of some very minor items. The design can be unified (i.e. to be used for varying site conditions) or specific for one site.

A number of recent publications by the IAEA addresses several advanced SMR designs (for the larger part consisting out of designs which are still under development) which are currently under development in the world [22] [23]. In this thesis only the designs which have reached the preliminary design stage, are intended for land-based installations and have electrical power generation as their primary purpose, will be considered. One possible exception with regard to the primary purpose is the MYRRHA reactor described in paragraph 5.7. Currently the reactor design only makes reference to the future possibility of electrical power generation and its primary purposes are transmutation of high-level nuclear waste and generation of medical isotopes. However, recent studies have pointed out that using the technology underlying the MYRRHA reactor for electricity generation could potentially be economically feasible [88].

In summary, the following designs will not be considered in this thesis:

- Design efforts which, as of February 2012, had not progressed into the preliminary design phase.
- Design concepts which, as of February 2012, had entered the conceptual design phase, but for which the required technical information is not available.
- Design concepts for which the development programmes had been discontinued as per February 2012
- Barge-mounted design concepts planned for deployment in remote regions. Due to the scope of this thesis being high income economies, this line of designs is believed to be of lesser interest.

The categorisation of the advanced SMRs considered in this paper is done according to the basic characteristics of their reactor design. Considering the limited availability of empirical data for most of the reactor technology lines only one design will be evaluated per reactor type in order to avoid making too specific ‘predictions’ based on uniform general assumptions. One exception to this rule will be made for PWR category in which the NuScale and mPower designs will be evaluated. This is done in order to give an overview of the effects of modularity on the cost of electricity.

An overview of the different reactor types and the selected designs, which had progressed into the preliminary design stage, under the restrictions described above is provided in Table 4.1.

Reactor Type	Design
Pressurized Water Reactors	NuScale, mPower
Advanced heavy water reactors	PHWR-220
High temperature gas cooled reactors	HTR-PM
Sodium cooled fast reactors	4S
Lead-bismuth cooled fast reactors.	SVBR-100
Molten Salt Reactors	FUJI
Accelerator Driven Systems	MYRRHA

Table 4.1

4.2 Research Questions

The competitiveness of the selected SMRs will be assessed based on three categories: passive safety features, proliferation resistance and cost of electricity. The corresponding questions this thesis aims to answer are:

I. To what degree do the passive safety features of the various reactor designs mitigate the chance of damage to reactor core in the case of an accident scenario?
II. To what degree does the reactor technology offer protection against the proliferation of nuclear material?
III. What is the range of the expected costs of electricity?

4.3 Methodology

Quantitative models, taken from literature and modified to perform under limited data availability, will be used in this thesis to attribute degrees of proficiency to each category for each reactor design. Paragraph 4.3.1 details how the level of passive safety will be assessed, paragraph 4.3.2 aims to offer insight into the analysis of the proliferation resistance and paragraph 4.3.3 contains a description of the SMR economics model.

4.3.1 Passive Safety Model

Based on the publicly available design parameter, it should be noted that most of the SMR reactors have relatively large primary coolant inventories in comparison to typical LRs. In combination with the fact that several designs also plan to utilize coolants with higher heat capacities than light water, this could mean that SMRs have a relatively higher volumetric heat capacity, which provides the reactor with a larger capability for heat uptake in the event the heat transfer to the secondary coolant loop is lost. In the context of this thesis, this ‘thermal inertia’ can be considered to be combined the heat capacity of the primary coolant and/or moderator, passive heat removal systems and additional high heat capacity in-core materials, relative to the amount of heat being generated by the reactor core per unit of time. In other words, it defines how quickly the temperature of the in-core environment increases when a loss-of-secondary-coolant accident occurs. A reactor can be considered to have a relatively high thermal inertia when the temperature increase of the in-core environment is relatively low.

The formula for the thermal inertia can be determined by using either, the volume and the pressure of the primary coolant, or the mass of the primary coolant. In line with the NEA [29], in the event the mass of the primary coolant is known the expression for the thermal inertia (in K⁄s) becomes:

$$\frac{\delta T_{av}}{\delta t} = \frac{P_{th}}{\left[m_1 C_{p,1} + m_2 C_{p,2} + m_3 C_{p,3}\right]}$$

Equation 4.1

In event the volume of the primary coolant, moderator, or passive cooling tank is given, the mass can be derived as follows:

$$m = V\rho$$

Equation 4.2

In Equation 4.2 and Equation 4.2, P_{th} is the reactor thermal output in kW_{th}, m_1 represents the primary coolant/moderator mass in kg, $C_{p,1}$ stands for the heat capacity of the primary coolant/moderator at the average in-core temperature (kJ/kgK). In the event the reactor possesses a PHRS in the form of a reservoir, its coolant mass m_2 and its corresponding heat capacity $C_{p,2}$ need to be taken into consideration as well. Similarly, if the reactor contains in-core structures with a high heat capacity, such as graphite, their mass m_3 and heat capacity $C_{p,3}$ also needs to be taken into account. Furthermore, V equals the primary coolant/moderator volume in m^3 and ρ is the density of the coolant/moderator (kg/m^3).

Additionally, as discussed in paragraph 3.2, SMRs are also reported to possess a relatively high surface area to volume ratio which would allow them to mitigate excess heat to the environment at an accelerated pace. In order to test this hypothesis, surface area to volume ratio (SA:V) of the various reactor pressure vessels will be determined. Due to the lack of detailed schematics, we will assume that we are dealing with a cylindrically shaped reactor vessel. The formula for the SA:V of a cylindrical reactor vessel can be determined to equal:

$$SA{:}V = \frac{2\pi r(r+h)}{\pi r^2 h} = \frac{2(r+h)}{rh}$$

Equation 4.3

In Equation 4.3, $SA{:}V$ (m^{-1}) is a function of the cylinder's radius r and height h. Both the radius r and the height h are given in metres.

4.3.2 Proliferation Resistance Model

In recent years several models have been suggested for quantifying and comparing the proliferation resistance across reactor designs. One of the most recent methods is the Figure Of Merit (FOM), as described by Bathke et al. (2010) [89], which provides a rough indicator of the usefulness of spent nuclear fuel for nuclear weapons production through the entire lifespan of the radioactive decay process by means of three characteristics: (1) the bare critical mass M of the spent product material (either in purified or unpurified metal form), (2) its corresponding heat content and (3) the irradiation dose rate incident on a surface located at 1 metre distance from a $0.2M$ mass of waste, a seemingly arbitrary number adopted in FOM literature. Although highly transparent, this method is unsuitable for the selection of SMRs under investigation in this thesis. The computation of the dose rate for example, requires knowledge on the exact isotopic composition of a spent fuel sample over a certain period of time. A spent fuel analysis of this level of detail is considered beyond the scope of this thesis.

The methodology developed by Pierpoint (2008) [53] for early stage fuel cycle analysis is more in line with the provisional development state of some of the SMR designs and only requires knowledge of fissile isotope composition at the end of the irradiation period. The complete framework is composed of eleven metrics which were designed to evaluate the proliferation resistance of, among other things, the once-through, the mixed-oxide (MOX) and the advanced burner reactor cycles. The metrics can roughly be divided into three sections: the material characteristics, the material handling characteristics and the facility characteristics. Each metric has its own corresponding utility function, which is subsequently weighted in accordance with a proliferation threat type (national or subnational).

The focus of this thesis is on the assessment of the proliferation resistance of the reactor designs and not a proliferation analysis of the entire nuclear fuel cycle, of which the reactor is a part. Therefore portions of the methodology offer very little additional insight and are considered to be of limited contribution to this work. Regarding the NPP segment of the nuclear fuel cycle, it is reported in Pierpoint (2008) [53] that once the nuclear material begins its 'reactor step', which is nuclear fission inside the reactor core, the proliferation resistance is at its highest, with the harsh, radioactive environment providing a sizeable inherent barrier to the theft of nuclear material.

Taking into consideration the aforementioned barriers to fuel removal from the reactor core, it is assumed when assessing the proliferation resistance of a reactor design that the parts, of the NPP segment of the nuclear fuel cycle, in which theft of nuclear material is most likely to occur is during the inventory stage in the front-end and during the wet-cooling spent fuel phase following the end of each irradiation cycle. In order to determine how the fresh and spent fuel rods perform regarding their proliferation resistant, several metrics were taken from Pierpoint (2008) [53].

4.3.2.1 Fresh Fuel Proliferation Resistance

Regarding fresh fuel, two metrics need to be used in unison to measure the proliferation resistance. These are the spontaneous fission metric in the case of plutonium based fuel and the enrichment metric in the case of uranium-based fuel. The former can be used to assesses an inherent proliferation resistance mechanism of plutonium; spontaneous fission. Plutonium isotopes with an even number of protons and an even number of neutrons tend to have a shorter spontaneous fission half-life. Therefore, the rate of spontaneous fission of plutonium increases with the ratio of even isotopes. A higher rate of spontaneous fission could lead to a pre-initiation of the nuclear chain reaction which could cause the weapon to have a low or 'fizzle' yield. As reported by Cochran and Paine (1995) [51], it should be noted that even a low yield nuclear weapon can still have an explosive force in excess of 1 kiloton and could therefore still be regarded as a coveted addition to the arsenal of any terrorist. In line with Pierpoint (2008) [53], the utility function corresponding to the spontaneous fission rate is given by:

$$u_{SF}(x_{SF}) = \begin{cases} \frac{1}{2}[1 - \exp(-3.5(x_{SF})^{1.8})] & 0 < x_{SF} < 0.6 \\ 0.07 + \exp(6x_{SF} - 4.8) & 0.6 \leq x_{SF} < 0.8 \\ 1, & otherwise \end{cases}$$

Equation 4.4

In Equation 4.4, the proportion of even plutonium isotopes is a proxy for the spontaneous fission rate. Therefore, the utility of spontaneous fission $u_{SF}(x_{SF})$ is taken to be a function of the fraction of even plutonium isotopes x_{SF}. In Pierpoint (2008) [53], the boundary conditions 0.6 and 0.8 were not chosen arbitrarily but set in accordance with expert opinions. Plutonium consisting for 60 per cent or more out of even-number isotopes is reported to severely hinder a successful detonation sequence. An even-isotopic composition of 80 per cent or more was considered to render the material ineffective for weapons manufacture, which is why a perfect score of 1 is attributed to any plutonium with a x_{SF} value of 0.8 or higher.

The spontaneous fission metric is inadequate to assess the material quality of uranium; the fission probability per decay for U235 being low enough to exclude it a serious hindrance. Moreover, the technical obstacles that need to be overcome in order to create a high-yield uranium bomb are much smaller than for a high-yield plutonium bomb. Therefore, in order to assess the material quality of uranium the fuel enrichment grade represents a more meaningful input factor. In Pierpoint (2008) [53] an expression for the uranium enrichment utility function is given. However, as stated by the author, the gradual decrease in proliferation resistance for lower levels of enrichment is based on the assumption that the potential proliferator has no access to enrichment technology. If access to enrichment facilities can somehow be acquired the 5-12 per cent enrichment interval becomes much more attractive, considering that a sizeable part of the enrichment process has already been completed. Furthermore, as can be seen in Figure 4.1, the proliferation resistance was taken to decrease less rapidly in the 20 – 50 per cent interval due to the relatively small additional enrichment time requirements. As can be seen in Figure 4.1, from the 50 per cent enrichment mark onwards, the proliferation resistance starts to decrease more rapidly again. This is in recognition of the fact that at these enrichment levels the uranium could be used to manufacture a low-yield nuclear weapon without any additional enrichment. The minimum proliferation resistance utility of zero was attributed to the so-called 'weapons-grade' enrichment level. For U-235 this value equals 93.5 per cent [90].

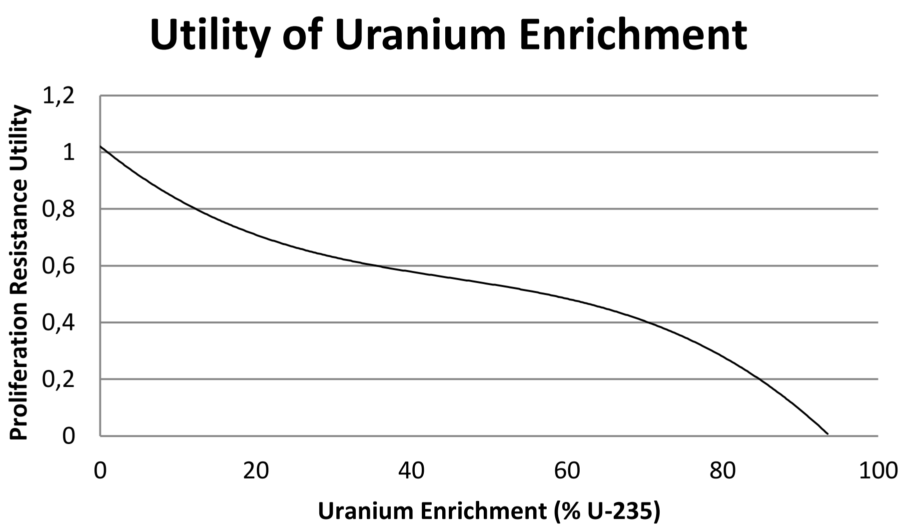

Figure 4.1| The proliferation resistance of fresh uranium fuel rods as a function of the fuel enrichment

$$u_{UE}(x_{UE}) = 1 - 3 \cdot 10^{-6}(x_{UE})^3 + 4 \cdot 10^{-4}(x_{UE})^2 - 2.25 \cdot 10^{-2} x_{UE}$$

Equation 4.5

Equation 4.5 represents the corresponding utility function of the proliferation resistance of enriched uranium u_{UE} as a function of the enrichment grade of the uranium fuel (u_{UE}). The formula was fitted in excel and adjusted to have a maximum value of one. Furthermore, it should be noted that Equation 4.5 has a value smaller than one for all non-zero uranium enrichment levels.

4.3.2.2 Spent Fuel Proliferation Resistance

In line with Pierpoint (2008) [53], the metric which will be used to assess the proliferation resistance of the spent fuel is the concentration metric. This is a utility function that accounts for the amount of nuclear material in the spent fuel in terms of SQs as discussed in paragraph 3.3 and shown in Table 3.1. The higher the significant quantities of plutonium and uranium present per kg of heavy metal, the lower the amount of effort a potential proliferator has to expend in order acquire the material needed to construct a nuclear weapon. Therefore the concentration metric utility function is a measure of likelihood that a potential perpetrator will be able to successfully avert detection while displacing nuclear material. In case of a low concentration the potential proliferator will require either more time, more associates or more theft attempts to acquire the desired amount of material. All of these factors increase the likelihood of the perpetrator being caught or his attempts being thwarted. Therefore, the chances of successfully circumventing antiproliferation barriers increase exponentially with an increased number of SQs in the spent fuel. This is why the input factor of the utility function is given in SQs per unit of spent fuel. The formula is given by:

$$u_{MC}(x_{MC}) = \begin{cases} 1, & x_{MC} < 0.01 \\ exp\left[-20.5\left(\frac{x_{MC}}{x_{MC,max}}\right)\right] & x_{MC} > 0.01 \end{cases}$$

Equation 4.6

In Equation 4.6, the utility function of the material concentration u_{MC} is given as a function of the cumulative number of plutonium SQs per metric tonne of heavy metal x_{MC}. In line with Pierpoint (2008) [53], the maximum material concentration was taken to be 94 SQs/MTHM, which is the concentration of 'super-grade' plutonium. As can be derived from Equation 4.6 the proliferation resistance of the spent fuel exponentially decreases with an increase in significant quantity. This is intuitively correct, considering that an increase in nuclear material could be argued to have a more profound effect on the proliferation resistance in the low concentration region than in the high concentration region.

4.3.3 Small Modular Reactor Economics Model

4.3.3.1 Levelized Cost of Electricity

The primary focus of the small modular reactor economics model is to derive an estimate of the Levelized Cost of Electricity (LCOE) for a specific SMR design. The LCOE can be defined as the cost of electricity per kWh when including the cost of capital, debt service, O&M and fuel. It is given by the following expression, which is based on the formula used in reference [91].

$$C_{ee} = \frac{\sum_{n=1}^{n_{lr}} \frac{1}{(1+r_d)^n}\left[F_{cr}c_i + c_{fom} + 8760L_f c_f\left(1+p_f\right)^n + c_{vom}(1+p_{vom})^n\right]}{\sum_{n=1}^{n_{ltm}} \frac{8760L_f}{(1+r_d)^n}} + \frac{\sum_{n=n_{lr}}^{n_{ltm}} \frac{1}{(1+r_d)^n}\left[8760L_f c_f\left(1+p_f\right)^n + c_{vom}(1+p_{vom})^n\right]}{\sum_{n=1}^{n_{ltm}} \frac{8760L_f}{(1+r_d)^n}}$$

Equation 4.7

In this expression C_{ee} is the cost of produced electrical energy in US$/kWh, c_i represents the specific overnight costs in US$/kWe, F_{cr} the fixed charge rate, c_{com} the constant operations and maintenance cost in US$/kWy, c_{vom} the variable operations and maintenance cost in US$/kWy, c_f the fuel costs in US$/kWh, n_{lr} the years of loan repayment, n_{ltm} the operational plant lifetime given in years, rd the discount rate, p_f the average foreseen fuel price changes during the operational lifetime, p_{vom} the escalation cost of variable annual operations and maintenance cost and L_f the life time load factor. In this expression it is assumed that the operating license is not expanded at the end of the estimated operational lifetime.

The main goal of assessing the costs according to the methodology described above is to obtain an independent levelized cost of electricity for SMRs under investigation in this thesis. This will be done by evaluating the overnight capital costs of a reference LRs, and deriving cost approximations for the fuel costs and fixed and variable O&M costs. For the evolutionary designs assessed in this thesis an attempt will be made to base the SMR overnight capital cost estimate on actual construction data. The innovative designs, for which in most cases the construction experience is lacking, will be assessed based on the actual construction costs of larger-sized plants of similar design published in peer-reviewed papers. When these are not available, estimates will be acquired from independent scientific or technical sources. This approach is in line with a recent publication of the NEA [29].

It should be noted that the available cost data on nuclear power plants has a high degree of uncertainty. The NEA defines this uncertainty as the *"implicit impact of non-quantifiable factors"* [29]. An overview of how the economic model used in this thesis deals with the uncertainty resulting from these factors is given in paragraph4.3.3.2. In paragraph 4.3.3.3 the method employed for scaling the cost characteristics of the reference reactors is discussed. It should be noted that the expressions used to reflect the scaling law are only approximations based on simplifications. Furthermore, the effects of the economies of scale on individual investment items (such as reactor equipment and miscellaneous plant equipment) and the benefits of modularity in terms of the design complexity are based on experience with specific types of NPPs. In paragraph 4.3.3.4 the financial parameters used in this thesis; the discount rate p_d and the fixed charge rate F_{cr} are discussed. Paragraph 4.3.3.5 revolves around the operations and maintenance costs which will be derived as a function of the rated power output. A cost analysis of a standard once-through nuclear fuel cycle is encountered in paragraph 0 as well as an analysis of why this cycle was chosen. Finally, paragraph 4.3.3.7 presents the details on how the load factor will be valued in the model and paragraph 4.3.3.8 contains a discussion on the operational plant lifetime.

4.3.3.2 Base Estimate and Uncertainty Analysis

The U.S. Government Accountability Office defines Cost Estimating as the combined science and art of assessing the future cost of something based on known historical data that is adjusted to reflect new material, technology, software languages and development teams [92]. A more concise statement is that a cost estimate is a well-formulated assessment of the probable construction cost of a specific building project [93]. In effect, a cost estimate establishes a base line for the expected development of the cost level at different stages of a project and cost engineering is the area of the engineering practice where judgement and experience are utilized in unison with scientific principles and techniques to resolve the problems of cost estimation and cost control.

In industrial construction projects, like the construction of a NPP, it is customary to produce a cost estimate projection based on available data. It is in this context that one often comes across the term contingency, which is related to the availability of data and experience. During the early design stages these factors are often in short supply. Take for example the 4th generation NPP designs. It can be expected that, because there is little to no construction experience building plant components (including pricing and scheduling), there will be large risks associated with their licencing and specialized design [94].

The EMWG, in their 'Cost Estimating Guidelines for Generation IV Nuclear Energy Systems' [95], formally define contingency as: *"An adder to account for uncertainty in the cost estimate (...). Contingency includes an allowance for indeterminate elements and should be related to the level of design, degree of technological advancement, and the quality/reliability pricing level of given components (...). Contingency does not include any allowance for potential changes from external factors, such as changing government regulations, major design changes or project scope changes, catastrophic events (force majeure), labour strikes, extreme weather conditions, varying site conditions, or project funding (financial) limitations. A contingency can be also applied to the interest during construction and the capacity factor to account for uncertainty in the reactor design/construction schedule and reactor performance, respectively."*

Although this provides the reader with an overview of what factors to take into considering when determining the contingency factor, it does not define its meaning. Reference [96] states that the contingency of a cumulative probability density function gives the probability of achieving the base estimate. This makes it possible to determine the required amount which needs to be added or subtracted to acquire the preferred probability of cost underrun or overrun. When taking into consideration that the cost estimates for NPPs are seldom underestimated, one is able to define the contingency factor as an additional amount that must be added to the base estimate to achieve the minimum desired amount of certainty. There exists a range of estimating techniques, which can be used to derive a value for the project cost contingency. An overview of the available methods is given in reference [97] and a discussion of why probability distribution were selected in this thesis in favour of point estimates can be found in Appendix 10.2.

The Monte Carlo Method (MCM) is able to cope with the shortcoming of uncertainty resulting from imperfect information by applying random number theory on the input variables over the course of a certain amount of repetitions (iterations). The MCM requires knowledge of certain upper and lower bounds in combination with knowledge on the outline of the probability density function. Due to the MCM having knowledge of the probability of an occurrence, MCM is able to produce a much narrower probability distribution of the desired output because it does not attribute equal weights to the various scenarios.

As discussed in Appendix 10.4, cost estimates with lognormal distributions can be assigned a contingency equal to their standard deviation [98]. The main advantage of adopting this approach is that it partially eliminates the subjectivity of selecting a contingency level. Therefore, in this thesis, the standard deviation will be set to equal the contingency.

Integrated approaches to uncertainty analysis have not been a common practice in the previous generation of nuclear cost studies [63, 99]. However publications which are currently being reviewed have been noted to perceive it as a critical element in their investigation [100, 101]. Therefore, in this thesis, an uncertainty analysis by means of the Monte Carlo Method will be performed by assigning probability density functions to a host of input parameters. It is believed that this method provides a better approximation of the complexity and the ambiguity of the real world, because it enables the user to account for the various possible combination of input variables that could present themselves.

4.3.3.3 Specific Overnight Costs

A study conducted on the previous generation of LWRs build in the U.S. concluded that bottom-up, or engineering-based, cost estimates have a long history of not matching up with the actual construction costs [102]. Based on historical evidence, the conclusion is drawn that several of the previous generation of cost estimation attempts have displayed poor insight into the construction time and costs of NPP construction[10]. An explanation for this could be the definition of the contingency, as given in Appendix 10.4. This does not include force majeure and several other types of occurrences that have been known to set back nuclear construction projects. Changing government regulations, for example, might require extensive design changes regarding nuclear safety. In light of the Fukushima accident, a re-evaluation of safety features of SMRs could be expected. This could potentially increase the plant construction costs [29]. Leaving out such cost categories can have a detrimental effect on the accuracy of total investment costs. Therefore, a means of incorporating them into the specific overnight costs estimate should be devised.

In reference [94] a model was created in line with the suggestions made in reference [98], in which the capital account costs were assumed to be lognormally distributed.

The author validated his assumption by means of expert interviews[11]. The experts unanimously agreed that there exists a significantly higher probability that costs will be higher than the point estimate, than costs being lower than the point estimate. Russo [94], concluded that the model which fits the 'mental model' of total cost distribution best has the best estimate without contingency located at the 10th percentile, a standard deviation set to equal the contingency and the 90th percentile being equal to 110 per cent of the best estimate with contingency costs.

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[10] It should be noted that several sources indicate that NPPs have been built on time and within budget in Asia (e.g. [234]).

[11] These expert interviews were conducted by Russo in March 2010. The interviewed experts were: Kent Williams (ORNL), Chaim Braum (Stanford), Larry Papay and Reiner Kuhr (Shaw)

The validity of this model could be disputed, because the author only partially incorporates the cost overruns of the previous generation of LWRs build in the United States. Some of these are in excess of 300 per cent. Similar cost overruns are found in other publications [100]. Russo [94] does point out however, that the spread resulting from a distribution with a range of 300 per cent, combined with the point estimate placed at the 5th percentile, would defeat the purpose of deriving a point estimate in the first place. In this manner, the spread would simply be too large to have any estimation value. The Russo Model allows for the most probable outcomes to be confined to a relatively small spread, while allowing for significantly higher values at lower probabilities. Therefore, in this thesis, the Russo Model will be used to represent the overnight costs distribution.

An attempt will be made to derive a best estimate capital cost for each SMR design, by taking the cost estimate of a LR of similar design as a proxy. Subsequently, the specific overnight costs will be corrected for the economies of scale and the effects of learning, co-siting, modularity and design and shorter construction and unit timing. If a LR of similar design has not yet been constructed, a peer reviewed cost estimate will be used. The process schematic is displayed in Figure 4.2, in effect what is shown here is that the savings factor Ψ equals

$$\Psi = \vartheta_{ES} \cdot \vartheta_L \cdot \vartheta_{CS} \cdot \vartheta_{MD} \cdot \vartheta_{CT}$$

Where ϑ_{ES} represents the economies of scale, ϑ_L the learning effect, ϑ_{CS} the effect of co-siting, ϑ_{MD} the effect of modular design and ϑ_{CT} the effect of shorter construction and unit timing[12]. This approach is a slight modification of the approach taken in Carelli et al. (2010) [103]. Furthermore, in order to ensure that the various reactor rated power outputs do not interfere with analysis of how the various economies co-depend, an attempt will be made to roughly equalize the total rated NPP power output. Furthermore, a schematic overview of the specific overnight cost methodology is given in Figure 4.2.

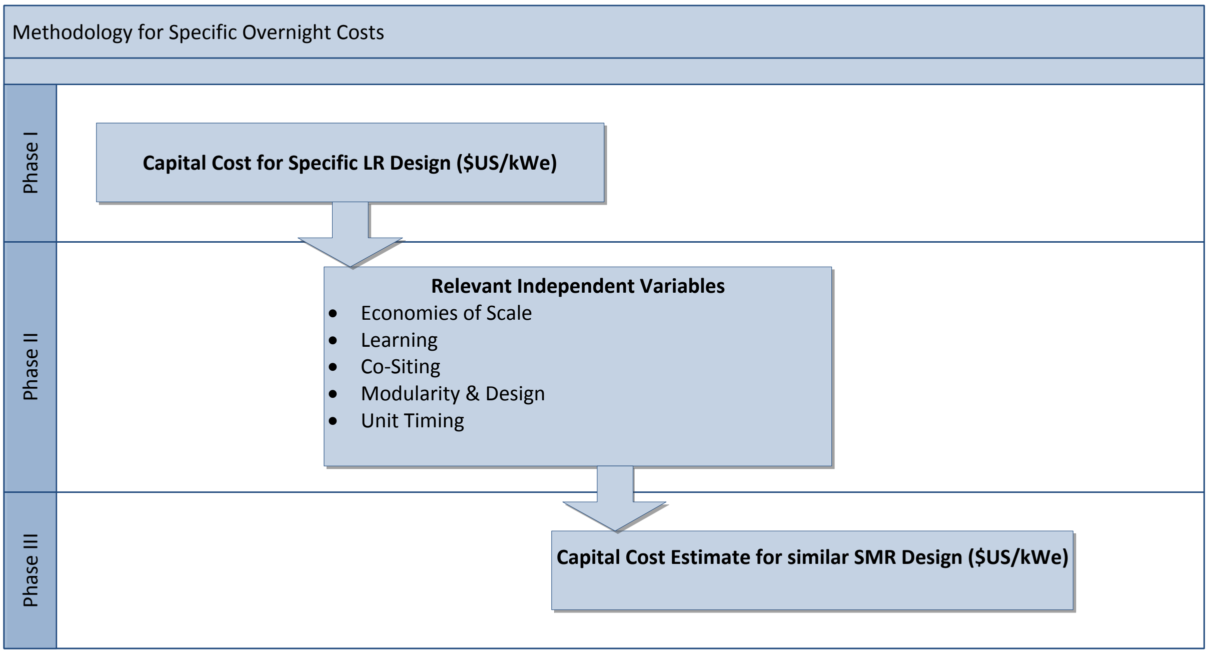

Figure 4.2| Schematic overview of the Specific Overnight Cost Methodology

4.3.3.3.1 Economies of Scale

NPPs experience falling average costs of production with increases in output. This is a result of the upfront capital costs representing a large share of the total fixed costs (e.g. [63]), this effect is referred to as 'economies of scale'. Its existence can be explained by such factors as the presence of unique set-up costs,

[12] The financial benefits of being able to add capacity in increments, as discussed in paragraph 4.3.3.3.5

such as siting activities and licensing, and poorer operating efficiency of smaller facilities as a result of lower thermal efficiency and less personnel specialization. In line with Carelli et al. (2010) [103], the economies of scale ϑ_{ES} can be derived by the following formula,

$$\vartheta_{ES} = \frac{S_{LR}}{S_{SMR}} \left(\frac{S_{SMR}}{S_{LR}}\right)^{\propto_{ES}}$$

with $\propto_{ES} < 1$

Equation 4.8

In Equation 4.8, S_{LR} equals the rated power output of the reference reactor, S_{SMR} represents the power output of a particular SMR design an $\propto_{ES}$ is a scale exponent, which relates the total capital costs of SMRs and LRs. The front term was added to correct for the differences in power output.

Considering that there are positive economies of scale, the value of the scale exponent $\propto_{ES}$ is smaller than one, it should also be noted that the closer $\propto_{ES}$ gets to zero, the larger the economies of scale effect becomes. A study conducted by Carelli et al. (2010) [103], identified that in most nuclear cost studies the scale exponent $\propto_{ES}$ is distributed between 0.5 and 0.7, with a mean value of 0.6. In line with their findings, in this thesis the scale exponent ϑ_{ES} will be assumed to be triangularly distributed with a minimum, most likely value and maximum of 0.5, 0.6 and 0.7 respectively.

4.3.3.3.2 Economies of Learning

SMRs are foreseen to be factory-fabricated before being transported to be assembled on-site. Although such a production line for NPPs has never existed, similar methods were used in the construction of marine propulsion in the Soviet Union in the period 1960-1990. An analysis of an OKBM study on the economic life-cycle of nuclear powered propulsion performed in 2003-2005 remarked that the successful operating experience exceeded 8000 reactor years [31]. In this 30 year timespan several hundred reactors were constructed and the serial production of reactor cores exhibited strong signs of economies of serial fabrication. It was concluded that cost reduction was at least 15 per cent for the second system, and approximately 5 per cent for the second, third and fourth unit, from the fifth unit onwards no further cost reductions were reported [31]. In this thesis the economies of learning by doing will be quantified in line with Carelli et al. (2010) [103], where the learning factor ϑ_L is expressed as:

$$\vartheta_L = \sum\nolimits_{n=1}^{f} \left[\frac{C_{eq} + C_{lab} + C_{mat}}{f} \cdot C_n\right]$$

with

$$C_{eq} = K_{eq} \cdot (N_{world} + N_n)^{\alpha}$$
$$C_{lab} = K_{lab} \cdot (N_{world} + 1)^{-\beta_2} \cdot (N_n)^{\beta_1}$$
$$C_{mat} = K_{mat}(N_n)^{\gamma}$$

Equation 4.9

In Equation 4.9, α equals the factory equipment learning rate, β_1 the production-site labour learning, β_2 the world labour learning, γ the material learning, n the reactor units index number, f the total number of reactor units on-site, C_n the cost escalation due to time frame, N_n progressive number of plants on-site, N_{world} total plants in the world (off-site). K_{eq} , K_{lab} and K_{mat} are the typical percentages for the equipment, labour and material costs.

In line with the EMWG Guidelines (2007) [95], the factory equipment learning rate is 0.94 per doubling, the production-site labour learning rate equals 0.90 per doubling and the material learning rate is 0.90 per doubling. The world labour learning rate is assumed to be 0.97, which is the lower limit of the range of future learning rates deemed plausible by reference [104]. Furthermore, this assumption is in line with reference [103], which states that the world learning rate should be smaller than the site learning rate. The corresponding rates for per doubling are:

$$\begin{cases} \alpha = \dfrac{\log(0.94)}{\log(2)} = -0.09 \\ \beta_1 = \gamma = \dfrac{\log(0.9)}{\log(2)} = -0.15 \\ \beta_2 = \dfrac{\log(0.97)}{\log(2)} = -0.04 \end{cases}$$

In line with Carelli et al. (2010) [103], the cost escalation rate C_n is taken to be equal to one and N_{world} is considered to be zero, because in all cases but one we are dealing with a FOAK (first of a kind) NPP. The typical percentage values for K_{eq} , K_{lab} and K_{mat} are taken from the Energy Economic Data Base (EEDB) [105], a database which was created to provide representative and consistent electric power generating, station technical and cost information to the U.S. Department of Energy (DOE). Since the last entry value were added in 1989 the costs for building NPPs have increased, but in this thesis it is assumed that the proportions of the cost categories labour, equipment and material are still accurate reflections of the cost breakdown of modern day NPPs. K_{eq} , K_{lab} and K_{mat} are taken to equal 0.10, 0.70 and 0.20 respectively [105].

4.3.3.3.3 Economies of Co-siting

It is well known that the incremental addition of reactor cores to an existing NPP site leads to so called co-siting economies, which can be defined as the existence of certain cost advantages related to siting (e.g. land rights and grid connections). This is caused by the fact that these indivisible costs have already been accounted for, which means they are of no concern to any future reactor additions. Therefore, the larger the number of reactor units per site, the smaller the required indirect investments per unit.

A new construction projects could benefits from having several installed reactor units in a similar fashion. As pointed out by Carelli et al. (2010) [103], a multiple reactor core design has the potential to distribute the non-construction related costs, or indirect, cost such as the supportive labour, construction equipment, insurance and office costs more evenly. These indirect costs form a significant portion of the total construction costs and because they only need to be incurred once, theoretically an infinite number of modules per site would result in the near complete mitigation of the indirect cost component. A more realistic way to model the co-siting effect is as follows [103]:

$$\vartheta_{CS} = \frac{1 + (n-1) \cdot (1 - F_{IND})}{n}$$

Equation 4.10

Based on the previous examples of non-recurring costs, Equation 4.10 represents the assumption that the total asymptotic savings would equal the indirect costs after the addition of n^{th} marginal unit, such that eventually the marginal costs would equal the direct construction costs, as shown by the expression:

$$\lim_{n\to\infty} (\vartheta_{CS}) = 1 - F_{IND} = F_{DIR}$$

In these expression, F_{DIR} and F_{IND} represent the proportional size of the direct costs and indirect costs respectively. Values for the percentages of indirect costs vary per source. The EEDB [105] reports that indirect cost fraction went from 30 per cent in 1978 to 53 per cent in 1987. An analysis performed by Cohen [106] concludes that this cost increase can largely be explained by increased professional labour costs for design, construction and quality control engineering following an increase in regulatory restrictions. Carelli et al. (2010) [103] utilizes an indirect cost proportion of 34 per cent, situated in the range proposed by the EEDB. Continued increases in regulatory restrictions could cause the indirect cost percentage to rise, it should be noted that in the wake of the Fukushima disaster a number of disadvantages of co-siting were also revealed. In recognition of these facts it was decided to represent the indirect costs as a triangular distribution with a minimum of 0.30 and a maximum of 0.40. To err on the side of caution, the most likely value was taken to be 0.34, in line with Carelli et al. (2010) [103].

4.3.3.3.4 Economies of Modularity and Design

As addressed in paragraph 4.3.3.3, by the end of 1990s, following several prominent nuclear accidents, NPP designs had become much more complicated as a result of the complexity of the addition of multiple engineering patches and active safety systems. For example, features such as high pressure water injection systems would keep the reactor core flooded in the case of a loss of coolant accident, but they drove NPP construction costs up significantly. A possible solution to this problem was found in the form of the integral design approach, which revolves around eliminating the components that require the presence of an active system. Removing the high pressure water injection system from the design, for example, would require the removal of all large primary cooling pipes. A typical integral design incorporates the reactor core, steam generator and pressurizer into a single common pressure vessel [107]. An integral reactor design, therefore, is expected to have lower construction costs and maintenance requirements, while, at the same time, not requiring operator intervention or actively engineered safety systems to mitigate the consequences of an accident scenario. The primary drawback of the integral design approach is that above a certain rated power integral reactors would become prohibitively difficult to manufacture and transport. Ingersoll (2009) [108] estimates this limit to be in the 300-400 MWe range. Therefore, integral reactors are necessarily made modular in an attempt to compensate for economies of scale by means of economies of replication ('mass production').

The use of the integral design approach could lead to cost reductions per MWe as a result of a reduction in the requirement of cubic metres of concrete, tons of steel and the number of forged safety components at the expense of economies of scale[13]. The complete integration of primary components inside the pressure vessel avoids an intricate piping network in the balance op plant, which dramatically enhances the safety level and increases the compactness of the plant design. This might improve overall plant security as a result of its smaller ground imprint and reduced skyline, which makes the reactor a smaller target for a terrorist attack by airplane, especially if the reactor core is built partially underground as is foreseen in several designs.

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[13] A possible objection is that the amount of concrete and steel required to construct four 400 MWe is most likely larger than the amounts required to construct one 1600 MWe reactor, but this is already covered by the negative economies of scale discussed in paragraph 4.3.3.3.1

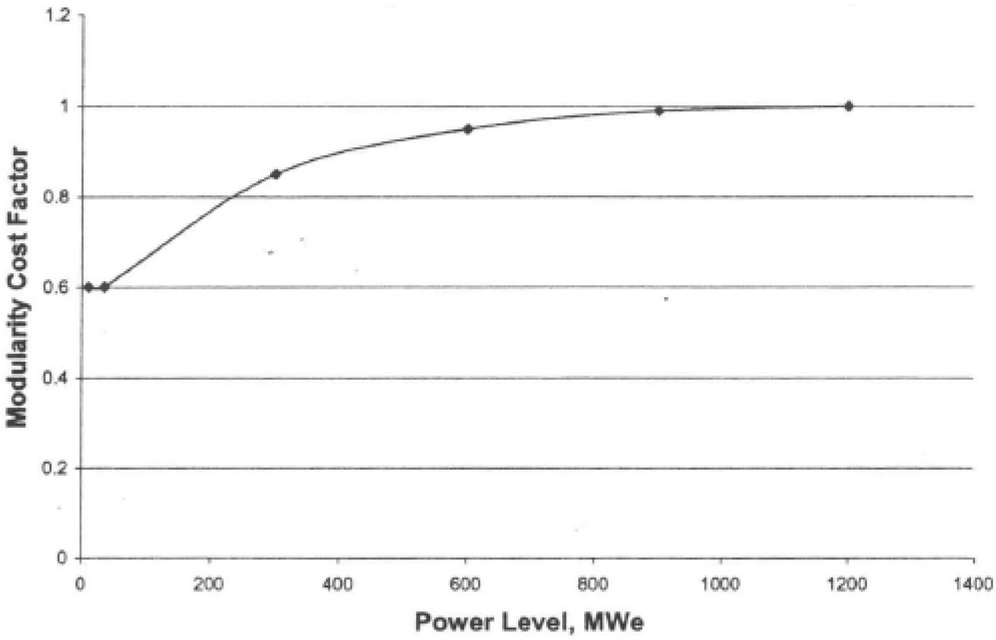

Figure 4.3| The potential cost advantages which can be attained by modular design features as a function of the power output

Taken from Reid (2003) [109]

The abovementioned implies that the size of the design simplifications (modularity effect) ϑ_{MD} is proportional to the size reduction of the reactor. A study conducted by the Oak Ridge National Laboratory (ORNL) attempted to quantify this relationship for NPPs by using available data on the impacts of modularity on the construction cost of housing and coal fired power plants [109]. Some basic conclusions were that modular-built, 2000 square feet homes with a factory fabrication percentage of 85 per cent, can achieve a modular-construction cost ratio of 0.85. For mobile homes, which have a factory fabrication percentage of 98 per cent, the cost ratio was found to be as low as 0.61. These findings were validated by assessing the modularity induced savings for 300 MWe pulverized coal-fired power plant and a fluidized-bed boiler plant design. The corresponding cost ratios were found to be 0.86 and 0.87, which is in accordance with the values found for modular-built housing. Figure 4.3, is based on the assumption that: (1) above 900 MWe, very little cost reduction can be achieved by means of modular-construction, therefore the cost ratio is taken to be 0.99 at 900 MWe and 1.0 at 1200 MWe, (2) at 600 MWe the cost ratio roughly equals 0.95, which is based on a private conversation with a reactor vendor described in Reid (2003) [109], (3) at 300 MWe the cost ratio is taken to be 0.85, which is in line with the cost ratio for modular housing at 85 per cent factory fabrication, (3) at power levels below 35 MWe the reactor is assumed to be completely factory fabricated, with the corresponding cost ratio taken to be as low as 0.6. After fitting the data in Excel the following polynomial series was found:

$$\vartheta_{MD} = 4 \cdot 10^{-10}(P_r)^3 - 10^{-6}(P_r)^2 + 0{,}0012(P_r) + 0.581$$

Equation 4.11

Equation 4.11 gives modular design cost ratio of ϑ_{MD} as a function of the rated power P_r in MWe

4.3.3.3.5 Economies of Construction Schedule and Unit Timing

It is reported by Carelli et al. (2007) [110], that as a result of a shorter construction timescale and incremental capacity additions allowing for a better fit between supply and demand, the cumulative expenditures will be approximately 6 per cent less for a NPP composed out of multiple SMR modules. This saving is compiled of two interrelated components: (1) the shorter construction time and (2) considering that SMR investments take place in smaller increments due to the relatively limited scope of their construction intervals, the plant capacity can be better adapted to the changing market conditions. It is reported by Carelli et al. (2010) [103], that especially the latter can have far-stretching implications on the generation costs, revenues and financial costs.

Firstly, the possibility of a reduced construction time signifies that the investment timing can be shortened, without limiting the capacity or losing revenue. This shorter construction time could result in a higher net present value. In general it can be argued that for a given size, multiple SMRs could have lower financial costs than a single LR. Additionally, in a growth market the SMR construction can be sequenced or concentrated. The former indicates that the final SMR will be added to the grid at the same time as the LR, while the latter suggests that all SMR are simultaneously constructed resulting in an earlier grid connection than the LR. Although it is reported that the total investment costs could in fact be larger, both scenarios allow the operator to acquire earlier and larger revenues that would have been foregone in the case of a LR [103].

