Comparative review of the benefits and flexibility of small modular reactor designs

Prepared by: Keith Featherstone
Student Number: FTHKEI001

Supervisor: Professor CT Gaunt

October 2019

Dissertation presented for the degree of

MPhil degree specialising in Nuclear Technology

Department of Electrical Engineering

University of Cape Town

DECLARATION

I know the meaning of plagiarism and declare that all the work in the document, save for that which is properly acknowledged, is my own. This thesis/dissertation has been submitted to the Turnitin module and I confirm that my supervisor has seen my report and any concerns revealed by such have been resolved with my supervisor.

Signed by candidate

4/11/2019.

Keith Gordon Featherstone

Department of Electrical Engineering,

University of Cape Town,

South Africa

ETHICS ASSESSMENT FORM

Application for Approval of Ethics in Research (EiR) Projects
Faculty of Engineering and the Built Environment, University of Cape Town

APPLICATION FORM

Please Note:
Any person planning to undertake research in the Faculty of Engineering and the Built Environment (EBE) at the University of Cape Town is required to complete this form **before** collecting or analysing data. The objective of submitting this application *prior* to embarking on research is to ensure that the highest ethical standards in research, conducted under the auspices of the EBE Faculty, are met. Please ensure that you have read, and understood the **EBE Ethics in Research Handbook** (available from the UCT EBE, Research Ethics website) prior to completing this application form: http://www.ebe.uct.ac.za/ebe/research/ethics1

APPLICANT'S DETAILS		
Name of principal researcher, student or external applicant		Keith Featherstone
Department		EBE – Electrical Engineering
Preferred email address of applicant:		FeatherK@eskom.co.za
If Student	Your Degree: e.g., MSc, PhD, etc.	MPhil (EM027)
	Credit Value of Research: e.g., 60/120/180/360 etc.	60
	Name of Supervisor (if supervised):	Prof CT Gaunt
If this is a researchcontract, indicate the source of funding/sponsorship		Click here to enter text.
Project Title		Comparative review of the benefits and flexibility of a small modular high temperature reactor design

I hereby undertake to carry out my research in such a way that:

- there is no apparent legal objection to the nature or the method of research; and
- the research will not compromise staff or students or the other responsibilities of the University;
- the stated objective will be achieved, and the findings will have a high degree of validity;
- limitations and alternative interpretations will be considered;
- the findings could be subject to peer review and publicly available; and
- I will comply with the conventions of copyright and avoid any practice that would constitute plagiarism.

SIGNED BY	Full name	Signature	Date
Principal Researcher/ Student/External applicant	KG Featherstone	Signed by candidate	**18 May 2018**

APPLICATION APPROVED BY	Full name	Signature	Date
Supervisor (where applicable)	GAUNT.	Signature Removed	21 May 2018
HOD (or delegated nominee) Final authority for all applicants who have answered NO to all questions in Section1; and for all Undergraduateresearch (Including Honours).	Olabisi Falowo	Signature Removed	06/06/18
Chair : Faculty EIR Committee For applicants other than undergraduate students who have answered YES to any of the above questions.			

Page **1** of **1**

ABSTRACT

Over the past few years, there has been a sustained interest in the development of small modular reactors (SMRs) evident by the number of global initiatives focused on SMR development. This desktop study was performed to review the viability of SMRs based on their benefits and flexibility, focusing predominantly on the light water NuScale and the gas cooled AHTR designs. In assessing the level of safety, the typical general design and safety criteria were reviewed to establish a basis to compare the NuScale and AHTR designs. The need for flexibility to support grid operators and the ability of a nuclear plant to load follow were reviewed to confirm their flexibility. The principal of cogeneration and the feasibility for cogeneration and energy storage with SMRs was explored to determine the potential industrial application. Finally, the technical readiness and uncertainties, the potential market and economic competitiveness of SMRs were reviewed.

The review established that SMRs with safety performance levels exceeding those of current reactor designs are definitely viable. The ability to prevent fuel failure through passive cooling simplifies the design by eliminating the need for complex safety systems and reduces the constraints associated with siting, opening up energy markets where previously nuclear reactors would not have been viable. Their flexibility and the ability to add additional units over time enable them to integrate into any size electrical network and a variety of energy markets. As a clean energy source, SMRs are well suited to support strategies to reduce greenhouse gas emissions and replace fossil-based energy sources. SMRs operating at high temperatures have the added option of considering thermal storage as a means to provide additional flexibility.

The biggest uncertainty in the deployment of SMRs is associated with the regulatory and licencing processes. However, there is a large potential market for SMRs and the lower capital cost per unit, the shorter period until a revenue stream is established and the ability to stagger the financial impact of additional units are expected to make SMRs easier to finance than large nuclear units. This preliminary review concluded that SMRs are definitely viable, but until a SMR design has been successfully licenced, constructed and operated, the uncertainty associated with the licencing of a new technology and the potential for long delays during construction are likely to prevent any large-scale deployment in the near future.

ACKNOWLEDGEMENTS

I am extremely grateful of the advice and assistance provided by my supervisor, Prof CT Gaunt who supported me during my thesis and provided me with the necessary overall direction and insight. In addition, I must thank Dr Steph Steyn and David Nicholls for their guidance in the respective areas of nuclear fuel and the ESKOM PBMR and AHTR research initiatives.

TABLE OF CONTENTS

Page

LIST OF FIGURES AND TABLES

Page

ABBREVIATIONS and ACRONYMS

ACPR	- Advanced Chinese Pressurised Water Reactor
AHTR	- Advanced High Temperature Reactor - ESKOM
ANSI	- American National Standards Institute
APR	- Advanced Power Reactor (Pressurised water) - Korea
AVR	- Arbeitsgemeinschaft Versuchsreaktor - German pebble bed reactor
BWR	- Boiling Water Reactor
C	- Celsius
CAREM	- small modular pressurised water reactor prototype - Argentina
CDF	- core damage frequency
CHF	- critical heat flux
DHR	- diverse heat removal system
DHRHX	- diverse heat removal heat exchanger
DHRIV	- diverse heat removal isolation valve
DNBR	- departure from nucleate boiling ratio
DR	- design requirement
ECCS	- emergency core cooling system
EPR	- European Pressurised Reactor - Framatome
Eskom	- Electricity Supply Commission – South Africa power utility
EU	- European Union
F	- Fahrenheit
F_Q	- heat flux hot channel factor
F_{dH}	- enthalpy rise hot channel factor
FIMA	- fissions per initial (heavy) metal atom
FOAK	- first of a kind
GEN	- generator
GWd/MTU	- gigawatt days per metric ton of uranium
GWe	- gigawatt
HPC	- high pressure compressor
HTGR	- high temperature gas reactor
HTR-PM	- High Temperature (gas) Reactor - China
IAEA	- International Atomic Energy Agency
IHX	- intermediate heat exchanger
IMSR	- Integral Molten Salt Reactor, Terrestrial Energy Inc. - Canada

INL	- Idaho National Laboratory
IRENA	- International Renewable Energy Agency
IRSN	- Institute for Radiation Protection and Nuclear Security
KLT	- Water Cooled, marine based SMR - Russia
LOCA	- loss of coolant accident
LCOE	- levelised cost of energy
LEAD	- Learning and Demonstration Plant (Prior to FOAK)
LPC	- low pressure compressor
LWR	- light water reactor
MFIV	- main feedwater isolation valve
min	- minute
MOU	- memorandum of understanding
MPa	- mega Pascal
MS-HT	- molten salt high temperature tank
MSIV	- main steam isolation valve
MS-LT	- molten salt low temperature tank
MWe	- megawatt electrical
MW	- megawatt
MWt	- megawatt thermal
NEA	- Nuclear Energy Agency
NEI	- Nuclear Energy Institute
NNR	- National Nuclear Regulator (South Africa)
NOAK	- next of a kind
NRC	- Nuclear Regulatory Commission. (USA)
NSSS	- nuclear steam supply system
NuScale	- Small modular pressurised water reactor – NuScale Power - USA
OECD	- Organisation for Economic Co-operation and Development
OR	- operational requirement
PBMR	- pebble bed modular reactor
PCHE	- printed circuit heat exchanger
pcm	- per cent mille (one one-thousandth of a percent)
PCT	- peak cladding temperature
Pn	- power nominal
PWR	- pressurised water reactor
sec	- second
SF	- safety function

SMART	- System-integrated Modular Advanced Reactor (PWR) - Korea
SMR	- Small Modular Reactor
SMR-160	- Small modular pressurised water reactor – Holtec - USA
SVBR-100	- Lead-bismuth cooled reactor - Russia
THTR300	- Thorium High Temperature Reactor - Germany
TRISO	- Tri-structural isotropic particle fuel
US	- United States of America
VHTR	- Very high temperature (gas) reactor
VVER	- series of pressurised water reactors developed in Russia
WNA	- World Nuclear Association
WNN	- World Nuclear News

CHAPTER ONE

INTRODUCTION

1.1 Introduction

There has been an increase in global interest in small modular reactors due to their postulated benefits. Their smaller size and flexibility are claimed to be the solution for integration into networks with alternative energy sources, including those with high penetration of renewables, as they can be deployed either as a single module or in a multi-module plant with suitable load following capabilities.

By contrast, the generating capacity (electrical output) of commercial nuclear power plants has tended to increase over time. The capacity of the initial commercial nuclear power plant designs were less than 100 MWe (Calder Hall commissioned in 1956 was 49 MWe) but this escalated quickly and by the mid-1980s nuclear power plants were typically rated around 1000 MWe, while the latest standard plant designs being built are significantly larger [IAEA, 2017]. This was not specific to a reactor type, but was evident across all the commercial nuclear reactor technologies. Some of the reactors that are currently being constructed around the world include the French EPR (1600 MWe), the Russian VVER (1200 MWe) and the Korean APR (1400 MWe).

The increase in capacity was a natural economic response in what could be referred to as the classical centralised generation public utility era. The overnight capital cost per megawatt of installed capacity could be reduced for larger units, which together with the low primary energy cost for nuclear fuel resulted in lower marginal cost of electricity produced from larger nuclear plants. It was therefore more cost effective to build larger plants, which the large centralised utilities could afford, but the plants had to be operated with a high utilization factor to recover the significant capital investment required, often referred to as “base load” operation. This also resulted in the designs of most of the nuclear plants that are in service being focused and optimised for base load operation.