4.3.3.4 Discount Rate and the Fixed Charge Rate

SMRs are designed with competitiveness in mind. Therefore, they should be a viable option for utility companies operating in present day and future deregulated electricity markets. Investment decisions undertaken by the utility companies operating in these markets are often based on maximizing the return on investment while taking into consideration the acceptable levels of risk and regulatory constraints. Under these restrictions it is not only important that an investment project is shown to be cost effective, but also that the expected financial return is high enough to offset the risk undertaken by the investors. Assessing the possible future returns is a modelling exercise which requires intimate market knowledge. Although the construction of a complete market model is deemed beyond the scope of this thesis, a competitive discount rate will be used to determine the economic attractiveness of a typical SMR design. The aim of this approach is to assess SMRs according to the financial requirements that a typical privately owned company would adhere to. This thesis expands on the knowledge base by incorporating an alternate financial prioritizing mechanism than the one incorporated in conventional deregulated market models (e.g. MIT (2009) [111]). An attempt will be made to construct the financial parameters by means of the bottom-up method using published market data.

The discount rate r_d is a means of valuating a series of costs and benefits which are to occur at a future time *t* in the present. There are numerous ways of deriving a value for the discount rate and in most cases the techniques are based more on experience than on actual science. However, as a result of the discount rate strongly influencing the outcome of an economic analysis, the method of derivation merits some consideration. Two factors that are believed to play a role are the risk premium, which is the difference in average real return between equity and government bonds, and the risk-free rate of which the proxy is usually the expected return on the aforementioned government bonds with an expiration date matching that of the investment project under consideration. In the Capital Asset Pricing Model (CAPM), which forms the basis of modern portfolio theory, the discount rate is considered to be the necessary return required to offset the risk of obtaining an undesirable investment outcome. As a result of many investment portfolios consisting out of both debt and equity, a widely deployed method for approximating the discount rate is the Weighted Average Cost of Capital (WACC), represented by Equation 4.12 (e.g. [112]):

$$r_d = WACC = i_e \frac{E}{D+E} + i_b(1 - T_e)\frac{D}{D+E}$$

Equation 4.12

Here, i_e equals the cost of equity, i_b equals the cost of bonds (debt), T_e stands for the corporate tax rate, which is 26.3 per cent in Sweden [113] and $\frac{E}{D+E}$ and $\frac{D}{D+E}$ represent the ratio of equity and debt respectively.

4.3.3.4.1 Cost of Equity

The WACC can be considered to be a blend of the cost of equity and the after tax cost of debt. Although, the cost of equity, which can be considered a theoretical payment made to the providers of equity, varies from nuclear cost study to nuclear cost study due to its dependence on several national market characteristics, in most cost studies its assumed value is usually in the 12 to 15 per cent range [114].

In this thesis, the cost of equity will be determined by means of the relation between risk and expected return observed in the CAPM using historical Swedish market data. In the CAPM the expected rate of return of an equity investment, in an efficient market portfolio can be derived using Equation 4.13 (e.g. [115]):

$$i_e = R_f + \beta \cdot (E(R_m) - R_f)$$

Equation 4.13

Where R_f stands for the risk free rate, β is an approximation of the sensitivity of the asset's return to variation in the market return and $(E(R_m) - R_f)$ is the expected risk premium above the risk-free rate on a market portfolio. In a recent study by Sörenssen (2011) [116], it was established that in the period 1998-2010, the expected equity risk premium varied between 3.5 and 5.4 per cent. The analysis was performed using a dataset of 13 surveys from PricewaterhouseCoopers, which conducts an annual survey among corporate finance firms, stockbrokers, fund managers and insurance companies. The distribution will be considered to be normally distributed, with the outliers representing the 5^{th} and 95^{th} percentile. Furthermore, β is a stock or portfolio specific variable. Therefore, in order to form a proxy for the β value of a nuclear power investment three β values of exchange traded funds are compared. These are PowerShares Global Nuclear Energy (PKN, $\beta = 1.22$), Market Vectors Uranium + Nuclear Energy (NLR, $\beta = 1.32$) and iShares Global Nuclear Energy Fund (NUCL, $\beta = 1.09$). In line with these findings $\beta = [1.09, 1.32]$.

A risk free investment can be defined as an investment in which there is no risk of default and in which there is no reinvestment risk over the length of the investment. The first restriction limits the scope of the securities to government bonds, while the second requires the security to be of a similar timeline, or at least as close as possible, to the timeline of the intended investment project. Taking into consideration the typical payback period of a NPP, the long term government bond could be considered to be a good proxy for the risk-free rate. Coincidentally, following the loss of the United States its AAA credit rating and the currency turmoil in European Union, Sweden is currently considered to be one of the safest investments in Europe, if not in the world [117]. The central bank of Sweden (Sveriges Riksbank) provides access to a database on the forward rated of 10-year Swedish government bonds, in the period from April 2002 onwards [118]. The time interval can be taken in days, weeks, months, quarters and years.

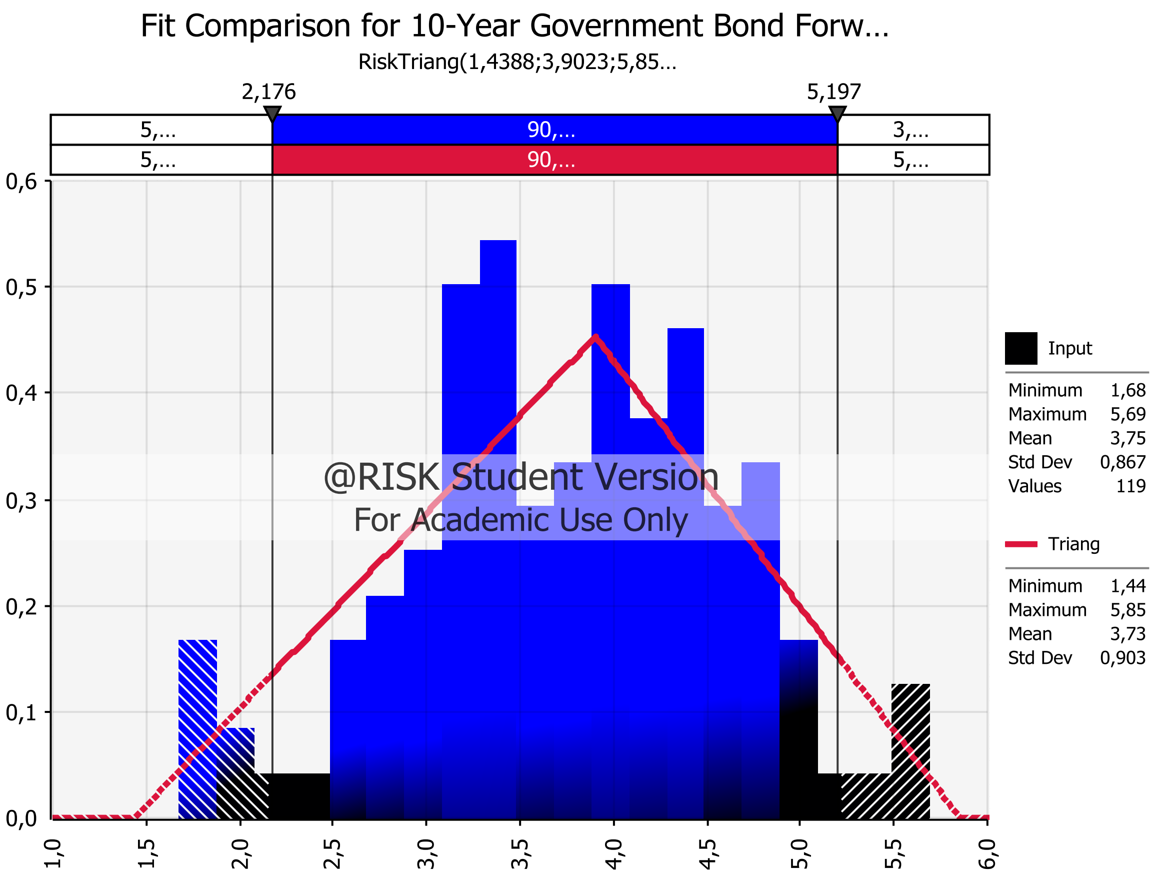

Figure 4.4| Probability distribution of the return on a 10-year Government Bond

In line with Sörenssen (2011) [116] the yield to maturity on a 10-year bond was taken in this thesis to be the best available proxy for the risk-free rate. In order to acquire a decent sample size, the time interval was taken in months, from April 2002 till March 2012. The distribution is displayed in Figure 4.4

4.3.3.4.2 Cost of Debt

The cost of debt does not only vary from market to market but also from company to company and is typically perceived to be a function of the change in a company's debt to equity ratio. Rothwell [100] reports that although the relation between the cost of capital and the ratio between the investment size and the enterprise value is a complex one, it can be represented in simplified manner as follows.

$$i_b = R_f e^{\psi\left(\frac{KC}{PV}\right)}$$

Equation 4.14

Here, R_f represents the risk free rate, ψ is the proportional rate of change in i_b and $\frac{KC}{PV}$ portrays the ratio of total capital investment costs (KC) and the net worth of a NPP owner/operator (PV). The proportional rate of change ψ is related to the Absolute Risk Aversion (ARA), which is a part of the economic theory of risk.

As described in Appendix 10.4, it is a reasonable assumption to set the contingency equal to the standard deviation of a log-normal distribution. However, the EPRI risk guidelines given in Rothwell (2004) [98], indicates a contingency range and not a point estimate. This implies that the uncertainty estimate is subject to some uncertainty as well. The proportional rate ψ can therefore be seen as a ratio of the market uncertainty sentiment and the contingency estimate. For example, if the market's risk perception were to equal the upper bound of the EPRI contingency range for a preliminary estimate, 30 per cent, than ψ would equal 1.64. This is the quotient of market risk perception and the value of 18.3 per cent assumed to be a representative contingency for a preliminary estimate in this thesis.

Finally, the ratio $\frac{KC}{PV}$ of the total cost of capital and the firm net worth can assume a range of value. The market capitalization rate can be used as a proxy for the market's opinion on net worth of a company. In the range of potential SMR customers, there are firms with a market capitalization of 110 billion US$ and 85 billion US$, representing EDF and E.ON respectively [119], but also smaller firms, which might have a market value just above the estimated total capital costs of a SMR. To simulate the effects of the firm size on the cost of debt interval of 0.05 to 0.5 will be taken for the ratio $\frac{KC}{PV}$.

4.3.3.4.3 Fixed Charge Rate

In accordance with Bunn et al. (2003) [120], the fixed charge rate F_{cr} is the fraction of the initial investment that is collected each year to reimburse the construction costs and other up-front costs by using a return on investment. In line with the recent literature on NPP construction in deregulated electricity markets (e.g. Locatelli and Mancini (2010a) [114]), it is assumed that the SMR will be privately owned and financed. Firstly, as previously mentioned this would imply that every SMR is financed by a combination of debt and equity. Secondly, private companies are obligated to pay corporate income tax, but they are also entitled to subtract deductible expenses such as dividends paid on bonds and depreciation costs. The formula for the fixed charge rate F_{cr} is given by the following formula [120]:

$$F_{cr} = \frac{1}{1-T_e}\left[\frac{(1-b)i_e}{1-(1+i_e)^{-t_d}} + \frac{di_b}{1-(1+i_b)^{-t_d}} - \frac{T_e}{n} - di_b t_e\right]$$

Equation 4.15

In Equation 4.15, b is equal to the share of debt in the financing structure, T_e being equal to the corporate tax rate, i_e being equal to the cost of equity, i_b represents the cost of debt and t_d indicates the amount of time in which the plant investment is depreciated. It is reported to be common practice to depreciate the plant in approximately half the design lifetime for tax purposes [120]. Considering the large variation in the design lifetimes could interfere with the implicit ceteris paribus assumption, a depreciation period of 15 years will be assumed, which is in line with Du and Parsons (2009) [61].

Although, the fixed charge might seem like a discounted variant of the WACC, an important difference is that the debt and equity percentages are not necessarily the same. This is because the fixed charge rate is a project specific value, while the WACC is typically a company specific variable. When taking into account the advantages that can be obtained from the tax exemptions of debt, it is to be expected that every plant owner operator will try to maximize its debt to equity ratio. Therefore, a debt percentage will be taken to lie in a 50 to 80 per cent interval.

4.3.3.5 Fixed and Variable O&M Costs

The Operations and Maintenance (O&M) costs are the costs that need to be incurred to ensure the continued operation of the NPP in question. In this thesis the O&M costs are divided into the fixed O&M costs c_{fom} and the variable O&M costs c_{vom}. It should be noted that the prefixes 'fixed' and 'variable' are not defined in the traditional manner, considering that for NPPs most of the costs that fall into these categories do not depend on the amount of electricity produced.

4.3.3.5.1 Fixed O&M Costs

The fixed O&M costs c_{fom} are an expression for the provision that is set aside at start of the plant operational lifetime to cover the costs of the dismantling and decommission (D&D). Locatelli and Mancini (2010b) [121] provide a complete overview of all the cost drivers and the effect they have on the total D&D costs. For the purpose of distinguishing between the D&D cost scenarios of the various designs, two cost categories are more likely to contain variations than the others. These are: (1) the dismantling activities and (2) the waste processing, storage and disposal.

The expression for the size of the annual provision, or annuity, which needs to be made in order to cover the end-of-lifetime expenses, is given in line with Bunn et al.(2003) [120] to be:

$$c_{fom} = F_{mu} F_s F_{ts} c_{ddl} F_{dd}$$

Equation 4.16

In Equation 4.16, F_{mu} stands for the multiple unit correction factor, F_s portrays the economies of scale, F_{ts} is the correction factor for the cost reduction due to technical savings, c_{ddl} equals the specific D&D costs of a LR in US$_{2011}$/kW(e) and F_{dd} represents the annuity factor.

A study performed by Locatelli and Mancini (2010b) [121], concludes that, with respect to the economies of scale F_s, the decommissioning costs for small-medium reactors are a factor 3.09 higher than those for their larger size counterparts. Similar to the construction process, reactor sites with multiple units that require decommissioning experience co-siting savings. Therefore, the cost of decommissioning two reactors on one site is less than the cost of decommissioning two reactors on two sites. The second unit decommissioning cost was found by Locatelli and Mancini (2010b) [121] to be only 2.16 times as high due some non-recursive costs. Furthermore, it was assumed that the D&D costs for any additional power units would be the same as for the second unit.

The abovementioned numbers were computed for a 335 MWe reactor, which is larger in terms of power output than all of the reactor modules under investigation in this thesis. However, this figure will be assumed to be representative for the entire range of SMR output values. It should be noted that this value might be slightly optimistic when taking into consideration that an exponential increase in the specific decommissioning costs is reported following a decrease in reactor output [121]. Combining the abovementioned cost figures results in the following empirical formula for F_{mu}:

$$F_{mu} = \frac{3.09 + 2.16(N-1)}{3.09N}$$

Equation 4.17

In Equation 4.17, N equals the number of power modules installed on site. The numerical factors are the economies of scale corresponding to the D&D costs of the first and second module as established in Locatelli and Mancini (2010a) [121].

F_{mu} could also turn out to be a rather conservative estimate due to the lack of data concerning the decommissioning of power plants with more than 2 reactor units. It could be argued, for example, that the numerical factor will continue to diminish as N increases as a result of additional non-recurring fixed costs. For example, the IAEA envisions that the decommissioning of SMRs could follow the same outline as the construction process [23]. The reactors could be sent back to the factory in the same way they arrived on-site, in assembled form. Subsequently, all reactor cores would be mass disassembled at the factory site. It can be argued that the disassembly and recycling of the components of a decommissioned NPP at a centralised facility will result in lower costs than an on-site D&D operation as a result of the favourable economies of scale following from a mass dismantling operation. This effect however, has not be quantified as of yet. Moreover, it will be assumed that the slightly optimistic value for F_s and the slightly pessimistic value for F_{mu} cancel each other out.

The technical savings factor F_{ts} is an indication of how the use of innovative design features can reduce the overall D&D costs. In particular the reduction of radioactive components resulting from integral design features could reduce the amount of required waste processing and advanced (passive) safety features could reduce the amount of dismantling activity required. In line with Locatelli and Mancini (2010a) [114] the value for F_{ts} is taken to equal 0.81. Furthermore, 500 US$$_{2001}$ /kW(e) is given to be a good approximation for c_{ddl} [122] which, when corrected for inflation, results in 635 US$$_{2011}$ /kW(e)[14].

Finally, the annual provisions made to cover the D&D expenses are usually deposited into a fund. As can be derived from Equation 4.18, the size of the annuity can be determined by means of the annuity factor F_{dd}. This factor takes into consideration the expected return on investment, which enables the derivation of the minimum annual provision which will be required to reach the aspired fund value at the end of the specified period of time. In line with Bunn et al. (2003) [120] the formula for the annuity factor is taken to be:

$$F_{dd} = \frac{R_f}{(1+R_f)^{t_d}-1}$$

Equation 4.18

In Equation 4.18, R_f equals the risk free interest rate and t describes the interval over which the annuity is paid. Following common practice, this period is taken to equal the period in which the construction costs are remunerated, which was taken to be 15 years as described in Paragraph 4.3.3.4.

4.3.3.5.2 Variable O&M Costs

The variable operating costs c_{vom}, given in US$$_{2011}$/kWe, can be divided into the labour costs and the miscellaneous costs. The labour costs are the product of the amount of employees and average labour rate per employee plus benefits, as shown in Equation 4.19 [100].

$$c_{vom} = \frac{p_L L + M}{P_r}$$

Equation 4.19

In Equation 4.19, p_L equals the annual labour rate, L the required number of employees, Mrepresents the miscellaneous costs and P_r the rated pwer. Rothwell (2011 [100]) attempted to fit the required number of U.S. NPP employees L to the rated power level P_r by using publicly available light water NPP employment records. It was determined that a 'semi-log' model, as given by Equation 4.20 would have the most appropriate functional form.

$$\ln(L) = \alpha + \epsilon(P_r \cdot 10^{-3})$$

Equation 4.20

In this formula α represents the minimum number of employees and ϵ represents the rate of growth per MWe of added capacity P_r. After fitting the functional form to the available data, it was determined that the distribution and the data had the highest correlation if α and ϵ were taken to be 5.55 and 0.87 respectively. This implies that the minimum amount of employees in an operational U.S NPP is roughly 257 and that per 100 MWe additional capacity, approximately 23 employees are added to the labour pool. The general form of the formula for the minimal number of employees can therefore be expressed as [100].

$$\ln(L) = 5.55 + 0.87(P_r \cdot 10^{-3})$$

Equation 4.21

In Equation 4.21 the amount of personnel L is a function of the rated power output P_r in MWe. In line with Rothwell (2011) [100], the average annual labour rate is taken to be 80,000 US$$_{2011}$/yr. The total labour costs have been reported to be approximately 67 per cent of the variable O&M costs, with the remaining 33 per cent being expenditures on maintenance and material supplies [123]. These are referred to as the

[14] The U.S. Bureau of Labor Statistics Inflation Calculator (http://www.bls.gov/data/inflation_calculator.htm)

miscellaneous costs M. However, the same source also indicated that since the accident at Three Miles Island the miscellaneous costs M have been approximately 30 per cent understated, as a result of significant increases in insurance costs and regulatory fees. In line with Rothwell (2011) [100], the following adjustment should be made:

$$M = \frac{33}{67 \cdot 0.7} p_L L \cong 0.70 p_L L$$

As shown above the miscellaneous costs M are in fact 70 per cent of the labour costs instead of 50 per cent. Combining this results in the following expression for the variable O&M costs c_{vom}:

$$c_{vom} \cong \frac{1.70 p_L L}{P_r}$$

Equation 4.22

It is assumed in Equation 4.22 that the relations and conditions regarding the division of O&M costs has a general validity and is applicable for all high-income economies. Furthermore, due to the unavailability of employment data for several of the designs still under development, it is assumed that - due to the employment levels in nuclear power plants being strongly influenced by national and international regulations - they will not differ that much across the various designs. It should be noted that. reductions in O&M costs in SMRs are occasionally taken to come from reduced staffing requirements, However, some national regulations require a certain minimum amount of security staffing, which is independent of plant capacity. Although this could halt the spread of SMRs designed to serve the needs of isolated communities in sparsely populated areas, such conditions are hardly encountered in the high income economies this thesis aims to serve [29].

4.3.3.6 Nuclear Fuel Cycle Costs

4.3.3.6.1 Nuclear Fuel Cycle with Direct Disposal

The Nuclear Fuel Cycle, as shown in Figure 4.5, can be defined as a series of processes which are concerned with the preparation, use or disposal of nuclear fuel. Of these processes, the subset which is related to the creation of nuclear fuel out of uranium ore is referred to as the front end of the nuclear fuel cycle. Similarly, the processes following the end of irradiation period in the reactor core; temporary storage, reprocessing and geological deposition or final storage, are referred to as the back-end of the nuclear fuel cycle.

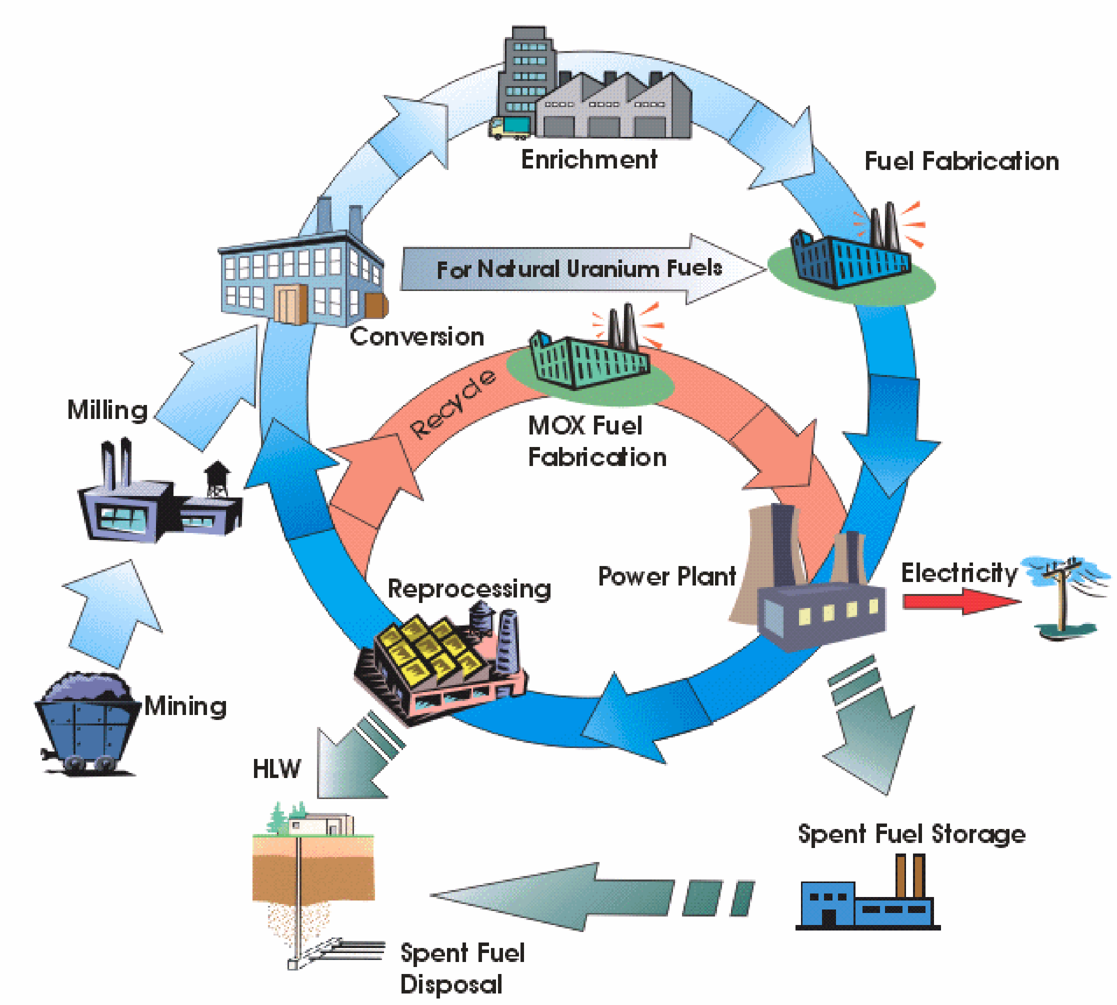

Figure 4.5| Flow diagram giving a schematic representation of the once-through and the MOX fuel cycles

Taken from the IAEA Website [124]

In the power plant stage, the accounting of nuclear fuel requires precise knowledge on lead times in-between fuel loading from the first to the last core fuel load. For the purpose of simplifying the analysis it will be assumed in this thesis that the fuel costs are spread out evenly over the lifetime of the reactor. Unlike fossil fuel based plants, which expense fuel costs following plant operation, nuclear fuel is paid for over time. Therefore, it should be treated as a capital cost which is continuously written off, irrespective of whether the plant is in use or not. Following Rothwell (2011) [100], the following formula will be used to determine the thermal reactor fuel costs c_f in US$_{2011}$/kWh.

$$c_f = \frac{\left[C_{FFC} + \frac{C_{IS}}{(1+r_d)^{t_{is}}} + \frac{C_{DS}}{(1+r_d)^{t_{ds}}}\right] F_C}{24 \cdot 10^3 B\eta}$$

Equation 4.23

In Equation 4.23, C_{FFC} represents the cost of the front end of the fuel cycle (US$_{2011}$/kgHM), C_{IS} stands for the interim storage costs, C_{DS} equals the direct disposal costs, r_d is the discount rate, t_{is} is the time in-between fuel discharge from the reactor and the fulfilment of the interim storage payments and t_{ds} is the lag time until the irradiated fuel is placed into geological disposal. Furthermore, B is the burn up rate measured in MWd per kg of heavy metal (kgHM) and η is the thermal efficiency. It should be noted that the lag time t_{ds} represents an important consideration the evaluation of the direct disposal costs. When considering that, in the case of first fuel batches, disposal might not occur for several decades it might seem excessive to reserve an amount equal to C_{DS} at discharge. However, a substantial part of the activities involved with the creation of a repository occur early on, such as site work and the construction process of the repository and its corresponding infrastructure placement. Therefore, the costs for disposing of the spent fuel will be taken to occur immediately after the fuel is discharged from the reactor core. Subsequently, $t_{ds} = 0$. Bunn et al. (2003), C_{IS} is reported to equal 200 US$_{2003}$/kgHM. Corrected for inflation, in this thesis C_{IS} is taken to equal 244 US$_{2011}$/kgHM[15]. Furthermore, F_C represents the carrying charge factor, which is given by:

$$F_c = \frac{\frac{\tau}{365L_f} r_d (1 + r_d)^{\frac{\tau}{365L_f}}}{(1 + r_d)^{\frac{\tau}{365L_f}} - 1}$$

Equation 4.24

In Equation 4.24, τ is the number of Effective Full Power Days (EFPDs); a measure for length of the irradiation cycle, L_f is the load factor and r_d is the discount rate.

The front-end of the uranium fuel cycle is usually viewed as being composed of four distinct market segments, these are: (1) uranium mining and milling, (2) uranium conversion, (3) uranium enrichment and (4) nuclear fuel fabrication. When combining the first two segments, the price per kilogram of heavy metal (US$/kgHM) as complied by the value additions per segment, can be described as follows:

$$C_{FFC} = \frac{R_{NU} \cdot C_{MMUC}}{(1 + r_d)^{-t_{uc}}} + \frac{N_{SWU}.C_{SWU}}{(1 + r_d)^{-t_s}} + \frac{C_{FAB}}{(1 + r_d)^{-t_f}}$$

Equation 4.25

In Equation 4.25, R_{NU} is the ratio of natural uranium input to enriched uranium output, C_{MMUC} is the cost of U_3O_8 plus the cost of converting it to uraniumhexafluoride (UF_6), N_{SWU} is the number of Separate Work Units required, C_{SWU} is the cost of UF_6 enriching, t_{uc} is the lead time from uranium purchase to reactor loading t_s is the lead time from conversion to reactor loading, t_f is the lead time from fabrication to reactor loading, C_{FAB} is the cost of fuel fabrication and r_d equals the discount rate.

Following Bunn et al. (2003) the values for t_{uc}, t_s and t_f will be taken to be 2.0, 1.0 and 0.5 years respectively. Furthermore, P_{UF6} and P_{FAB} have been reported to be historically stable at v 8 US$_{2011}$/kgHM and 250 US$_{2011}$/kgHM respectively [125]. The value for R_{NU} can be determined by means of the expression.

$$R_{NU} = \frac{(\chi_P - \chi_T)}{(\chi_F - \chi_T)}$$

Equation 4.26

In Equation 4.22, χ_P equals the percentage of U-235 in the product, χ_F is the percentage of U-235 in the feed and χ_T is the percentage of U-235 in the tail. The amount of separate work units (SWUs) can be calculated by means of the following formula:

$$N_{SWU} = V(\chi_P) - V(\chi_T) - R_{NU} \cdot [V(\chi_F) - V(\chi_T)]$$

Equation 4.27

In which

$$V(\chi_i) = (2\chi_i - 1) \ln \left[\frac{\chi_i}{(1 - \chi_i)}\right]$$

---

[15] The U.S. Bureau of Labor Statistics Inflation Calculator (http://www.bls.gov/data/inflation_calculator.htm)

$$\text{with } i = F, P, T$$

Equation 4.28

In the abovementioned formulas $V(\chi)$ is dubbed the 'value function' with $V(\chi) = \langle 0,1 \rangle$. Typical values are 0.711 per cent for the feed χ_F and 5 per cent or less for the product χ_P, which is the limit for low enriched fuel. The only variable which does not follow readily from either the natural enrichment percentages or an industrial practice is the tail percentage χ_T (also referred to as the tail assay), which is usually determined by means of optimisation. One such approximation is given by Equation 4.29, in which the optimal tail assay is given by the logarithmic function [100]:

$$\ln(\chi_T) = -6.085 + 0.468(\ln \varphi) - 0.0074(\ln \varphi)^2$$

Equation 4.29

Where,

$$\varphi = \frac{C_{SWU}}{C_{MMUC}}$$

Equation 4.30

As can be derived from Equation 4.30, the optimal tail assay is determined by the relation between the price per kilogram of SWU and the price per kilogram of UF_6. It follows readily form Equation 4.29, that if there is an increase in P_{SWU} relative to P_{UF6}, the optimal tail assay decreases. This can be explained by that, when there is a relative price increase in the enrichment process, one want to minimize the amount of SWU required. This implies that the product enrichment per SWU should be as high as possible and that the tail assay should therefore be minimized. Similarly, if there is a relative increase in the price of UF_6, it is in one's best interest to minimize the additional UF_6 purchase by leaving the tail assay (which is fed back into the enrichment facility) at a slightly higher enrichment level. The result is that running more SWU cycles is favoured over purchasing more UF_6.

The price of uranium mining and milling, which will henceforth be referred to as the uranium spot price, has known several periods of spot market price volatility since it first became possible to privately own uranium deposits. For the purpose of providing an insight into the uranium market, a short history will be given. A more detailed account can be found in, what is referred to as, the 'Red Book' of which the most recent version was published in 2009 [126].

Following the introduction of uranium onto the free market, the price development was relatively uneventful until the oil crisis of 1973 which caused the price of uranium to skyrocket to 287 $US\$_{2011}$/kgU in 1976 [100]. Subsequently, following the accident at Three Miles Island in 1979 several NPPs were retired which resulted in several electric utility companies leaving the market. From this period until the end of the 1990s there was systematic overproduction of uranium, which resulted in a near-continuous price decrease. In addition to this, the fall of the USSR opened up the extensive Russian plutonium market to western companies, which could be utilised in the form of MOX fuel; a partial substitute for traditional uranium fuel rods. A short price upheaval occurred following a reduction in exploration and exploitation activities and the bankruptcy of a major uranium trading company. In 2001 the uranium price began to rebound from its historical low streak for a variety of reasons. Several of these reasons were: (1) problems in various nuclear fuel production facilities, (2) the weakness of the US$, (3) the expansion of nuclear power in Asia and (4) the increased awareness among governments of the possibilities of nuclear power for CO_2 mitigation [126]. Since the peak price hit roughly 300 $US\$_{2011}$/kgU in June 2007, the price has come down again following a consumption shift to MOX fuel, general market corrections and the reluctance of the traditional buyers to trade at that price level [126]. As of June 2011 the uranium spot price has resided around the 110 $US\$_{2011}$/kgU.

In line with Rothwell [100], in this thesis the spread in the uranium spot price between 1990 and 2010 will be taken to be representative for all future uranium spot price scenarios. Figure 4.6 represents the distribution fitted to a dataset of uranium metal prices by Rothwell [100]. As can be derived from this figure the mode lies around 40 US$$_{2011}$/kg-U. However, due to the highest reported value equalling 339 US$$_{2011}$/kg-U, significant skewness is added to the distribution, the best fit is a Pearson distribution with a mean of approximately 72 US$$_{2011}$/kgU. When combining this mean value with the point value taken for the UF_6 conversion cost (8 US$$_{2011}$/kgU), the mean value for C_{MMUC} roughly equals 80 US$$_{2011}$/kgU. This is a bit lower than the often cited 2009 MIT study [63], where the point estimate used for the combined cost of uranium and the conversion to UF_6 equals 93 US$$_{2011}$/kgU[16], but still was still considered to lay within an acceptable range. The difference could be explained by assuming that a contingency factor was added to the uranium price in the MIT study.

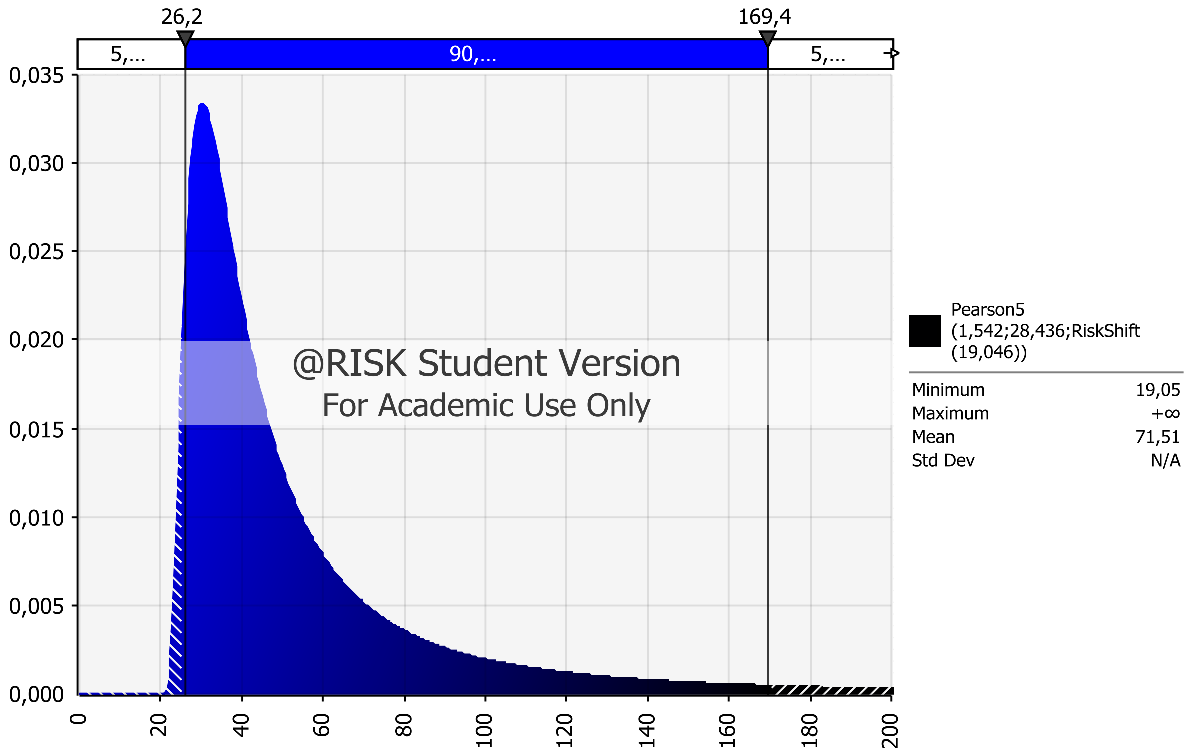

Figure 4.6| Probability distribution for the spot market price of U3O8 in US$$_{2011}$/kgHM

Increasing the percentage of fissile U-235 in natural uranium from its initial value of roughly 0.7 per cent to fuel-grade specifications is called uranium enrichment. There are two techniques in use for commercial enrichment; gaseous diffusion and gas centrifuge. It has been reported that in the period 1989-2009 the spot price for natural uranium, measured in SWUs, increased from 80 US$$_{2008}$/kgHM[17] tot 160 US$$_{2008}$/kgHM [127]. One possible explanation for this trend is the continued operation of the less economic gaseous diffusion plants. The reason they are less economic is that their production costs are 85 per cent dependent on the electricity price [127]. The electricity prices almost consistently increased in the specified period, which explains why their production costs have been undercut by the more efficient gas centrifuges. However, as a result of several barriers to entry (e.g. increasing returns to scale, predatory pricing) several of the inefficient gas diffusion plants have, as of yet not been retired. It is expected that within a decade, this inefficient capacity will be retired and the high artificial price level will drop significantly. Artificial in this

[16]U.S. Department of Labor Statistics Inflation Calculator (http://www.bls.gov/data/inflation_calculator.htm)

[17] US$/kgHM is a customary unit in nuclear fuel-cycle literature and refers to the acquisition cost of an item or service normalized in terms of the heavy metal content of nuclear fuel at the instance before it was loaded into the reactor core (fresh fuel). Therefore, for example, the cost of storing an arbitrary quantity of nuclear fuel is often expressed in terms the amount of fresh fuel that was required to generate it.

context refers to the entry barriers; in an efficient market new entrants could have forced the retirement of the diffusion technology much sooner. However, it should be noted that even if cartel formation could be minimized, new market entry would still be constrained due to non-proliferation considerations.

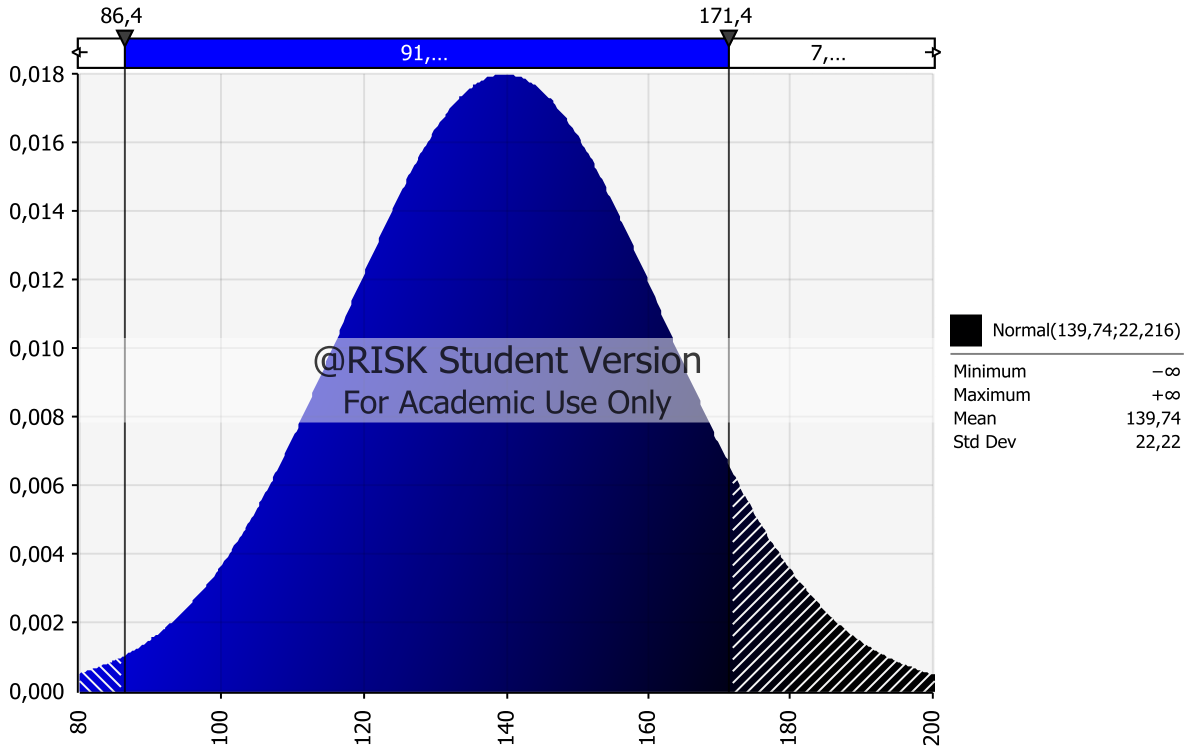

Figure 4.7| Probability distribution of the SWU spot market price in US$$_{2011}$/kgHM

In this thesis the SWU spot price in US$$_{2011}$/kgHM in the period 1990-2010, as encountered in Rothwell (2011) [100], will be used. The reported maximum value of 174 US$$_{2011}$/kgHM is approximately twice as high as the minimum value of 83 US$$_{2011}$/kgHM, which is consistent with the analysis in Rothwell (2009) [127] and in line with the assumptions of 174 US$$_{2011}$/kg-SWU made in the MIT (2009) study (after converting $2007 to $2011) [63]. As shown in Figure 4.7, it was determined that the distribution best fitting the available data is a normal distribution with a mean of 139 US$$_{2011}$/kgHM and a standard deviation of 22 US$$_{2011}$/kgHM.

4.3.3.6.2 Direct Disposal versus Waste Reprocessing

The debate on whether to directly dispose of our nuclear fuel or attempt to reprocess it and close the nuclear fuel cycle has been on-going for some time now. Bunn et al. (2003) [120] provides an extensive overview of the financial aspects of this decision process. One of the insights is that even at a reprocessing price of 1000 U.S. dollar per kg of heavy metal (US$$_{2003}$/kgHM), under the central estimates and key fuel cycle parameters used in that study, the waste reprocessing option will be more expensive than direct disposal until the uranium price exceeds 360 US$$_{2003}$/kgHM, an event which was evaluated as unlikely to occur in the coming decades.

Several countries are known to actively reprocess spent fuel. The main contenders in this market are France and the UK with reprocessing capacities of 1700 and 2400 tonnes HM/year respectively [128]. The main goal, reprocessing the low enriched uranium and plutonium into MOX fuel assemblies which can be used to fuel light water reactors, is reported to only save roughly 15 per cent of uranium source while creating additional proliferation risk [33]. Furthermore, the article also estimates that in order to make reprocessing a valid economic alternative the price of natural uranium needs to rise by a factor three to four, which is not expected to happen within this century [34]. Additionally, adopting the once-through cycle as the norm could put a halt to the current global increase of separated plutonium stockpiles, which were estimated to be roughly 255.5 tonnes in 2010 [129]. Finally, the existence of reprocessing facilities, which could be used by countries with emerging nuclear programs for a clandestine nuclear weapons program could generated additional public resistance.

Therefore, in this thesis we will assume that the nuclear waste will undergo geological disposal. Bunn et al. (2003) [120] reports on the necessity to have a temporary (or interim) storage facility where the spent fuel can be kept for several decades until the geological disposal sites are finished. However, several SMRs are reported to come equipped with large spent fuel storage pools, enough to accommodate all the spent fuel produced during the reactor design lifetime, which renders interim storage obsolete (e.g. mPower [130]). Furthermore, the FUJI by merit of its high conversion ratio and the fission products remaining dissolved in fuel, also requires no interim storage [23]. In addition, the MYRRHA is reported to come equipped with internal interim spent fuel storage inside the primary vessel, which allows for the use of spent fuel decay heat [131]. For all the additional reactors it will be assumed that the spent fuel will be moved into interim storage after 10 years of cooling in on-site spent fuel pool.

This approach is in line with current developments in Sweden as nuclear officials applied for a licensing application to build a geological vault in municipality of Osthammar, which is to assume operation in 2025 [132]. Sweden evaluated the cost for high level nuclear waste to be in-between 300 and 350 US$$_{1998}$ per kgHM. After correcting for inflation and assuming that this cost estimate is representative for the Osthammer site, the costs of direct disposal will lay somewhere within the 414-483 US$$_{2011}$/kgHM interval for the spent fuel from LWR reactors.

Based on estimates made for the Yucca Mountain Complex, it is estimated in Bunn et al. (2003) that the cost composition of the direct disposal option is made up out of three categories: 19 per cent of the costs incurred are related to the composition of the waste in the form of heat emission, 53 per cent of the costs are influenced by the volume, mass and packaging and 28 per cent of the costs are not affected by the waste type. For simplicities sake it will be assumed that the spent fuel will arrive in containers of equal volume and that the mass difference resulting from the different isotopic composition of the spent fuel is negligible. The differences in decay heat emanating from the spent fuel canisters of the different reactor types could be taken into consideration, but as demonstrated in Becker et. al (2007) [49], these only differ significantly in the first 100 years, following that the major compositional differences are caused by the different concentrations of short-lived isotopes. The reason that this variable influences the cost is that the heat output of the waste packages determines how close the units can be placed to one another without violating the repository's maximum temperature constraint. Therefore, it can be argued that the true limiting factor on the amount of spent fuel that can be stored within the confinements of the repository in not driven by the physical volume of the waste, but rather by heat output of the spent fuel canisters. Based on the above and assuming that the amount of spent fuel which can be fit into a given area of repository and the decay heat of a fuel canister is a linear relation, the direct disposal costs of spent fuel can be determined as follows:

$$C_{DS} = (1 - \phi - \varrho)C_{DS,LWR} + \phi C_{DS,LWR} + \varrho \frac{Q_{SMR}}{Q_{LWR}} C_{DS,LWR}$$

Equation 4.31

In Equation 4.31 the costs of direct disposal C_{DS} in US$$_{2011}$/kgHM are related to the fraction of the costs (ϕ) which is related to amount of decay heat, and the fraction of the costs which is related to the volume of spent fuel (ϱ). Furthermore, $C_{DS,LWR}$ represents the direct disposal costs of fuel incident from a light water reactor design and Q_{SMR} and Q_{LWR} represent the quantities of spent fuel from a specific SMR design and a typical light-water reactor[18], respectively in kgHM/GWe$_y$. An overview of the isotopic composition of the spent fuel, as generated with IAEA's Nuclear Fuel Cycle Simulation System [124] is given in Appendix 10.7.1.

4.3.3.7 Load Factor

At the end of 2010 only 27 out of 441 operational reactors had a cumulative load factor of greater than 90 per cent. Furthermore, only the top 100 plants had a cumulative load factor of more than 85 per cent [133]. A recently developed MIT model, used to determine the capacity factor risk, was also able to model the expected capacity factor level by using the global historical dataset [99].Within the context of this paper, the capacity factor was found to equal the load factor and the base estimate was established to be 74 per cent. If the estimate was constructed using only data from OECD countries it improved by approximately 1 per cent, while if estimates were derived by only using the reactor performance data since the year 2000, an improvement of 4 per cent was measured. They continued by stating that a mean lifetime capacity of 85 to 90 per cent could only be achieved by focussing exclusively on a small subset of the data, ignoring all other available data. Of particular interest was the remark: *"It is equally wrong to naively treat all datapoints as equally informative as it is to naively focus on only some of the datapoints and ignore the others. But we have not seen a careful justification for high estimates of the mean capacity factor that seriously confront the potential information available in the full data set"* [99]. Their findings on the expected lifetime capacity factor are in line with the probability function fitted to the historical lifetime capacity distribution by using the IAEA database as seen in Figure 4.8 [133].

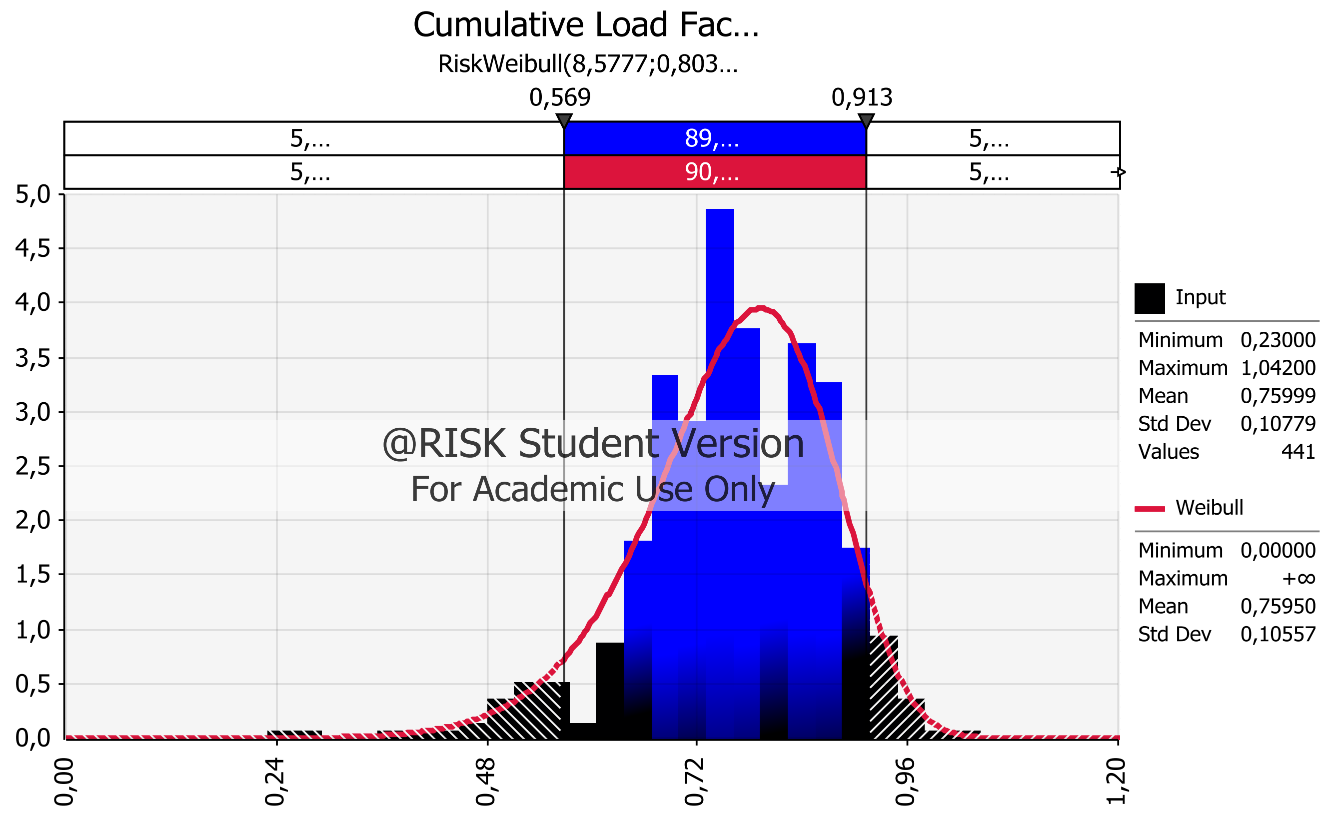

Figure 4.8| Probability density function of the cumulative load factors of 441 reactors operating in 2012

4.3.3.8 Operational Plant Lifetime

In some countries, such as the United States, reactors are licensed to operate for a certain lifetime (e.g. 40 years) after which they can be approved for a lifetime extension (e.g. 20 year) by their national nuclear safety authority. In many other countries however, there exists no such time limit on the length of operation. In

[18] 1 GWe PWR, 33 per cent efficiency, 100% load factor, 3.5% fuel enrichment and a burn-up 50 GWd/tHM

France, for example, the reactors undergo an in-depth licensing inspection procedure every 10 years. The French nuclear safety authority (ASN) evaluates on a per case basis if a reactor is able to operate for more than 30 years. After a reactor has received approval for post-30-year-operation, the ASN considers the 40 year benchmark, and all subsequent benchmarks, as regular inspections [8].

The Situation in Sweden shows similarities to the one in France. The profile of the Swedish nuclear power industry as given by the IAEA states that all existing Swedish NPPs have been licensed for an unlimited operational lifetime [134]. Limits to the operational lifetime and its function as control station may be imposed on a specific reactor however, but if a specific NPP is not in breach with any of the legally binding safety requirements, a license extension will, in principal, always be approved. In line with the ASN, the Swedish Radiation Safety Authority (SSM) is legally required to conduct periodic safety inspections at every major nuclear installation at every 10 year operational milestone. The purpose of these inspections is to determine whether the installation is still operating within the confines of the licensing conditions and current operations standards. Furthermore, the inspection also determines whether the development of the safety features is in line with the current safety requirements. In the event that an NPP fails to comply with the conditions attached to the license or a safety standard is some not implemented properly, the government or the SSM is authorized (whichever body issued the license) to either limit or revoke the licence under the Act on Nuclear Activities (1984:3). Reasons, other than revoking a license based on safety issues (as was the case with the Barsebäck 1 and 2), require a special law. A recent IAEA review on the Swedish nuclear and radiation safety framework concluded that the regulatory system is based on good practices [135].

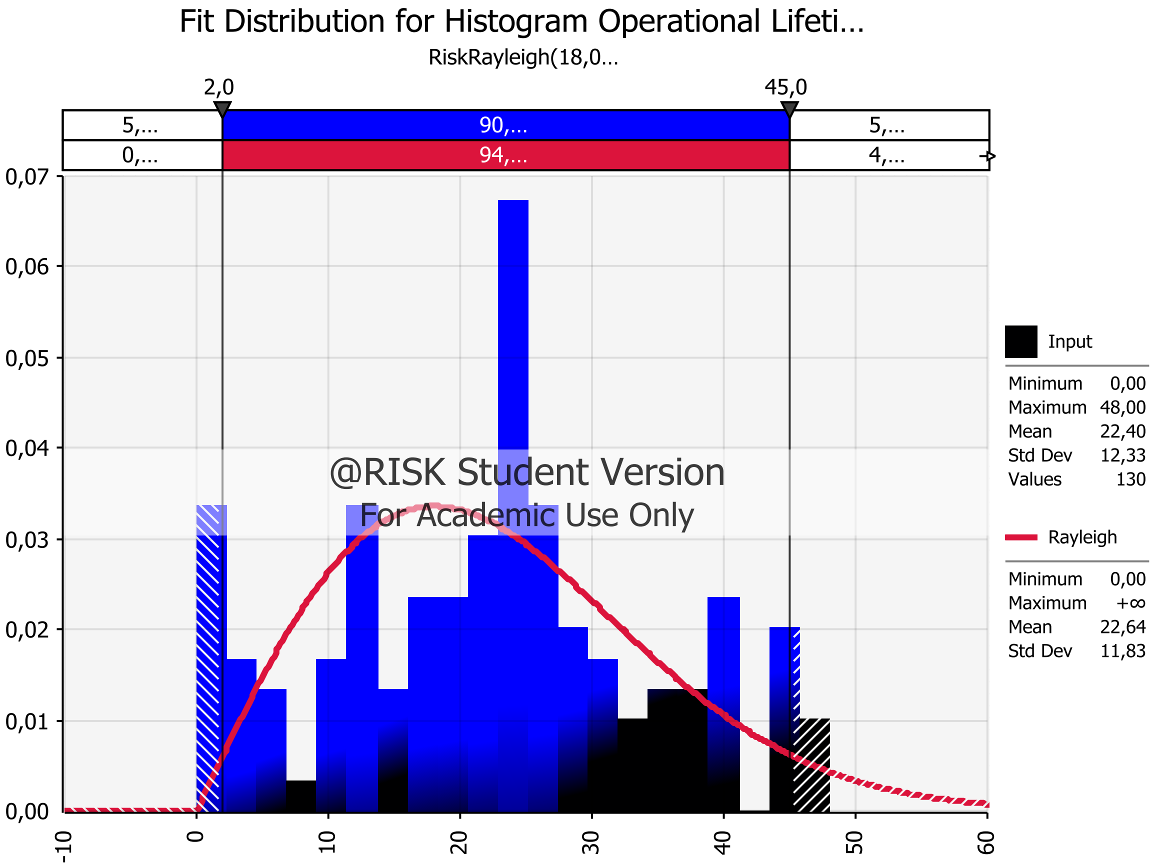

Figure 4.9| Probability distribution of the age of the existing reactor fleet

Currently lifetime extensions are being prioritized over new capacity additions by nuclear operators such as EDF [8]. The original assumption was that NPPs would have a ‘guaranteed’ technical lifetime of 40 years, with the inclusion of large margins on the major passive structures and components. As of 2011, Sweden has made significant investments in its nuclear fleet in order to improve their safety, reduce their environmental impact and extend their lifetime. Moreover, these continued modernisations are expected to continue in order to ensure that the remaining 10 reactors can be operated until the end of their 40 year design lifetime and possibly even beyond [134]. Considering that lifetime extensions have no formal meaning within the context of the Swedish licensing system and, as reported in paragraph 3.6, the nuclear phase-out has been annulled there are currently no political or legal barriers against the extended operation of the existing reactor fleet.

On a critical note, one should be aware that an operational lifetime of 40 years or more has not been the golden standard in NPP operational lifetime thus far. If one assesses the likelihood based on the age distribution of currently operating reactors a less optimistic view presents itself. It was reported in the World Nuclear Industry Status Report 2010-2011 [8] for example that as of April 2011, only 12 out of 437 worldwide operating reactors had exceeded the age of 40 years. Furthermore, a total of 165 operational reactors had reached an age of 30 years or more.

However, one can divine from the shape of the age pyramid, of which an updated version is available at the Power Reactor Information System (PRIS) section of the IAEA website, that this number will most likely increase in the coming years [12].

The retirement age of the 130 reactors that have already been shut down does little to change this view. As can be seen in Figure 4.9, the mean age of the retired reactor fleet lies around 23 years, which is lower than the mean age of 26 year of the current operational reactor fleet. However, a total of 32 of these reactors passed the 30 year operational lifetime benchmark and a considerable part of the reactors which only operated for a couple of years were first generation NPPs. Therefore, on the one hand it should be acknowledged that it is impossible to guarantee that a newly constructed NPP will reach the end of its design lifetime. On the other hand, assuming that the historical retirement statistics, largely based on first and second generation designs, applies to third and fourth generation designs, could be equally misleading. This view is reinforced by the fact the majority of the 16 which have thus far operated for more than 40 years were Magnox reactors [8]; a smaller-sized reactors (50-225 MWe) like the reactors under investigation in this thesis.

Most sources agree that the currently operating NPP were designed for at least 30 year of operation (e.g. [56]). This means that the mean retired reactor age currently lies at approximately 75 per cent of the design lifetime. Taking the current mean retirement age (75 per cent of the design lifetime) as the lower limit and the technical lifetime as the upper limit serves two purposes. Firstly, one acknowledges the possibility of an untimely retirement and secondly one acknowledges that due to increasing operational experience the average global retirement age could increase. Therefore, in this thesis, reactors with a technical lifetime of 40 years will have an expected operational lifetime of 30-40 years and reactors with a technical lifetime of 60 years will have an expected operational lifetime of 45-60 years. This is line with the MIT 'Future of Nuclear Power' series, where the expected operational lifetime is 25-40 years [111, 63]. Due to lack of detailed information on the probability density function, the distribution will be considered to be flat.

5 Description Reactor Technologies

5.1 Pressurised Water Reactor

5.1.1 NuScale and mPower Power Modules

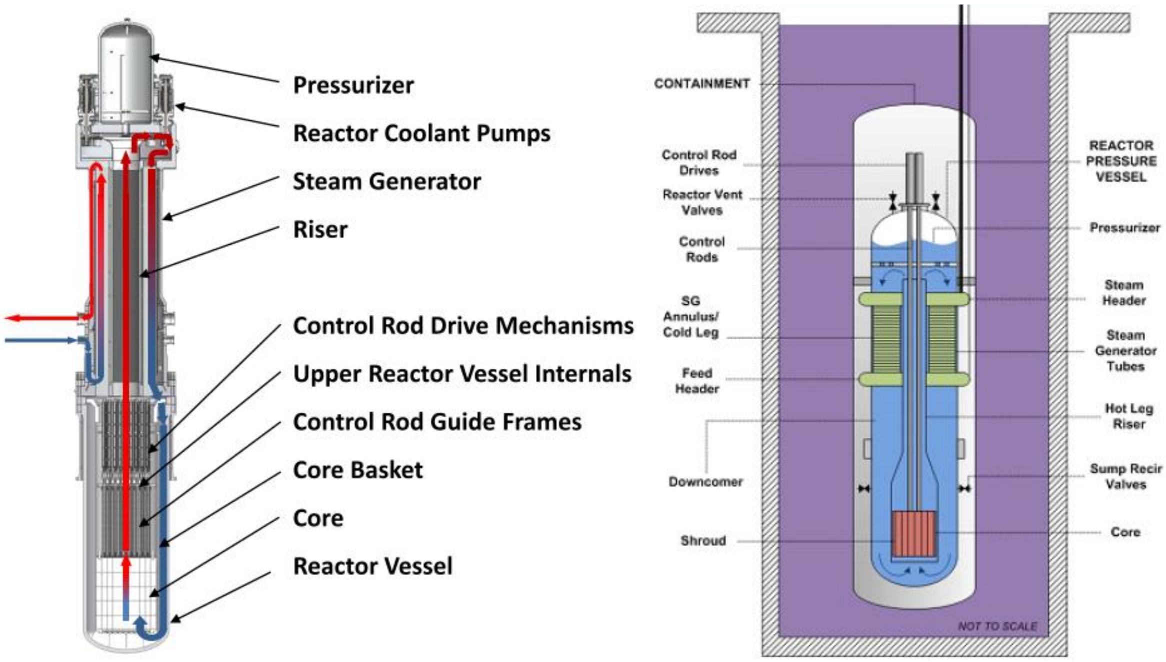

Figure 5.1| Schematic representation of mPower (left) and the NuScale (right) power modules

Courtesy of The Babcock & Wilcox Company and NuScale Power

Light water SMRs can be considered to be evolutionary designs, whose design improvements are drawn from numerous years of operational experience with light water reactor technology. Similar to their larger generation III counterparts, such as the Westinghouse AP1000, light water SMRs incorporate passive safety features which operate on physical principles such as gravity and natural convection into their design. These systems can be argued to be preferential to their more common pump driven counterparts for backup cooling purposes during accident scenarios, partially due to the elimination of the requirement for prompt actions by an operator [28]. What sets SMRs apart from larger-sized counterparts is that they improve upon existing design practices by incorporating features for reduced operation and maintenance complexity while simultaneously simplifying the overall design limiting the number of components. This is a hallmark technique of the integral design approach, as described in paragraph 4.3.3.3.4. By using in-vessel steam generators and a more compact containment, integral PWRs aim to depart from the established practice of periodical in-service inspections, aiming to significantly increase the required interval in-between inspections. This proclaimed increase in overall system reliability simultaneously acts as one of the major licensing obstacles that needs to be overcome, because any departure from the scope and periodicity of maintenance and inspection routines requires an explicit justification. This is especially the case when novel components and features are concerned [136].

As reported by Kessides (2012) [137], when considering that light water SMRs are based on light water reactor technology, the dominant reactor-type in the modern nuclear reactor landscape, they can be considered to have the lowest degree of technological risk attached to their development, with several designs nearing commercial deployment (as discussed in paragraph 3.1). To investigate the effects of economies of scale, serial production and learning on the cost of electricity two light-water-moderated SMR designs will be analysed in this thesis, the 180 MWe mPower Reactor and the 45 MWe NuScale Reactor. The mPower reactor, designed by Babcock & Wilcox is an integral pressurized water cooled reactor module, which can be incrementally added due to its modular design. Its pressure containment vessel, which is located in a subterranean containment building, houses all major components including the reactor core, the coolant pumps, the pressurizer and the control rods. Another prominent design feature is that it could operate for more than 4 years without refuelling. The NuScale Power Reactor, with its 45 MWe power output, departs even further from current industrial size-related practice. Additionally, it is completely passively cooled and its pressure containment vessel contains the reactor core and the steam generator. Furthermore, the NuScale module is designed to be deployed in series of up to 12 units and is housed in a subterranean containment building similar to the mPower module. The primary difference between both containment structures is that NuScale reactor modules are immersed in a water filled pool, providing additional cooling capacity under accident scenario conditions.

5.1.2 Safety Features

5.1.2.1 Safety Systems

Light water SMRs are expected to have lower decay heat levels, which would render them less dependent on extensive cooling mechanisms post-shutdown. This could be especially important when viewed within the context of Fukushima disaster. It is reported by Rosner and Goldberg (2011) [28] that concerning natural disasters, light water SMRs have several distinct advantages over conventional water cooled LRs:

1) The aforementioned lower dependency on emergency cooling systems due to the lower decay heat levels. This could severely reduce or even completely annihilate the necessity for electrically powered emergency cooling (when used in conjunction with natural convection cooling).

2) The suspension of the pressure containment vessel inside a pool of water, largely located underground, dampens the effects of rapid earth movement, enhancing the resistance to earthquakes of the reactor system.

3) Furthermore, the current designs feature large underground pools for designated spent fuel storage. These rigid structures are designed to further limit the probability of a dry-out in the spent fuel pool.

4) The integral design comes equipped with a self-pressurization mechanism, which has a stabilizing effect on the reactor; variations in void-fraction are stabilized by a pressure response from a dome pressure-feedback mechanism. This eliminates the need for the heaters and sprinklers which are encountered in conventional PWR systems.

Although the reactor designs do have the abovementioned features in common there are also some differences. Such as that in the event of a Loss Of Coolant Accident (LOCA) scenario following a black-out, the mPower module can only count on its primary coolant circuit for heat dissipation.

Although it provides no additional barriers to reactor core dry-out in comparison to other Generation III designs, such as the EPR, it should be noted that the reactor power to coolant volume ratio has improved somewhat and the core uncovering time has been increased by increasing the Pressure Containment Vessels' (PCVs) height, while giving it a smaller diameter. In effect, this increases the distance from the top of the fuel assemblies to the water/inert gas interface. Therefore, there is additional time to restore power before reactor core meltdown is initiated.

The NuScale Power module takes an alternate approach by submerging each individual containment vessel in large pool of water. It is reported by the vendor that containment vessels are only held in place by seismic supports on the side, which is possible due to their only slightly negative buoyancy [138]. The PCV is kept in a highly pressurized vacuum state; this prevents convection heat losses during normal operation. In the event of a LOCA, two additionally benefits of vacuum operation are the improved steam condensation rate and resistance to combustible hydrogen mixing as a result of no oxygen being present. The mPower on the other hand has an advantage in containment of the fission products and actinides, due to its higher degree of in-vessel systems integration with even the control rod drives being located within the containment vessel.

Furthermore, each mPower reactor module comes equipped with its own containment building, adding an additional containment barrier.

5.1.2.2 Coolant Characteristics

In order to determine the thermal inertia, the average heat capacity of light water moderator at the prevalent in-core pressure and temperature needs to be determined. In addition, for the mPower, the corresponding density is also required considering that only the moderator volume is known.

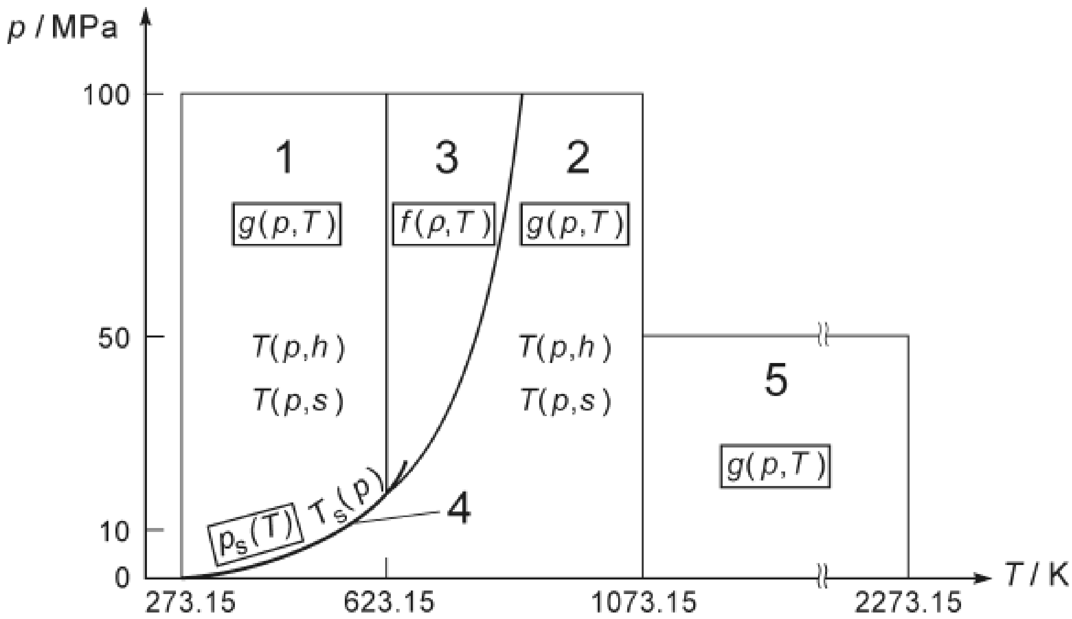

Figure 5.2| Regions of the IAPWS-IF97 possessing a corresponding set of equations

Taken from IAPWS-IF97 [139]

The most accurate formulation of the properties of water and steam is given by the *IAPWS Formulation 1995 for the Thermodynamic Properties of Ordinairy Water Substance for General and Scientific Use*, also known as the IAPWS-1995 [140]. Although considered to be extremely precise and of extraordinary quality, the reason it was found to be unsuitable for industrial use was that it used the density as an independent variable which made the computing times to lengthy for practical applications. In response to this shortcoming, the International Association for the Properties of Water and Steam (IAPWS) adopted a new formulation, the IAPWS Industrial Formulation 1997 (IAPWS-IF97). It is valid throughout the temperature region $0°\mathrm{C} \leq \mathrm{T} \leq 800°\mathrm{C}$, with $\rho \leq 100$ MPa and has been extended for high-temperature applications. As noted in IAEA-TECDOC-1496 [141], the most striking difference with the IAPWS-1995 is that the structure of the basic equations has been optimized by subdividing the temperature range into five regions (Figure 5.2). Taking into consideration the design characteristics, as given in Appendix 10.5, both the coolant systems of the mPower and NuScale modules operate in region 1.

The full set of equations for this region can be found in Reference [139]. However for simplicities sake, the relevant values for ρ and C_p were determined by means of the Water97_v13.xla Add-in for MS Excel, which provides a set of function for computing various thermodynamic and transport properties of water and steam using the IAPWS-IF97 industrial standard [142].

5.1.3 Proliferation Resistance

Water cooled SMRs share a large degree of their proliferation resistance features with their larger-sized counterparts. However, even the technologically less exotic water-cooled SMRs show some minor improvements that aim to prevent the theft of nuclear material. One thing that sets water-cooled SMRs apart from conventional LWR designs is their long fuel irradiation period. As shown in Appendix 10.5, the NuScale and mPower modules have a fuel cycle of 1644 and 732 Effective Full Power Days (EFPDs) respectively. This puts them ahead of advanced LR designs such as the EPR, which is reported to have a refuelling interval of 12 to 24 month [143]; a figure which includes reactor downtime and can therefore be assumed to be considerably less. The lengthy nature of the fuel cycle renders all intermediate access unnecessary and therefore unauthorized. The advantage of such an infrequent necessity to access to the reactor core during operation is that the monitoring of the fissile material is simplified.

As reported in Rosner and Goldberg [28], due to both the NuScale and the mPower reactors currently being under investigation by the U.S. Nuclear Regulatory Committee, all but the most trivial design characteristics are proprietary. The quantity of spent fuel quantity per GWe_y does vary from LR designs as a result of its dependence on the burn-up rate B and the efficiency η. Although intuitively it might seem that the spent fuel quantity per unit of electricity produced increases following from the lower power density which could result in a lower burn-up, this is not necessarily the case. The NuScale for example is reported to have a burn-up of well over 50 MWd per kilogram of heavy metal [29]. Furthermore, due to the smaller overall NPP size, less fuel rods are required, due to the increased irradiation cycle length less fuel shipments are required and the lower power output implies that the quantity of nuclear fuel per transport will also be smaller. The significance of the size of the fresh and spent fuel batches lies in that the transport section was considered to be the weakest point in the nuclear fuel cycle by Pierpoint (2008) [53], as discussed in paragraph 3.3.

5.1.4 Economics

In recent years several articles have been published on the competitiveness of SMRs, most of which report on the competitiveness of water cooled SMRs (e.g. [136]). Often such overviews, based on light water technology, conclude that the specific capital costs of light water SMRs are likely to be higher than those of large plants. An example is given in an OECD study, which concludes that four 300 MWe integral type PWRs, based on nuclear propulsion technology, built on the same site and Nth of a kind (NOAK) models, could have roughly 10 to 40 per cent higher specific overnight capital costs than a single 1200 MWe size PWR [136]. However, in addition to this, it is reported by the same source that the specific overnight capital costs of five 300 MWe module NPP would only by 7 to 38 per cent higher than the costs of a single 1500 MWe size LR. This implies that a sufficient number of reactor modules could offset the loss of economies of scale.

In this thesis, the reference for the specific overnight costs of a modern 1 GWe light water reactor will be taken to equal the value suggested in Du and Parsons (2009) [61], which is 4000 $US\$_{2007}$/kWe, or 4339 $US\$_{2011}$/kWe[19] when corrected for inflation. It should be noted that this figure is lower than the upper limit post-Fukushima figure of 6000 $US\$_{2011}$/kWe reported by UBS [70] and some authors (e.g. [137]). Therefore, based on the intuition that the events at Fukushima could have an upward effect on the overnight costs, a second overnight cost scenario is included in which regulatory turbulence and stricter safety requirements result in a specific overnight cost of 6000 $US\$_{2011}$/kWe for a 1 GWe PWR.

5.2 Sodium Cooled Fast SMRs

5.2.1 4S

The Super-Safe, Small and Simple, or 4S SMR, is an integral pool-type sodium-cooled fast reactor which is currently being developed by Toshiba and the Central Research Institute of Electric Power Industry (CRIEPI) in Japan. Although the Japanese government has declared that no new nuclear reactors will be built [144], this is not expected to be the end of the 4S' development. The 4S was scheduled to submit for design approval with the U.S. Nuclear Regulatory Commission in the third quarter of 2012 and the Toshiba

[19]U.S. Department of Labor Statistics Inflation Calculator (http://www.bls.gov/data/inflation_calculator.htm/)

Corporation is working with the town of Galena (Alaska) as a potential Combined Construction and Operating License partner [145].

At its current development stage it is envisioned to become available in two editions; a 30 MWth design generating approximately 10 MWe and a larger-size 135 MWth variant producing an estimated 50 MWe. In this thesis, the latter variant will be used to estimate the relative economic competiveness of the 4S SMR. As reported in IAEA-TECDOC-1536 [23] the safety features and proliferation aspects of the two designs can be considered to be identical.

Although the 4S operates in a fast neutron spectrum it is not a breeder reactor, owing to its relatively low conversion ratio of 0.53 (50 MWe) [146] which places it on par most LWRs. This is a peculiar design choice considering that fast neutron reactors are typically designed to breed fissile isotopes from fertile blanket material in addition to their function as power stations. However, in a similar fashion to the decoupling of the fuel breeding and power generation approach in the THORIM-NES philosophy (paragraph 5.6.1) the advantages of breeding were foregone for reduced design complexity.

In the course of its 30 year designed lifetime an adjustable reflector system slowly compensates for the loss in reactivity as a result of burn-up. This enables the 4S to operate without having to be refuelled during its entire lifetime. However, the documentation on a 2011 IAEA technical meeting reveals that once-per-lifetime refuelling is only available for the 10 MWe design and that the 50 MWe design will require refuelling every 10 years [147]. Additionally, the reactor power output can be adjusted by means of controlling the feed water. This leads to a change in the coolant temperature, which subsequently results in reactivity changes. The ability to self-adjust the power output makes the 4S ideally suited for load following and enables it to be configured to produce a variety of other products besides electricity such as desalinated water and hydrogen [148].

Although the smaller power output of the 4S implies that it might be used as a ship-based NPP, the 4S is solely designed for land-based operation. This is due to the containment building being designed to be placed below ground to limit unauthorized access and presenting a smaller target to airborne threats. It should be noted that the subterranean placement only applies to the nuclear island and that the steam turbine system and any additional facilities are located above ground level, which could limit the effectiveness of this anti-terrorism measure. In contrast to several of the other designs discussed in this thesis, commercial interest in this reactor can be dated back as far as 2003. In this year a 10 MWe Toshiba 4S NPP was suggested to power the small rural Alaskan community of Galena [36].

As with the mPower and NuScale Power modules, the major components are located within the reactor vessel, which validates the use of the prefix ‘integral’. These components are: the internal heat exchanger (IHX), the primary electromagnetic (EM) pumps, the movable reflectors and the control rod which functions as the ultimate shutdown system. The reactor is designed to utilize metal fuel, which has higher heat conductivity than conventional oxide fuels. Considering that there is currently no infrastructure for the industrial scale reprocessing of metal fuel, the initial phase of spent 4S reactor fuel is reported to be meant for geological deposition [23]. This is in line with the once-through cycle assumption made in this thesis.

5.2.2 Safety Features

5.2.2.1 Safety Systems

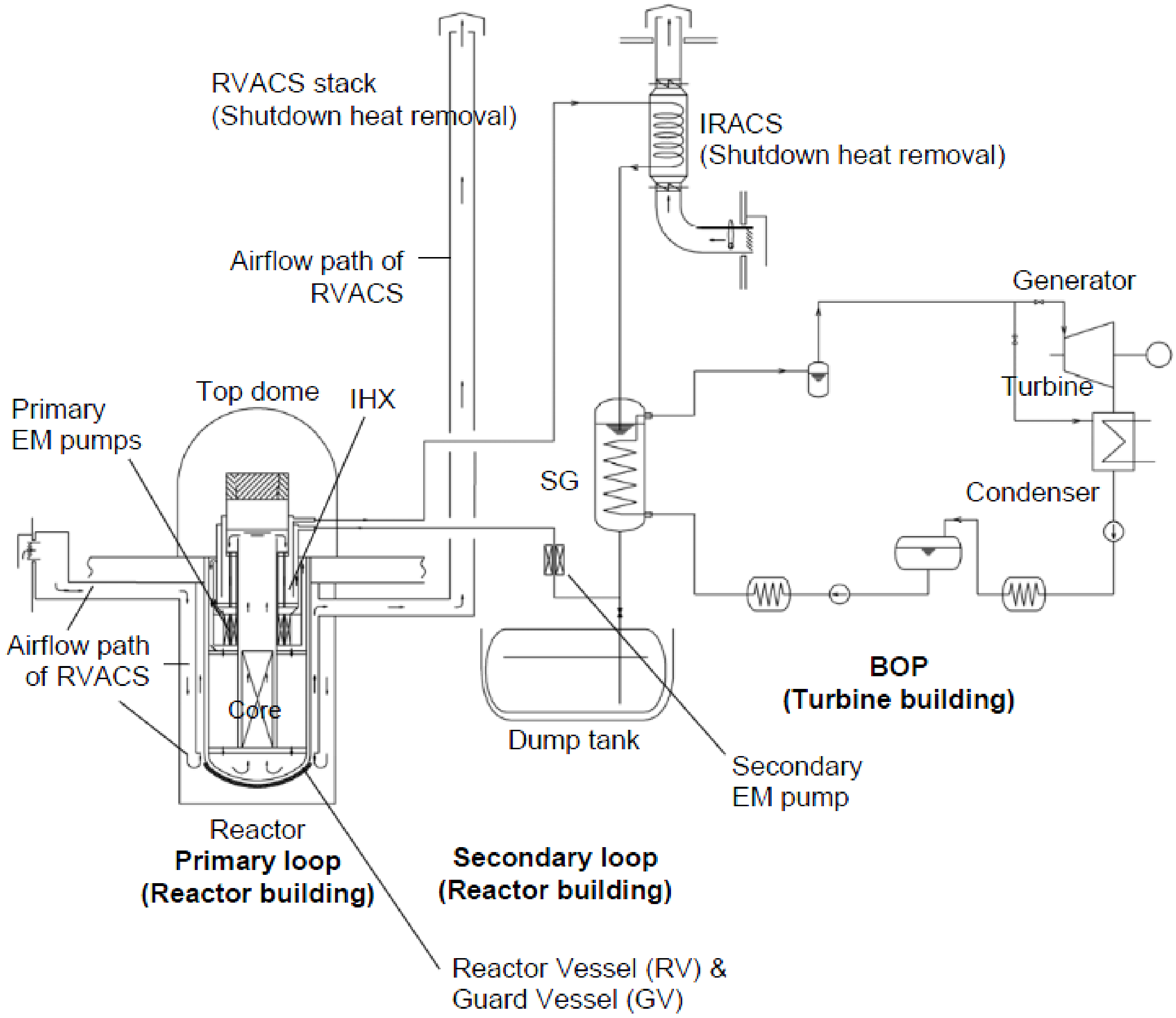

Figure 5.3| Schematic overview of the passive safety systems encountered in the 4S design

Taken from IAEA-TECDOC-1536 [23]

It can be argued that the design philosophy of the 4S places the highest emphasis on passive safety features of all the SMRs investigated in this thesis. The ultimate goal of the 4S safety design is equally unique in its ambition to completely remove the need for evacuation as an emergency response measure [23]. To achieve this goal all of the active and passive safety measures are designed with the following design guidelines in mind: (1) reducing the probability of component failure, (2) preventing core damage during an accident scenario and (3) confining the radioactive material in the case of an accident scenario.

The first item is achieved primarily by the eliminating active systems, feedback control systems and utilizing static devices such as EM pumps to provide circulation.

The second, preventing core damage in case of abnormal operation, is facilitated by two independent active shutdown systems, the enhanced safety of metal-alloy fuel, all reactivity coefficients being negative and two fully passive shutdown systems. The active shutdown systems include the operator initiated drop of several of the reflector panels and the presence of an ultimate shutdown rod. In comparison to uranium oxide fuel, metal alloy fuel decreases the odds of meltdown due to its higher conductivity which limits the accumulation of heat. Furthermore, the reactivity coefficient determines the effect a change in temperature has on the neutron multiplication factor, which in turn determines the reactivity of the system. It is well known that the various temperature reactivity relations (such as Doppler broadening and thermal expansion) have an important effect on the operation and ultimately the system safety (e.g. [149]). When any one reactivity coefficient is negative, an increase in temperature leads to a decrease in reactivity and vice versa. This entails that any reactor system with an overall negative temperature reactivity coefficient (a licensing requirement in the U.S.) has some form of self-regulatory control. Therefore, when a reactor can be designed in which all reactivity coefficients are negative, every deviation from the original reactivity state, will prompt an immediate and opposite response, resulting in increased stability.

Additionally, the 4S comes equipped with two independent passive safety systems; the reactor vessel auxiliary cooling system (RVACS) and the intermediate reactor auxiliary cooling system (IRACS). As can be derived from Figure 5.3, the RVACS removes heat from the surface of the guard vessel (a layer surrounding the PCV) by means of the natural circulation of air. The system is completely passive and always in operation, also during rated power operation. In order to obtain the necessary airflow, each reactor contains two pathways. The IRACS which, as shown in Figure 5.3, is responsible for removing heat from the secondary coolant loop is an electrically operated system, which is inactive during normal shutdown operation. It is reported of being capable of removing heat through the natural convection of both air and sodium in postulated accident scenarios [23].

The third guideline of radioactive material confinement is attained by a mixture of several conventional and unconventional features. The more conventional ones are the barriers obtained by the fuel cladding and the reactor vessel. The more unconventional barriers are aimed at containing the sodium coolant. This is necessary due to the sodium having the disadvantage that it spontaneously combusts upon impact with oxygen and reacts explosive upon contact with water. Therefore, intricate barriers to prevent the leakage of sodium or reduce the impact of a leakage event are present in the design. The first one is the presence of a separation layer between the PCV and the guard vessel, which is wide enough to allow for in-service inspection but narrow enough to prevent the internal heat exchanger inlet from emerging from the sodium coolant, in the event of a leak, and halting the natural circulation process. The cavity in-between both vessel layers is filled with nitrogen gas [147]. A similar cavity exists in-between the double wall steam generator, which is filled with mesh and helium. This system provides both inner and outer tube failure monitoring, the wire mesh and being used for moisture detection and the presence of a helium detector in the intermediate sodium loop. In the event a sodium-water reaction does occur, the increased pressure in the steam generator would rupture a membrane that would cause the contents of the secondary coolant loop to drain into the dump tank located below the steam generator, as shown in Figure 5.3.

5.2.2.2 Coolant Characteristics

5.2.2.2.1 Overview

Although sodium has been in use as a coolant for research and power generating systems for more than half a century, in contrast to light water systems, its development for use in industrial scale applications has met severe technical difficulty. In *Global Energy Assessment 2012* [2] a rather unsuccessful account is presented of the development of sodium-cooled fast reactor technology, which can be summarized as follows. The Japanese 280 MWe Monju reactor for instance, was shut down only a few months after is attained criticality in 1995 due to a sodium fire and was only restarted briefly, 15 years later in 2010, before being shut down again. A similar fate was met by the French 1.2 GWe Superphénix, which was out of operation for repairs so often that it only attained a cumulative lifetime capacity factor of 7 per during its operational lifetime from 1985 to 1996. Even the Russian BN-600, which has been comparatively successful with its cumulative capacity factor of 74 per cent [150], has had to withstand 15 sodium fires during its first 23 years of operation.

The benefits of continuing research are clear however as the attractiveness of sodium as a fast reactor coolant is well known. Several recurring benefits attributed to sodium are: (1) its corrosion inertness, making it compatible with traditional structural materials and conventional fuel compounds at high temperature, (2) the potential for low operating pressure and (3) its excellent thermophysical properties, ensuring the effective removal of heat under both natural and forced flow regimes [151]. There are some issues however (see below) that need to be addressed in the design, but the countries which are involved in its development, including France, Japan, Russia and the U.S., deem it to be a promising material for fast reactor cooling.