This approach to constructing large nuclear power plants is still being followed in regions where there is either: a clear policy decision associated with energy security or greenhouse gas reduction, a specific national energy strategy; or a large unserved or growing demand for energy. This is evident in countries like China, India, Russia, Korea, United Arab

Emirates and Turkey, which are examples of countries currently constructing large nuclear plants, either by the centralised generating utility or under a signed power purchase agreement.

In other regions, the future of existing nuclear power plants is continuously being questioned. Unlike the Chernobyl accident that occurred in April 1986, there was a much larger international political reaction, driven by environmental concerns, following the Fukushima accident of March 2011. By May 2012, Japan had shut down all 54 nuclear power plants for safety assessments. Only nine had returned to service by December 2018 [WNA, 2019], complying with the more stringent Japanese regulations that took effect after the Fukushima accident. In addition, Japan, in its latest Energy Plan, approved July 2018, has committed to target the percentage energy from nuclear to 20-22% in the 2030 energy mix [WNA, 2019]. Germany reversed its nuclear policy and announced in May 2011 that all German nuclear power plants will be shut down by 2022, seven of which were still in operation in 2018. As a result, Germany remains dependant on coal to generate a third of its energy and is off track in being able to meet its 2020 emission targets [Tamma, 2018]. France committed in July 2015 to transition their energy mix, reducing the contribution of nuclear from over 75% to 50% by 2025 [WNA, 2016]. French President Emmanuel Macron, who was elected on a program to reduce the country's reliance on nuclear, has since backtracked on this commitment, and is using the increase in carbon dioxide releases as the reason. Other governments are also being cautious about making firm commitments on the phase out or reduction of nuclear as an energy source [Tamma, 2018].

In addition, the world focus on greenhouse gas reduction and their respective carbon footprints, has led to the significant deployment of renewables (primarily wind). Many governments introduced subsidies in various forms to incentivise this shift to renewable energy sources like wind and solar, while the benefit of nuclear as a zero carbon emitting energy source was generally ignored. These renewable sources are totally dependent on the weather patterns, making them “not-dispatchable” as they are not available to the system operator for maintaining the balance between generation and demand. As a result, additional flexible dispatchable generation sources are required to manage the intermittency of the renewable generators (such as hydro, gas, coal or nuclear) and /or forms of energy storage (such as batteries, thermal storage, or pump storage).

If renewables operated in a wholesale market without subsidised prices, their economic viability would be less obvious as the open market price for electricity would be lowest when all the renewables are operating. There is therefore definitely a threshold level of penetration of renewables that a network can accommodate, especially if they are paid subsidies, to ensure ongoing reliability and sustainability. This essentially is what happened in South Australia which ended up in prolonged power outages when their substantial wind generation capacity was unavailable due to unseasonal weather changes. The excessive wind generating capacity was driven by lucrative feed-in tariffs that over time resulted in the closure of most of the alternative flexible generation capacity due to financial non-viability. The recovery has required the installation of significant flexible generation and storage options and has significantly driven up the price of electricity in the region. On 31 May 2018, China's National Development and Reform Commission, Ministry of Finance and National Energy Board issued a statement [NDRC, 2018] stopping subsidies for utility-scale solar projects and reducing feed-in-tariffs [Baker J, 2018]. During July 2018 in Ontario Canada, the Conservative government stopped over 700 clean energy contracts that were in the early stages, in order to save the province's ratepayers millions [Green Tech Media, 2018].

Most nuclear power plants have traditionally been operated as baseload sources due to their high fixed costs and low variable costs. Adjustments to maintain the balance between generation and demand were left to other technologies, in most cases with lower fixed costs but generally much higher variable costs like gas plants (pumped storage and coal in the case of South Africa). This situation is no longer the norm and in many countries, the old traditional baseload plants are now required to be flexible to be able to change their output to compensate for variations in power demand and the availability of intermittent generating sources. In regions with intensive renewables generating resources, large nuclear plants are also at risk of becoming stranded, financially unviable assets due to the economics associated with not operating in baseload mode. Generally, there are no tariff adjustments for the benefit that large nuclear power plants actually provide associated with the reliability in supply, the lack of carbon emission and the ability to provide flexible capacity. This was evident following the US District of Columbia Capacity Auction held in May 2018 where policy makers in Ohio and Pennsylvania have been urged to take action to prevent the premature closure of large baseload nuclear power plants [NEI, 2018].

It is a general misconception that nuclear power plants are technically incapable of ramping and load following. In fact, most nuclear power plants were designed to have relatively fast ramp rates and the ability to operate at different power levels. However, besides the economic benefit, operating constantly at full power is simpler and less demanding on the fuel and plant equipment. The ramping and load following capabilities of a plant are typically determined by regulatory limits and requirements associated with the fuel and the design of the reactor core. France is an example where nuclear generating capacity contributes more that 70% of the installed capacity and load-following is actively performed by nuclear units which reduce their output on a daily basis to meet the load profile [Lokhov, 2011, page 18]. It demonstrates that nuclear plants are able to operate through a range of their total capacity, though most of the older designs cannot operate for prolonged periods at low power without restrictions on the rate at which they can return to full power. This ability to reduce power and return back to original power in a short time frame is well suited for daily load following and integration with solar plants where the output varies on a daily basis but not for wind which can blow strongly for days.

It would appear, therefore, that in regions striving to meet the world Sustainable Development Goal number 7 (affordable and clean energy), there is definitely a role for large baseload nuclear power plants with their high reliability, to replace aging fossil plants, especially as nuclear is the lowest carbon emitter of all the traditional dispatchable energy sources.

In regions where there is high penetration of renewables, predominantly wind, there is a need for technologies that are more flexible over a longer time period, which according to the NEA is ideally suited to small modular nuclear power plants, as their advanced designs provide options for co-generation to improve their economics in these emerging markets [NEA, 2016, page 10].

However, although nuclear remains a very attractive clean energy source without carbon dioxide emissions, there remains significant opposition to nuclear in any form, which needs to be considered for all new nuclear projects.

1.2 Small Modular Reactor Technology Trend

Small Modular Reactors (SMRs) are characterised by their lower output, typically 300 MWe or less, and a modular design that enables major components to be fabricated under factory conditions and transported as a complete module to the site for installation, achieving the economic benefits of series production and short construction times [IAEA, 2018(a) page 8], [NEA, 2016, page 9].

Over the past few years, there has been a sustained interest in the development of SMRs, evident by the number of global initiatives focused on their development. Advanced SMR designs are claimed to offer many advantages and cater for more diverse markets than those in which large nuclear power plants operate. Their smaller size makes them attractive for developing countries with small electrical grids and the supply can increased over time by adding additional units as the demand increases. Multi-module versions of SMRs can be considered comparable to large nuclear units by providing the same overall capacity and may offer the generating flexibility that energy markets with large amounts of renewable capacity require. They have the added potential to be economically competitive by constructing in a series approach taking advantage of their less complex design, shorter construction durations, the accelerated learning rates and the short time span within which the first units become operational. The number of companies and private investors showing an interest in SMRs indicates a shift from the traditional government (state owned utilities) indicating some form of entrepreneurial goal stemming from their potential economic benefit and the predicted demand.

Globally there are more than 50 initiatives focused on SMR development, but only a handful of demonstration SMRs are under construction and about a dozen are likely to be deployable in the near term as their development is well advanced [IAEA, 2018(a)]. Although many of the concepts have not yet actually been finalised, they all can be classified into one of the following three generic types: light water reactors (PWR), liquid metal or molten salt reactors, and graphite moderated high temperature gas reactors.

Light water SMRs are moderated and cooled by ordinary water and have the lowest technological risk as they are similar to the majority of nuclear reactors currently in operation. The proposed fuel assemblies, although different in size are similar to existing fuel designs so no major development of the fuel is required. Within this type there are two variations: those that have a more conventional approach with a pressure vessel and external steam generators (e.g. KLT - Russia, ACPR - China, and SMR-160 - US), and those with an integral design where the complete steam supply system is inside the reactor pressure vessel (e.g. NuScale – US, CAREM – Argentina, and SMART - Korea).

Liquid metal or molten salt reactors are those cooled by a liquid metal or molten salt that is a solid at room temperatures but has high thermal conductivity and high boiling points such as sodium, lead-bismuth or a form of chloride or fluoride salt. The majority of these reactors do not have a moderator and operate on fast neutrons which makes them typically smaller and simpler that the light water variants. They typically have better fuel performance and can operate for longer periods between refuelling, although they require fuel enrichment between 15 to 20%, which is usually derived from blending down weapons grade plutonium or reprocessing used light water reactor fuel. There is a unique variant of the molten salt reactor type where the fuel is dissolved in the coolant (molten salt) and only reacts as it flows through the "core" which is designed with a moderator to enable the nuclear reaction to occur [IAEA, 2018(a), page 220]. All these SMRs have core outlet temperatures in excess of 500º C and operate at or near atmospheric pressure. Although historically there have been numerous research reactor prototypes of this type, a revised licencing framework and safety case approach is likely to be needed. Examples of SMR projects of this type include ThorCon (international consortium), the SVBR-100 (Russia) and the "Stable salt Reactor" (United Kingdom) while the IMSR (Canada) is a molten salt variant with the fuel dissolved in the coolant.