5.2.2.2.2 Sodium Burning

The first issue related to the use of sodium coolant is its high chemical activity. Pure metallic sodium does not exist in nature due to the tendency of alkali metals to react with water and form alkali hydroxides. The reaction is strongly exothermic and the generated heat ignites the hydrogen that is produced as a by-product of the sodium hydroxide reaction, resulting in an explosion.

Flammable material or substance	Burn rate, kg/(m^2·h)	Energy release, kJ/kg
Benzene	160–200	41870
Diesel fuel	150	41870
Wood	54	13800
Residual fuel oil	126	38700
Rubber	40	33500
Sodium	42–63	10900

Table 5.1

Taken from Poplavskii et al. (2004) [152]

The exact temperature at which sodium ignites is dependent on a number of factors, among others the impurity content and the air humidity, but the ignition temperature is reported to lie within the 140°C to 320°C range [153]. Therefore, at the operating temperature typically encountered in sodium cooled systems, the sodium spontaneously ignites when it comes into contact with air. However, as can be derived from Table 5.1, the combustion process of sodium has a relatively low intensity in comparison to other flammable substances. Therefore, as long as the sodium spills is in the order of magnitude of tens of litres (which in 2004 comprised roughly 80 per cent of the reported sodium leaks [152]), the fire is directly approachable and can be extinguished with a powder composition. This is assuming of course that the fire in nonradioactive in nature, but as will be discussed below this might not always be the case.

5.2.2.2.3 Radioactivity

During normal operation radioactive substances are formed in the coolant as a result of the activation of impurities present in the sodium coolant, activation of the coolant itself and the escape of fission products through the holes and cavities in the fuel cladding, activation of the construction materials and some minor corrosion. It is reported that in the event a sodium fire results in a radioactive release, the presence of the isotopes Na-22, Na-24, I-131, Cs-134 and Cs-137 can be assumed. All other isotopes are reported to either be present in inconsequential amounts or have a relatively short half-life [152]. Of the aforementioned radioisotopes, Na-24 is considered to be the most important, determining the radiation environment during operation under accident conditions. The reason that Na-22 plays such an insignificant role in comparison to Na-24 can be understood by studying their respective formation chains:

$$^{23}_{11}Na(n,\gamma)^{24}_{11}Na \xrightarrow{\beta^-} {}^{24}_{12}Mg$$

$$^{23}_{11}Na(n,2n)^{22}_{11}Na \xrightarrow{\beta^+} {}^{22}_{10}Ne$$

Sodium, which owes its reactivity to a single atom in its outer shell, does not readily give up two electrons, the requirement for the neutron producing reaction involved in the creation of Na-22. Therefore, even though

Na-22 has a half-life of roughly 2.6 years, it reported that even after 50 years of exposure to a fast neutron flux, Na-24 remains the primary isotope [153]. Only 10 days after reactor shutdown, when most of the Na-24 with a half-life of 15 hours has decayed, does Na-22 become the primary sodium isotope.

The activation of sodium in the primary circuit, in combination with sodium's high chemical activity with water and air also gives rise to the three circuit design which characterizes all sodium cooled reactor designs, including the 4S. As can be seen in Figure 5.3, these are: (1) the primary circuit which runs through the core and contains the radioactive sodium, (2) the secondary non-radioactive sodium loop, which is connected to the primary loop via the IHX and finally, (3) the tertiary power production loop containing the water. In this manner, the fault tree leading up to a radioactive sodium-water explosion is extended, reducing its probability of occurrence.

5.2.2.2.4 Thermophysical Properties

In order to determine the thermal inertia of the 4S, the heat capacity of sodium at the prevalent in-core conditions, as found in Appendix 10.5, need to be determined. An overview of the thermophysical properties of sodium liquid, including equations of state for the heat capacity C_p, can be found in Fink and Leibowitz (1995) [154]. However, for the sake of simplicity, in this thesis the empirical formulas suggested by the IAEA will be used [155]. The expression given by the IAEA [155] to represent the heat capacity C_p as a function of the temperature T equals:

$$C_p = \frac{(38.12 - 0.069 \times 10^6 \cdot T^{-2} - 19.493 \times 10^{-3} \cdot T + 10.24 \times 10^{-6} \cdot T^2)}{22.99}$$

Equation 5.1

In Equation 5.1, the heat capacity C_p, given in kJ/kgK, again is a function of the temperature T given in Kelvin K. Furthermore, although it is not explicitly stated by in reference [155], it is assumed that both expressions are valid in the temperature range in which the 4S reactor operates. Considering that the aforementioned reference was drafted for use in nuclear engineering, this assumption seems valid.

5.2.3 Proliferation Resistance

The IAEA [23] concluded that the foremost features of the 4S preventing the spread of nuclear material were: (1) the relative low enrichment grade of its uranium fuel, which contains less than 20 weight per cent of U-235, (2) the relatively low plutonium contents of the spent fuel and (3) the lack of available metal-alloy reprocessing technologies, which allow for the separation of plutonium and minor actinides from the spent fuel. Of the abovementioned features only the final argument appears to have any validity. Firstly, the fuel enrichment, reaching 12 and 18 per cent respectively in the core region and periphery respectively, is relatively high in comparison to the enrichment grades encountered in the other U-235 fuelled reactors. Secondly, as can be derived from Appendix 10.7.1, the plutonium content in the 4S' spent fuel is relatively high in comparison to the plutonium content of the LWR designs investigated in this thesis, with more than double the amount of SQs per GWe·year relative to the EPR.

Although the enrichment and spent fuel characteristics of the 4S can be considered relatively unfavourable, the 10 year refuelling intervals combined with the low maintenance requirements and the once–through approach do offer substantial physical barriers to the theft of nuclear material. Furthermore, both the 10 MWe and 50 MWe designs utilize the limited amount of 18 fuel subassemblies which simplifies monitoring. However, the proliferation barrier that sets the 4S most apart from the other designs in this thesis is its reported lack of facilities and equipment to discharge fuel assemblies [23]. This includes the inability to dismantle discharged fuel assemblies, separate the fuel pins and access the nuclear material inside. The current consensus on the fuel handling protocol for the 4S is that a fuel-handling machine will temporarily be made available to dispense of spent fuel subassemblies. It should be noted that this consensus was reached in 2007, when both the 10 MWe and 50 MWe reactor designs were still considered to have an initial fuel load that required no reloading. More recent material published by Toshiba presents a slightly altered configuration for the large MWe unit, which will require refuelling once every 10 year [147]. Therefore, it cannot be claimed with absolute certainty that the 50 MWe design under investigation in this thesis will not come equipped with processing facilities to prepare spent fuel intermediate and final storage. In the event a mobile apparatus, shared by multiple plant sites could be used for spent fuel processing, unauthorized fuel handling would become more difficult due to the limited availability of the equipment to perform the necessary proceedings.

5.2.4 Economics

In Nitta (2010) [156], the cost for an Advanced Liquid Metal Reactor (ALMR) is estimated to be 4304 US$$_{2007}$/kWe for a 2-Module 1288 MWe SFR. It should be noted that this is a relatively low in comparison to the overnight costs of the similar-sized Superphénix, which reportedly cost 34.4 billion FF$_{1996}$ [2] . Taking into consideration historical exchange rates, inflation and the rated reactor output of 1.2 GWe this equals approximately 8000 US$$_{2011}$/kWe. Therefore, an overnight cost as low as stated above might seem farfetched. An important distinction that needs to be made between the ALMR and the Superphénix is that the ALMR is not necessarily a breeder reactor, reportedly possessing a conversion ratio as low as 0.6 [157]. This gives it a great degree of technological similarity to conventional light water reactor technology and therefore a relatively lower degree of additional technical complexity. As abovementioned, owing to its low conversion ratio the 4S is also a burner reactor. Therefore the ALMR was chosen as the reference reactor.

In order to estimate the ALMR construction costs Nitta (2010) [156] primarily utilized cost estimates given by developers and updated some of the design independent cost categories to be more in line with the developments in the LWR market, such as the electrical equipment. Furthermore, some of the expected cost differences between LWRs and SFRs were highlighted, such as the higher cost of the reactor plant equipment for the SFR due to the more complex internals. Some subcategories that were identified to contribute towards the increase in overnight capital costs are: (1) the presence of additional heat-transfer loops required for the passive safety measures, (2) the necessity of an intermediate cooling loop to prevent a primary (radioactive) sodium and water reactions, (3) the use of sodium instead of water and (4) the presence of a complex automated control system. Although the ALMR and the 4S are different SFR designs, the abovementioned holds true for 4S design as well. The complex automated control system, in this case, could refer to the mechanism that operates the movable reflectors.

Several cost categories were estimated by Nitta [156] to be potentially less expensive for SFRs than for LWRs, the most important one being the Buildings and Structures account, which was estimated to be roughly 25 per cent higher for LWRs than SFRs. This difference was attributed to the proposed changes in the shape of the containment building, which contains less concrete and steel and can therefore be constructed at a lower cost. The main driver for this cost reduction is given to be the innovative design of the containment building, which utilizes less concrete and steel due to the reactor vessel being partially built underground. Following the guidelines set out in the cost study performed by Du and Parsons (2009) [61], the specific overnight capital costs of the reference LWR were scaled up to 4000 US$$_{2007}$/kWe, which resulted in the specific overnight capital cost of the ALMR increasing to 4304 US$$_{2007}$/kWe. Corrected for inflation, this results in a LR SFR point estimate of 4669 US$$_{2011}$/kWe[20].

5.3 Lead-Bismuth Cooled Fast SMR

5.3.1 SVBR-75/100

There are currently multiple lead-bismuth eutectic (LBE) cooled fast reactors under development, most of which are in the 25 to 100 MWe range. Notable designs, which have been surfacing as of late, are the

[20] U.S. Department of Labor Statistics Inflation Calculator (http://www.bls.gov/data/inflation_calculator.htm/)

Hyperion Power Module (U.S.), the PASCAR (South Korea), the SSTAR (U.S.) and the SVBR-100 (Russia). What the above mentioned concepts have in common is that they are all pool-type reactors utilizing an indirect Rankine steam cycle for electricity generation. Furthermore, all of the aforementioned reactors are designed to be factory fabricated and fuelled, operated at relatively low temperatures and intended for continuous operation without on-site refuelling [158]. Furthermore, all of these reactors claim to have advantages regarding their economics, passive safety features and proliferation resistance.

The concept which is in the most advanced design stage is the SVBR-75/100, as displayed in Figure 5.3. It is largely based on the 80 years of operational experience acquired through the use of LBE cooled fast reactors in Soviet Alpha Class Submarines [159]. It possesses the shortest irradiation cycle of the abovementioned alternatives, only 7 to 8 years, and is the only design which does not utilize natural circulation in its primary coolant loop.

5.3.2 Safety Features

An extensive analyse of the SVBR-75/100, henceforth referred to as the SVBR-100, conducted by the IAEA [23] concluded that the reactor design possesses a robust set of safety features. These enable it to cope with single equipment failures, unrelated errors by operating personnel and/or combinations of both which result in critical failure paths, amongst them scenarios which could be prompted by malevolent action. Furthermore, the design displays a high degree of safety system redundancy, one of the pillars of the defence-in-depth approach.

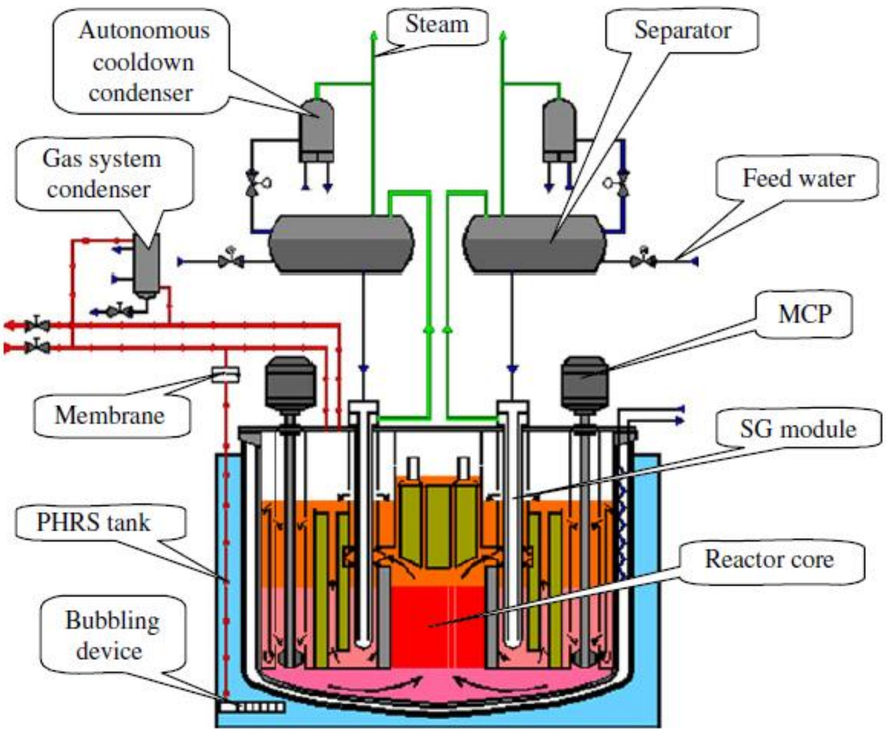

Figure 5.4| Hydraulic diagram of the SVBR-75/100 reactor including the safety systems

Taken from IAEA-TECDOC-1536 [23]

5.3.2.1 Safety Systems

An in-depth analysis of the various safety features of the SVBR-100 can be found in IAEA-TECDOC-1536 [23]. The emergency shutdown system consists of a set of 6 emergency protection (EP) rods and is supplemented by a group of 13 reactivity compensation (RC) rods. The EP rods are installed in the 'dry channels' and do not contain any drivers on the reactor lid. Their passive insertion is gravity dependant and they are held in place by springs and electromagnetic locks which are composed of an alloy that melts at a certain predetermined temperature associated with emergency overheating. Alternatively, they also respond to a control system signal. The complementary set of RC rods also comes equipped a set of springs and electromagnetic locks and their insertion is prompted by the same set of conditions. However, they have the added feature of being weighed with either uranium or tungsten, which prevents them from floating upwards in the LBE coolant, which might occur during certain accident scenarios. It should be noted that the RC rods should therefore be considered to be a secondary defence mechanism following failure of the EP rods.

The SVBR-100 comes equipped with two heat removal channels, which are each individually capable of removing 3 per cent of the rated reactor power. Each channel contains an autonomous cool down condenser and a separator. It is cooled by a water feed entering the separator and a control valve operated drainage pipeline, which opens if the pressure in the separator exceeds normal operational parameters. During normal operation this autonomous circuit is only used in start-up and after cooling events. When operated in waiting mode however, the condenser, which is flooded with water, will release its content once the steam pressure exceeds some predetermined value. The condensate subsequently drains into the separator, which exposes the full surface of the heat exchanger. The resulting steam condensation allows the steam pressure in the circuit to drop.

A passive heat removal system (PHRS) in the form of a water tank is included to allow for passive heat transport through the reactor vessel wall in the event of a simultaneous failure of all the reactor safety systems. The PHRS tank enables the heat dissipation from the reactor core through the evaporation of water from the tank by boiling. The steam generated in the process is release into the atmosphere through air tubes. The water inventory in the reactor tank is reported to be of sufficient quantity to allow for 5 days of continual heat removal, before the increased decay heat results in reactor core damage [23]. During normal operation, the PHRS tank removes no more than 0.2 per cent of the rated power output and simultaneously performs the function of neutron shielding. Additionally, the presence of the PHRS tank allows for repair and maintenance activity of the secondary heat removal circuit without compromising the overall reactor safety level.

In the event of a stream generation (SG) module leak, which is responsible for the autonomous heat removal from the system, reactor core overpressurization could occur following the ingress of steam-water mixture into the coolant pool. The design is able to cope with this problem by means of the upward flow of LBE, which prevents the steam-water mixture from descending with the cooled LBE stream in the primary circuit. Two emergency water cooled condensers attached to the gas system are responsible for relieving the pressure. These are enough to keep the system coolant pressure around its 0.5 MPa design parameter in the event of small SG leaks. Larger SG leaks, which are beyond the gas condenser systems design parameters, will cause a membrane connecting the gas system to the PHRS tank to rupture. The pressure required to rupture the membrane (currently 1 MPa), is set below the computed core-damaging pressure level. Following membrane rupture, the steam-water mixture will flow into the PHRS tank where the majority will condense. As a result, a large part the volatile radionuclide present in the cover gas will be trapped in the PHRS tank, with only a minor fraction reaching the atmosphere through filtered ventilation shaft. As discussed in paragraph 5.3.2.2.3, the radioactive release is expected not to exceed acceptable radiation levels.

5.3.2.2 Coolant Characteristics

Liquid metal cooled reactors using lead-alloy possess several beneficial characteristics that makes them better suited for nuclear power generation in comparison to their light water-cooled counterparts. A selection of the advantages of lead coolant, as encountered in reference [158], are listed below.

5.3.2.2.1 Overview

LBE is a eutectic alloy of lead (45 per cent mass) and bismuth (55 per cent mass). When used as a coolant it possesses several features that might result in favourable neutronics when operating in a fast neutron spectrum which is often considered a requirement for breeding, fuel conversion and actinide transmutation. These are: (1) the small capturing cross-section for fast neutrons ensuring little parasitic neutron losses, (2) a high scattering cross-section, which results in a minimal neutron escape probability, (3) very little energy loss per collision as a result of the high atomic numbers of the coolant elements resulting in very little spectrum softening due to moderation, and (4) a high boiling temperature ensuring little reactivity effects due to the voiding of boiling coolant.

Regarding the requirements LBE places on the structural materials used in the reactor design, the following attributes are reported: (1) manageable corrosion and mechanical degradation of the containment and structural materials resulting in a decent lifetime of the equipment, and (2) a high degree of liquid metal stability as a result of little chemical interaction with the secondary coolant and air as well as little formation of spallation products.

In terms of the thermal hydraulics the following can be said about LBE: (1) it places only moderate power requirements on the circulation of liquid metal, and (2) LBE has a relatively high heat transfer coefficient which allows the heat exchanger to be comparably small [158].

Finally after years of development, the general consensus on the safety aspects of LBE coolant is that: (1) its operation only requires simple and reliable safety measures, and (2) the foreseeable chemical and radioactive hazards are controllable. These statements will be ratified in the paragraph 5.3.2.2.2 and paragraph 5.3.2.2.3.

5.3.2.2.2 Lead Corrosion Processes

The corrosive properties of lead and the resulting search for structural materials with sufficient corrosion resistant properties has been considered to be the main barrier to the widespread application of lead cooled reactor technology [153]. Lead or LBE coolant has been reported to possess undesirable effects such as material dissolving, embrittlement, thermal mass transport and inter-granular penetration of the structural material. The best known structural materials resistant to the aforementioned effects are chromium steel and to a lesser degree austenitic steels, which suffers from the high solubility of the nickel alloy.

Typically, corrosion of the structural materials presents itself as local effect, which appears at temperatures over 550 °C after several hundred operating hours under suboptimal conditions. Suboptimal operating conditions can be summarized as: (1) the use of structural material of inferior quality steel, which contains impurities and unbalances in the alloy, (2) a lack of coolant quality control, and (3) erratic coolant flow regimes throughout the reactor core. Under these circumstances a material corrosion rate of 2.55 mm/year has been reported [153].

A means of increasing the corrosion resistance of the structural materials was devised in form of adding dissolved oxygen to the coolant. It was discovered that the corrosion resistance, and there with the lifetime of the structural materials, was to a large extent dependent on the concentration of dissolved oxygen in the coolant and that after reaching a certain concentration the corrosion process was essentially halted following the creation of a protective oxide film on the surface of the PCV. An additional discovery was that at higher temperatures, alloying the steel with a 1 to 3.5 mass per cent of silicon further enhanced the resistive properties of the structural material [153].

The preservation of this anti-corrosive oxide film (Fe_2O_4) on the steel surface is of the utmost importance and therefore, considering that this layer degrades during operation, a mechanism for maintaining the layer is required. Continued addition of lead oxide (PbO) to the coolant could solve the problem of the oxide film degradation, but imposes a new technological problem which a reactor design using LBE would need to overcome; the formation of slag in the cooling circuit which might result in blockages in the coolant pumps. It has been reported that a dissolved oxygen content of in-between $5 \cdot 10^{-6}$ and 10^{-3} by weight percentage should suffice to maintain the oxide film, while preventing the slag from clogging the coolant pumps [153].

5.3.2.2.3 Radioactivity

Another problem plaguing the LBE cooled fast reactors was providing sufficient radiation safety for submarine personnel in the case of an accidental spillage of LBE coolant as well as for the personnel involved with the subsequent repair activities. According to Zrodnikov et al. 2006 [160], it is necessary to consider this accident scenario as a result of the release of the α-emitting Po-210 radionuclide, which forms in both lead and LBE coolant.

The natural isotope of bismuth, which has an abundance of approximately 100 per cent, undergoes the following nuclear reaction as a result of neutron capture:

$$^{209}_{83}Bi\ (n,\gamma)\ ^{210}_{83}Bi \xrightarrow{\beta^-} {}^{210}_{84}Po \xrightarrow{\alpha} {}^{206}_{82}Pb$$

Po-210, which has a half-life of 138 days, undergoes α-decay and is a volatile liquid at the operating temperatures encountered in the SVBR-100. It partially migrates to the cover gas where it aerosolises. Leakage of coolant or cover gas can therefore result in contamination issues. Similar problems exist in the case of lead coolant, where Po-210 is created following Bi-209 formation as a result of neutron capture in Pb-208, which is 52.3 per cent abundant. The relevant absorption and decay chain is portrayed below:

$$^{208}_{82}Pb(n,\gamma)^{209}_{82}Pb \xrightarrow{\beta^-} {}^{209}_{83}Bi(n,\gamma)^{210}_{83}Bi \xrightarrow{\beta^-} {}^{210}_{84}Po \xrightarrow{\alpha} {}^{206}_{82}Pb$$

It should be noted that the latter decay chain results in significantly less Po-210, approximately 3 orders of magnitude less. However, the fraction of the polonium that manages to migrate out of the coolant bulk to the cover gas is also higher by roughly 2 orders of magnitude as a result of the higher coolant temperature. Equilibrium activity at the end of the reactor service life has been reported to be approximately 10 Ci/kg for lead-bismuth coolant while only being $5 \cdot 10^{-4}$ Ci/kg for lead coolant [153]. The difference can be explained by the fact that in the case of lead coolant the decay chain is only dependent on Pb-208, while for LBE coolant reactivity is influenced by both the Pb-208 and the Bi-209 isotopes.

Recent reports analysing the hazards associated with polonium creation in lead-bismuth cooled reactors are generally positive. A study conducted by Pankratov et al. (2004) [161] concluded, for example that if the gas removal system in the first loop were to become unsealed, a staff member who resided in the reactor room for 1 hour, would receive a radiation dose load of approximately 38 mSv as a result of- α-emitting Po-210 inhalation. This is nearly forty times the amount which a member of the public, who is not employed in the nuclear industry, is allowed to receive from industrial ionizing radiation per year in the U.S. However, it should be noted that this is still within the total occupational dose limit for reactor personnel of 50 mSv per year [162] and also below the acute dose level at which the first biological effects of radiation exposure are reported to manifest (50 mSv) [149]. Furthermore, this computation is based on several conservative assumptions such as a small volume reactor room and an inoperable ventilation system. In summary, the equivalent dose which received during an accident scenario could be within range of the current occupational dose limits [162].

5.3.2.2.4 Thermophysical Properties

In order to determine the thermal inertia of the SVBR-100, two heat sinks need to be taken into consideration: (1) the primary LBE coolant and (2) the PHRS light water tank. The required properties of the latter where again determined by means the IAPWS-IF97, as discussed in paragraph 5.1.2.2. Considering that the primary coolant inventory was encountered as a volume, for LBE both the density and heat capacity are required.

A recent study by the Nuclear Energy Agency (NEA) on the thermophysical properties of lead bismuth coolant reported that the density of LBE can be approximated by the empirical formula [158]:

$$\rho = 11096 - 1.3236 \cdot T$$

Equation 5.2

In Equation 5.2, ρ is given in kg/m^3 and the temperature T is given in Kelvin. It should be noted that the expression is only a function of the temperature T, due to the comparably small additional compression a liquid undergoes following an increase in pressure. Furthermore, the deviation of the density, as computed by this equation, is reported to not exceed the values reported in various literature studies by more than 0.8 per cent [158].

The availability of information regarding the heat capacity of LBE is reported to be rather limited, with only three independent sources being mentioned in the NEA study [158]. Despite an agreement existing between the various sources on the behaviour of LBE in between the melting temperatures of bismuth and lead, 540 and 600 K respectively, the temperature dependence at higher temperatures is heavily debated. This leads to large differences in the various heat capacity development paths. In an attempt to account for the similarities between LBE and its component materials Pb and Bi on the one hand, and to find an expression that would satisfy the experimental data to as large an extent as possible on the other hand, the following expression was eventually decided upon to produce the best fit for the heat capacity of LBE [158]:

$$C_p = 159 - 2.72 \cdot 10^{-2} \cdot T + 7.12 \cdot 10^{-6} \cdot T^2$$

Equation 5.3

In Equation 5.3, the heat capacity of LBE is given in J/kg·K and the temperature T is given in Kelvin.

5.3.3 Proliferation Resistance

In IAEA-TECDOC-1536 the proliferation resistance features of the SVBR-100 shows a large degree of similarity with the 4S in particular and many of the other reactor designs discussed in this thesis in general [23]. The first of these features is that the initial uranium fuel load does not exceed the 20 weight per cent enrichment level, exempting it from the 'direct use materials' status set by the IAEA (e.g. Cochran & Paine (1995) [51]). Furthermore, regardless of whether the fuel is reprocessed at the end of its irradiation period, the fission products and minor actinides will remain confined to the plutonium, the high spontaneous fission rate of these nuclides results in a high radiotoxicity of the spent fuel, which makes it dangerous to handle. Moreover, this could aid in the detection of unregistered nuclear materials.

A design feature that does set it apart from conventional LWR designs is its long fuel irradiation period. As reported, the SVBR-100 is not refuelled on-site in favour of a complete core overhaul every 8 years. The lengthy nature of the fuel cycle renders all intermediate access unnecessary and therefore unauthorized. The advantage of such weld-sealed operation is that the monitoring of the fissile material is simplified. A core overhaul, although the name might suggest otherwise, only involves replacing the fuel rods, it does not extend to the LBE coolant. It is reported that the original coolant is not only capable of lasting through several fuel irradiation cycles, but also through multiple reactor lifetimes. Even after 1000 years of irradiation, the long-lived radioactivity of LBE coolant is still reported to be lower than the radioactivity of natural uranium ore [23]. This could significantly lower the end of lifetime radioactive materials inventory.

5.3.4 Economics

Until recently, fast reactor designs were considered to be more expensive to build than conventional NPPs, which resulted in the estimates of the costs of electricity also being higher. However, a number of recent Russian publications have estimated that the construction costs of a 1625 MWe NPP utilizing SVBR-100 power modules could be lower than the cost of a single-core larger-sized reactor design [160]. As was the case with the 4S, sources that predict cost figures similar to or lower than the figures published for conventional light water reactors merit some scepticism. In this case, nearly all publications on the SVBR-100 originate from the Institute of Physics and Power Engineering, which is coincidentally also one of the main participants in the SVBR-75/100 project. Therefore, all cost studies can be considered as published by beneficiary, which behoves some scepticism regarding their height.

Regardless, stated reasons for the lower costs of electricity are that: (1) the chemically inert LBE does not require an intermediate coolant loop; which tends to raise the costs for liquid metal reactors, (2) the liquid metal coolant makes the reactor safety less dependent on complicated safety systems, and (3) the extended 8 year fuel cycle lowers the reactor downtime due to refuelling [163].

As can be derived from Table 5.2, the economics of the SVBR 75/100 – as calculated by Zrodnikov et al. (2005) - are not only expected to be superior to those of several thermal and fast reactor designs, but also to those of a 10-unit sized PGU-325 natural gas fired power plant. One possible reason that the specific overnight costs of a 16-module SVBR plant are reported to outperform those of a larger-sized reactor are the expected far-reaching effects of learning, serial fabrication and the other economies described in paragraph 4.3.3.3. Considering that the reference size of an SMR NPP in thesis is taken to be smaller than 1625 MWe, it is expected that the specific capital costs will deteriorate somewhat as a result of diminished advantages due to extensive modularity. Furthermore, it should be noted that the reported figures are based on preliminary design data, which might merit some scepticism concerning their validity. In addition, the reported costs are based on a Russian construction scenario, a country with lower average wages than the high income economies this thesis is intended for.

Parameter name and unit	NPP based on SVBR-75/100	NPP based on VVER-1500	NPP based on VVER-1000	NPP based on BN-1800	TEPP based on PGU-325
Installed power of the unit (MWe)	1625	1550	1068	1780	325
Number of units at the plant	2	2	2	2	10
Share of electric power used to operate the plant (%)	4.5	5.7	6.43	4.6	4.5
Net power plant unit efficiency (%)	34.6	34.4	33.3	43.6	44.4
Specific capital cost ($/kW)	610[a] 550[b]	625	819.3	860	600
Design-based cost of produced electricity (cent/kW h)	1.3	1.35	2.02	1.6	1.75

[a] The additional margin cost of ~17% (over the normative one) has been introduced that is 60% of the cost of the reactor installation's equipment.
[b] With due account of realizing the opportunities to changeover to the over-heated steam or to increase the temperature of the fuel elements claddings up to 650 °C.

Table 5.2

Taken from Zrodnikov et al. (2006) [160]

The VVER-1000 series is a Russian designed PWR, which is described to closely resemble a standard Western PWR [149]. It is therefore assumed that a VVER-1000 constructed in a high income economy will have overnight costs comparable to a new 1000 MWe PWR. As discussed in paragraph 5.1.4, the point estimate for the overnight cost of a new conventional light water reactor is taken to be 4339 US$$_{2011}$/kWe. After adjusting the VVER-1000 specific capital costs to the expected price level, the corresponding specific capital cost of the 16 module SVBR-100 reference plant become 3231 US$$_{2011}$/kWe, which will be further rescaled to fit the intended power plant size.

5.4 High Temperature Gas-Cooled SMRs

5.4.1 HTR-PM

High-Temperature Gas Cooled Reactors (HTGRs) are a nuclear reactor subtype, which utilize graphite as a moderating material and helium as a coolant. They are fuelled by ceramic coated particles, also referred to as fuel pebbles, which contain many small fuel kernels. Currently, maximum core outlet temperatures are estimated to be in-between the 700 °C and 900 °C, but it is estimated that higher temperatures will eventually become possible following the development of more advanced structural materials [164].

The High Temperature Reactor – Pebblebed Modular (HTR-PM) is a 250 MWth reactor, which is predicted to generate 105 MWe. This final output estimate was drafted in 2006 and a 2 times 250 MWth plant is currently under construction in China. Although it has been estimated that this setup could achieve capital costs similar to those of a current 2nd generation PWR [165], it has been indicated that this is only the first step towards a much grander setup; a 600 MWe Multi-Module HTR-PM. Model calculations in this thesis will be based on this design.

Figure 5.5| Design Schematic of the HTR-PM Module

Taken from IAEA TWG-GCR-22 (2011) [166]

As can be derived from Figure 5.5, a single HTR-PM module is comprised out of a steam generator (lower right) and a reactor core (upper right), located in separate pressure vessels and connected by means of a horizontal hot gas duct. These three components are located within a concrete containment structure. Helium circulation is realized through a helium blower mounted in the top of the steam generator pressure vessel. It enters the reactor core at 250 °C, through borings in the outer side of the reflector, and following a single pass through the core, it heated up to a temperature of 750 °C at the outlet. Designing the cold channels to pass through the graphite performs essential cooling to the metal outer wall of the reactor pressure vessel, which provides structural support to the ceramic internals.

A necessary adjustment was made concerning the position of the control rods in the HTR-PM, which are located inside the graphite reflector close to the core, instead of being in the middle. This is the result of the disorganised nature of the core fuel pebble loading. These spherical fuel elements, which have a diameter of 60 mm, contain roughly 12000 ceramic coated particles of nearly 1 mm in size and are uniformly spread out across a graphite matrix of 50 mm. The outer 10 mm shell is a fuel-free zone constructed out of pure graphite for structural and chemical protection purposes [164, 167]. Each ceramic coated particle is composed of a UO_2 kernel of 200 – 600 µm and is surrounded by three layers of pyrolythic carbon (PyC) and one layer of silicon carbine (SiC), which is where the ceramic coated particles name is derived from; TRISO fuel, which stands for TriStructural-Isotropic fuel. Each spherical fuel element is reported to contain 7.0 grams of heavy metal and in order to achieve a reasonably uniform distribution of fissile material the fuel is operated in a multi-pass scheme, which means the fuel pebbles are continuously circulated [22].

5.4.2 Safety Features

5.4.2.1 Safety Systems

The HTR-PMs design philosophy is typical for a generation III+ nuclear reactor, considering that it aims to realize high safety levels, whilst simultaneously reducing the overall level of design complexity. In fact IAEA-TECDOC-1485 [22] states, that the overall plant safety should be of the standard that would make any emergency measures outside of the plant boundary technically unnecessary, or at the very least redundant. Within this context this would imply that it could be minimized in favour of other redundant systems. Furthermore, the HTR-PM design also adhered to the defence-in-depth approach and the basic safety concept can be considered to be three-pronged.

The first safety concept is the confinement of the radioactive material by means of multiple barriers within the fuel elements. It should be recalled that every fuel kernel is coated with three PyC layers and one SiC layer. Subsequently, these ceramic fuel particles are proportionally dispersed within a graphite matrix, which in turn is covered by a fuel-free graphite shell. It has been established that the fuel elements used in the HTR-PM are able to withstand temperatures of up to 1620 °C [164]. Both the IAEA and the Institute of Nuclear and New Energy Technology of Tsinghua University consider accident scenario's in which these temperatures could manifest themselves extremely unlikely [164, 22]. The fuel itself is only considered the first barrier however, the more traditional pressure vessel and containment building being counted as the second and third barriers to radioactive material release.

When accident conditions do arise, passive processes are reportedly capable of removing the decay heat. If, during an accident scenario, the circulation of helium is halted, the low reactor power density and the large heat capacity of the in-core graphite structures enable the heat to be dissipated towards the external environment by means of conduction and radiative transfer within the internal core structures. It is believed that these processes will prevent the fuel from reaching critical temperatures [22]. Finally, it is reported that as a result of the fuel and moderator temperature reactivity coefficients being negative, that under all foreseeable accident scenarios, a cease of helium circulation will result in an autonomous shut down [22]. It is recognised that this approach seems counter-intuitive, but considering that a temperature increase decreases the overall reactivity and that the thermal conductivity supersedes the build-up rate of heat in the reactor core, this assessment appears to be valid when based upon the design specifications.

Following the 2010 downfall of the South African Pebble Bed Modular Reactor (PBMR) project, a critical assessment of why the development of a reactor with such commendable safety features could have failed outright was performed [168]. Although two of the tree lessons that can be learned from the project refer to the opportunity cost of pursuing nuclear power and the accountability for public money, the third refers the state of development of HTGR technology. It is stated that:" *[...] any further attempts to commercialize HTGRs must be based on a clear understanding of why earlier attempts have failed and with a high level of confidence that the earlier problems have been fully overcome*" [168]. The PBMR was largely designed around the German 46 MWth AVR Pebble Bed Reactor, which operated in-between 1967 and 1988, reaching coolant temperatures of up to 990 °C. It is concluded by Moormann (2008) [169] that the unresolved safety issues of HTGR technology are insufficiently published. In the case of the HTR-PM, it should be considered a possibility that the powerful attraction HTGR technology has on nuclear scientists, in combination with technological overconfidence resulting from withholding certain critical issues, could result in another PBMR-type situation. Therefore an overview of the unresolved issues is deemed appropriate.

The primary circuit of the AVR is reported to be heavily contaminated with the fission products Sr-90 and Cs-137. While the exact amount is unknown, it has been estimated to amount to a significant fraction of the core inventory [169]. This contamination has been contributed to faulty fuel element design. However, Moormann (2008) [169] claims that the AVR contamination was primarily caused by core temperatures in excess of its design specifications. Irrespective of its cause, in-core contamination in HTRs is a problem inherent to the use of TRISO fuel, which cannot contain metals at the same degree of proficiency with which it is able to contain noble gasses. Moormann (2008) [169] reports that, when their specific temperature limits are exceeded during long term operation, metals diffuse through the fuel kernel, coating layers and graphite containment. These problems do not occur in intact conventional LWR fuel elements up to temperatures of 2500°C, because the cladding remains at temperature below 600 °C. At these temperatures release by means of diffusion can be excluded. These findings are backed up by experiments conducted at the HFR Petten, which confirm the relation between an increase in core temperature and the release of metallic fission products from the fuel elements into the primary circuit [170].

In conclusion, the abovementioned sources indicate that pebble bed HTR technology as pioneered by the German AVR design, still has several unresolved technical issues that the HTR-PM designers will have to address before it can be brought online. These issues are: (1) occurrence of core temperature in excess of the design specifications, the reasons for which are ill-understood, (2) inadequate retention of metallic fission products by the TRISO fuel coating, especially under high temperature conditions during long term operation and (3) safety, maintenance and decontamination related problems resulting from the possible build-up of metallic fission products in the primary circuit [169].

5.4.2.2 Coolant Characteristics

5.4.2.2.1 Overview

The use of a single-phase coolant, like the inert gas helium, has several advantages. For one, the possibility of flashing; the situation in which a quantity of coolant vapour created as a result of the high ambient temperature in the reactor vessel is sufficient to initiate a combustion reaction, can be ignored. Furthermore, pump cavitation; the formation and subsequent implosion of small bubbles in a liquid coolant, which tends to be a significant cause of wear for turbines, cannot occur. Additionally, helium coolant is not susceptible to neutronic reactions, nor are there any chemical reactions between the coolant and the fuel.

5.4.2.2.2 Helium Corrosion Processes

Although Zhang and Sun (2007) [165] and the IAEA [22] claim that the HTR-PM will have a core outlet temperature of roughly 750 °C, recent studies, such as Zhang et al. (2009) [164], claim that the core outlet temperature may reach temperatures of up to 950°C. At these elevated temperatures the impurities present in the helium coolant, such as H_2, H_2O, CH_4, CO, CO_2 and O_2, could interact with the structural nickel base metallic materials. In the past decades multiple alloys have been developed that could be used for high temperature applications, of which several could be useable in the internal heat exchanger (IHX) and primary circuit. What these materials have in common is that they form a continuous, self-replenishing barrier on the material surface, which functions as a protective layer. An extensive analysis of the two most promising structural materials, alloy 617 and alloy 230, was performed in Cabet and Rouillard (2009) [171]. The article reveals that alloy 230 in particular can be associated with high thermal stability and mechanical strength, although reference is made on how it performs relative to alloy 617. It is reported however, that for both materials, under specific high temperature operational conditions, the surface chromium oxide layer is irreversibly reduced.

Argonne National Laboratory (ANL) [172] reports that, for alloy 617, corrosion is minimal below 475°C as the alloy, which contains a high chromium content, forms a thin protective chromia scale on its surface. At 900°C the thin chromia scale is still present, but there are also signs of internal oxidation and the presence of precipitate-free (uncovered) zones at the grain boundaries. Both are considered undesirable because they have detrimental effects on the mechanical and corrosion properties of an alloy. Above the 900°C mark, alloy 617 is described to be subject to rapid carburization or decarburization depending on the impurity composition of the helium gas [172]. Alloy 230 is described to destabilize in a similar fashion [171]. The rapid onset of carburization or decarburization is prompted by a critical temperature being reached, which is found to be dependent on the composition of the alloy and partial pressure of CO. The ANL report documents that at the critical temperature the CO becomes relatively more stable than the metal oxide layer. Therefore, above the critical temperature the H_2 impurities reduce the chromia layer forming H_2O, which subsequently reacts with the carbon present in the metal alloy to form CO [172].

Cabet and Rouillard (2009) [171] indicates that there are several ways in which helium impurities induced high temperature corrosion can be prevented. The first is actively controlling the helium chemistry in order to suppress the reactions that lead to chromia layer reduction. It is declared that this effect can be achieved by increasing the partial pressure of CO. It should be noted though, that any new atmospheric composition should also be compatible with other in-core high temperature materials, such as the carbon structures. A second solution is further optimization of the alloys composition. Different chromia layer forming materials have different oxidation states as a result of the different combinations of minor reactive elements they are composed of.

5.4.2.2.3 Thermophysical Properties

One aspect of the HTR-PM design, which is not that forthcoming, is the primary coolant volume/mass, which is required to determine the thermal inertia.

The mass of the helium in-circulation was indirectly determined to be approximately 3000 kg per 105 MWe reactor system, based on the following indicator: "*The helium purification system of HTR-PM was designed for a helium flow rate of 150 kg/h, corresponding with a 5% of the helium inventory in primary circuit*. " [173]. The heat capacity C_p of helium is reported to be constant across the temperature range at 5.195 kJ/kg·K [174].

In addition to the heat uptake of the helium gas, the HTR-PM is designed to dissipate decay heat by utilizing the large heat capacity of its in-core graphite structures [22]. Therefore, it would not be justifiable to base the thermal inertia of the HTR-PM on the heat capacity of the primary helium coolant alone. In light of this, the heat capacity of the in-core graphite structures will be added to the heat capacity of the primary coolant circuit. The heat capacity of solid graphite was found to be constant at 0.71 kg/kJ·K [21].

5.4.3 Proliferation Resistance

There are several references that state that the HTR-PM, operated under the fuel cycle described in paragraph 4.3.3.6.1, could have reduced spent fuel attractiveness. In part, this is due to the most important proliferation resistance feature of the VHTR being the fuel itself [175]. In order to acquire a significant quantity of nuclear material, as described in paragraph 3.3, one would have to process several metric tons of encased fuel particles. In addition to this, the high design burn-up of 80 GWd/tonne U generates plutonium with less favourable isotopic composition for use as weapons material than fully burned conventional light and heavy water reactors. The isotopic fraction of Pu-239 in spent PBMR fuel being as low as 46 per cent according to the ORNL [175].

In addition, the IAEA reports that all the reprocessing techniques available to acquire the fissile material are considered sophisticated and costly [22]. Ougouag et al. (2006) [176] expands on this by explaining that the principal difficulty lies not with the isotopic composition of the spent fuel, which is reported to be similar to the spent fuel encountered in conventional light water reactors, but in the chemically inert nature of the graphite and silicon carbide structural materials. It is reported that very few chemical agents are capable of

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[21] Taken from the Engineering Toolbox: http://www.engineeringtoolbox.com/specific-heat-solids-d_154.html

dissolving these outer layers effectively and efficiently. It is noted though, that at great costs, a combination of physical and chemical processes is capable of breaching the containment layers. Although the feasibility of separating spent TRISO fuel pebbles is therefore recognised, it is also stated that any industrial scale operation would come at a great expense [176]. Therefore, from the point of view of nuclear safeguards, the fissile contents of TRISO fuel pebbles should not be considered irretrievable.