As the name implies, graphite moderated high temperature gas reactors are designed with graphite as the moderator and an inert gas, typically helium, as the coolant. The fuel is in the form of TRISO (tri-structural isotropic) coated particles combined with graphite and silicon carbide into "pebbles" (balls) or "prism" blocks which are stable to over 1600º C. The capability of operating at high coolant gas outlet temperatures of up to 1000º C, provides the option of coupling the reactor directly to a Brayton cycle gas turbine generator set with possible efficiencies in excess of 50%. High temperature gas reactors have in the past been built and operated successfully for many years [Scheuermann et al, 2017, page 13] such as Peach Bottom Unit 1 (Pennsylvania), a 115 MWt plant that operated from 1966 to 1974 and Fort St Vrain (Colorado) a 842 MWt plant which operated from 1974 to 1989. Both were helium-cooled and used prism block fuel in the reactor. In Germany the AVR (Jülich) which operated from 1966 to 1988 at 46 MWt and the THTR300 (Schmehausen) which operated from 1983 to 1989 at 750 MWt, were also helium-cooled but used pebble fuel. Although there is a recent revival in interest, a revised licencing framework and safety case approach will be needed to address the desired higher temperatures. The HTR-PM (230 MWe) is in the final stages of construction in China, and the AHTR concept is being considered by Eskom after shelving the PBMR design due to the inability to obtain a committed end user.

There are clearly numerous different SMR designs, each with its own unique characteristics and associated advantages and challenges, but they all share similar concepts and claim to have the benefit of passive safety. The question remains whether the benefits and flexibility of small modular reactor designs make them viable. This paper examines this question, focusing predominantly on the strengths of SMRs because they have not been documented in the context of energy policies for countries like South Africa and to limit the scope of this minor dissertation.

Two examples of SMR technologies have been chosen for comparison: the light water NuScale concept that is easily compared with a large PWR and the Eskom AHTR concept with increased peaking capability through the use of thermal energy storage.

1.3 Key research questions

Considering the background described above, the aim of the research addressed in this dissertation is to try and answer the questions on the viability of SMRs by exploring the following aspects:

- Can the passive design features of SMRs result in an improved level of safety, what are the associated benefits and how does it impact the SMR design, licencing, and costs?
- What are the implications and constraints on the fuel and reactor core associated with load follow operation?
- Are there viable co-generation options for SMRs and is thermal storage a viable consideration?
- What are the technological uncertainties and economic considerations associated with SMRs?

1.4 Research methodology and structure of the dissertation

The research was a desktop study that entailed the identification and analysis of relevant and available literature and research material. The report is divided into six chapters with chapter one providing the background to the key research questions and the framework of the report. A separate chapter is then dedicated to each of the main research questions. The criteria on which the comparison is performed are identified throughout the text and included in brackets, for example {Safety Function 2.1}.

Chapter two outlines the typical internationally accepted approach required to demonstrate safety of a Nuclear Power Plant, and defines the requirements against which a comparison of the NuScale and AHTR designs is performed. This chapter also explores the unique features of the NuScale and AHTR designs, the associated safety benefits, and concludes that both concepts, based on the currently available information, can be considered credible designs with enhanced passive safety features.

Chapter three provides an overview of the load following requirements, the impact that load following has on the nuclear reactor, the design features and operational controls that are applied. It concludes that modern reactor designs can provide large operational flexibility. The level of flexibility required by a system operator can have a significant impact on the load factor of the reactor. Alternate means of co-generation to enable the reactor to operate at a higher load factor can improve the financial viability of SMRs.

Chapter four explores the feasibility of cogeneration as a means to improve the economic viability of a SMR by supplying both electricity and thermal energy for industrial applications. The literature search identified reports and technical assessments that demonstrate the feasibility of SMRs to cogenerate process steam to compensate for the variability in the electrical output demanded by the grid. The practicality of cogeneration is specific to each application, but certain conditions must be present. These include the existence of a market for both products, the ability to co-site the nuclear and industrial facilities, the compatibility of the life-spans of both facilities and the impact of the variability of each product on the production of the other. One of the more complex issues is the number of stakeholders involved (Regulator, Plant Operator, System Operator, Process steam user, etc.). The AHTR concept of storing excess thermal energy that is later used to generate electricity has unique advantages as it has the potential to maximise the reactor load factor and its flexibility for the system operator, but it must still be demonstrated to be financially viable.

Chapter five reviews the technological uncertainties and the economic viability of SMRs. The literature indicates a large potential market for SMRs and there does not appear to be any technological barriers for the NuScale and HTGR concepts. There would appear to be an upper temperature limit of around 1000^{o} C for HTGR designs, which could influence the AHTR concept. The future of small modular reactors will ultimately be determined by market competition and the successful licencing and operation of FOAK plants.

The smaller unit size, the significantly fewer components, and the shorter construction period of SMRs will reduce the overnight capital costs required per unit if compared with a large conventional nuclear unit. This lower capital cost together with the shorter period until a revenue stream is established and the ability to stagger the financial impact of additional units is expected to make both individual SMRs and multi-unit SMR sites easier to finance than large nuclear units.

However, most parties are cautious about being the first to invest in a first-of-a-kind technology, and want them to be built and tested elsewhere. Until a SMR design has been successfully licenced, constructed and operated, the uncertainty associated with the licencing of a new technology and the potential for long delays is likely to prevent any large-scale deployment in the near future.

Chapter six discusses the results of the previous chapters and draws the conclusion that SMRs are definitely viable. They have safety performance levels far exceeding those of current reactor designs. Their flexibility and the ability to add additional units over time, enables them to integrate into any size electrical network and a variety of energy markets. In addition, as a clean energy source SMRs are also well suited to support strategies to reduce greenhouse gas emissions. Although this is only a preliminary review, and further study and assessment is warranted, it concludes that the benefits associated with SMRs, should have an important role to play in the future of nuclear energy. The actual deployment of any SMR design will however rely on the successful licencing, construction and operation of a FOAK plant and their economic viability within the target energy market.

CHAPTER TWO

DESIGN SAFETY FEATURES OF SMALL MODULAR REACTORS

In this chapter, the typical general design criteria for safety are reviewed and the safety features of the NuScale and AHTR concepts are assessed, leading to a comparison of each in terms of the general design criteria.

2.1 Review of the typical Regulatory Safety Criteria

The regulatory safety requirements governing the design and operation of nuclear reactors have always focused on the ability through safety analysis to demonstrate the overall safety of a plant and provide reasonable assurance that the plant can be operated without undue risk to the health and safety of the public. This requires the operator to demonstrate through analysis that the associated safety criteria for all anticipated plant conditions based on their frequency of occurrence and consequence, are respected.

The American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurised Water Reactor Plants [ANSI, 1973, page 2] expanded on the General Design criteria for PWRs and defined four categories of occurrences that can be summarised as:

- Condition I - Normal Operation: Occurrences that are anticipated to occur regularly during plant operation, which the plant must accommodate without the need for automatic or manual protective action.
- Condition II - Incidents of Moderate Frequency: Occurrences that are not anticipated to occur more than once a year, which the plant must be designed to tolerate, by shutting down if necessary, and being capable of returning to service.
- Condition III - Infrequent Incidents: Occurrences that could occur during the life of a plant. If they occur, the plant design must limit the damage to a small fraction of the fuel elements, and with no impact on the surrounding area.
- Condition IV - Limiting Faults: Occurrences that are not expected to occur, but are considered due to the potential significance of their occurrence. The plant response to these occurrences must limit the release of radioactive material to within regulatory limits.

The analysis of Condition I occurrences demonstrates that a plant can operate with the defined flexibility without challenging any protection set points. The analysis of Condition II and III occurrences verifies the design assumptions for the reactor protection system including the appropriateness of the set point values. The analysis of Condition IV occurrences, the most drastic conditions that must be designed against, represent the limiting design case and they verify the design of the required safeguard systems.

The roles of the required safeguard systems [Rasmussen, 1975, page 25] are to perform the following Safety Functions (SF):

- Reactor Trip: to stop the fission process, terminating additional heat input. {SF 2.1 - Control of reactivity}
- Emergency Core Cooling: to cool the core, thereby maintaining the core geometry and keeping the release of radioactive material from the fuel to low levels. {SF 2.2 - Control of heat removal}
- Containment Integrity: to reduce the radioactivity released from the fuel into the containment and prevent the radioactivity within the containment from being dispersed into the environment. {SF 2.3 - Control / containment of radioactive material}

The likelihood of a reactor vessel failure is considered negligible due to the high quality requirements applied during the design, manufacture and operation. However, due to the severity of the consequences of a postulated loss-of-coolant accident (LOCA) from a leak or break in the primary circuit piping external to the reactor vessel, the US general design criteria included a specific requirement [US NRC, 2019, section 46] that light-water reactors plant designs using uranium oxide fuel contained within cylindrical zircaloy cladding must include an emergency core cooling system (ECCS). In the late 1970s when these requirements were originally defined, it was generally accepted that an ECCS was necessary to minimise fuel failure and to avoid fuel dispersion in order to minimise the radiological risk to the public. This resulted in specific performance criteria for the emergency core cooling system response. It also specified criteria that can be considered an additional safety function associated with the maximum amount of cladding oxidization and hydrogen generation from the chemical reaction of the cladding with water or steam {SF 2.4 - Control of chemical attack}.

Although the regulatory environments differ between countries, most countries adopted the approach developed in the US by the American Energy Corporation (now called the Nuclear Regulatory Commission) in 1973 [NEA, 2003, page 45]. In order to demonstrate adequate safety and to comply with the general design requirements [US NRC, 2019, Appendix A] and the internationally recommended acceptance criteria [IAEA, 2003, Page 8], reactor designs had to include many redundant safety systems, including:

- Redundant systems to ensure emergency core cooling {Design Requirement (DR) 2.1 - Redundant safety and safeguard systems};
- A containment building to contain the possible release of all the energy stored in the plant systems and to limit the spread of radioactive material {DR 2.2 - Containment building};
- Redundant onsite backup electrical systems as the safeguard systems required electrical power to operate {DR 2.3 - Redundant on-site Ac power sources};
- Measures to address all the natural phenomena and environmental conditions associated with the site {DR 2.4 - External hazards}.

This deterministic approach is well defined in specific design codes and standards and encouraged through the industry guidance on Safety Analysis for Nuclear Power Plants [IAEA, 2009].