It should be noted that a preliminary investigation performed by General Atomics concluded that the decay heat generated by spent fuel of the Gas Turbine High Temperature Reactor, a HTR with similar design features as the HTR-PM, was assessed to be approximately 50 per cent of the decay heat of a conventional LWR [177]. Although this is a favourable characteristic in terms of safety, it reduces the proliferation resistance somewhat, because it implies that spent fuel is less radiotoxic and therefore lowers the material requirements of handling the material.

5.4.4 Economics

An economic analysis of the HTR-PM design was conducted by Zhang and Sun (2007) [165] to determine the price differences in the major cost components between a 2-module HTR-PM and a second generation PWR. The conclusion was that the increased cost of the RPVs and reactor internals could be compensated for 50 per cent by the reduced system complexity. Furthermore, it was indicated that the remainder of the cost increase could be compensated by: cost reductions in the turbine equipment, benefits related to modularization and a potential decrease in construction length in combination with cost reductions in on-site engineering. Furthermore, Zhang and Sun (2007) [165] estimated that the specific costs of a multi-module HTR-PM would be somewhere in the 90 to 120 per cent range in comparison to the specific overnight costs that would need to be incurred for a PWR design. As can be seen in Appendix 10.8.2, the independent estimate derived in this thesis is located within these parameters.

Notwithstanding the negative economies of scale, two items were identified that could potentially have favourable effects on the specific capital costs, the first one being the specific costs of the turbine plant equipment, which could be 25 per cent lower. The reason for this was given to be that the higher core output temperature of the HTR-PM can achieve higher efficiencies utilizing conventional steam turbines than PWR designs. Additionally, Zhang and Sun (2007) [165] reported that the quotations received from vendors for the 200 MWe demonstration plant were roughly 75 per cent of the specific costs that would have been incurred for a PWR reactor design. It can be argued that this is a conservative estimate, following that additional cost reduction could be expected in multi-module plants where a larger-sized turbine-generator system, with additional economies of scale, could be utilized. Furthermore, cost advantages could also be attained in the buildings and structures, project management and owner's cost divisions. Following that in a multi-module plant costs are spread out of multiple modules, it was reported that this could result in a 20 per cent lower specific overnight cost for the HTR-PM in comparison to conventional PWR designs.

In line with the modelling principles outlined in paragraph 4.3.3.3, the specific capital costs of the HTR-PM will be approached by adjusting the specific capital costs of a larger sized reference plant to quantify the benefits and disadvantageous of smaller unit size. The reference plant was taken to be the 1152 MWe HTGR described in Choi (2011) [178]. Subsequently, the specific capital costs of a 1300 MWe PWR were derived by taking the average specific capital costs for a 1000 MWe LWR as given in Du and Parsons (2009) [61] in US$$_{2011}$, and adjusting for the economies of scale ϑ_{ES} by means of Equation 4.8. Subsequently the ratio between the HTGR and the LWR, as found in Table 5.3 was taken to be representative for the HTR-PM and a PWR. As shown in Appendix 10.8.2, the specific overnight costs for a 1152 MWe HTGR were estimated to be 3569 US$$_{2011}$/kWe.

Capital and O&M cost estimate of nuclear power plants.[a]

	1300 MWe LWR (GIF, 2007)	1400 MWe ARR	1152 MWe HTGR
Direct cost (M$)	1596	1990	1092
Indirect cost (M$)	907	610	483
Contingency (M$)	374	544	313
Decommissioning cost (k$/yr)	1700	2300	1200
Unit capital cost ($/kWe)	2213	2246	1639
Annual O&M expenses (M$)	94	90	36

[a] In 2007 dollars.

Table 5.3

Taken from Choi (2011) [178]

5.5 Heavy Water Reactors

5.5.1 PHWR-220

Pressurized Heavy Water Reactors (PHWRs) are a reactor type that was initially developed in Canada due to the lack of enrichment facilities that necessitated the use of natural uranium as a reactor fuel. In LWRs, the use of natural uranium fuel would not result in a self-sustaining chain reaction; this is due to the large neutron absorption cross section of the hydrogen pair in each water molecule. Heavy-water reactors can operate on natural uranium because deuterium (D or ^{2}H) has a smaller absorption cross section then the hydrogen in light water. However, considering that deuterium is twice as heavy, on the average less energy is lost per collision, which means that more collisions are needed to achieve thermal neutron energies in comparison to H_2O. This means that if the reactor core where constructed in the same manner as a PWR, the vessel would have to be significantly larger than in a conventional LWR. The Canadians were able to circumvent this problem by adopting the pressure tube concept. This allowed the calandria; the large horizontal cylindrical tank containing the moderator to remain at atmospheric pressures. The design was dubbed the CANDU reactor, or CANadian Deuterium Uranium reactor [149].

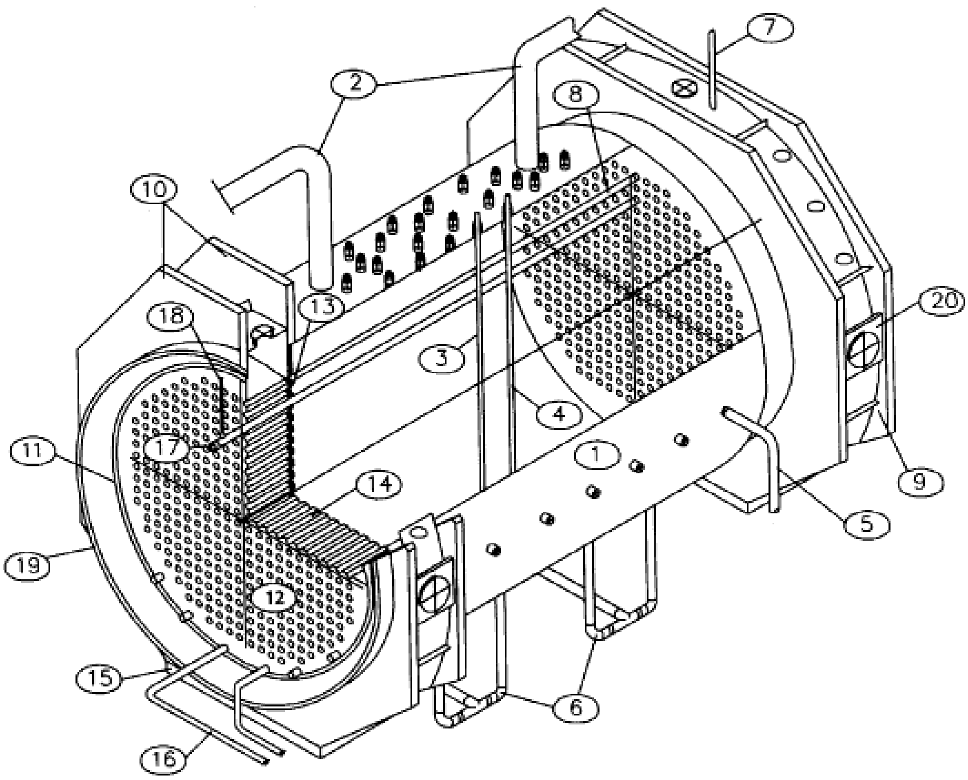


Figure 5.6| Schematic overview of a PHWR-220 calandria vessel

Taken from Bajaj and Gore (2006) [179]

In essence, the PHWR-220 does not differ that much from a CANDU reactor. As can be derived from Figure 5.6, in line with the CANDU, the fuel channels are positioned along the horizontal axis. The heavy water moderator is located in the unpressurized calandria, which is pierced by 306 pressure tubes. These pressure tubes contain both the fuel rods and the heavy water coolant, which is circulated by the primary pumps towards heat exchangers, located on opposite sides of the calandria. Here the heat is transferred to the light water working fluid. It is reported that flow direction in adjacent pressure tubes is opposite and that the fuel, which is comprised of bundles made up out of relatively few fuel rods, is placed inside these pressure tubes. An advantage of this set-up is the relative ease of access to the fuel, which allows for online refuelling; new fuel bundles can be pushed in at one end of the fuel bundle while the spent fuel bundle emerges at the other end. The fact that these reactors do not need to be shutdown in order to refuel could allow them to achieve higher load factors than reactors that do require periodic shutdowns.

The Indian Nuclear Energy Program, which had a limited amount of resources and infrastructure at its disposal, quickly adopted PHWR technology in its quest for self-reliance and energy security. Like Canada at the time, India possessed the indigenous industrial capabilities and fabrication technologies to develop and produce pressure-tubes, which are mechanically less complex than high pressure reactor vessels. Furthermore, India possessed the ability to produce heavy water. It has been reported that the Indian PHWR design has been gradually improved over the years, adopting technological innovations, feedback from experience, both domestic and abroad and lessons learned from reactor incidents, while at the same time satisfying ever changing regulatory requirements and cost considerations [179].

5.5.2 Safety Features

5.5.2.1 Safety Systems

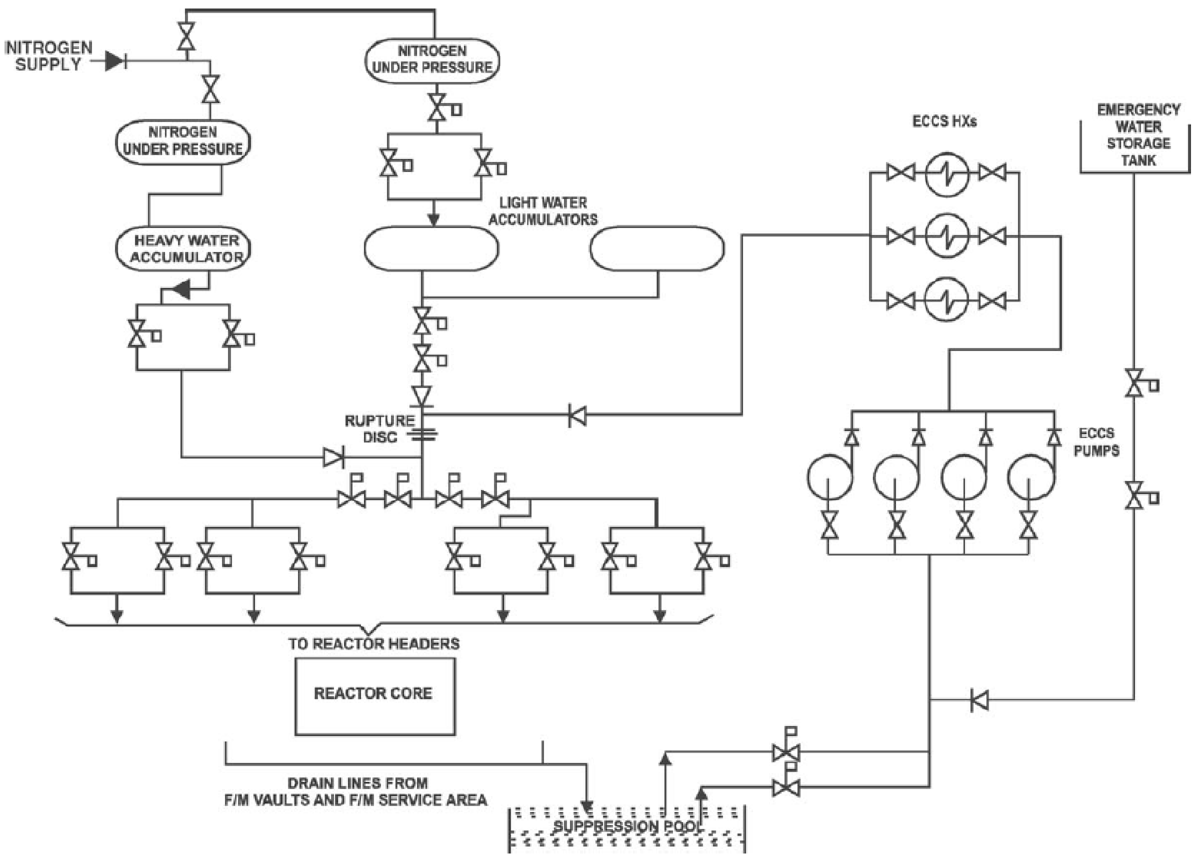


Figure 5.7| Hydraulic diagram of the safety systems encountered in the PHWR-220

Taken from Bajaj and Gore (2006) [179]

The PHWR-220 is a conventional reactor design of which the first version achieved criticality in 1983 [180]. Therefore, it should come as no surprise that the most recent version of the PHWR-220 shares a great deal of its safety features with other conventional reactor designs, such as the latest generation of PWRs.

The safety systems of the PHWR-220 can be subdivided into three categories, the Reactivity Control and Shutdown Systems (RCSS), the Emergency Core Cooling System (ECCS) and the Containment.

The RCSS is designed for power control purposes and it consist of 4 cobalt/stainless steel regulating rods for power maneuverability, 8 cobal/stainless steel absorber rods which can be withdrawn to provide the reactor with the excess reactivity required to compensate for the xenon buildup (xenon override) and 2 cadmium sandwiched stainless steel rods to be used in the event that a quick power reduction is necessary [181]. Additionally, in support of RCSS an automatic liquid poison system is present. Furthermore, under accident conditions the PHWR-220 has two independent shutdown systems at its disposal, each one equipped with the ability to facilitate a full shutdown. The primary shutdown system, the first line of defence in the event of a shutdown demand, is composed of 14 mechanically driven cadmium rods. The secondary shutdown system consists of 12 vertical tubes filled with liquid poison. These tubes penetrate directly into the calandria allowing for a fast filling of the heavy water moderator with liquid poisons.

It is reported by Bajaj and Gore (2006) [179] that the ECCS of the PHWR-220 incorporates a high pressure heavy water injection system, an intermediate pressure light water injection system and a low-pressure long-term circulation. As shown in Figure 5.7, both the high pressure heavy water injection and intermediate pressure light water injection are provided with water by a system of accumulators pressurized by a common nitrogen supply tank. In the event that the system pressure falls further, the low pressure light water injection system is activated. Once the primary objective of refilling the core has been achieved, recirculation is initiated which will be used for long term heat removal. Suction is provided by a suppression pool; a large body or light water in located the basement of the containment building. The heat from this pool, in turn is removed by an active process water cooling system, which is in turn connected to the ultimate heat sink, a reservoir/canal [179]. Although not officially part of the ECCS, the existence of the calandria vault cooling systems should also be recognised. Much in the same way as the PHRS of the SVBR-100, described in paragraph 5.3.2.1, it serves to shield the reactor personnel from attenuating neutrons and core gamma rays and take up heat from the heavy water moderator. The PHWR calandria vault if filled with demineralized water. Furthermore, it is designed to start removing heat load as soon as the overall moderator temperature increases by 5°C [181].

As can be seen in Figure 5.7, coolant injection by the ECCS only utilizes two of the four possible headers. Which couple is selected is dependent on the size and the location of the break. It should be noted that very small breaks can be handled without involving the ECCS as the spilled heavy water can be collected by suitably located drainage pipes. The pipes transport the heavy water to a drainage tank from where it is pumped back into the primary heat circuit after it has been cooled in a heat exchanger and passed through a purifier.

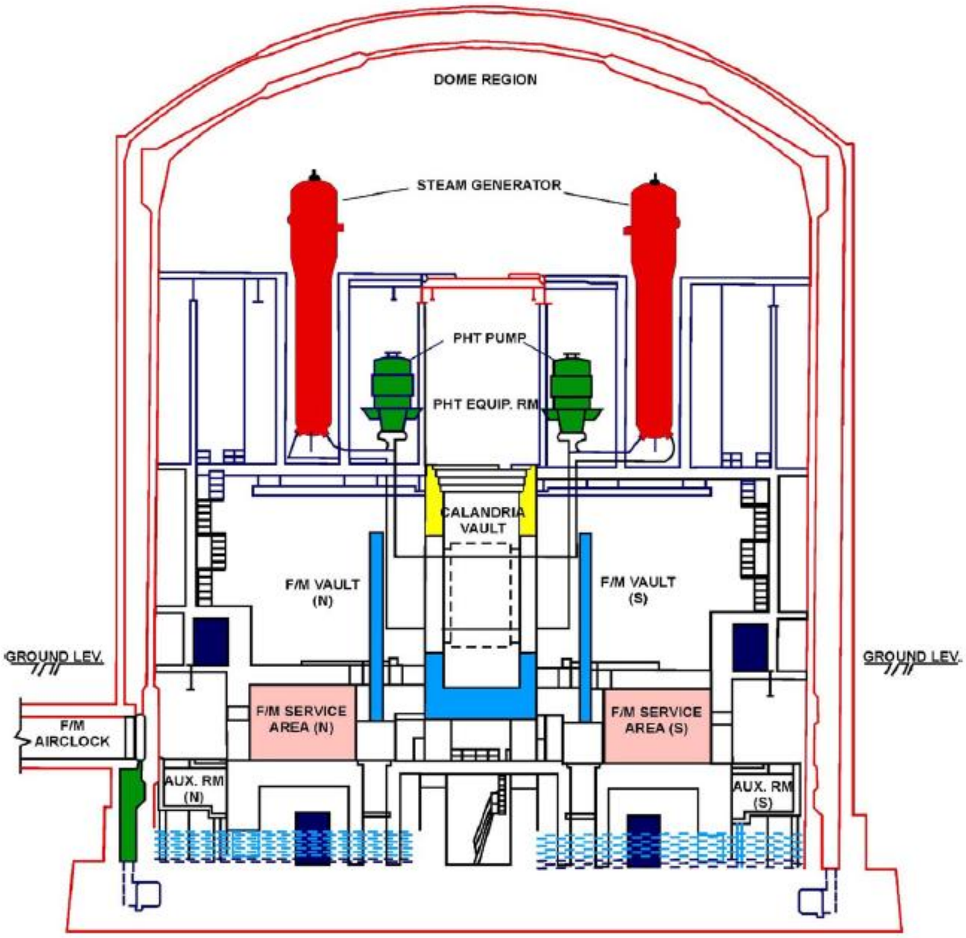


Figure 5.8| Cross-section of a PHWR-220 containment building

Taken from Bajaj and Gore (2006) [179]

As can be derived from Figure 5.8, the PHWR-220 utilizes a double containment principle. As would be the case in a single-layered containment building, the space between the primary and secondary containment envelopes is maintained at below-atmospheric pressure levels. It is claimed by Bajaj and Gore (2006) [179] that this set-up considerably reduces the release of radioactive material to the environment in the event of an accident scenario. Both the containment structures are made of concrete and the inner-lining of the primary containment vessel is covered with epoxy coating to increase the leak-tightness and simplify the decontamination process. An additional benefit of employing the double containment principle is that further leak-tightness enhancements, such as the addition of a steel-liner, are no longer considered necessary.

The abovementioned suppression pool plays a pivotal role in the pressure suppression system designed to limit the pressure in the containment building in the event of a LOCA or a steam line break. As described in Bajaj and Gore (2006) [179], the primary containment building is divided into two sections accident based volumes, a drywell V1 and a wet well V2. These two volumes are separated by leak-tight walls and floors and connected by means of a vent system, which passes through a suppression pool containing light water. During certain accident conditions, a pressure rise in V1 will cause steam-air mixture to start flowing towards the pressure pool where steam will condense. Subsequently the escaping air will pass into volume V2. System operation is fully passive and non-active during normal operation. Additionally, the pressure suppression is also part of the long-term emergency core-cooling arrangements. In order to lower the temperature and therewith the pressure of the reactor vessel during accident conditions, a series of air coolers are situated at various positions spread out across volume V1 of the containment building. The coolers are powered by on-site diesel generators and the heat is removed by means of an assured process water supply. At low pressure levels where conventional cooling does not suffice, further depressurization is achieved by means of controlled gas discharges through a series of filters.

5.5.2.2 Coolant Characteristics

5.5.2.2.1 Thermophysical Properties

In order to assess the thermal inertia of the PHWR-220, the heat capacities of the heavy water coolant, the heavy water moderator and the light water coolant pool are required. Again, the required light water properties are determined by means of the IAPWS-IF97 methodology, discussed in paragraph 5.1.2.2.

Furthermore, similar to the thermophysical properties of light water, the IAPWS has summarized the properties of heavy water. The most resent version is the *Revised Release on the IAPS Formulation for the Thermodynamic Properties of the Heavy Water Substance* (2005) [182]. However, the fundamental equation, a Helmholtz function of the variables temperature and density, is obstructed from being widely used by the same computational barriers as the IAPWS-95 discussed in paragraph 5.1.2.2. Durmayaz (1997) [183] confirms that a set of independent values consisting of density and temperature is cumbersome to work with given that they are different than the independent variables commonly used to analyse thermal-hydraulic systems. Therefore an alternative approach based on approximate functions is suggested by the author.

In this thesis, a computer program (D20 V1.0 [184]) is used for the computation of the specific heat capacity of heavy water which is based on the piecewise continuous approximation functions as given in Durmayaz (1997) [183].

5.5.3 Proliferation Resistance

It is reported by Parent (2000) [185] that the widely held view is that PHWR possess worse proliferation resistance characteristics than LWRs. There are two main arguments for this statement. Firstly, as a result of the fact that PHWRs do not need to be shutdown in order to be refuelled, the movement of nuclear material is more difficult to monitor in comparison to LWRs, which require a shutdown before fuel rods can be replaced. Secondly, as shown in Appendix 10.7.1, PHWRs are known to produce more plutonium per GWe_y than LWRs. In addition to that, as a result of the low burn-up, the plutonium in the spent fuel has a more favourable isotopic composition for use in nuclear weapons owing to its higher percentage of odd-numbered plutonium isotopes, as discussed in paragraph 4.3.2.1.

5.5.4 Economics

The IAEA estimates that heavy-water reactors are approximately 15 per cent more expensive than light water reactors under the same licensing and siting criteria [186]. However, more recent sources often claim that the differences in overnight costs between heavy water and light water designs are insignificant (e.g. [185]). Considering that the overnight capital costs for a conventional large-sized LWR were taken to equal 4339 $US\$_{2011}$/kWe in paragraph 5.1.4, the specific overnight costs for a new large-size HWR will be assumed to equal the same.

5.6 Molten Salt SMRs

5.6.1 FUJI

The FUJI is a simplified Molten Salt Reactor (MSR) designed for closed Th-U233 fuel cycle operation. It is envisioned to be the power generation component of the so-called THORIum Molten salt Nuclear Energy System (THORIM-NES). A fuel cycle concept which is radically different from the fuel cycles utilized by the reactor types discussed thus far. The major differences between the FUJI and the more conventional innovative reactor designs are that it: (1) uses thorium instead of uranium as the fertile element, which breeds the fissile U-233, (2) contains a fissile molten salt mixture instead of solid fuel elements, and (3) largely separates power production from fuel breeding.

Initially, power production can be undertaken using either U-235 or Pu-239. Later on a specific type of Accelerator Driven System (ADS), called an Accelerator Molten Salt Breeder is envisioned to be added to the fuel cycle, which is devoted entirely to breeding U-233 by means of thorium spallation. A complete overview of the THORIM-NES philosophy is given in Furukawa et al. (2008) [187].

The FUJI Reactor is based on the molten salt reactor experiment, which was conducted at Oak Ridge National Laboratory between 1965 and 1969. Since its closure it has been followed up by a large amount of conceptual design studies aimed at optimizing the design (e.g. [188]). There have been significant advancements since these early attempts and the FUJI reactor has improved on previous designs by being simpler, self-sustaining in terms of fissile fuel use, size-flexible, and having a reduced chemical processing need, which went from quasi-continuous to once every 2000 EFPDs, which comes down to once in every 7.3 years when taking into account the predicted load factor of 0.75. Refuelling is still performed on a quasi-continuous basis with 2 kg of U-233 being supplied to the core every 30 EFPDs and 67 kg of Th being added every 150 EFPDs.

In the original Molten Salt Breeder Reactor, both the core boundaries in the radial and the axial directions were conceived to be under-moderated in order to increase the breeding yield in the blanket material. This however, resulted in a large maximum-to-average flux ratio; which severely limits the lifetime of the graphite moderator due to the high neutron flux near the reactor core centre. It is reported in Mitachi et al. (2007) [189] that the neutron flux can be flattened by introducing multiple core-regions. Considering that the neutron flux is typically high near the core's centre and low near the core boundary, in order to achieve a level flux distribution the neutron multiplication factor k_∞ should be decreased near the core centre while being increased near the core boundary. In the FUJI this is done by varying the fuel volume passing through the graphite moderator. Therefore, the higher flux core region is designed to be under-moderated. Due to the higher fuel volume per volume of graphite less thermal neutrons will be generated lowering the fission-neutron flux in the core centre. Because an increase of k_∞ near the core boundary region would result in greater neutron leakage, a third core region was introduced, in-between the core centre and the periphery. The flatting of the neutron flux, resulting from this power distribution mechanism has been reported to have eliminated the need for core-graphite replacements during the reactor's design lifetime [189].

The FUJI-U3 design, under investigation in this thesis (see Figure 5.9), has a power output of 450 MWth and a rated power of 200 MWe. The fissile and fissionable isotopes, as well as the fission products and the minor actinides resulting from neutron capture are contained within the molten salt as ionic compounds. This molten salt mixture is forced to circulate through the primary coolant loop, but only achieves criticality once it enters the core region which is a nearly solid block of graphite functioning as the moderator and the reflector. Following the resulting fission events the fuel is heated as it passes through the reactor core. Once the fuel is no longer surrounded by the graphite core the molten fuel salt ceases to be critical. Subsequently, it travels through the heat exchanger before it is recirculated back into the core region. Following heat transfer, the heat is transported through a secondary coolant salt loop filled with $NaBF_4$-NaF, before it enters a super-critical Rankine cycle where steam expansion results in electricity generation with a thermal efficiency of roughly 44 per cent [190].

5.6.2 Safety Features

5.6.2.1 Safety Systems

Design based analysis of the FUJI reactor has been subdivided into two categories; single component/operator error and static component failure. It has been reported that none of the foreseen failure paths have been found to result in accident scenarios with significant consequences [23]. The foremost safety properties of the FUJI reactor are related to its operating conditions and the chemical aspects of the molten fuel salt. The system pressure on the first two coolant loops, for example, is lower than 0.5 MPa which limits the mechanical strain on the system. Additionally, the fuel salt is chemically inert; it does not react with oxygen or moisture in the air and additionally does not suffer any damage from radiation exposure.

Furthermore, the boiling point of the FLiBe fuel salt, an acronym for a fuel salt based upon a Fluoride, Lithium and Beryllium composition, is reported to be 1673 K [191].

Considering that the operating temperature of the system does not surpass the 973 K under normal operating conditions, there is a large margin for temperature rises. In addition to this, the primary circuit and the power producing circuit are separated by means of a secondary loop, containing pure molten salt. This severely limits the possibility of pressure build-up in the (radioactive) primary loop due to steam ingress or water evaporation [23].

In the event that molten salt leaks from the primary circuit into the reactor vessel, the absence of a reflector prevents it from achieving criticality. Although the yield of delayed neutrons per fission event was found to be smaller for U-233 than for U-235, the neutron lifetime, also known as the mean neutron generation time, is longer for U-233 than for U-235. Combined with excellent reactor controllability characteristics, due to the large negative prompt temperature coefficient of the molten salt fuel, the power flux is not very prone to reactivity swings. This effect can be attributed to the molten state of the reactor fuel. When the temperature in the fuel rises, the fuel salt composition is expanded. The result is that the amount of fissile material present in the core region is reduced which reduces the number of fission events. The tendency of the FUJI to remain close to its design specifications is further enhanced by the large heat capacity of the graphite, which slows any rise in temperature, in combination with the large negative temperature reactivity coefficient it is believed that this provides the operator with the means to exert sufficient control over the reactor system.

When using graphite one needs to be mindful of the possibility of a graphite fire. Like with all fires, next to presence of a combustible material, two other conditions need to be met: sufficient flow of oxygen and the availability of an external heat source. In the event of a primary circuit breach, the threat of ingress of air/oxygen is removed by filling the high temperature containment with nitrogen gas. Furthermore, the probability of a breach in containment is reduced due to the high boiling temperature of the molten salt mixture, which limits the chance of over-pressurization due to molten salt vapours. In the event of air passing into the high pressure containment from the outside, the fuel salt is drained into the drainage tank, effectively removing the heat source.

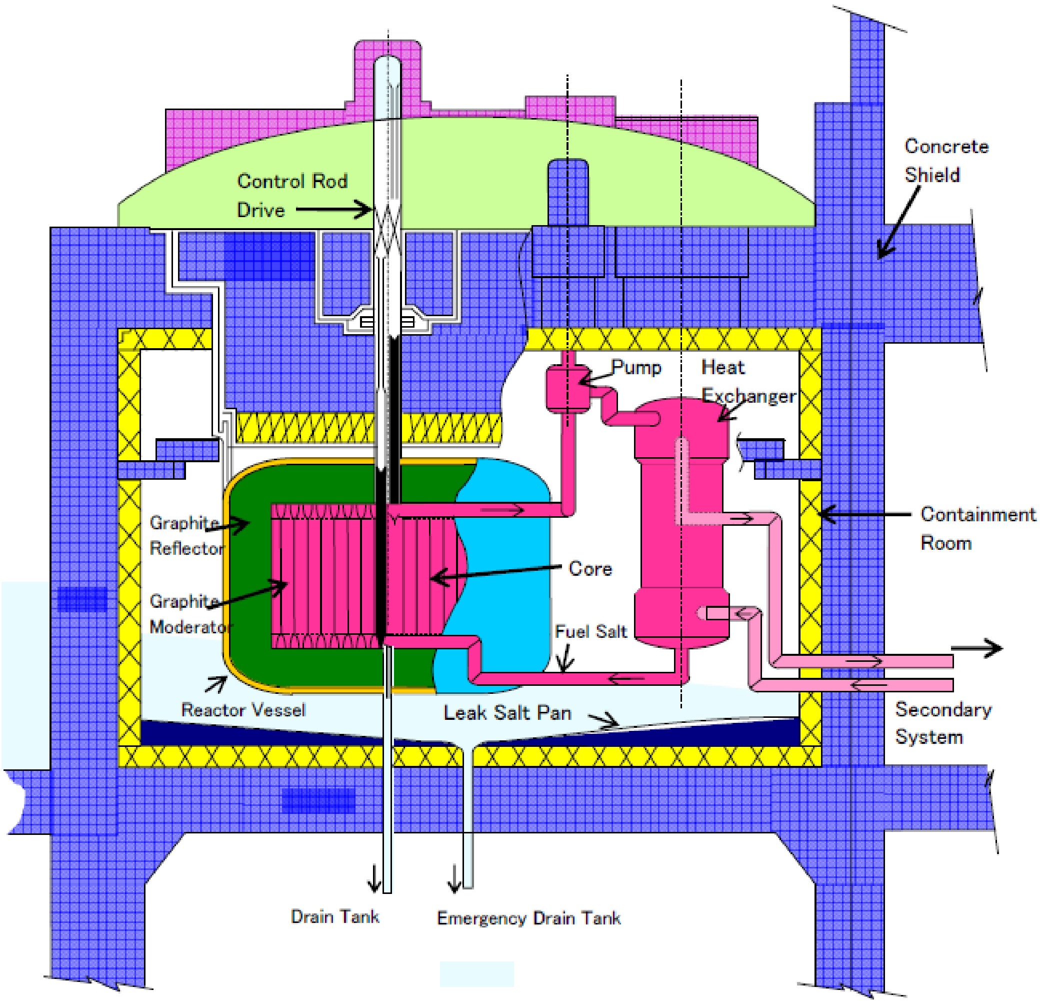


Figure 5.9| Schematic overview of the containment building of a FUJI-U3, including the reactor vessel and heat exchanger

Taken from IAEA (2007) [23]

An additional benefit of molten fuel salt is that the fuel salt composition can be continuously altered if need be. Therefore, the excess reactivity of the fuel can be kept at a minimum. Subsequently, the control rod (there is only one) only requires a small reactivity worth, which defines its ability to remove excess reactivity. Its limited contribution draws into question the necessity of adding a control rod in the first place.

Additional indicators encountered in the literature on the safety of the FUJI SMR are related to the mechanical advantages of liquid fuel operation. These include features such as the ability for on-line fission product removal, which allows for the capture of gaseous fission products, such as xenon, krypton and tellurium inside activated charcoal beds or similar structures. This could potentially limit the consequences of a LOCA. Also, following that the primary circuit is confined inside high temperature graphite containment (about 810 K), there is no need for external devices such as heaters, insulator and neutron shields. It is reported by Furukawa et al. (2012) [191] that this could substantially simplify processes such as remote maintenance, inspection and repair, which in turn could result in higher safety levels due to further limitations on the possibility of personnel being subject to radiation exposure. As shown in Figure 5.9, the bottom of the containment building is shaped like a funnel, which allows any fuel salt that breaches the reactor vessel to be transported into a drain tank. During an emergency event, the entire content of the primary circuit can be drained into a borated water pool cooled with a vapour condenser. Separating the primary circuit from the drainage tank is a freeze seal valve, which melts in the event of excessive heat build-up. Subsequently, the FLiBe salt is pulled out of the reactor core by means of gravity. Experimentation with freeze seal valve technology ran alongside the Molten Salt Reactor Experiment at ORNL, it was concluded that all designs performed ‘satisfactory’, each seal being frozen and melted 100 times [192].

5.6.2.2 Coolant Characteristics

5.6.2.2.1 Overview

Many candidate salts are currently being investigated for use as coolant in Molten Salt Reactors. Most studies are inclined to err on the side of caution when it comes to their recommendation, stating that their goal is merely to establish for which candidates further research could be merited [193].

In Williams et al. (2006) [193], the primary distinction is made between ZrF_4-salts and BeF_2-salts. Generally speaking, ZrF_4-salts are reported to have a higher potential in terms of low costs and special properties such as low vapour pressure and a low melting point, while BeF_2-salts are considered to have superior nuclear properties such as being particularly disinclined to absorb neutrons (neutron transparent). Considering that the FUJI SMR utilizes FLiBe as its molten salt coolant, the focus will henceforth be on LiF-BeF_2.

An extensive overview of the characteristics regarding the use of molten salt coolant is given in Sohal et al. (2010) [194]. FLiBe is reported to be radiolytically and thermally stable at temperatures in excess of 1000°C, which is well above the 700°C core outlet temperature as shown in Appendix 10.5. The use of fluoride based salts in particular was thoroughly researched during the initial build up period as well as throughout the lifespan of the Molten Salt Reactor Experiment. It has been reported that there was no detectable degradation of the fuel salts as a result of either exposure to radiation or heat [193].

Dissociation of fluoride was reported at temperatures above 800 °C in the fission power density range between 80 and1000 W/cm^3 fuel as a result of the recoiling radiation and fission products [195]. However, the recombination rate of the dissociated species was reported to be reasonably swift as a result of the molten state of the fuel, which was argued to be the reason that no elemental F_2 would exist long enough to escape the molten salt. Furthermore, it is reported by Sohal et al. (2010) [194] that molten fluoride salts are not intrinsically corrosive for the alloys typically encountered in containment vessel alloys.

The mechanism underlying the lack of corrosive behaviour of the fluoride molten salt and the containment vessel alloys is related to the difference in Gibbs free energy between the chosen molten salt and the choice metal. Molten salt alloy combinations which have relatively similar Gibbs free energies are more prone to undergo corrosion reactions than combinations of latter for which this ‘Gibbs free energy gap’ is relatively large. Therefore, these combinations of materials are more suitable for forming alloys to be used in the containment vessel wall. As pointed out in Olsen (2009) [196] and shown graphically in , molybdenum seems to be a better options for a containment wall materials than for example Ni. However, there are other factors that need to be taken into consideration. Olsen (2009) [196] elaborates that material such as the Ni-Mo based Hastelloy B, which would supersede any other Ni-based alloy in terms of corrosion resistance is unusable in high temperature applications because it becomes extremely brittle in-between 650°C and 815°C. Therefore, when alkali metals such as Li, Na and K are present as cations then materials such as Ni are desirable for use in the containment wall due to the relatively large size of the Gibbs free energy gap.

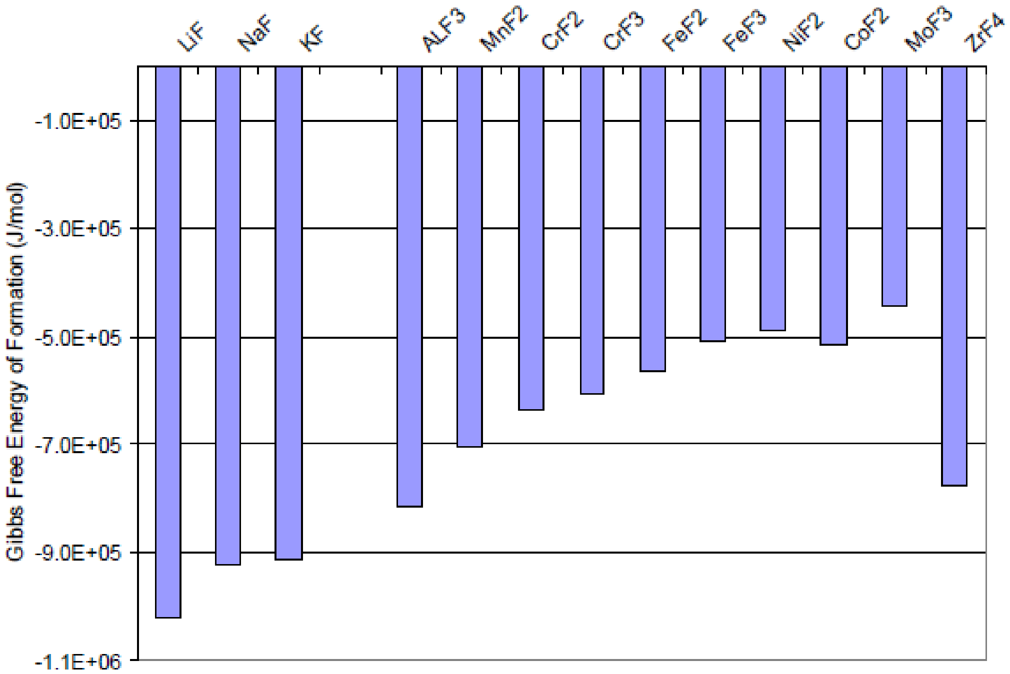


Figure 5.10| Gibbs free formation energy of various molten salts (left) and the various compounds they can form with the various containment vessel alloy metals

Taken from Olsen (2009) [196]

While the alkali and alkaline earth metals are considered to be weak oxidants and therefore stable, when external oxidants such as UF_4 are added, corrosion will increase in response to the increase in oxidation potential. However, when modern structural materials such as Hastelloy N are used, the addition of UF_4 to the FLiBe will still only result in a negligible corrosion rate of 0.002 mm/y [194]. Therefore it is concluded by Sohal et al. (2010) [194] that Hastelloy N is sufficiently corrosion resistant at temperatures up to 700°C. It is noted however that there is insufficient data to project what will happen at temperatures above 750°C and that in the event of core overheating some form of redox control might be necessary in the coolant system.

5.6.2.2.2 Thermal Inertia

In order to determine the thermal inertia of the FUJI SMR both the thermophysical properties of the primary coolant salt (FLiBe) and the graphite moderator/reflector need to be taken into consideration. Furthermore, also the FLiBe density corresponding with the in-core temperature is also required due to the molten salt inventory being given as a volume.

In Ignat'ev et al. (2006) [197] the density of a LiF-BeF_2 mixture was measured by means of the hydrostatic method, it was concluded that up to 670°C there is a linear relation between the density of the molten salt composition and the temperature. In the temperature interval 670 -700°C a small departure from linearity was measured, with the experimental noise indicating an apparent increase in density. From approximately 730°C onwards, the density is reported to decrease again. The explanation offered for this phenomena by Ignat'ev et al. (2006) [197] is that above 670°C gas bubbles start to form in the molten salt. Initially these gas bubbles predominantly form at the surface where they consequently increase the expulsion force and therewith the density of the molten salt mixture. At temperatures above 730°C, the gas formation in the bulk becomes definitive and subsequently decreasing the effective density. In light of this discovery, Ignat'ev et al. (2006) [197] concludes that in order to accurately reflect the effects of the gas formation in the bulk, the density expression would need to be separated. Therefore, the density of the LiF-BeF_2 molten salt mixture can be ascertained by means of the following expressions, which is a revised version of the formula given in Ignat'ev et al. (2006) [197]:

$$\begin{cases} \rho = 2518 - 0.406 \cdot T & T \leq 973\ K \\ \rho = 2763.7 - 0.687 \cdot T & T > 900\ K \end{cases}$$

Equation 5.4

In Equation 5.4, the density of the molten salt mixture ρ is a function of the temperature T in K, the measurement error is reported to be roughly 0.9 per cent and is ignored in above given expression.

A recent publication by the Idaho National Laboratory concludes that there are no simple correlations for the heat capacity of FliBe salt [194]. Although several empirical formulas, relating the heat capacity to the temperature, are known to exist these are reported to deviate roughly 10-20 per cent from the measured values. A possible explanation for these deviations, given by Williams et al. (2006) [193], is that the experimental values were determined with relatively crude calorimeters and that the more the refined measurement techniques are within 2 per cent of the values predicted by theory. In any case, it is also reported that the variation of heat capacity with temperature is small and is therefore often neglected in preliminary estimates. It is reported in Sohal et al. (2010) [194] that the heat capacity of FliBe equals 2415.78 J/kg·K at 973 K, which coincidentally is equal to the fuel salt temperature when it exits the core-region. This value will be assumed to be constant throughout the entire primary coolant loop. Additionally, in line with the discussing in paragraph 5.4.2.2.3, the presence of the graphite moderator is also considered when determining the mitigation of the heat build-up during a secondary loop LOCA.

5.6.3 Proliferation Resistance

The FUJI MSR was designed with proliferation resistance in mind, with partial contributions from the chosen fuel cycle, the liquid nuclear fuel and the innovative reactor design. The most prominent barriers to the spread of nuclear material, as reported in IAEA TECDOC-1536 [23], are discussed below.

The FUJI reactor utilizes a single phase fluid molten salt, in which both the blanket and the fissile material is absorbed as an ionic compound. The fissile contents is typically less than 2 per cent by weight, regardless of whether U-233 or plutonium is used this severely limits its attractiveness for use in nuclear weapons. Furthermore, explained in paragraph 5.6.1, there is a distinct separation between breeding and power generation in THORIM-NES. This allows the excess reactivity; the deviation of the k_{eff} from the critical state, to be small. Any unauthorized molten salt removal can therefore be detected. Furthermore, owing to the larger than unity conversion ratio the fissile material additions that need to be made to the molten salt fuel mixture are limited [189]. This limits the necessity of regular refuelling, which decreases the frequency of fuel transports. As discussed in paragraph 3.3, this reduces the potential for material diversion from an otherwise relatively attractive source.

The U-233 in the fuel salt mixture contains roughly 500 ppm of U-232 and its daughter nuclides. These isotopes are products of the decay chain of Th-232 and several of its daughter nuclides, especially Tl-208, are especially strong γ-ray emitters (2.6 MeV for Tl-208). The resulting radioactivity poses a fundamental barrier to the proliferation of nuclear material; in order to divert one SQ of U-233 (8 kg), one would require approximately 250 litres of fuel salt. It is reported that such a quantity radiates a dose of 1 Sv/hour at 50 cm distance. At such a dose rate 50 per cent of the potential proliferator would start vomit within 2 hours of exposure, including a deterioration of the physical attributes [149]. The safe handling of the material would require a protective lead slab of approximately 20 cm in thickness, making the molten salt extremely difficult to divert without remote handling technology.

$$^{232}_{90}Th(n,\gamma)^{233}_{90}Th \xrightarrow{\beta^-} {}^{233}_{91}Pa \xrightarrow{\beta^-} {}^{233}_{92}U$$

In theory it is possible for a potential proliferator to obtain pure U-233, this can be done by means of continuous removal of trace amounts of Pa-233. As displayed in the above given transmutation chain, protactinium is formed as intermediate step, its half time of 27 days is long enough to enable its separation from the molten salt, allowing pure U-233 generation, and short enough to allow for steady production. This indirect U-233 production mechanism suffers from the same issues as direct U-233 removal; a high radioactivity and a relatively short half-life. It has been reported that 2 months after shutdown as much as 75 per cent of the Pa-233 has decayed to U-233 which subsequently blends with the strong gamma emitting U-232. Furthermore, it has been noted that no less than 50 tons of fuel salt would be required to obtain 1 SQ of U-233. In addition, it should be noted that the presence of U-232, a blessing with respect to the proliferation resistance, is also typically regarded as a safety issue for the back end of the fuel cycle [198].

By virtue of the relatively low atomic number of its nuclear fuel, the Th-U233 fuel cycle does not produce any significant quantities of plutonium or minor actinides (such as Cm, Np and Am) in significant quantities. It has been reported that the average Pu and Am/Cm production in the FUJI reactor is 0.5 kg and 0.3 g per GWe·y respectively [187]. The corresponding figures for a typical LWR are 230 kg and 25 kg [187]. If need be, the fuel composition of the FUJI could even be altered to transmute these elements, as is intended for the FUJI-Pu. Furthermore, owing to its high conversion ratio only very limited amounts of fissile material have to be loaded; 2 kg of U-233 every 30 EFPD and 67 kg of Th every 150 EFPD [23].

The low inventory, following the limited loading of fuel material as well as the recycling of the U-233 for reuse in a new reactor and the possibility of Pu being incinerated in FUJI-Pu reactors, inspection and verification could be simplified. In addition, the strong gamma ray emission of the U-232 decay chain would facilitate fuel monitoring. Lastly, in case the reactor needs to be taken out of operation, the ability to store the molten fuel salt in a sealed containment tank can be considered an advantage as well.

5.6.4 Economics

The economics of the FUJI reactor, when taken to be a part of the THORIMS-NES philosophy, are described by Furukawa et al. (2008) [187] to possess several favourable characteristics in comparison to those encountered in conventional LWRs. Regarding the capital costs, the conclusion was indefinitive, stating that both the MSR and the LWR have their own advantages and disadvantages. For example, the FUJI has three fluid loops, which implies a higher material cost. However, it is also expected to have a higher thermal efficiency, a lower operating pressure and a simplified design in comparison to a conventional LWR. Furthermore, following the removal of all accident scenarios related to the use of solid nuclear fuel, the number of safety systems can be brought down allowing the size of the containment building to be reduced.

In Moir (2002) [199] an update was given on the cost breakdown of a 1000 MWe MSR, as described by the ORNL in 1980, which was subsequently compared to the expected capital costs of an equally sized PWRs and coal fired power plant. It concluded that the total capital cost required for a MSR and LWR were 1584 M US$$_{2002}$ and 1448 M US$$_{2002}$ respectively. Corrected for inflation, these values equal 1981 M US$$_{2011}$ and 1811 M US$$_{2011}$. As described in paragraph 5.1.4, in line with Du and Parsons (2009) [61], the expected specific overnight costs of a 1 GWe PWR are expected to equal 4339 US$$_{2011}$/kWe. Considering that the reference PWR and reference MSR equal rated power output levels, and assuming that the overnight cost ratio is still valid, the specific overnight cost for the reference MSR are taken to equal 4746 US$$_{2011}$/kWe.

Comparative estimates between the MSR and the LWR, such as the one given by Furukawa et al. (2008) [187], expect that MSR fuel-cycle costs will be lower and (variable) O&M costs will be roughly the same. The expected lower fuel-cycle costs are based on the lower average annual fuel loading following the higher conversion ratio of molten salt fuel. With regards to the O&M cost, it is estimated that the advantages and disadvantages roughly cancel each other out. On the one hand, the MSR is not expected to require any refuelling outages which, even with the help of advanced optimization software, still require LWRs to suspend their operation for roughly 1-3 weeks [200]. On the other hand, although its downtime is shorter, the MSR does require remote maintenance due to the radioactive molten salt circulation being partially located outside of the reactor vessel.

5.6.4.1 Molten Salt Reactor Fuel Cycle

As discussed above, MSRs use molten fuel salt, which makes it stand apart from all contemporary solid fuel reactors. Originally these reactors were conceived to be thermal, graphite moderated reactors with a high emphasis on breeding. However, in most recent publications, the preference has shifted towards scenarios in which the power generation and breeding functions are distinctively separated (THORIMS-NES [191]). Molten fluoride nuclear fuel salts, in the form of lithium fluoride (LiF) and beryllium fluoride (BeF_2), are considered the most suitable working fluid. This solution, better known as FLiBe, is a single phase fluid which is chemically stable and non-combustible at ambient pressure. Furthermore, molten fuel is not damaged by radiation like solid fuels and additionally has a trilateral functionality, working as a medium for the nuclear reaction, the heat transfer and the chemical reprocessing [191].

Owing to the great versatility of the molten fuel salt, there are multiple types of FUJI concept reactors are being developed. All of these designs have a rated power output of 100-200 MWe and most of them are geared towards the Th–U233 fuel cycle, which is generally considered to be more proliferation resistant than the U238-Pu fuel cycle. Furthermore, several sources report the possibility of utilizing the MSRs fuel flexibility to dispose of existing plutonium stocks [201]. For example, in an appropriate reactor environment plutonium could be transmuted into U-233; a fissile material considered to be less suitable for nuclear weapons. Furthermore, U-233 on average produces about 10% more neutrons per fission, making it better suited as a nuclear fuel.

The abovementioned fuel flexibility is the result of that the fissile material choice does not have a significant impact on the properties of the fuel salt. Therefore, any reactor running on an arbitrary fissile material combination would be capable of efficient power generation. Although this simplifies the fuel economics, considering that to some extent the fuel costs are independent of the fuel choice, the fuel burn-up is dependent on the fissile material and therefore the design.

The MSR is often considered to be the one of the last design to become commercially available, with deployment seldom reported to be earlier than 2030 [202]. It can be assumed that by the time that the FUJI-U3 becomes available reprocessing will have become an economically viable alternative to the once-through fuel cycle. Moreover, the FUJI-U3's molten fuel mixture and the possibility of a greater than unity core breeding ratio, imply that it was designed to operate in a closed fuel cycle. This design feature has several effects on the fuel cycle costs. For one, the fuel enrichment step does not have to be taken into consideration. This is because the fissile material in the initial load is recovered from the irradiated fuel of a secondary FUJI-U3 reactor. In Bunn et al. (2003) [120], it is assumed that the spent fuel is acquired free of charge, because the providing reactor stands to benefit from the spent fuel exchange by not having to incur disposal costs. This might give the FUJI an unfair competitive advantage and the spent fuel will most likely not possess the required isotopic composition. This is because the plutonium, fission products and TRUs will first need to be separated from the uranium.

For simplicity it will be assumed that at the deployment date of the FUJI-U3 there will be an active U-233 market, at which the U-233 required for fuel salt fabrication can be purchased at the market price of UREX reprocessing described in OECD (2006) [203] plus the costs of UF_6 conversion. The description of the MSR fuel cycle given in Engel et al. (1980) [188], clarifies that when computing the front end fuel cycle costs one needs to take into consideration both the initial molten salt load and periodical fissile feeding. Given that both factors combined will result in a mathematical expression of unpractical length, the expression for the front-end fuel cycle costs of the FUJI-U3 c_f^{MSR},will be subdivided into three sections: the initial fuel loading c_{ini}^{MSR}, the fissile U-233 feed c_{U-233}^{MSR} and the fertile Th-232 feed c_{Th-232}^{MSR}. The expression proposed in this thesis for accounting for the molten salt fuel cycle is the following:

$$c_f^{MSR} = c_{ini}^{MSR} + c_{U-233}^{MSR} + c_{Th-232}^{MSR}$$

Equation 5.5

in which

$$c_{ini}^{MSR} = \frac{x_{\mathrm{UF4}}\left(\frac{C_{UF4}}{(1+r_d)^{-t_{hmc}}}\right) + x_{\mathrm{ThF4}}\left(\frac{R_{Th/U}\cdot C_{MMUC}}{((1+r_d)^{-t_{hmc}})}\right) + \frac{C_{fc}}{(1+r_d)^{-t_{uc}}}}{24\cdot 10^3 B\eta} + \frac{x_{\mathrm{LiF}}\left(\frac{C_{\mathrm{Li}}}{(1+r_d)^{-t_{lmc}}}\right) + x_{\mathrm{BeF2}}\left(\frac{C_{Be}}{(1+r_d)^{-t_{lmc}}}\right)}{24\cdot 10^3 B\eta}$$

Equation 5.6

and,

$$c_{U-233}^{MSR} = \frac{\left[x_{\mathrm{UF4,u}}\left(\frac{C_{UF4}}{(1+r_d)^{-t_{hmc}}}\right) + x_{\mathrm{LiF,u}}\left(\frac{C_{\mathrm{Li}}}{(1+r_d)^{-t_{lmc}}}\right) + \frac{C_{fc}}{(1+r_d)^{-t_{fc}}}\right] M_u F_{cu}}{8760\cdot 10^3 B\eta}$$

Equation 5.7[22]

and,

$$c_{Th-232}^{MSR} = \frac{\left[x_{\mathrm{ThF4,th}}\left(R_{Th/U}\frac{C_{MMUC}}{((1+r_d)^{-t_s})}\right) + x_{\mathrm{LiF,th}}\left(\frac{C_{\mathrm{Li}}}{(1+r_d)^{-t_{lmc}}}\right) + x_{\mathrm{BeF2,th}}\left(\frac{C_{Be}}{(1+r_d)^{-t_{lmc}}}\right) + \frac{C_{fc}}{(1+r_d)^{-t_{fc}}}\right] M_{th} F_{cth}}{8760\cdot 10^3 B\eta}$$

Equations 5.8

In Equation 5.6 through Equations 5.8, C_{UF4} is the cost of U-233 fluoride, which was taken to equal the cost of UREX reprocessing as given in OECD (2006) [203] in US$$_{2011}$/kgHM plus the cost of uranium conversion as described in paragraph 4.3.3.6.1. Moreover, $R_{Th/U}$ is the assumed ratio between the spot market prices of Thorium and U-235. This variable is necessary because as of this writing there is no active demand for thorium metal and a proxy needs to be used. Its value is taken to equal the ratio between the cost of uranium and thorium as given in Engel et al. (1980) [188]. Furthermore, C_{MMUC}, C_{Li} and C_{Be} represent the U-235 spot market price plus conversion costs and the spot market prices of lithium and beryllium respectively, all given in US$$_{2011}$/kgHM. C_{uc} and C_{thc} equal the U-233 and thorium conversion cost in U.S. dollars per kilogram of fuel salt (US$$_{2011}$/kgFS) and the fractions $x_{\mathrm{LiF,u}}$, $x_{\mathrm{LiF,th}}$, $x_{\mathrm{BeF2,th}}$, $x_{\mathrm{ThF4,th}}$ and $x_{\mathrm{UF4,u}}$ are the mass percentages of the lithium fluoride component in the uranium and thorium fissile feedings, followed by the fractions of beryllium fluoride, thorium fluoride and uranium fluoride in their corresponding fuel mixture. Furthermore, the fractions x_{UF4}, x_{ThF4}, x_{LiF} and x_{BeF2} signify the fuel salt composition of the initial fuel salt load. All fractions are given in kilograms of fuel salt per kilogram of heavy metal (kgFS/kgHM). Based on the molar percentages given in various publications (e.g. [23]), the initial fuel salt composition expressed in mass percentages can be calculated to be as given in Table 5.4.

Initial Fuel Salt Load		Composition (mass%)
Lithium Fluoride	LiF	29,2%
Beryllium Fluoride	BeF2	11,8%
Thorium Fluoride	ThF4	57,8%
Uranium Fluoride	UF4	1,2%

Table 5.4

[22] Equation 5.7 and Equations 5.8 are variants of the expression for liquid-metal breeder reactors used in Bunn et al. (2003) [120].

Moreover, M_U and M_{Th} represent the annual loading of of U-233 and thorium in kilograms of fuel salt per MWe year (kgFS/MWe·y), F_c is the carrying time of the reactor fuel, as described in paragraph 4.3.3.6.1 and given in Equation 4.24. Again, B and η stand for the fuel burn-up and the thermal efficiency respectively, r_d is the discount rate and, t_{hmc}, t_{lmc} t_{uc} and t_{thc} equal the lead times for heavy metal reprocessing and conversion, light metal processing and conversion, U-233 fuel salt fabrication and thorium fuel salt fabrication.

U-233 Fuel Salt		**Composition (mol%)**	**Composition (mass%)**	**Mass Additions**	
Lithium Fluoride	LiF	73,00%	18,5%	0,60	kg
Uranium Fluoride	UF3	27,00%	81,5%	2,65	kg
Total				3,3	kg

Table 5.5

Thorium Fuel Salt		**Composition (mol%)**	**Composition (mass%)**	**Mass Additions**	
Beryllium Fluoride	BeF 2	72,00%	39,2%	964,75	kg
Thorium Fluoride	ThF 4	16,00%	57,1%	1404,93	kg
Lithium Fluoride	LiF	12,00%	3,6%	88,95	kg
Total				2459	kg

Table 5.6

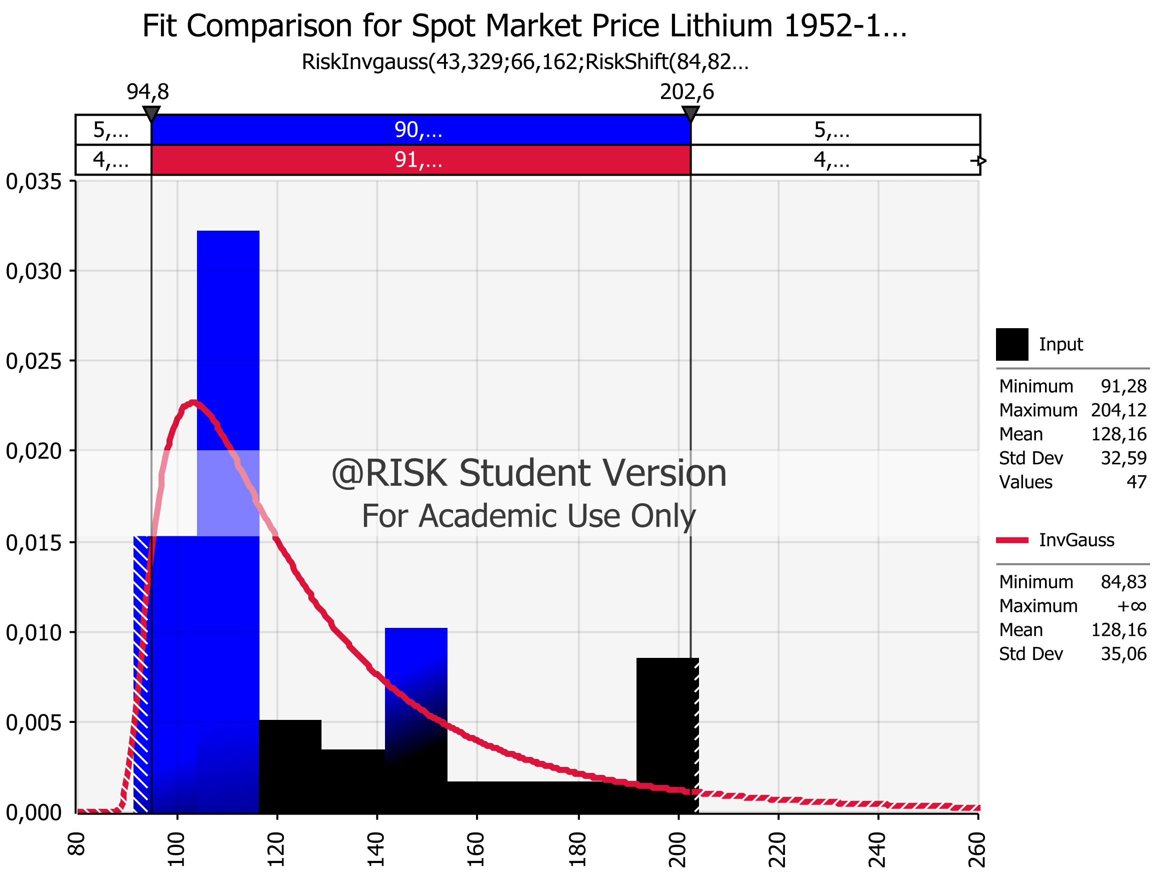
The U-233 and thorium fuel salt conversion costs C_{uc} and C_{thc} are taken to be equal to those in Engel et al. (1980) [188], corrected for inflation, these are 123 US$$_{2011}$/kgFS. In line with Choi (2011) [178], C_{UF4} is taken to equal 240 US$$_{2011}$/kgHM[23] and has a lead time of 60 months, the uranium conversion costs are taken to be equal to those in the U-235 cycle described in paragraph 4.3.3.6.1. The construction lead times for molten fuel salt are assumed not to differ substantially from the lead times of fuel rod construction (apart from the UREX Reprocessing). Therefore, $t_{uc} = t_{thc} = 0.5$, $t_{lmc} = 1.0$ and $t_{hmc} = 5.0$.

In IAEA-TECDOC-1536 [23] it is reported that 2 kg of U-233 is supplied to the core in the form LiF-UF_4 (73-27 mol %) every 30 EFPDs and 67 kg of thorium is added to the fuel salt mixture in the form LiF-BeF_2-ThF_4 ((72-16-12 mol %) every 150 EFPDs. It is computed that this requires an addition of 3.1 kg of U-233 fuel salt (Table 5.5) and 2459 kg of Th-232 fuel salt (Table 5.6

When taking into consideration that for the FUJI-U3 an effective full power day equals 200 MWd, it can be calculated that the annual loading of U-233 fuel salt M_U and thorium fuel salt M_{Th} are equal to 0.19 kg/MWe·y and 29.91 kg/MWe·y respectively.

Data on the spot prices for lithium and beryllium metal was gathered from the U.S. Department of the Interior [204]. Although contemporary spot market data on these metals is unavailable, there is little reason to assume that the spot price distribution for beryllium has changed significantly since 1998. The spot price distribution for lithium could be misrepresented as result of the emergence of commercial lithium-ion battery technology in the previous decade. Although, it could be argued that the increased demand has resulted in an increase in known resources as well, levelling out the upward price tendency. Both spot price distributions are shown in Figure 5.11.

[23] This equals the market price of UREX reprocessing plus the market price of uranium conversion



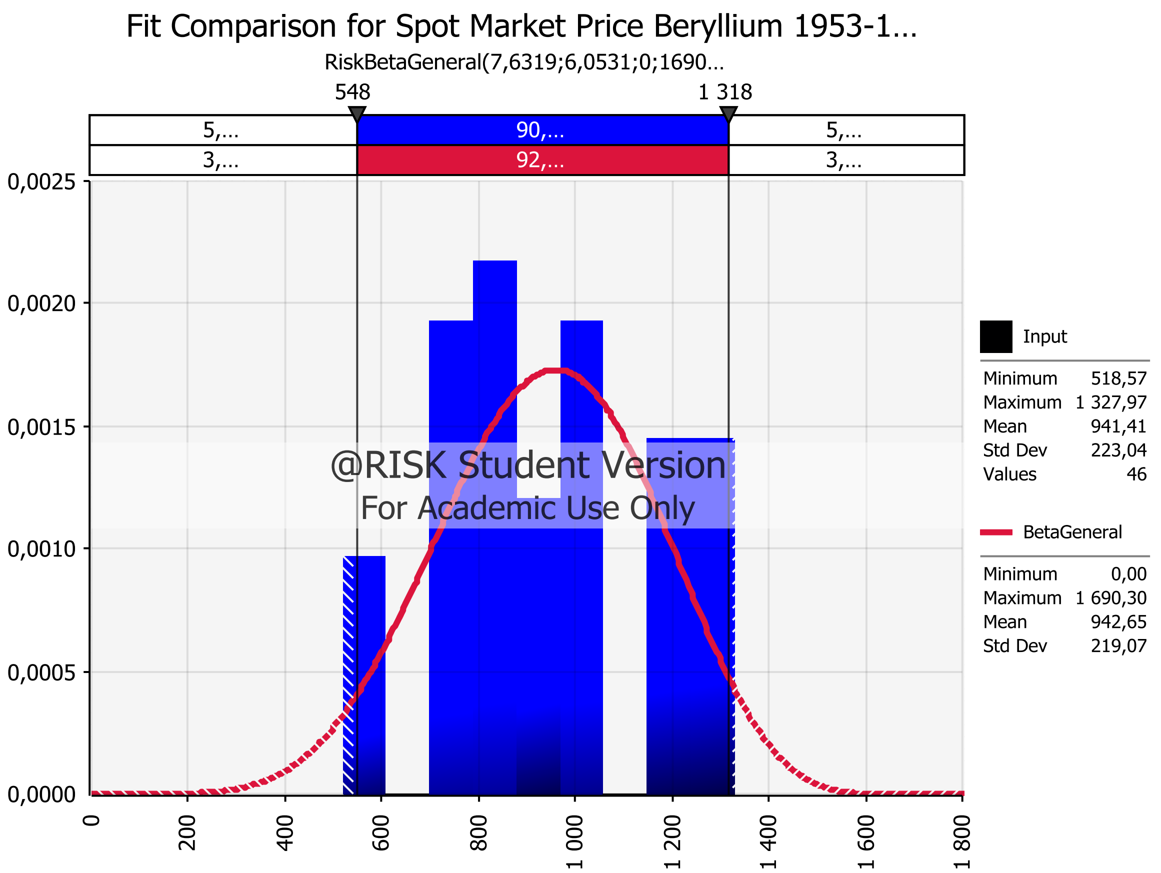


Figure 5.11| Probability distributions for the spot market prices of Lithium and Beryllium given in U.S. dollars per kilogram of metal

5.7 Accelerator Driven System

5.7.1 MYRRHA

MYRRHA, which is an acronym for Multipurpose hYbrid Research Reactor for High-tech Applications, is an experimental Accelerator Driven System (ADS) designed to demonstrate the feasibility of transmutation in hybrid reactors. As described in Aït Abderrahim et al. (2010a) [205], it consists of a proton accelerator with a beam energy of 600 MeV and an intensity of 2.5 mA, which in turn is coupled to a liquid LBE target. As shown in Figure 5.12, the LBE target is located in the centre of the subcritical reactor core region, which is cooled by a separate LBE circuit. When an accelerated proton hits the target the reaction induces a spallation reaction, which deforms the nucleus and releases a number of up to 30 neutrons. Subsequently, these neutrons induce fission events in the subcritical core region, which is comprised of MOX fuel assemblies, with a plutonium content of roughly 30 to 35 per cent [206]. The amplified neutron flux resulting from this series of interactions subsequently transmutes the minor actinides loaded in the 8 in-pile sections (IPSs) shown in Figure 5.12. Although, as will be discussed in paragraph 5.7.4, the MYRRHA has multiple loading scenarios, in this thesis it will be assumed that the MYRRHA will be exclusively used for transmutation, which is can be defined in this context as the transformation of long lived minor actinides into shorter lived fission products.

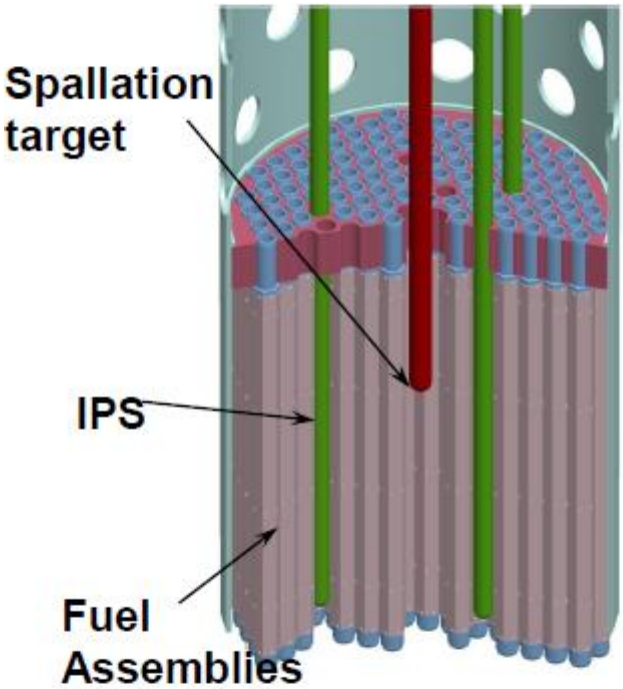


Figure 5.12| Schematic overview of the structures present inside the MYRRHA's core region

Taken from Popescu (2012) [207]

Transmutation can be induced by bombarding nuclei directly with protons, however this direct interaction is reported to be highly uneconomic [208]. The reason for this is that charged particles, such as protons, have to overcome the Coulomb barrier; a repulsive force which resists the joining of same-sign electric charges. Therefore, hybrid reactor systems, such as MYRRHA, use indirect transmutation in which a spallation target acts as a neutron-producing intermediary. The progeny of these first generation neutrons interacts with the fuel assemblies in the multiplier medium, generating further fission neutrons usable for transmutation purposes.

As a result of their lack of electric charge, neutrons are considered to be a more suitable projectile for prompting transmutation than protons, considering that as a result they do not have to be accelerated to as high energies as the protons. The process, by means of which these neutrons are generated, spallation, is reported to be a highly energy efficient method [88]. Firstly, for direct transmutation, the proton energy requirement is reported to be optimal in the 600-800 MeV region [88], which is in line with the 600 MeV beam energy of the MYRRHA, while direct proton-driven transmutation is reported to require energies in the 1-2 GeV range [208].

The abovementioned preference for indirect, neutron-based, transmutation implies that the reactor core that spawned the nuclear waste in the first place would also be a suitable environment for transmutation, were it not for the poor neutron economy of conventional light water reactors. Minor actinide burners can be set up in a dedicated capacity by utilizing the intense neutron fluxes that can be generated by means of spallation; a nuclear reaction that occurs when a relativistic light particle hits a heavy nucleus. As described in Mongelli et al. (2005) [209], spallation can be roughly divided into two distinct steps. The initial phase, also known as the intra-nuclear cascade, is when incoming nucleons undergo a series of scattering interactions with the nucleons in the target. This initial set of nucleon-nucleon scattering interaction results in the ejection of nucleons, which can set in motion additional intra-nuclear cascades, and leaves the recoiling nucleus in an excited state. De-excitation of this nucleus occurs either through the emission of light nuclei or through fission.

ADSs could be outfitted with the same power generation equipment as conventional light water reactors. However, operating in designated power generating mode could come at a significant penalty with regard to energy conversion performance because of the parasitic losses that need to be incurred to power the accelerator. It is reported in Aït Abderrahim et al. (2010b) [210] that the electrical efficiency of an ADS, with a multiplication factor of 0.95[24], will be roughly 12 per cent lower. This means that for a similar power output, an ADS will produce roughly 14 per cent more high level waste.

Additionally, one of the fundamental design challenges is the MYRRHA's spallation target, which can be set up in either a sealed or an open accelerator beam tube configuration. It is reported that all currently operating spallation target facilities utilize either one or multiple seals, which are referred to as 'windows', to separate the beam vacuum from the target surroundings [210]. However, on the major shortcomings of this design is that high heating and thermal stress tend to limit the current density of the accelerator. An additional consequence is that as a result of being the most structurally loaded the beam tube window needs to be easily replaceable in the event of a rupture [206]. Although several methods have been identified for achieving nearly flat proton profiles, significantly higher current densities could be achieved if window could be removed altogether. A windowless team tube can be defined as a system in which the spallation target and the beam line share a common vacuum [206].

The primary difficulty associated with the design of the windowless system is that a satisfactory vacuum level needs to be maintained in order to prevent plasma formation. In other words, a sufficiently high pumping capacity needs to be maintained, while simultaneously, the outgassing occurring in the LBE, containing volatile spallation products, needs to be kept in check. Although the design of the spallation target has been the topic of much debate, a recent paper by Aït Abderrahim et al. (2012) [211] concludes that the latest design, the MYRRHA/FASTEF utilizes the windowed spallation target configuration, as implied above one of the foremost considerations were design simplifications.

An article on the overall technological readiness of Accelerator Driven Systems, to which the SCK·CEN contributed, was published in 2010 [210]. The summarized findings were that the basic technology required to build a demonstration facility was already available, with only some development needed to increase the overall system reliability. For implementation on an industrial scale, it was reported that although many of the key technologies had been demonstrated, additional work was needed. Among other things the beam

[24] Accelerator Driven Systems such as MYRRHA can be designed operate in subcritical mode. This means that their multiplication factor; the number of fission neutrons in one generation divided by the number of fission neutrons in the preceding generation, is lower than one. The implications of this are discussed in paragraph 5.7.2.

quality, halo control and the reliability of sub-systems still required some improvement. Furthermore, it was deemed of the utmost importance that before any serious thought was given to power production, lessons were learned from operational experience of previous generations of ADS facilities, primarily because of the abovementioned need for improved components, sub-systems and overall system level reliability.

5.7.2 Safety Features

It is reported that the MYRRHA/FASTEF, in which FASTEF stands for Fast Spectrum Transmutation Experimental Facility, is designed to be capable of both sub-critical and critical operation [211]. In contrast to previous configurations, this would necessitate the addition of a reactivity control and scram systems.

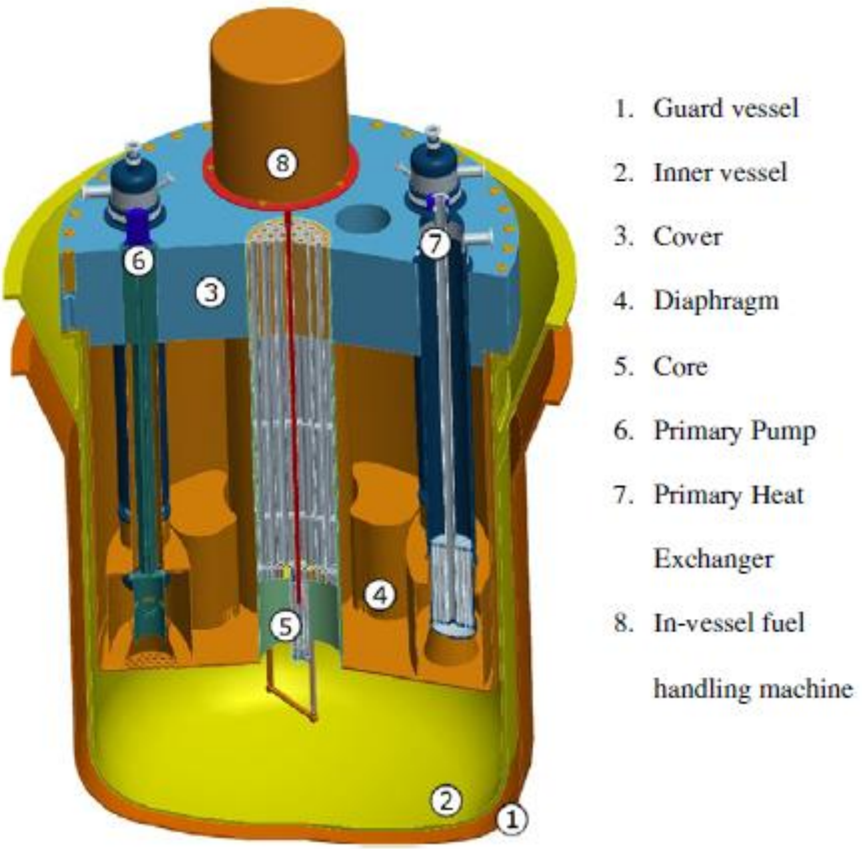


Figure 5.13| Schematic overview of the MYRRHA's reactor pressure vessel

Taken from Aït Abderrahim (2012) [211]

As can be derived from Figure 5.13, the MYRRHA/FASTEF is an integral pool type reactor, owing to the fact that all primary systems are housed within the reactor pressure vessel. To prevent release of radioactive material, the reactor vessel is envisioned to be comprised of an inner and an outer vessel, with the inner vessel containing the LBE and the outer vessel acting as a secondary containment layer in the event of a breach in the inner containment wall. The reactor cover supports all the in-vessel system components including two primary heat exchangers and two primary pumps, which are responsible for circulating the LBE. Each heat exchanger uses pressurized light water as its secondary coolant and, although lead does react with light water under normal conditions the heat exchanger is double walled to prevent the pressurized water from overflowing into the LBE pool, which could result in a breach of containment. Both heat exchangers combined are reportedly capable of removing 110 per cent of the rated core heat output, so that the removal of the heat produced by the additional in-vessel sources is also accounted for [211]. Both the LBE and the pressurized water flows are driven by forced circulation.

In the event MYRRHA experiences a LOCA while in a subcritical operation, the reactor can bring to bear its foremost safety system; the ability to facilitate a shutdown by discontinuing accelerator beam operation. For operating in critical mode shutdown control rods are added. To ensure that the reactor is capable of removing the decay heat in the event of a primary coolant pump failure, a dedicated system capable of removing 7 per cent of the rated core power by means of a passive operating mechanism is envisaged [211]. Additionally, the same source foresees a Reactor Vault Air Cooling System (RVACS) capable of fully cooling the reactor through natural convection, which is capable of full decay heat removal in the event of a loss of secondary coolant accident. The RVACS is intended to consist of a series of pipes placed in the proximity of the outer reactor vessel wall. All these individual pipes are connected to a header which directs their airflow towards a chimney. For completeness, the heat transport path would be that a rise in reactor temperature leads to a rise in thermal radiation emission, which is radiated through the walls and induces convection in the chimney.

The above-mentioned safety systems cover the focal points of the ADSs safety policy as detailed in Aït Abderrahim et al. (2010b) [210]. The key issues that are reported to be in need of addressing are: (1) provision of adequate cooling of the target, (2) maintaining the structural integrity of the target system, (3) containing the radioactive inventory, and (4) accommodating accelerator induced transients. Of these four points only the last one is not addressed in the most recent MYRRHA configuration report [211]. This is in line with the expectations however, considering that one of the areas, identified in Aït Abderrahim et al. (2010b) [210] to be in need of further research and development, is the overall system reliability. This includes designing the target in such a manner that it is able to handle accelerator trips, start-up-transients and other potential malfunctions.

5.7.2.1 Coolant Characteristics

The MYRRHA reactor is cooled by the Lead-Bismuth Eutectic, as described in paragraph 5.3.2.2.

5.7.3 Proliferation Resistance

Kemp (2005) [212] concludes that particle accelerators offer a feasible route towards the acquisition of material that could be used for the constructing of nuclear weapons. It is stated that the more advanced ADSs, such as the linear proton accelerator employed in MYRRHA, is an economically attractive alternative although it is also noted that nuclear proliferation using ADSs is most likely more cumbersome than using conventional nuclear reactors. In addition it was indicated that there exist possibilities for acquiring the necessary components from foreign sources; the relative novelty of ADS technology being the reason that in several countries no regulatory body is notified when ADS system components are acquired [212]. It seems this situation has not been rectified yet, as it is stated in INFCIRC/254/Part 2 which contains Guidelines for Transfer of Nuclear-Related Dual-Use Equipment, Materials, Software and Related Technology mentions that the document *'does not control accelerators that are component parts of devices designed for purposes other than electron beam or X-ray radiation'* [213]. It is stated that one can operate an accelerator design of reduced complexity, for example a non-multiplying configuration, in combination with the ambition to design the relatively simple 'gun-type' nuclear device using U-233 acquired from irradiating thorium. This would allow for the circumvention of many proliferation barriers by exploiting particle accelerator technology, even by less technologically advanced proliferators [212].

On the other hand, it is reported that spent ADS fuel reaches the same radio-toxicity as coal ash after only 500 years of storage [214]. Furthermore, an international team of scientist and engineers, referred to as the MYRRHA International Review Team (MIRT) [215], has recognised that the MYRRHA design team has given serious though to the non-proliferation issues in accordance with guidelines set out by the Pacific Northwest National Laboratory commissioned by the U.S. Department of Energy [216]. In comparison to the current research reactor operated by the SCK·CEN, the BR2, the MYRRHA is assessed to perform roughly equal in terms of proliferation resistance. The MYRRHA outperforms the BR2 in terms of its in-core content of plutonium-oxide fuel, which is less. However, the BR2 has a lower material throughput in comparison to the MYRRHA, which equals a reduction in opportunities for theft or diversion [215]. A non-proliferation assessment performed by the SCK·CEN itself, also compared the MYRRHA to the BR2, which was argued to be the most logical reference plant due to the overlap in its intended user profile [217]. In line with the MIRT, it concludes that MYRRHA is more proliferation resistant to theft by sub-national groups owing to its use of reactor-grade plutonium in comparison to the HEU used in the BR2. However, in comparison to the BR2, MYRRHA also provides relatively ample opportunity for the diversion of nuclear fuel for a clandestine weapons programme as a result of its higher nuclear material throughput. Therefore, the report notes, that in order to acquire a final assessment of the proliferation risks of the MYRRHA, in comparison to designs such as the BR2, the political situation of the host state should also be taken into consideration. Recommendations for further developments are that, in correspondence with this analysis, the focus should be on devising additional barriers to material and facility access.

5.7.4 Economics

An important distinction between the MYRRHA and the other reactor designs being investigated in this thesis is that the MYRRHA is not primarily intended for the generation of electricity. The current application catalogue as encountered on the MYRRHA website contains a variety of purposes, such as: (1) demonstrating the ADS technology for transmutation of long-lived radioactive waste, (2) aiding in the development of fast spectrum reactors, (3) production of neutron irradiated silicon and (4) radioisotope production for nuclear medicine [206]. Considering that none of these applications have the output unit kWh(e), the relative costs of the MYRRHA will be expressed in kWh(th). An estimate of the construction costs of the MYRRHA was released by the SCK CEN in 2009, totalling 960 M€$_{2009}$ [206], which includes a contingency of 192 M€$_{2009}$.

The cost breakdown of this figure in terms of labour, equipment and material is given in Popescu (2012) [207]. Considering that MYRRHA has a deviating primary purpose, these figures were used in establishing the ultimate capital cost reductions following from the economies of learning as described in paragraph 4.3.3.3.2. The percentage of indirect capital costs was assumed to be the same as for the SMR NPPs. For continuity purposes the price in €$_{2009}$ was converted to US\$$_{2009}$ using the average exchange rate over 2009. This figure was given by the IRS to be equal to 0.748 €/US\$ [218]. Subsequently, this figure was converted into US\$$_{2011}$ by means the CPI inflation calculator of the U.S. Bureau of Labor Statistics, resulting in a specific overnight cost point estimate equal to 13'456 US\$$_{2011}$/kWe. Although no such plans are currently known to exist, it will be assumed for comparative purposes that the MYRRHA will utilize a multi-module-set with 6 modules in total.

5.7.4.1 Accelerator Driven Systems using MOX

Mixed Oxide (MOX) fuel is a mixture of uranium oxides and plutonium (possibly mixed with other transuranics). MOX fuel is typically encountered with a plutonium heavy metal content of less than 12 per cent [219], with the rest of the core content being comprised of depleted or natural uranium. The current generation of MOX fuelled reactors is only partially filled with MOX fuel rods, the rest of the fuel being conventional uranium oxide rods in order to meet the safety margins. The advantage of employing systems with larger percentages of traditional uranium oxide rods (>90 per cent) is that they are easier to license, because the fuel composition more closely resembles the low-enriched fuel encountered in conventional LWR designs. The disadvantage of a having higher uranium content is that such a system also breeds plutonium and minor actinides, which limits the actual plutonium depletion. It has been reported that higher percentages of MOX fuel rods (~30 per cent) would still allow a reactor to operate within the safety margins, while simultaneously destroying some plutonium [219].

The MYRRHA reactor is envisioned to operate on a full core of MOX, which is even more advantageous, viewed from a waste disposal perspective. As a result of the MYRRHA fuel cycle being fundamentally different from the one utilized in SMRs which use more conventional designs, the fuel cost of operating an ADS was redefined following the example of Bunn et al. (2003) [120].

$$C_{FFC}^{MOX} = \frac{x_u}{(1-f_m)}\frac{C_{DU}}{(1+r_d)^{-t_{du}}} + \frac{C_{mf}}{(1+r_d)^{-t_f}}$$

Equation 5.9

In Equation 5.9, x_u is the percentage of depleted uranium in fresh MOX fuel, f_m is the fraction of depleted uranium lost during the chemical conversion and fabrication process. Furthermore, the cost of depleted uranium is assumed to be zero, in line with Bunn et al. (2003) [120] and MIT [220], except for the chemical conversion which will be taken to equal the chemical conversion costs for freshly mined uranium, 8 US$$_{2011}$/kgHM, as encountered in Equation 4.25 and the UREX Reprocessing required to separate the uranium from the other spent fuel materials as described in paragraph 5.6.4.1. Furthermore C_{mf} equals the cost of blending and manufacturing MOX. t_{du} and t_f are the lead times of the chemical conversion and the fabrication process respectively. In the MYRHHA reference case, $x_u = 0.6177$, as reported by Malambu et al. (2011) [221], and $f_m = 0.005$. The cost of fuel fabrication C_{mf} will be taken to equal the cost of MOX-EU as given in OECD (2006) [203], the values of which corresponds with the assumption made in Rothwell (2011) [100]. C_{mf} is taken to have a triangular distribution with 1100, 1375 and 1650 US$/kgHM representing the lower bound, nominal value and upper bound respectively. Furthermore, $t_{du} = t_{hmc} = 5.0$, as described in paragraph 5.6.4.1, and $t_f = 0.5$ in line with the description given in paragraph 4.3.3.6.1.