The IAEA recommended target core damage frequency {DR 2.5 – Probabilistic risk frequencies} for existing nuclear power plants is a frequency of occurrence of severe core damage that is below 10E–4 events per plant operating year and for future plants a goal of not more than 10E–5 severe core damage events per plant operating year [IAEA, 1999, page 11]. The definition of what constituted core damage differs considerably with reactor technology. For example for light-water nuclear power reactors core damage is defined as a local fuel temperature above 1204º C, the acceptance criteria for the ECCS operation [US NRC, 2019, Section 46 1b]. Others have more general definitions such as prolonged core uncover or long-term cooling. The limits associated with core damage event frequency required by different regulators vary between 10E-4 and 10E-6 per year {DR 2.5a} but the requirements for new plants are consistently stricter than for existing ones, and the use of a probabilistic risk (safety) assessment is becoming mandatory as opposed to indicative [SSM, 2011, page 23]. Similarly, the regulatory targets for the probability of a large early release event occurring vary between 10E-4 and 10E-7 per year {DR 2.5b}.

The Fukushima Daiichi accident demonstrated the need for reliable safety system performance for extended periods of time, especially residual heat removal, and the need to prevent off-site releases once fuel failure has occurred. It also resulted in the "Design Extension Condition" concept further improving the safety of nuclear plants by considering additional postulated accident conditions and multiple equipment failures that are not considered as part of the design basis [IAEA, 2016, page 24]. These requirements are in principle addressed through the established design safety requirements, but emphasise the importance of the long-term availability of the safety and safeguard systems {DR 2.6 – Assured long term cooling}.

Due to the unique characteristics associated with most SMR designs, the licencing and certification of the SMR design will have to justify the ability of the plant to meet the generally accepted safety functions and safety requirements, and where necessary, justify new methodologies and appropriate acceptance criteria.

2.2 Typical Operational and Design Criteria

The design of a nuclear power plant must ensure the ability to remove the thermal energy (heat) generated within the fuel during the fission process. The thermal energy is transferred to the coolant as it circulates through the reactor core, resulting in a temperature increase in the coolant. A temperature gradient from the centre of the fuel through the fuel cladding and to the cooling medium is required for the transfer to occur. This results in a temperature profile within the fuel rods, with the fuel centre experiencing the highest temperature. The main contributors to the resultant fuel temperature are the amount of heat being generated in the fuel, the ability of the coolant to remove the heat and the thermal conductivity of the fuel and the cladding material. If the fuel temperature exceeds the fuel melting temperature or the cladding material exceeds its temperature limits, fuel failure can occur providing a pathway for fission products to be released from the fuel. The amount of failed fuel rods, referred to as the failed fuel fraction [DR 2.7 - Fuel failure fraction}, defines the inventory of gaseous and solid fission products that could be released to the environment.

If the coolant does not remove all the heat generated in the core, the temperature of the fuel and the cladding will rise. In a PWR when inadequate cooling occurs, a water/steam mixture forms within the coolant. The mixing produced by the steam bubbles in the coolant initially improves the heat transfer. However, above a critical heat flux (CHF) level the bubbles coalesce forming a continuous steam film against the cladding that severely decreases the heat transfer capability. This departure from nucleate boiling can occur anywhere in the core as the amount of heat produced by the fuel varies both radially and axially throughout the core in relation to the flux shape [Olander, 2009].

The radial flux profile can vary significantly due to the influence that the enrichment, burnup history and location of each individual fuel rod in the core can have on the power that a rod produces [IAEA, 2005, page 46]. In the axial direction, the coolant flow subjects the fuel rods to a temperature gradient along their length, influencing the temperature related reactivity coefficients and the power being produced. In addition, the position of the control rods can significantly alter the radial and axial flux shapes due to their high neutron absorbing characteristics.

The safety analysis of the reactor core must demonstrate that the design criteria associated with fuel integrity for the different categories of occurrences are respected throughout the reactor core [ANSI, 1973]. Many of the inputs to the safety analysis involve fuel and reactor operating parameters that vary over time. As a result, the input assumptions that are used in the safety analysis are translated into Operational Requirements that must either be validated during the design of the core for each operating cycle, or be physical operational limitations to ensure the plant is always operated within the design and the safety analysis.

Condition 1 and II occurrences must not result in any damage to the fuel cladding [ASME, 1973, page3]. To demonstrate this the reactor coolant system must be able to prevent the CHF being reached anywhere in the core, with sufficient margin. The margin associated with the CHF is calculated on local conditions and is expressed as the departure from nucleate boiling ratio (DNBR). DNBR is the ratio of the heat flux needed to cause departure from nucleate boiling to the actual local heat flux of a fuel rod at a specific location. In order to achieve this, core design limits are established for overall core power, the maximum variation in local power distribution and the minimum DNBR [IAEA, 2005]. The local power distribution or peaking limits are expressed as limits on total heat flux and enthalpy rise:

- The heat flux hot channel factor (F_Q) is the ratio of maximum localised heat flux on the surface of a fuel rod anywhere in the core to the average heat flux of all the rods in the core.

- The enthalpy rise hot channel factor, (F_{dH}) is the ratio of power generated by the rod producing the most power anywhere in the core to the average power generated by all the rods in the core.

Based on the above, to prevent fuel damage and plant operation outside the configuration defined by the safety analysis assumptions, Operational Requirements for a PWR include the position of the control rods {OR 2.1 - Control rod insertion limits}, the neutron flux shape {OR 2.2 - Axial flux profile}, fuel peaking limits {OR 2.3 - Peaking limits} and a minimum DNBR ratio {OR 2.4 - DNBR limit}.