6 Results and Discussion

6.1 Passive Safety Model

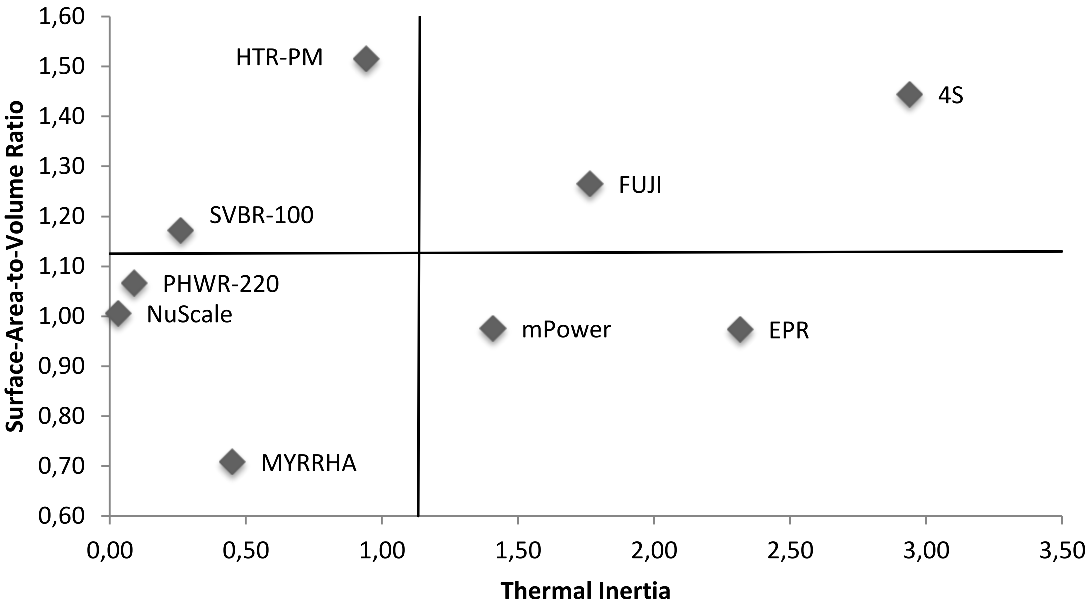


Figure 6.1| Comparisson of the Thermal Interia set out against the Surface-Area-to-Volume Ratio per reactor design

Figure 6.1| Comparisson of the Thermal Interia set out against the Surface-Area-to-Volume Ratio per reactor designFigure 6.1 gives an overview of the performance of the evaluated reactor designs regarding their thermal inertia and SA:V. The horizontal and vertical lines represent the average values per category of the selected reactor designs and should only be taken as relative performance indicators. In terms of passive safety the safest reactors are those with a high SA:V and a low thermal inertia score. Therefore, of the selected reactors, two reactors were identified to perform above average in both areas: the HTR-PM and the SVBR-100

The composition of thermal inertia per reactor design as described in paragraph 4.3.1 can be found in Appendix 10.6.2. The final value represent thermal inertia after taking into account the presence or absence of PHRS and/or graphite in-core structures, which possess a significant heat capacity. This segregation was made to enable the identification and extent of the contributions made by the different relevant reactor components. The EPR, which was determined to possess a thermal inertia of 2.3 K/s, can be considered to be the reference value for a state-of-the-art LR.

Information on the volume of coolant in the primary loop was typically readily available from literature, with the exception of the NuScale. Therefore, the NuScale's thermal inertia should be interpreted as a conservative estimate, considering that it is based entirely upon the thermal capacity of the Passive Heat Removal System (PHRS); the large pool of light water in which the reactor is immersed. The other input variables for the NuScale and mPower light water reactors were acquired from corporate presentations [222] [223]. The PHWR-220 inputs were acquired from peer-reviewed literature [179] and the mass of the light water in the calandria vault was taken from an Indian National Magazine [224]. The helium mass content of the primary circuit of the HTR-PM was deduced from peer-reviewed literature as discussed in paragraph 5.4.2.2.3. Furthermore, in a recent technical meeting at the IAEA Headquarters in Vienna, Toshiba revealed that a full scale EM pump would circulate approximately 8 tons of liquid sodium [147]. Although not stated explicitly in any source, it is assumed in this thesis that the 'full scale design' referred to by Toshiba, is the initially planned 30 MWth (10 MWe) design and that the sodium inventory is proportional to the thermal power output, which would result in a sodium inventory of roughly 36 tons for the 135 MWth (50 MWe) variant. Additionally, the input data for the SVBR-100 was taken in its entirety from IAEA-TECDOC-1536 [23], including the volume of the PHRS tank. The data for the FUJI was acquired from the same source, with the addition of the mass of the in-core graphite moderator taken from the Molten Salt Energy Technology Forum [225]. Finally, the required variables for the MYRRHA reactor were found on the MYRRHA home page section of SCK·CEN website [206].

It should be noted that heat mitigation using a light water heat sink, maintained at atmospheric temperature and pressure, only has a limited lifespan due to evaporation of the reservoir and that it is designed with preventing core damage over a limited period of time (in-between 5 and 30 days) only. However, it is claimed by some designers that after the water in immersion pool has evaporated the reactors power output has been sufficiently decreased to be cooled indefinitely by air [222]. Furthermore, full passive air cooling is the fall-back mechanism of choice for the 4S and MYRRHA reactors, which makes it particularly hard to make a sensible statement regarding their thermal inertia. After all, the environment can be considered to be an infinite heat sink, which would make the thermal inertia infinitely large. Therefore, in order to maintain comparability, the thermal inertia will be by means of parameters established in paragraph 4.3.1 and the inability to incorporate the atmosphere will be considered a shortcoming of the model.

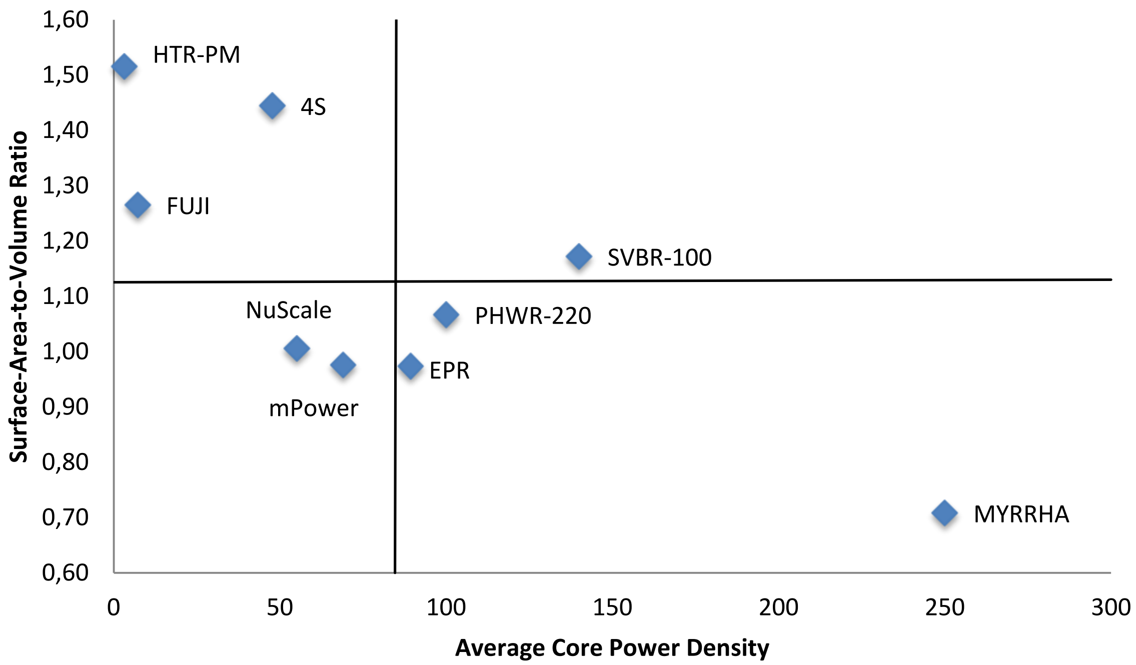


Figure 6.2| Comparisson of the Average Core Power Density set out against the Surface-Area-to-Volume Ratio per reactor design

A somewhat different perspective is acquired when comparing the SA:V with the average core power density, as shown in **Fout! Verwijzingsbron niet gevonden.**Figure 6.2 In terms of passive safety the best performing reactors are those with a relatively low core power density in addition to a relatively low SA:V. The foremost difference between Figure 6.1 and Figure 6.2 is that the effects of a large (high heat capacity) coolant inventory are negated. As a result the SVBR-100 loses the advance of its large volume liquid metal pool and conversely the relatively small coolant inventories of the FUJI and 4S are excused. Furthermore, the dominant position of the HTR-PM is solidified and MYRRHA replaces the EPR as the reactor design with the lowest overall passive safety under the specified parameters. With regard to the MYRRHA design, the low SA:V power density ratio is a surprising outcome, considering that the designers contemplated the use of a fully passive convective air chimney to cool the surface of the RPV.

The differences in the above given figures should be interpreted as the difference between the passive safety during a Loss-Of-Secondary-Coolant scenario and a Loss-Of-Primary-Coolant Scenario. On the one hand, in the event that secondary cooling is lost, primary coolant might still be available as a heat sink, which merits favouring the thermal inertia over the average core power density. One the other hand, when a loss of primary coolant scenario occurs, the average core power density can be considered the parameter of choice. It should be noted here that the thermal inertia of the second coolant loop has little relevance due to a lack of significant heat transfer as a result of the unavailability of the primary coolant medium.

Some preliminary conclusions that can be drawn from Figure 6.1 and Figure 6.2 are that significant improvements in thermal inertia can be realized by using additional heat removal systems and structures which benefit from an increased SA:V. The foremost example of this is the HTR-PM, which utilizes a passive safety system designed around the high-heat capacity of the graphite containing TRISO fuel pebbles and a large SA:V reactor vessel. The gaseous helium on the other hand has a relatively low density, which makes it unsuitable for the uptake of large amounts of heat, resulting in a primary coolant loop thermal inertia nearly 7 times lower than the reference EPR.

With regards to the thermal inertia, the NuScale is the best performing reactor design, a feat which can be explained by the presence of pool, in which the 12 reactor modules are submerged, with a light water inventory of roughly 15 million litres of water. In this regard a pattern is ascertainable when looking at the best performing reactors in this category. Next to the NuScale, also the PHWR-220 and the SVBR-100 display high levels of thermal inertia due to the presence of PHRS and in-core graphite. In addition to making the reactors safer, a design with a PHRS or in-core graphite could potentially also be cheaper. This is due to a corresponding reduction in size of, or complete removal of certain safety-level components. These could otherwise have been required to achieve similar overall safety levels. Although some additional costs need to be incurred for the construction of the PHRS, the structures are not typically located within the reactor vessel, which allows for the use of cheaper non-safety-level components.

In accordance with Turkenburg (2003) [25] it should be noted that a full evaluation of the passive safety features should, in addition to the parameters given above, also take into consideration: (1) the energy production per amount of fissile material (J/kg); (2) the maximum allowable increase of reactivity ($/sec); (3) various coefficients such as: the reactivity coefficient, the power coefficient and the void coefficient; and (4) the chemical and physical properties of the materials used in the reactor. That being said, a preliminary conclusion that can be drawn is that based upon the SA:V, the power density and the thermal inertia it can be argued that all SMRs could outperform the EPR in terms of passive safety, with the possible exception of the 4S and the MYRRHA. However, when taking into consideration that the 4S was only found to be lacking in terms of thermal inertia due to an inability to attribute meaningful values to an infinite environmental heat sink, this preliminary conclusion can be drawn into question.

6.2 Proliferation Resistance

Proliferation Resistance Model								
Reactor	**mPower**	**NuScale**	**PHWR-220**	**HTR-PM**	**4S**	**SVBR-100**	**FUJI**	**MYRRHA**
Type	*PWR*	*PWR*	*PHWR*	*HTGR*	*SFR*	*LFR*	*MSR*	*ADS*
Front-End Fuel Cycle	0,90	0,90	0,98	0,83	0,73	0,73	0,72	0,31
Back End Fuel Cycle	0,77	0,72	0,89	0,95	0,45	0,28	0,98	0,0004

Table 6.1

6.2.1 Fresh Fuel

The front-end fuel cycle values in **Table 6.1** are colour-coded to indicate which values can be compared to one another; this is only to a limited extent possible, considering that different types fissile isotopes are used as nuclear fuel. As can be found in Appendix 10.5, the mPower, NuScale, PHWR-220, HTR-PM and 4S all use enriched or natural U-235-based fuel, albeit in different oxide and alloy forms. FUJI and MYRRHA, on the other hand, utilize U-233 and MOX fuel respectively.

Bearing in mind that the PHWR-220 is fuelled by natural uranium, the fact that it possesses the strongest front-end fuel cycle proliferation resistance amongst the U-235 reactors, comes as no surprise. The proliferation resistance of the additional U-235 fuelled reactors decreases gradually until the 12 per cent enrichment mark, where in line with the argument given in paragraph 4.3.2.1, it can be argued that the ease with which the uranium can be enriched increases significantly. In this context it should be noted that the 12 per cent mark is chosen quasi-arbitrarily, based upon a recommendation by Pierpoint (2008) [53], and is therefore subject to debate. However, the only reactors utilizing fuel with an enrichment percentage that would make them qualify for consideration; the liquid metal cooled fast reactors, are sufficiently past the 12 per cent mark to argue that any parameter change would not severely alter the outcome.

MYRRHA, which utilises MOX fuel assemblies to transmute MA rods, was designated its own utility. This means that although the performance metric, 0 to 1, remains the same, the outcome cannot be compared to the values attained for the various uranium fuelled reactors, but should be judged on its own merit. Based on the relatively low score attained by MYRRHA, it can be argued the MA rods contain a relatively unfavourable plutonium composition for non-proliferation purposes, before being entered into the reactor core. It should be noted though, that the purpose of the transmutation process is to pacify harmful nuclear waste, which already implies that any fresh fuel will inherently have an unfavourable isotopic composition. Secondly, as can be seen in Appendix 10.7.2, the percentage of even-mass-numbered plutonium isotopes increases after irradiation. Therefore, it can be argued that the proliferation resistance is improved twofold: the first improvement being the transmutation of the longer lived major actinides into shorter lived minor actinides and the second improvement being the relatively less favourable isotopic composition of the remaining plutonium for weaponization purposes. For the U-233 fuelled FUJI, neither the uranium enrichment utility nor the plutonium enrichment utility applies. Furthermore, review of the existing literature revealed no proliferation resistance utility function for U-233, which can probably be accredited to the current lack of commercial interest in the Th-U233 fuel cycle. Therefore, in order to be able to provide quantification, the spent fuel metric was applied to the fresh fuel, which only makes it possible to state that the proliferation resistance increases after fuel irradiation has commenced.

6.2.2 Spent Fuel

As discussed in paragraph 4.3.2.2, the proliferation resistance of the spent fuel was determined by means of the specific SQ, a property which, in contrast to the fresh fuel model, allow for direct proliferation resistance comparison between the different reactor types. As with the fresh fuel model, the results can be found in **Table 6.1**. For the conventional and evolutionary designs based on existing light and heavy water technology; the PHWR-220 and the mPower and NuScale power modules respectively, the spent fuel profile was generated by inputting the reactor design specifications into the IAEA's Nuclear Fuel Cycle Simulation System [124]. For the innovative designs, including the HTR-PM, 4S, SVBR-100 and FUJI data was collected by means of a literature review. Considering that in the back-end of fuel cycle the same utility function is used for all reactor types, the colour-coding is identical. In addition to the specific SQ per reactor type which is used to determine the spent fuel proliferation resistance utility, two additional categories; the U233 and Plutonium inventories were added. This was done to possibly provide a marginal note as well as to correct for the load factors assumed in the references and the length of the irradiation cycle. A comprehensive overview of the isotopic composition of the spent fuel per reactor type, including the corresponding specific uranium and plutonium inventories is given in Appendix 10.7.1. Considering that power production is not a near term goal for MYRRHA, the plutonium inventories is given in SQs.

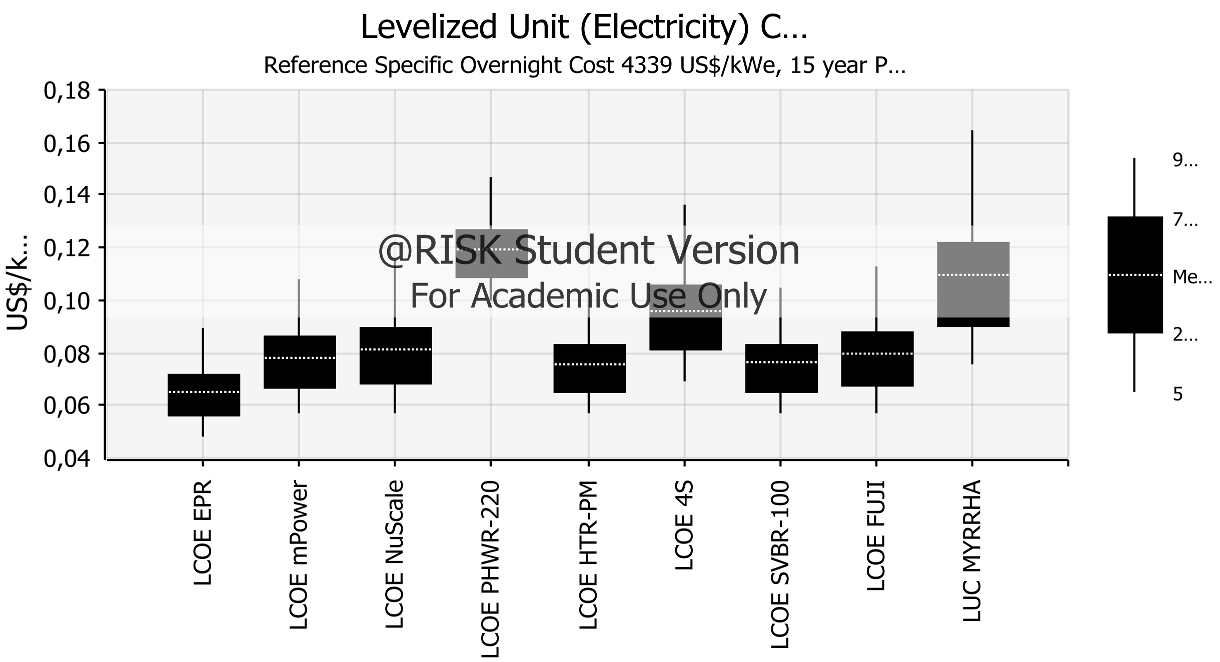
The isotopic composition of the HTR-PM was not encountered in any of the reviewed literature. Therefore, an approximation was made using the mass flow data of unprocessed used fuel for a Once-Through HTGR-UOX with a pebble bed configuration. The data, acquired from Piet et al. (2010) [226], contains several mismatches with the HTR-PM design. The foremost of which is a slightly higher enrichment percentage of 9.6 per cent (HTR-PM pebbles having an enrichment of only 8.9 per cent). However, considering the minuteness of the difference, it will be assumed that this will not severely impact the isotopic composition of the spent fuel. For the purpose of determining the amount of SQs in the spent fuel, the plutonium percentage in the transuranic discharge was taken to 85 per cent, in line with Parent (2003) [185], resulting in the isotopic composition found in Appendix 10.7.1. As previously cited, the resulting isotopic composition might be lacking somewhat in precision, but considering the reticent public information flow of the HTR-PM program, an approximation based on similar technological designs appears to be the only alternative.

Furthermore, the spent fuel compositions of the 4S and FUJI were readily available from IAEA-TECDOC-1536 [23]. The isotopic composition of the SVBR-100s spent fuel was encountered in Zrodnikov et al. (2009) [227], but was measured after a 7 year cooling period. This however, can be considered merely inconvenient when taking into consideration the lengthy half-lives of the actinides of interest. Moreover, the data on the isotopic composition of both the minor actinide and MOX fuel assemblies were acquired from Malambu et al. (20XX) [221].

It should be noted that, based on **Table 6.1** and Appendix 10.7.1, FUJI is the best performer both in terms of the spent fuel proliferation resistance utility and the additional indicators. This was to be expected considering, the low level of excess reactivity combined with the use of U-233 as fissile material, which breeds relatively minute amounts of plutonium under the prevalent in-core conditions. Two other reactors that can be argued to possess superior non-proliferation characteristics are the HTR-PM and the PHWR-220. The HTR-PM owes its low number of SQs per unit of spent fuel to the graphite fuel pebble casings, which represents the majority of the weight of the spent fuel. Further support for the superior non-proliferation properties of the HTR-PM are provided by the relatively modest plutonium inventory, which is the second smallest amongst the reactor designs under investigation.

The proliferation resistance valuation of the PHWR-220 is less straight forward. Although the reactor has a relatively good specific SQ per unit of spent fuel, it generates the largest amount of plutonium SQs per GWey. This can be attributed to the large quantities of U-238 in the spent fuel, which masks the relatively sizeable amount of plutonium generated by the PHWR. Of the two light water reactor designs, which will likely reach the commercial stage first, the NuScale has a lower plutonium inventory. This can be explained by its higher burn-up. The mPower on the other hand possesses a superior SQ per unit of spent fuel factor, which can largely be attributed to its higher conversion efficiency.

6.3 Small Modular Reactor Economics



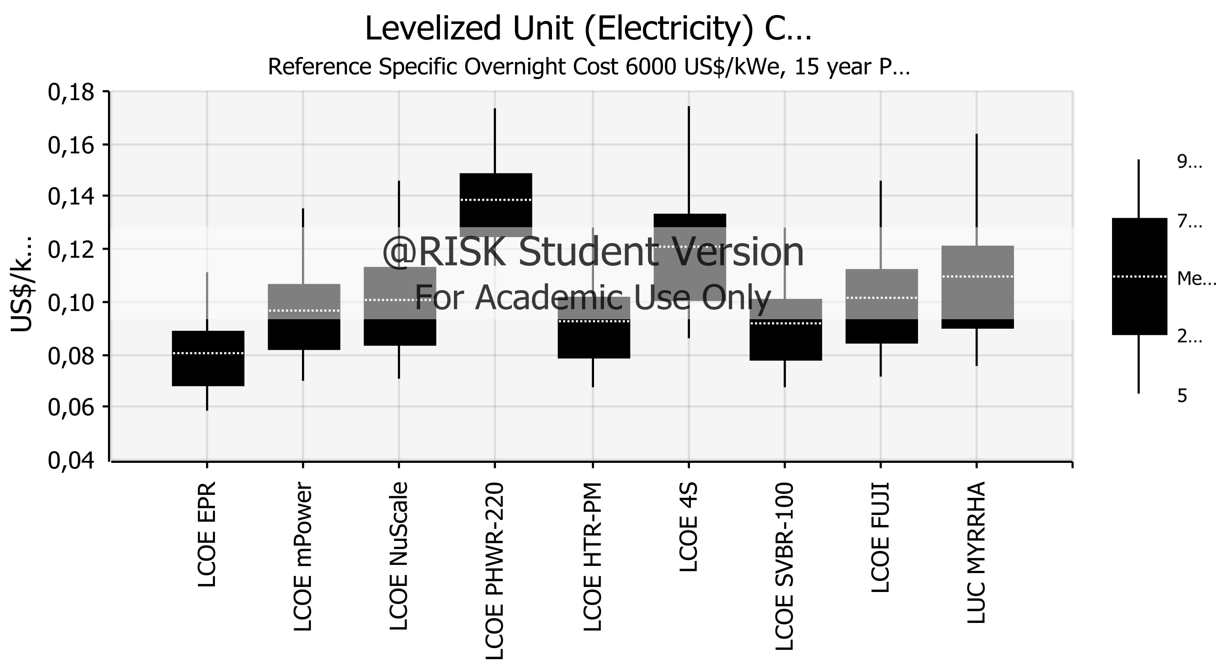


Figure 6.3| Estimated levelized cost of electricity for the reactor designs assuming that (1) the post-Fukushima overnight costs will nog change significantly and (2) that regulatory turbulence and increased safety requirements will result in higher specific overnight costs. The depreciation time is assumed to be 15 years in both cases.

A total of 30 specific probability distributions were identified for the 8 different reactor types under investigation. For the overnight costs, two scenarios were identified: one '*business-as-usual*' scenario in line with the conventions uncovered from previous nuclear cost studies (e.g. [61]) and a second based upon the intuition that following the evens of Fukushima the overall overnight costs will rise due to increased safety features and *'regulatory turbulence'*. It can be concluded from Figure 6.3 and Appendix 10.8.2 that of the SMR designs investigated in this thesis the HTR-PM is expected to have the lowest LCOE for the specified NPP output level of around 600 MWe. However, irrespective of various cost diminishing economies related to modular construction and design, none of the investigated reactor are able outperform the EPR in terms of LCOE. As can be derived from Figure 6.3, for several SMRs the differences in LCOE can be considered minute. Exceptions are the PHWR-220, 4S and HTR-PM, for which the differences are primarily caused by high geological disposal costs for the PHWR-220 and unfavourable and favourable expected overnight costs for the 4S and HTR-PM respectively. The latter serves as an indication that over- or underestimating the overnight costs can have severe effects on the expected LCOE, while the former illustrates the severe cost penalty incurred by reactor designs with high spent fuel volumes in the once-through cycle.

Furthermore, Figure 6.3 reveals that reactors based upon a similar reactor technologies, such as the mPower and the Nuscale, have roughly similar LCOEs with the PDF of the LCOE of the smaller output reactor design (NuScale) being more heavy-tailed, i.e. displaying a greater tendency to deviate from median values. It was with revealing the effects of progressive modularity on the cost of electricity in mind that these two LWRs with deviating technical specifications were selected. Therefore, according to the SMR Economics model proposed in this thesis, increasingly modular reactor designs may have a similar levelized lifetime cost of electricity in comparison to their larger counterparts.

It appears that although there is a large negative impact from the economies of scale, as can be seen in Appendix 10.8.2, the combined effect of the smaller positive impacts of the economies of learning, co-siting, modularity and design and unit timing are capable of outweighing this degenerate effect, as demonstrated by the PHWR-220. However, can be derived from Appendix 10.8.2 this outcome is by no means guaranteed10.8.1.3, even after combining the various positive economies listed in paragraph 4.3.3.3, the overall specific cost could be as much as 14 per cent higher when downsizing the reactor output. These conclusion were drawn only from cost scalings with a LR as a reference, therefore the 16-Module SVBR reference and the single MYRRHA core were not included. These findings are therefore not in line with Carelli et al. (2010) [103], where it was found that based upon relations encountered in literature the economics of scale of a modular 4-core IRIS reactor power plant could only have a slightly higher specific capital cost than a single core larger-sized reactor. A possible explanation for this difference is that IRIS reactor is a medium-sized reactor rather than an SMR owing to its power output of 335 MWe. This limits the negative effects of the economies of scale, while simultaneously including several modularity related advantages.

On a more general note, all of the reactors under investigation in this thesis seem to have significantly higher electricity generating costs than the existing reactor fleet as described in the 2010 edition of the *'Projected Costs of Generating Electricity'* [228]. This includes the EPR reference reactor. Modular construction and design only partially accounts for the specific capital cost increases of the reactor designs under investigation in this thesis. The major contributor is the strong documented overnight cost rise encountered in the actual costs of recent reactor constructions in Japan and Korea as reported by MIT [63].

In addition to the variations in specific costs, further differences in the cost of electricity are created due to the use of different fuel cycles. On the one hand there is the FUJI reactor, for which the expected fuel costs are as low as a fraction of a cent per kWh, as a result of relative resource abundance and the absence of costly processes such as fuel enrichment in the front-end of the fuel cycle. On the other hand there are the 4S and SVBR-100 reactors, which require relatively large quantities of spot-market uranium in order to acquire fuel with the necessary uranium enrichment. In combination with the extensive use of enrichment facilities the fuel costs are estimated to amount to roughly 2.6 cents per kWh(e) as shown in Appendix 10.8.2. Moreover, the fuel costs of the SVBR-100 are of such height that they compensate for the relatively favourable overnight costs. It should be noted that, considering that these reactors were designed for the U.S. and Russian markets respectively, they could possibly be fuelled with diluted weapons-grade uranium, which could reduce the fuel cost. However, for the purpose of comparability it was assumed that the uranium was acquired from the spot market.

Furthermore, as can be derived from Appendix 10.8.1.3, for most of the reactors under investigation, the height of the fuel cycle costs is predominantly influenced by either the front-end or the back-end fuel cycle costs. This can be explained by the inverse relationship which seems to exist between the fuel enrichment and the spent fuel quantity. Reactors with higher fuel enrichment typically also have a higher fuel burn-up, which results in less spent fuel and therefore lower spent fuel costs. However, higher fuel enrichment comes at additional costs at the front end of the fuel cycle. The HTR-PM, which has relatively high front-end and back-end costs, forms the only exception in this thesis. This can be explained by the presence of the graphite structural material in the TRISO fuel pebbles, for which there are currently no economically viable separation techniques, as reported in paragraph 5.3.3. Therefore, the graphite is assumed to be sent to spent fuel storage at the end of each irradiation cycle, adding to the back-end fuel cycle costs due to the cost of spent fuel storage being proportional to its volume

Under the assumptions made in thesis, the SMR with the lowest cost in US$/kWh was found to be the HTR-PM reactor, which could possibly generate electricity at a median price of 0.073 US$/kWh. This result is predominantly caused by the relatively low specific overnight costs reported by Choi (2011) [178], which could be a source of scepticism with respect to this result. The 4S is the direct opposite of the HTR-PM with regarding its cost characteristics. Due to its relatively small power output, which results in severely negative economies of scale, it is expected to have 14 per cent higher specific capital costs than its LR reference plant. In combination with the relative novelty of liquid metal reactor technology, which comes accompanied by a higher contingency rate, the results in a large uncertainty spread. The same argument can be made for MYRRHA, which has a singularly high contingency value attributed to its design as a result of its technical complexity. It should be noted that the economic output unit of MYRRHA is given in US$ per kWh thermal instead of in US$ per kWh electric. Although this might hinder economic comparison, such a comparison was deemed unfair due to the MYRRHA purely being a research reactor which was not designed with commercial power generation in mind.

Judging by the sensitivity analysis of the various reactor designs, as can be found in Appendix 10.8.1, the parameters exerting the strongest influence on the levelized lifetime cost of electricity are the specific overnight costs and the economies of scale in addition to the risk free rate and the risk premium, the latter two combined can be considered to be a proxy for the state of the economy. A separate statement can be made regarding the cumulative load factor, which in comparison to some studies has a conservatively median estimate of 76 per cent. As discussed in paragraph 4.3.3.7, it was decided to use an identical probability density function for all designs, which explains the common shape encountered across the various sensitivity charts in Appendix 10.8.1. Therefore, it offers very limited insight. However, it should be noted that an increase in the cumulative load factor can significant improve the LCOE, often approximately 0,01 US$/kWh when increased to 90 per cent.

With regard to the financial parameters, a higher risk free rate implies that an investor with a well-diversified portfolio requires a higher compensation for systemic risk, which occurs in periods where the risk of default is high (such as in a recession). A high risk premium suggests that the additional expected return an asset must yield above the risk free rate must be high. This implies that there is very little inclination amongst investors to acquire risky assets (again conform to a recession). Therefore, in line with intuition, when subject to a market-conform interest rates, it is best to initiate, and preferably also finish, construction when the investor sentiment is positive. Although this applies to both large and small SMRs, it is easier for SMR to take advantage of this, as a result of their expected shorter construction time being better suited for situations of imperfect knowledge. For the SVBR-100, the uranium spot market price was also found to exert a strong influence on the levelized lifetime cost of electricity. Although both the 4S and the SVBR-100 utilize highly enriched uranium fuel, this effect is more pronounced in the SVBR-100 due to its fuel cycle being nearly 40 per cent shorter.

Lastly, it should be noted that discount rate was constructed specifically for modelling nuclear assets under Swedish market conditions. The underlying CAPM can be modified to provide a crude equity valuation of a variety of industrial sectors in numerous national economies. It should be noted that in different markets nuclear equity could have a different risk beta β value, considering that the perceived riskiness of investments in nuclear power-generation technology is driven in part by cultural heritage. As described in paragraph 4.3.3.4.1, the values that were used to generate the parameters for the risk beta β in this thesis were taken from the U.S. financial market. These values were taken to be representative for the Swedish market based upon a similar overall public support for nuclear power a year after the Fukushima accident [229]. However, under the restriction that one possesses knowledge of the various probability density functions for the risk free rate, risk beta and the risk premium, one could obtain a fit for a different economy with relative ease.

7 Conclusion

In this thesis an attempt was made to construct a framework for comparing various SMR designs. Of the total population of SMRs, the majority is still in the early stage of development. Therefore, only one design was selected per reactor type, with the exception of light water-based SMR of which two designs are currently under licensing review. The assessment was based on the level of progression which has been made on the design; the minimum progression having been determined to be the preliminary design stage. The selection was subsequently analysed with regard to its passive safety, for which the thermal inertia in combination with the SA:V was considered to be a proxy; its proliferation resistance, for which several utility function were taken from literature; and its economic potential, for which the levelized cost of electricity was determined.

For a thorough evaluation, several additional criteria, such as the public acceptance of certain nuclear fuel cycles, should also be taken into consideration. In addition, for some of the criteria that were evaluated, it needs to be wondered whether they were developed to the fullest possible extent. The passive safety evaluation in particular was identified as not taking into consideration sufficient informative parameters to render any definitive conclusion premature.

Therefore, it was not deemed possible to come to a definitive conclusion about what SMR could be considered the most desirable by means of a multi-criteria analysis.

The proliferation resistance was split up in the front-end and the back-end of the fuel cycle. In the front end of the fuel cycle the best overall performers were found to be the reactors utilizing low enriched uranium, like the NuScale and mPower reactors, or natural uranium fuel, like the PHWR-220. This is due to the larger number of SWUs required to produce weapons-grade uranium out of low enriched uranium fuel. In addition, a natural uranium fuel cycle could imply the unavailability of enrichment facilities in general, which removes the possibility of their use in a clandestine operation.

For the back-end of the spent fuel cycle the FUJI reactor was ranked the most proliferation resistant reactor, with a nearly perfect score following the almost complete absence of plutonium in its spent fuel. Additionally, the HTR-PM, which benefits from a large mass of spent fuel, produces the second lowest amount of plutonium SQs per GWe·y. A large spent fuel mass is also the reason that the PHWR-220 rated relatively resistant to proliferation, which can be considered surprising due to the relatively large amounts of spent plutonium SQs in the spent fuel. The large mass of spent fuel the SQs are divided over however resulted in a relatively low plutonium percentage. Considering that plutonium has the potential for direct-use and reactor-grade uranium has to be enriched first, the proliferation resistance of the back-end of the fuel cycle was considered more important than front-end of the fuel cycle for all reactors save the MYRRHA. The latter utilizes plutonium as its fuel.

The passive safety was approximated by assessing how quickly the temperature of the materials in the reactor environment increases when the removal of heat from the primary circuit is interrupted. An analysis resulted in the identification of multiple structures which could aid in dissipation of heat. In this thesis (1) the primary coolant, (2) a passive heat removal system using radiative heat transfer, and (3) the presence of large in-core deposits with a high heat capacity, such as graphite, were included in the analysis. It was concluded that all but one reactor (the 4S) possessed a higher thermal inertia than a state-of-the-art large-sized reactor, the EPR. However, this is a contestable item. The best overall performer is the HTR-PM, this is due to its low average core power density and high degree of thermal inertia as a result of the large volumes of graphite present inside the reactor core. Other notable performers are the SVBR-100, which employs an intricate passive heat removal system, and the FUJI reactor, which contains a large volume of graphite moderator/reflector that doubles as heat sink, similar to the HTR-PM. Additionally, all of the abovementioned reactors have a relatively large SA:V in addition to a high degree of thermal inertia.

The economic potential was measured through the levelized cost of electricity, which includes all costs that need to be incurred during the reactor's lifetime, ranging from construction to decommissioning and decontamination. Considering that SMRs are partially designed to be affordable for privately held companies, competitive interest rates were approximated by means of the CAPM in favour of the government-backed lending schemes, which typically come at lower interest rates. Furthermore, variables such as the specific overnight costs, cumulative load factor and operational lifetime were not considered to be constant, but input as probability density functions, which is more in line with the uncertainty inherent to the cost estimation process of reactor designs. This is of particular importance due to the majority of the designs investigated in this thesis still requiring a first-of-a-kind construction experience.

The resulting cost distributions are generally higher than conventional cost estimates for nuclear power. This can be explained by the overall higher specific capital costs of SMRs, which have been widely documented to be the major cost component in the cost of electricity. Furthermore, SMRs are typically designed to operate co-generation mode, meaning that additional useful output in the form of heat credits or desalinated water is generated, which reduces the generating costs. These and similar possibilities for cost reductions were not taken into consideration in this thesis.

Of the SMR designs, when looking at the cost of electricity alone, the HTR-PM was found to have the lowest levelized cost of electricity. This can be solely attributed to the relatively low specific overnight costs estimate, in comparison to conventional light water reactor designs, published by various authors (e.g. [178]). It should be noted these estimates could prove to be optimistic as a result of the authors, or their sources, being directly or indirectly involved with the HTR-PMs development. However on a separate note, based on the three models discussed in this thesis, the HTR-PM can be considered to be a strong contender.

In the end only actual operational experience can determine whether the designs investigated in this thesis are feasible for participating in any meaningful way towards reducing our dependency on fossil fuels. History is littered with examples of overly optimistic prognoses of the role of nuclear power in our energy order. Although this does merit some scepticism towards 'miraculous new designs' offering unparalleled safety, negligible waste issues and electricity too cheap to meter, scepticism should ultimately not restrain us from developing and building any reactors of which it can be verified that they can perform at or above the governing standard.

8 Research Suggestions

8.1 Parameter (Re-)evaluation

As indicated in paragraph 6.1 the passive safety of the various reactor designs was not investigated to the fullest extent possible based upon the list provided in Turkenburg (2003) [25]. Therefore, a re-evaluation of the passive safety term would be in order if methodologies for the missing parameters were developed. Furthermore, several aspects of the nuclear power debate were completely left untouched in this thesis, such as the issues relating to high-level waste management. A possible expansion of this conceptual framework for example, would be an analysis of how long it would take for the spent fuel of the various reactor designs to attain the level of radioactivity associated with natural uranium.

8.2 Multi-Criteria Analysis

Locatelli and Mancini (2012) [230] attest to that selecting a nuclear reactor design among many alternatives is a multidimensional optimization problem, in which both financial and other ‘external’ variables need to be taken into account. An adaptation of a Multi-Attribute Decision Making model is given by Locatelli and Mancini (2012) [230], which address a multitude of economic considerations such as the grid vulnerability, various financial parameters and the effects on local industries and employment. However, it fails to address several items of which debates have indicated that they are serious points of public contention, such as the perceived reactor safety and the resistance to proliferation, as outlined in Turkenburg (2003) [25]. In this thesis, methods were outlined that could be used to quantify a reactor’s passive safety as well as provide a measure for its proliferation resistance. A reassessment of the Locatelli and Mancini model, potentially adopting the aforementioned methodologies could add an additional dimension to the model; a partial reflection of public support.

8.3 Co-generation

Several SMRs were specifically designed to be operated in co-generation mode. It is estimated that in this way the LUEC of a plant can be improved by 20 to 30% [136]. Although the benefits of co-generation can be accomplished by any NPP, regardless of its capacity, a recent inquiry [231] into the economic viability of SMRs in future European cogeneration markets has concluded that the main advantage SMRs have over LRs regarding cogeneration is their limited thermal capacity. Shropshire et al. [231] conclude that SMRs tends to be better aligned with the actual demand of process industries and that the greatest deployment potential is in the chemical/petroleum, paper, metal, and bioenergy markets with small capacities (50–250 MWth). However, it is noted that the actual competitiveness of SMRs against Coal-CHPs and Gas-CHPs is highly dependent on the development of the overnight capital and the total cost of capital in general (i.e. cost of debt during construction). It is expected that co-generation plays to the strengths of SMRs due to their ability to be placed closer to population centres as a result of their expected exclusion zones being smaller. However, a possible effect of the events at Fukushima is that the exclusion zone of LRs will be increased and, in terms of the danger they pose to the environment, 100 MWe and 1000 MWe can be argued to be quite similar. Therefore, the adoption of SMRs for use in their intended role is far from certain. As a result, further research into the deployment scenarios of the investigated SMR designs for simultaneous electricity production and district heating/desalination/hydrogen production could result in different views regarding the specific reactor economics.

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10 Appendices

10.1 Nomenclature

ADS	Accelerator Driven System
ALMR	Advanced Liquid Metal Reactor
ANL	Argonne National Laboratory
ASN	French Nuclear Safety Authority
CANDU	CANadian Deuterium Uranium (Reactor)
CAPM	Capital Asset Pricing Model
D&D	Decommissioning and Dismantling (Costs)
DoE	U.S. Department of Energy
ECCS	Emergency Core Cooling System
EEDB	Energy Economic Data Base
EFPD	Effective Full Power Days
EM	Electromagnetic
EMWG	Generation IV Economic Modelling Working Group
EP	Emergency Protection (Rods)
EPR	European Pressurized Reactor
FOM	Figure of Merit
HTGR	High-Temperature Gas Cooled Reactor
IAEA	International Atomic Energy Agency
IAPWS	International Association for the Properties of Water and Steam
IAPWS-IF97	IAPWS Industrial Formulation 1997
IEA	International Energy Agency
IHX	Internal Heat Exchanger
IRACS	Intermediate Reactor Auxiliary Cooling System
LBE	Lead Bismuth Eutectic
LCOE	Levelized Cost of Electricity
LOCA	Loss of Coolant Accident

LR	Large Reactor
LWR	Light Water Reactor
MA	Minor Actinide (Rods)
MCM	Monte Carlo Method
MOX	Mixed Oxide (Fuel)
MOX-EU	European Mixed Oxide (Fuel)
MSR	Molten Salt Reactor
NEA	Nuclear Energy Agency
NOAK	N^{th} of a kind
NPP	Nuclear Power Plant
NPS	New Policies Scenario
O&M	Operation and Maintenance (Costs)
OECD	Organisation for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PBMR	South African Pebble Bed Modular Reactor
PCV	Pressure Containment Vessel
PDF	Probability Density Function
PHRS	Passive Heat Removal System
PHWR	Pressurized Heavy Water Reactor
PRIS	IAEA Power Reactor Information System
PWR	Pressurized Water Reactor
RC	Reactivity Compensation (Rods)
RCSS	Reactivity Control and Shutdown System
RPV	Reactor Pressure Vessel
RVACS	Reactor Vessel Auxiliary Cooling System
SA:V	Surface-Area-to-Volume Ratio
SFR	Sodium-Cooled Fast Reactor
SG	Steam Generation (Module)
SMR	Small Modular Reactor

SQ	Significant Quantity
SSM	Swedish Nuclear Safety Authority
SWU	Separate Work Units
THORIM-NES	Thorium Molten Salt Nuclear Energy System
TRISO	TriStructural-Isotropic (Fuel)
TRU	Transuranic Elements
UREX	Uranium Extraction
WACC	Weighted Average Cost of Capital

10.2 @RISK

The analysis of the SMR Economics was performed by utilizing the @RISK Excel add-in. @RISK allows for the definition of uncertain cell values in Excel as probability distributions by means of command functions. These functions, each of which allows for different distribution types to be attributed to certain cell values, are added to the existing Excel function set. These distributions, referred to as PDFs within this thesis, can be added to any number of cells and/or formulas within a worksheet and can include expressions and cell references as arguments, which allows for sophisticated definitions of uncertainty by utilizing the interrelationships amongst cells. Furthermore, @RISK comes equipped with advanced capabilities for specifying and executing simulations in Excel models. Among others, Monte Carlo and Latin Hypercube sampling techniques are included and distributions specifying the outcome can be generated for any cell within the spreadsheet model. Adding to its ease of use is that all simulation options and model selection features can be entered by means of familiar Windows-styled menus, dialog boxes and conventional mouse operation. For more information, see reference [232]

10.3 Probability Distribution

Traditionally, cost estimates are portrayed as point estimates. This means that each cost element is represented by a single value which is deemed to be most likely to occur. The contingency is often a single percentage indiscriminately applied to all cost categories. The most important weaknesses of this technique are that the percentage is usually arbitrarily arrived at and therefore not project specific. Furthermore, even with the percentage addition the cost estimate is still a single figure prediction containing no information on the shape of the distribution. Moreover, it only includes an indication of the potential for downside risk; it does not indicate any potential for cost reduction. Although cost reductions are unlikely, not incorporating the possibility that they migh occur at all could lead to poor management during the projects progression [233].

Furthermore, point estimates severely limit the depth of the analysis that can be performed on the estimate and they limit the possibility of comparison between the various options. Therefore, considering that the construction of an NPP has many uncertainties and unknowns, which may directly or indirectly influence the costs, presenting the capital costs as a single point estimate might be an unrealistic representation of how the costs evolve over the length of the project. Reference [94] states that *"viewing the capital costs as a distribution speaks to the fact that there is uncertainty in the cost and that a point estimate cannot be accepted as an authoritative, final price.* Therefore, if one reports the capital costs as a distribution, it opens up the possibility for probabilistic analysis and comparison between two or more options, which are based on more than the difference between two point estimate values. It might, for example, be the case that as a result of the uncertainty ranges, two or more alternatives are within range of one another, which could indicate that the differences in estimated costs are statistically insignificant. Capital costs however, are not the only sources of uncertainty in the economic evaluation of NPPs, therefore various other variables might benefit from being presented as a distribution as well.

10.4 Contingency

Most studies using MCM for project cost estimates recommend setting the base estimate plus the contingency at the 50 per cent probability level (median) [98] [96]. This is based on the rationale that it provides the project teams with a target for which they have an equal probability of underrunning or overrunning the estimate.

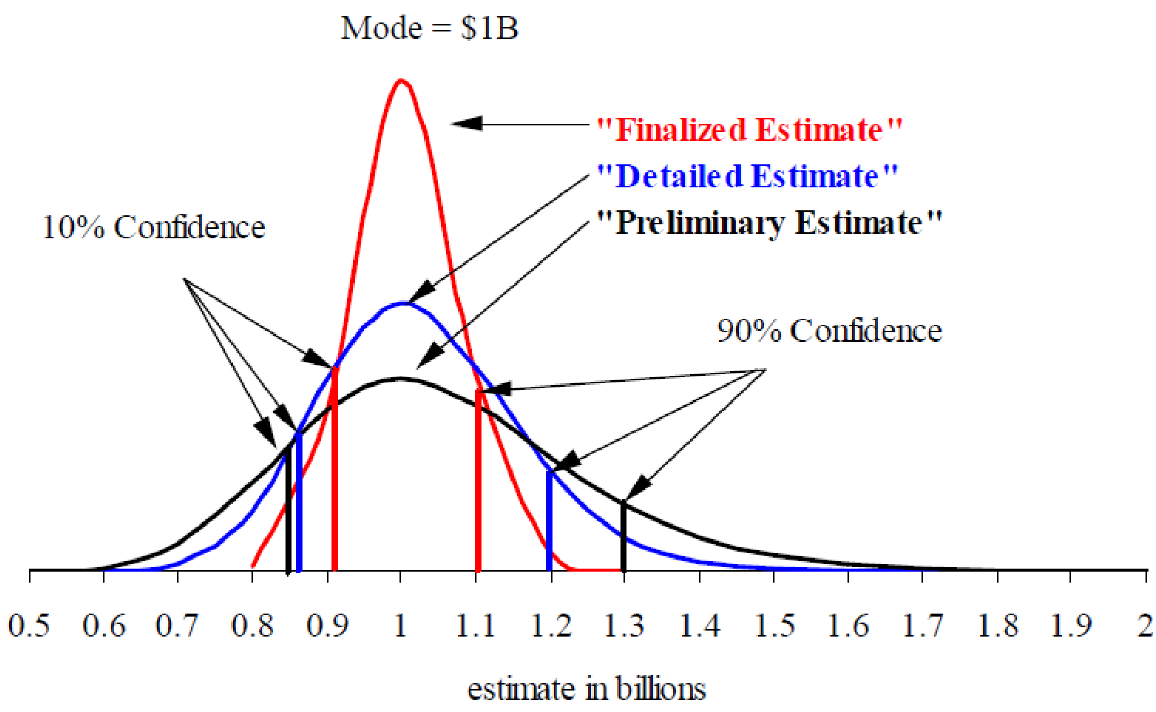


Figure 10.1| Probability distributions for estimates with varying levels of detail

Taken from Rothwell (2004) [98]

However, this profile does not correspond with contemporary nuclear construction experience, where cost overruns are known to occur more frequent than cost underruns. Therefore, the lower range is less in absolute terms than the higher range due the higher probability of overrun in comparison to the probability of underrun. Rothwell [2004] established that this corresponds to a lognormal distribution [98]. Figure 10.1, shows the probability density functions corresponding to several levels of uncertainty in the cost estimation process. Table 10.1 provides the corresponding values for the median, mean, variance and standard deviation after the mode has been normalised. The variance is given by Equation 10.1 and the standard deviation by Equation 10.2 [100]

	Mode	Median	Mean	Variance	Standard Deviation
Preliminary Estimate	1,000	1,033	1,049	3,4%	18,3%
Detailed Estimate	1,000	1,017	1,025	1,7%	13,1%
Finalized Estimate	1,000	1,005	1,008	0,5%	7,0%

Table 10.1

$$Var(X) = [\tilde{x} \cdot (\tilde{x} - 1)]$$

Equation 10.1

$$\sigma = \sqrt{Var(X)}$$

Equation 10.2

In these equations $Var(X)$ represents the variance of a random variable X, $\tilde{x}$ is an expression for the median and $\boldsymbol{\sigma}$ equals the standard deviation. The latter is of particular interest here, since it has been shown to fall within the contingency range of the Electric Power Research Institute (EPRI). The 18.3 per cent standard deviation derived for the preliminary estimate lies in the range of 15 to 30 per cent contingency suggested by the EPRI. Likewise, the 13.1 per cent standard deviation falls within the EPRI range of 10 to 20 per cent. The standard deviation for the finalized estimate is 7 per cent which is within the 5 to 10 per cent range given by EPRI.

10.5 Design Characteristics

Reactor	Symbol	Unit	EPR	mPower	NuScale	PHWR-220	HTR-PM	4S	SVBR-100	FUJI	MYRRHA
Type	-	-	PWR	PWR	PWR	PHWR	HTGR	SFR	LFR	MSR	ADS
Thermal Capacity	Pth	MWth	4250	530	160	754	250	135	280	450	100
Electrical Capacity	Pr	MWe	1550	180	45	235	105	50	101,5	200	-
Average Core Power Density		MW/m3	89	69	55	100	3,22	48	140	7,3	250
Thermal Efficiency	η	%	0,36	0,34	0,28	0,31	0,42	0,37	0,36	0,44	-
Design Lifetime	nd	years	60	60	60	40	40	30	60	30	40
Operational Lifetime	nl	years	52,5	52,5	52,5	35	35	30	52,5	30	35
Fuel	-	-	UO2	UO2	UO2	UO2	TRISO (UO2)	U-Zr alloy	UO2	UF4-ThF2	MOX
Fuel Cycle Duration	τ	EFPD	730	1644	732	300	706	3653	2200	2000	270
Enrichment	χp	%	0,050	0,050	0,050	0,007	0,089	0,159	0,163	-	0,3
System Pressure	p	MPa	15,5	13,1	10,7	8,532	7	0,08	9,5	0,5	0,1
Average Burn-Up	B	MWd/kgHM	70	40	62	6,7	90	90	66	100	100
# Modules	N	#	1	3	12	3	6	12	6	3	6

10.6 Passive Safety Model

10.6.1 Thermophysical Properties Primary Coolant Loop, PHRS and In-Core Structures

Reactor	Symbol	Unit	EPR	mPower	NuScale	PHWR-220	HTR-PM	4S	SVBR-100	FUJI	MYRRHA
Primary Coolant	-	-	Light Water	Light Water	Light Water	Heavy Water	Helium	Sodium	Lead-Bismuth	FLiBe	Lead-Bismuth
Core Outlet Temperature	Tout	K	601,1	594	561,9	566	1023	783	755	833	673
Core Inlet Temperature	Tin	K	568,5	570	520,9	522	523	628	593	973	543
Core Avg. Temperature	Tav	K	584,8	582	541,4	544	773	705,5	674	903	608
Primary Coolant Inventory	m	kg	-	-	-	100000	3000	36000	-	-	-
Primary Coolant Volume	V	m3	455	92	-	-	-	-	18	-	-
Density Coolant	ρ	kg/m3	695	702	778	-	-	-	10204	2151	10205
Heat Capacity Coolant	Cp	kJ/kgK	5,800	5,830	4,992	4,86	5,195	1,276	0,144	2,42	0,145
PHRS Operating Pressure	-	MPa	-	-	0,100	-	-	-	0,100	-	-
PHRS Coolant Density	-	kg/m3	-	-	998,236	-	-	-	998	-	-

PHRS Volume	-	m3	-	-	-	-	-	-	250	-	-
PHRS Inventory	-	kg	-	-	15.141.647	110000	352000,00	-	-	-	-
PHRS Heat Capacity	-	kJ/kgK	-	-	4,185	4,19	0,71	-	4,185	0,71	-
Calandria Vault	-	kg	-	-	-	400000	-	-	-	-	-
Calandria Heat Capacity	-	kJ/kgK	-	-	-	4,19	-	-	-	-	-

10.6.2 Surface-Area to-Volume Ratio and Thermal Inertia

Passive Safety Model										
Reactor	**Unit**	**EPR**	**mPower**	**NuScale**	**PHWR-220**	**HTR-PM**	**4S**	**SVBR-100**	**FUJI**	**MYRRHA**
Type		PWR	*PWR*	*PWR*	*PHWR*	*HTGR*	*SFR*	*LFR*	*MSR*	*ADS*
Primairy Coolant Loop	K/s	2,32	1,41	NA	1,55	16,04	2,94	10,59	3,28	0,45
With PHRS	K/s	-	-	0,03	0,09	-	-	0,26	-	-
With In-Core Graphite	K/s	-	-	-	-	0,94	-	-	1,76	-
Thermal Inertia	**K/s**	**2,32**	**1,41**	**0,03**	**0,09**	**0,94**	**2,94**	**0,26**	**1,76**	**0,45**
RPV Height	m	12,70	23,00	19,81	5,00	11,00	18,00	6,92	2,94	11,00
RPV Radius	m	2,45	2,25	2,21	3,00	1,50	1,50	2,27	3,42	3,80
Surface Area to Volume Ratio	**m^{-1}**	**0,97**	**0,98**	**1,01**	**1,07**	**1,52**	**1,44**	**1,17**	**1,27**	**0,71**

10.7 Proliferation Resistance

10.7.1 Spent Fuel Characteristics SMR NPPs

Spent **Fuel Contents**	**Unit**	**EPR**	**mPower**	**NuScale**	**PHWR-220**	**HTR-PM**	**4S**	**SVBR-100**	**FUJI**
Reference Load Factor		90%	90%	90%	90%	85%	95%	90%	90%
Rated Power Production	GWey	1,00	1,00	1,00	1,00	0,31	1,60	0,96	0,88
Reference Power Production	GWey	0,90	0,90	0,90	0,90	0,26	1,52	0,87	0,79

Isotopic Composition Spent Fuel									
U235	tonnes	0,04	0,34	0,09	0,39	0,050	1,36	0,864	0,00
U233	"	0,00	0,00	0,00	0,00	0,000	12,74	0,00	0,01
U234	"	0,00	0,00	0,00	0,00	0,000		0,0003	0,00
U236	"	0,09	0,14	0,13	0,11	0,016		0,1117	0,00
U238	"	11,63	22,44	17,20	154,97	1,238		7,0476	0,00
Pu238	"	0,01	0,00	0,01	0,00	0,023	0,65	0,0008	0,0005
Pu239	"	0,06	0,12	0,09	0,41			0,341	
Pu240	"	0,04	0,05	0,05	0,14			0,0191	
Pu241	"	0,02	0,03	0,03	0,03			0,0005	
Pu242	"	0,02	0,01	0,02	0,01			0	
Am241	"	0,00	0,00	0,00	0,00	0,004	0,00085	0,0003	0,00
Am242m	"	0,00	0,00	0,00	0,00			0	0,00
Am243	"	0,01	0,00	0,01	0,00			0	0,00
Cm242	"	0,00	0,00	0,00	0,00			0,000067	0,00
Cm243	"	0,00	0,00	0,00	0,00			0	0,00
Cm244	"	0,00	0,00	0,00	0,00			0	0,00
Cm245	"	0,00	0,00	0,00	0,00			0	0,00
Cm246	"	0,00	0,00	0,00	0,00			0	0,00
Np237	"	0,02	0,01	0,02	0,00	0,00	0,0162	0,0069	0,00
Total HM	"	11,93	23,16	17,65	156,06	1,33	14,77	8,39	0,01
Total FP	"	0,93	1,00	1,21	1,09	0,14	-	0,62	1,70
Total Misc.	"					41,5			
Grand Total	"	12,86	24,15	18,86	157,15	42,97	14,77	9,01	1,71
U-233 Inventory	SQs/GWey								0,17
Plutonium Inventory	SQs/GWey	20,46	29,51	28,79	81,60	10,86	53,41	52,05	
Specific SQ	SQs/tSF	1,59	1,22	1,53	0,52	0,25	3,62	5,78	0,10
Percentage Plutonium	-	0,011	0,009	0,011	0,0037	0,0005	0,044	0,040	0,00

10.7.2 Spent Fuel Characteristics MYRRHA

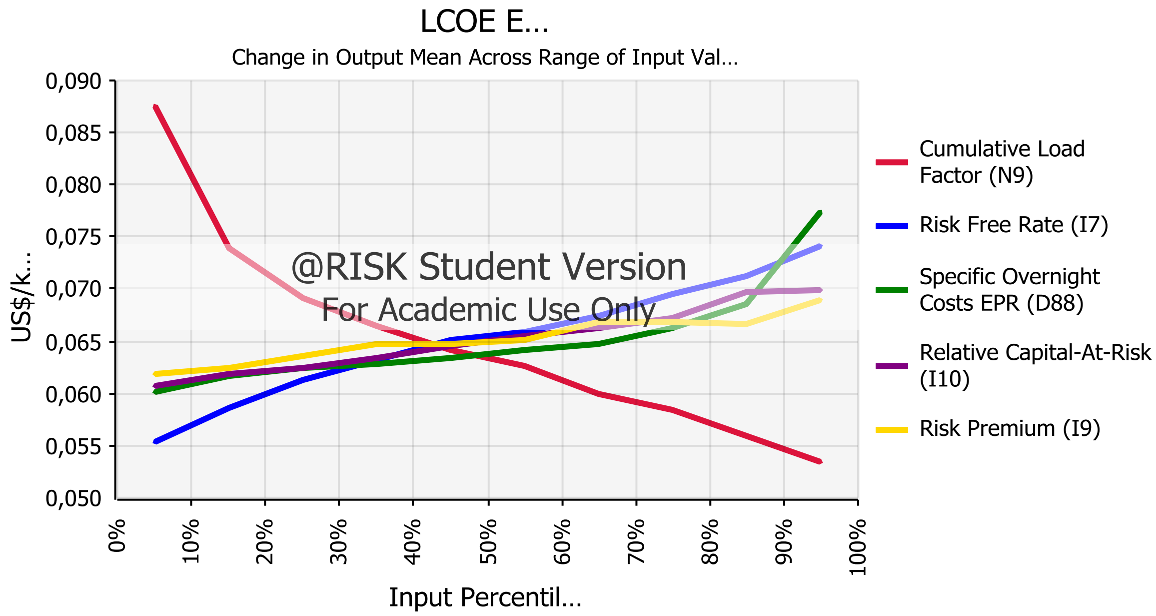
MYRRHA				
Spent Fuel Composition	Unit	MA Rods (kg)		MOX Rods (kg)
Isotopic Composition		Fresh Fuel	Spent Fuel	Fresh Fuel
U235	tonnes	0,00	0,00	1,51
U233	"	0,00	0,00	0,00
U234	"	0,00	0,00	0,01
U236	"	0,00	0,00	0,37
U238	"	0,00	0,00	372,46
Pu238	"	0,18	0,26	0,20
Pu239	"	1,37	1,18	99,18
Pu240	"	1,10	1,09	37,66
Pu241	"	0,48	0,41	14,34
Pu242	"	0,49	0,52	0,71
Am241	"	3,02	2,62	0,00
Am242m	"	0,00	0,06	
Am243	"	1,51	1,35	
Cm242	"	0,00	0,12	
Cm243	"	0,00	0,00	
Cm244	"	0,81	0,87	
Cm245	"	0,09	0,10	
Cm246	"	0,00	0,00	
Np237	"	0,00	0,00	
Total HM	"	9,05	8,58	526,45
Total FP	"	-		-
Total Misc.	"			
Grand Total	"	9,05	8,58	526,45
Plutonium Inventory	SQs	0,45	0,43	19,01

Specific SQ	SQs/tSF	49,99	50,31	36,11
Percentage Plutonium	-	0,400	0,402	0,29
Even-Even Isotope Fraction	-	0,49	0,54	0,25

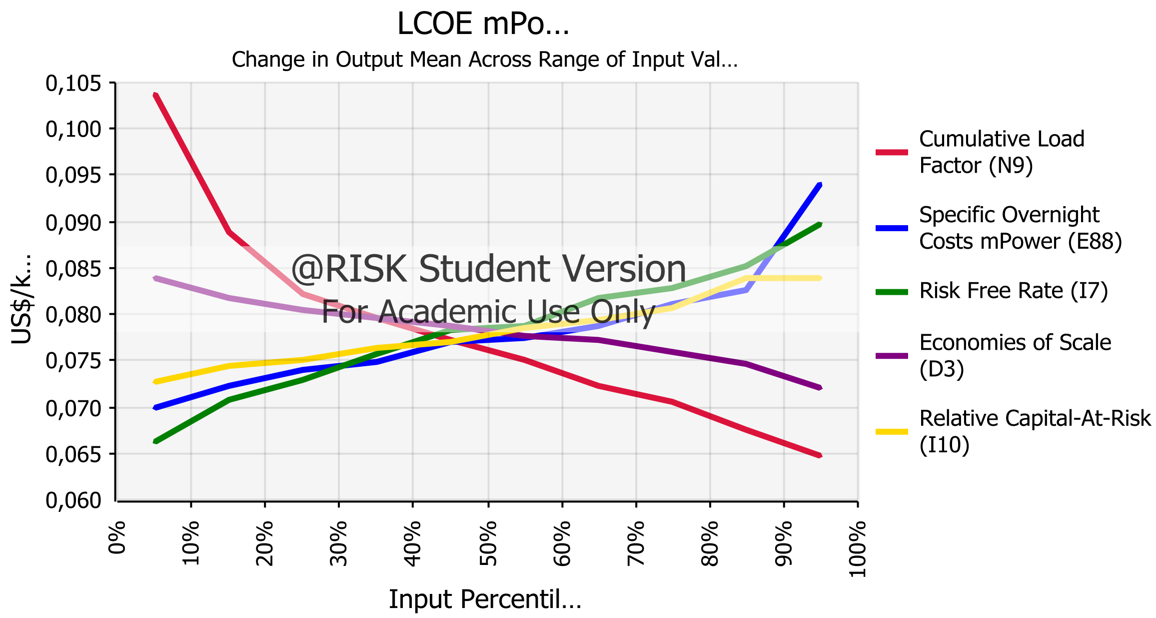
10.8 Small Modular Reactor Economics

10.8.1 Sensitivity (Business-As-Usual)

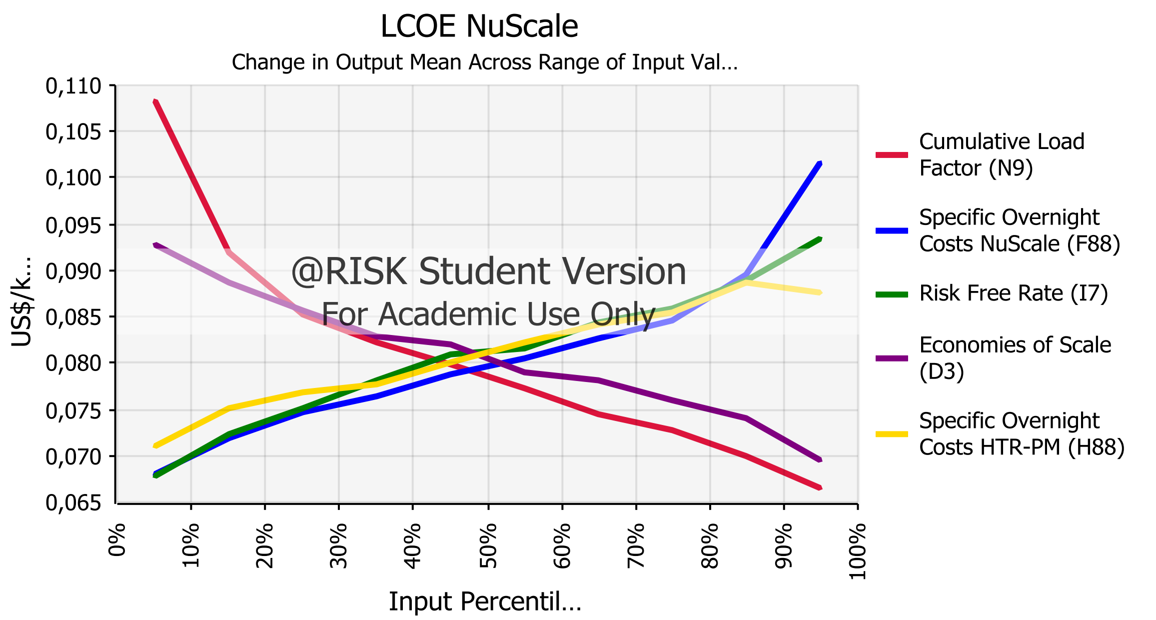
10.8.1.1 EPR



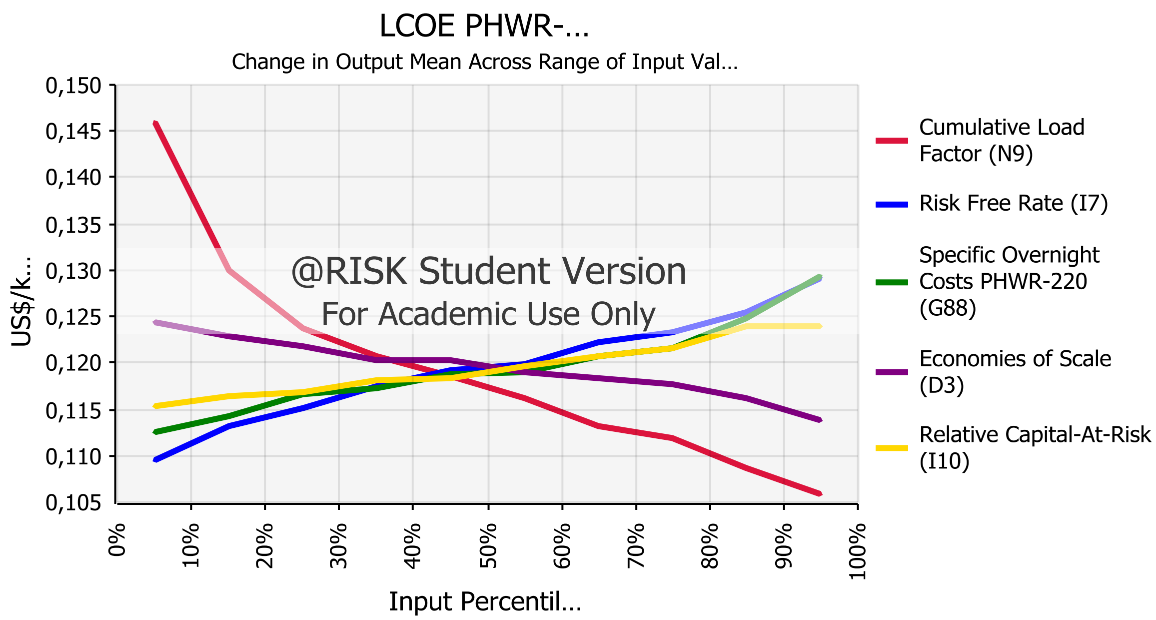
10.8.1.2 mPower



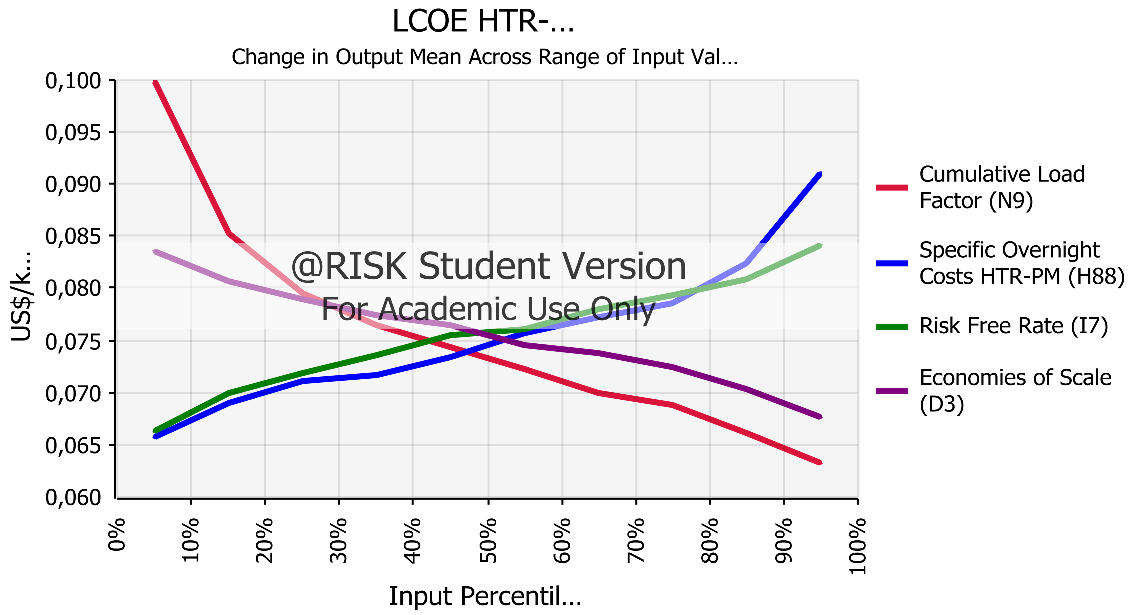
10.8.1.3 NuScale



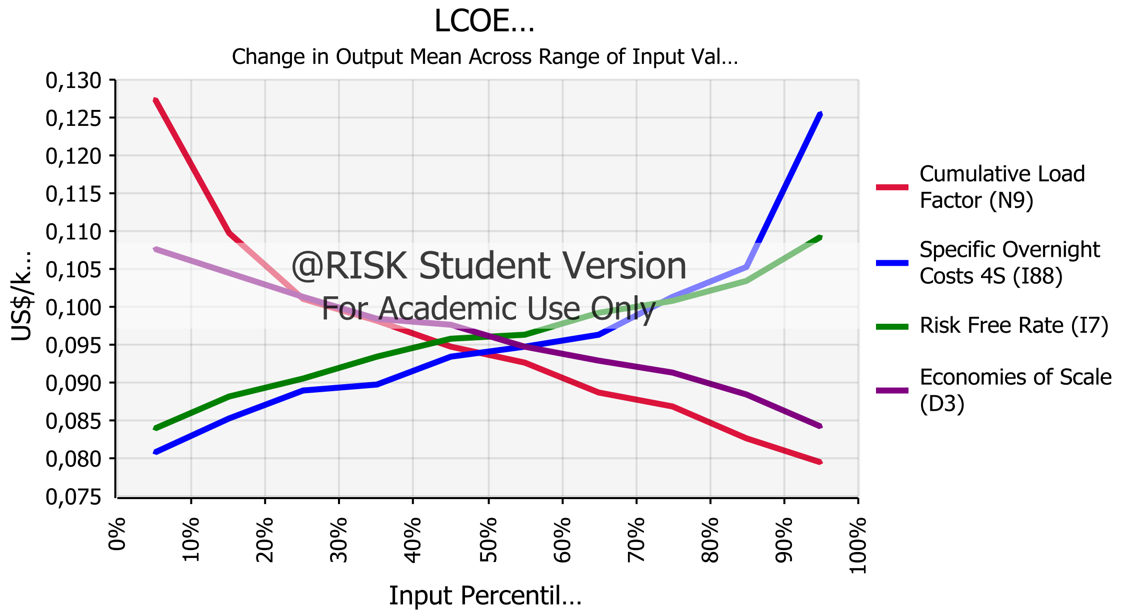
10.8.1.4 PHWR-220



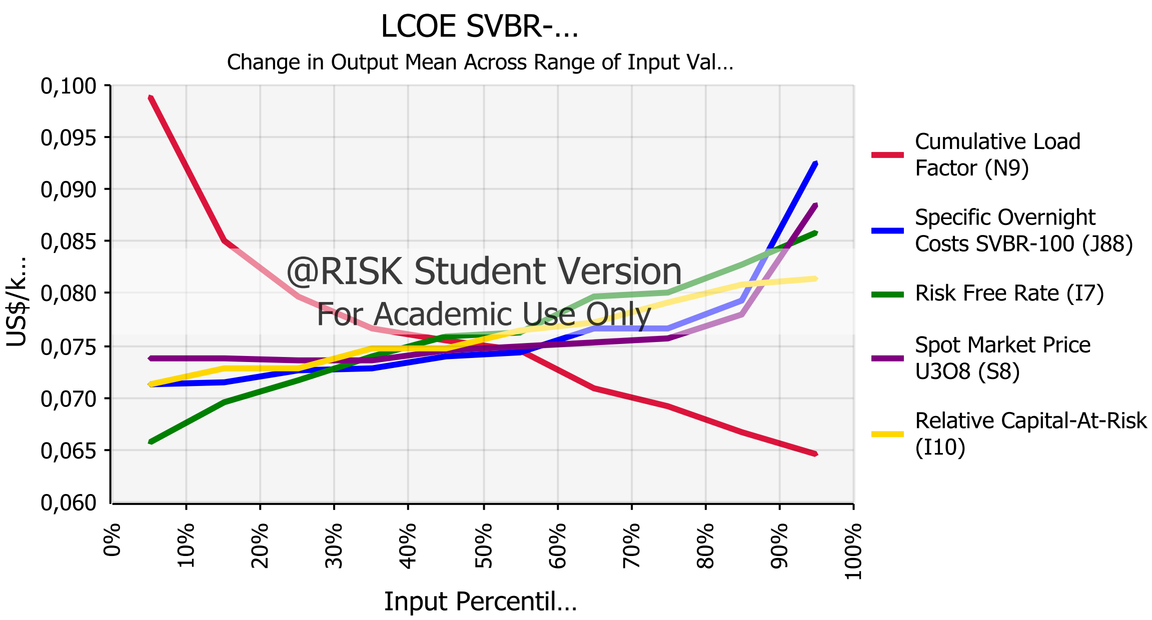
10.8.1.5 HTR-PM



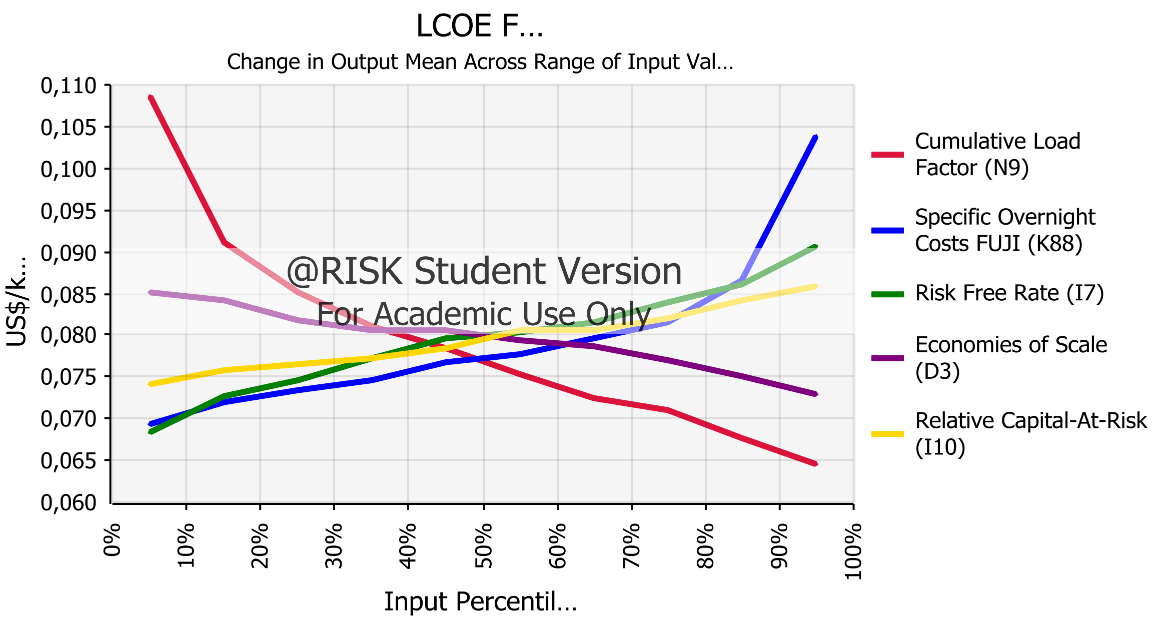
10.8.1.6 4S



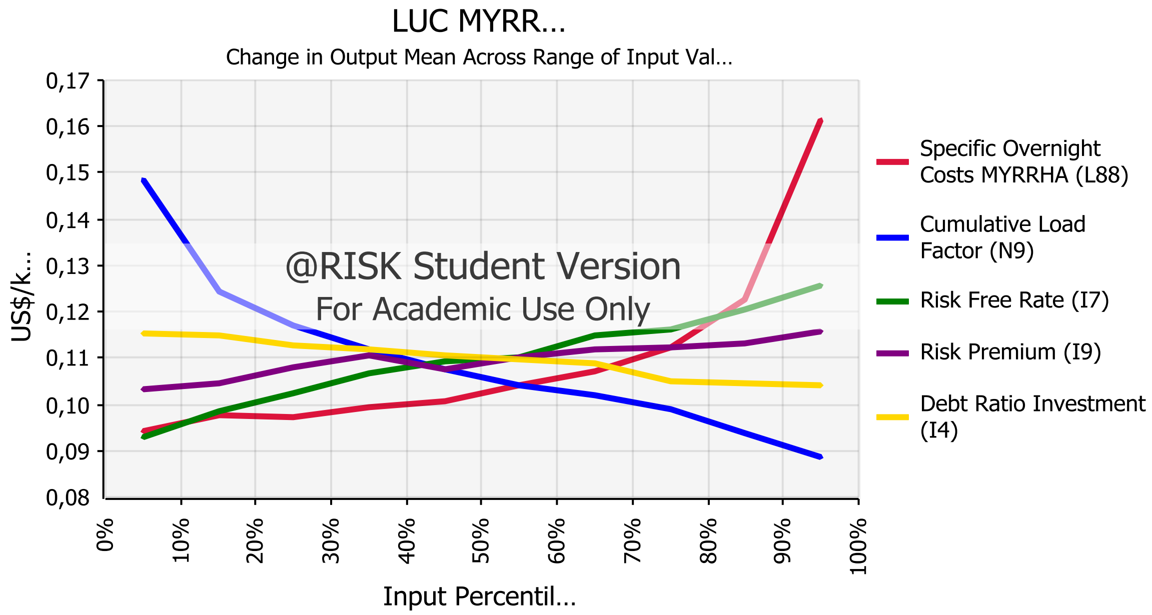
10.8.1.7 SVBR-100



10.8.1.8 FUJI



10.8.1.9 MYRRHA


LUC MYRR...
Change in Output Mean Across Range of Input Val...
0,17
0,16
0,15
0,14
0,13
0,12
0,11
0,10
0,09
0,08
US$/k...
@RISK Student Version
For Academic Use Only
0%
10%
20%
30%
40%
50%
60%
70%
80%
90%
100%
Input Percentil...
Specific Overnight Costs MYRRHA (L88)
Cumulative Load Factor (N9)
Risk Free Rate (I7)
Risk Premium (I9)
Debt Ratio Investment (I4)

10.8.2 Variables and Intermediate Results (Business-As-Usual)

Reactor	**Symbol**	**Unit**	**EPR**	**mPower**	**NuScale**	**PHWR-220**	**HTR-PM**	**4S**	**SVBR-100**	**FUJI**	**MYRRHA**
Type	-	-	PWR	PWR	PWR	AHWR	HTGR	SFR	LFR	MSR	ADS
Reference Plant	-		Large PWR	Large PWR	Large PWR	Large PHWR	Large HTGR	2-Module ALMR	16-Module SVBR	Large MSR	Single MYRRHA
Size Reference Plant	SLR	MWe	1000	1000	1000	1000	1152	644	101,5	1000	-
Number of Modules Reference Plant			1	1	1	1	1	2	16	1	1
Size NPP	SSR	MWe	1550	540	540	705	630	600	609	600	600
Overnight Cost Reference	ACLR	US$/kWe	4339	4339	4339	4339	3569	4669	3231	4747	13456
Economies of Scale	ϑ es	-	0,839	1,986	3,457	1,785	2,607	2,779	1,000	1,904	1
Economies of Learning	ϑ l	-	1,00	0,92	0,79	0,92	0,86	0,83	1,12	0,920	0,879
Economies of Co-siting	ϑ cs	-	1,00	0,77	0,68	0,77	0,71	0,83	1,05	0,77	0,71
Economies of Molarity & Design	ϑ md	-	1,00	0,77	0,63	0,81	0,70	0,64	1,00	0,78	1,00
Economies of Timing	ϑ ct	-	1,00	0,94	0,94	0,94	0,94	0,94	1,00	0,94	1,00
Contingency	%		0,131	0,131	0,131	0,070	0,131	0,183	0,183	0,183	0,250
Point Estimate with Contingency	ci	US$/kWe	3641	4392	4820	4185	3714	5342	3819	4710	8407
Total Overnight Costs		US$	5,64E+09	2,37E+09	2,60E+09	2,95E+09	2,34E+09	3,21E+09	2,33E+09	2,83E+09	5,04E+09
Overnight Cost Distribution		**US$/kWe**	**3545**	**4275**	**4692**	**4223**	**3615**	**5076**	**3629**	**4476**	**7518**
Savings Factor	Ψ	-	0,84	1,01	1,11	0,96	1,04	1,14	1,18	0,99	0,62
Rate of Change	ψ	-	1,53	1,53	1,53	1,43	1,53	1,64	1,64	1,64	1,20
Cost of Debt	ib	%	0,057	0,057	0,057	0,055	0,057	0,059	0,059	0,059	0,052
Fixed Charge Rate	**Fcr**	**-**	**0,111**	**0,111**	**0,111**	**0,111**	**0,111**	**0,112**	**0,112**	**0,112**	**0,109**
Discount Rate	rd	-	0,066	0,066	0,066	0,066	0,066	0,067	0,067	0,067	0,065
Annuity Factor	Fdd	-	0,051	0,051	0,051	0,051	0,051	0,051	0,051	0,051	0,051
Multiple Unit Factor	Fmu	-	1,00	0,80	0,72	0,80	0,75	0,72	0,75	0,80	0,75

Spec.D&D Costs SMR	cdds	US$/kWe	1589	1270	1151	1270	1191	1151	1191	1270	1191
Fixed O&M Costs	**cfom**	**US$/kWe**	**81**	**65**	**59**	**65**	**61**	**59**	**61**	**65**	**61**
Number of Employees	L	#	991	411	411	475	445	434	437	434	434
Specific Labor Costs	CL	US$/kWy	51	61	61	54	57	58	57	58	58
Specific Miscallenous Costs	M	US$/kWy	36	43	43	38	40	40	40	40	40
Variable O&M Costs	**cvom**	**US$/kWy**	**87**	**104**	**104**	**92**	**96**	**98**	**98**	**98**	**98**
Spent Fuel Production	-	tSF/GWy	12,9	24,2	18,9	157,1	481,5	10,4	14,7	0,3	10,9
Specific Spent Fuel Production	Qsmr	tSF/GWy	9,8	18,3	14,3	119,4	365,7	7,9	11,2	0,2	8,3
Number of Seperate Work Units	Nswu	#	7,25	7,25	7,16	0,00	15,13	29,87	30,66	0,00	0,00
Optimization Factor	ϕ	-	1,76	1,76	1,76	-	1,76	1,76	1,76	-	-
Enriched Uranium Input Ratio	Rnu	-	11,33	11,33	11,21	1,00	20,72	37,64	38,54	1,00	1,00
Enrichment Tail	χt	-	0,003	0,003	0,003	-	0,003	0,003	0,003	-	-
Value Function Tail	$V(\chi t)$	-	5,79	5,79	5,79	-	5,79	5,79	5,79	-	-
Value Function Feed	$V(\chi f)$	-	4,87	4,87	4,87	-	4,87	4,87	4,87	-	-
Value Function Product	$V(\chi p)$	-	2,65	2,65	2,66	-	1,91	1,13	1,10	-	-
Carrying Charge (Fissile Material)	Ffi	-	1,12	1,24	1,12	1,07	1,12	1,54	1,32	1,29	1,06
Carrying Charge (Fertile Material)	Ffe	-	-	-	-	-	-	-	-	1,02	-
Front End Fuel Cycle Cost	Cffc	US$/kgHM	2292	2292	2267	369	4224	7790	7981	369	368
Fuel Feeding Costs		US$/kgHM	-	-	-	-	-	-	-	€ 55	-
Interim Storage		0/1	1	0	1	1	1	1	1	0	0
Direct Disposal Costs	Cds	US$/kgHM	349	470	413	1899	5385	322	369	214	328
Fuel Cost	**cf**	**US$/kWh**	**0,005**	**0,008**	**0,007**	**0,048**	**0,011**	**0,010**	**0,015**	**0,001**	**0,0003**

Depreciation Period Interval	Σp<1,nlr>	US$/kWh	5520	6524	6793	8820	5873	7365	6152	6191	9276
Post-Depreciation Interval	Σp<nlr, nltm>	US$/kWh	611	834	774	1729	696	587	1006	357	433
Time Discount Factor	Σt<1, nltm>	-	96.668	96.668	96.668	90.208	89.646	85.066	95.830	85.066	91.481
Levelized Cost Of Electricity		**US$/kWh**	**0,063**	**0,076**	**0,078**	**0,117**	**0,073**	**0,093**	**0,075**	**0,077**	**0,106**