Condition III and IV occurrences can result in a limited amount of fuel damage. For condition III occurrences no more that 5% of the fuel rods can exceed the DNBR limit, all the cladding must respect the fuel rod cladding temperature limit and no fuel can exceed the fuel centreline melting temperature. For Condition IV occurrences (non-LOCA), up to 10% of the fuel rods can exceed the DNBR limit, all the cladding must respect the fuel rod cladding temperature limit, and 10% of the fuel by volume can exceed the fuel centreline melting temperature. These criteria prevent fuel cladding embrittlement and substantial volumetric changes of the fuel [Westinghouse, 1975, page 1-5]. The temperature at the centre of the fuel pellet at which fuel melting is assumed to occur is above 2644^{o} C {OR 2.5 - Fuel centre line temperature limit} and the peak fuel rod cladding temperature limit {OR 2.6a - Peak cladding temperature - non LOCA} is typically 1480^{o} C (2700^{o} F) [IAEA 2003, page 8].

The assessment of a LOCA for light water reactors (i.e. PWRs and BWRs) using fuel with zircaloy cladding must ensure the most severe postulated occurrence is analysed. In a PWR this involves a double-ended guillotine break of a primary coolant pipe between the reactor vessel and one of the main circulating pumps. The evolution of this LOCA (PWR) and the impact on the fuel [NEA, 2009, page 43] can be summarised as follows. Almost immediately after of the break occurs, departure from nucleate boiling occurs as the coolant is expelled from the reactor vessel. As the coolant is also the moderator, the loss of water from the core adds negative reactivity that rapidly shuts down the nuclear reaction. This initial blow down phase can last up to 30 seconds, until the interactions between the primary circuit pumps, the high pressure safety injection system, the accumulators and the break dynamics result in some water starting to re-enter the lower part of the reactor vessel. With the diminished cooling, the temperature profile within the fuel changes as the cladding heats up. During the refill period, while the lower part of the reactor vessel is filling with water, the decay heat being generated within the fuel causes the fuel to heat up in an adiabatic mode as there is no meaningful heat removal mechanism. The re-flood period starts after about 40 seconds from the time of the break when the core begins to fill from the bottom due to the accumulation of the water injected into the reactor. As the water level rises, the fuel rods are quenched from the bottom up. This generates a two-phase mixture and the rising steam provides cooling to the fuel rod surface area above the quench front. Eventually there is enough cooling to prevent further increase in cladding temperature, and the Peak Cladding Temperature (PCT) is reached. From here on, the temperature will continue to decrease as long as cooling capable of removing the decay heat is maintained.

During LOCA conditions other mechanisms can lead to fuel failure rate [NEA, 2009, page 31 and 33]. Cladding embrittlement can occur due to hydrogen pickup leading to cladding failure or fragmentation due to the stresses induced during the quenching phase. In addition, the zircaloy cladding oxidisation rate may increase significantly in the steam-water environment to the point that it becomes autocatalytic, leading to cladding and fuel melt. These fuel failure mechanisms can impact the geometry and the ability to cool the core. As a result, more penalising safety design criteria were established for LOCA conditions, the origins and rationale of which are explained in the NEA report [NEA, 2009, page 27], which can be summarised as follows:

- Peak cladding temperature {OR 2.6b - Peak cladding temperature, LOCA}. The fuel cladding temperature shall not exceed 1204^{o} C.
- Maximum cladding oxidation {subset of SF 2.4}. The total oxidation of the cladding shall not exceed 17% of the original cladding thickness.
- Maximum hydrogen generation {subset of SF 2.4}. The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the amount that would be generated if all of the metal in the cladding were to react.
- Coolable geometry {subset of SF 2.2}. The core geometry shall remain in a condition that allows cooling.
- Long-term cooling {same as DR 2.6}. The systems designed to remove the decay heat must be able to maintain the core temperature at an acceptably low value for an extended period of time.

The literature review confirms the stringent regulatory requirements associated with nuclear power plant designs. Specific regulations exist for common reactor types that dictate minimum safety functions and design criteria. Where they do not exist, new methodologies and appropriate acceptance criteria need to be agreed with the regulator. The safety analysis of the different categories of occurrences must verify that the criteria are respected. Operational requirements ensure the input assumptions used in these analyses are respected at all times during plant operations, and determine the operational flexibility of the plant.

2.3 Review of the safety aspects of the NuScale SMR design

An overview of the NuScale design was obtained from the Final Safety Analysis Report submitted as part of the application for design certification [NuScale, 2018, chapter 4]. The plant design is based on a pressurised light water reactor with a unique primary system design. The complete primary circuit is contained within the reactor vessel and the coolant circulates by natural convection eliminating the need for primary pumps and large primary pipes external to the reactor. The complete integrated reactor vessel is housed within a metal containment vessel with minimal penetrations, see figure 1. Both the containment and reactor vessels are factory manufactured. The complete Nuclear Steam Supply System (NSSS) is thus pre-manufactured, shipped to site and the complete module is installed in a reactor pool that is filled with water.

The reactor pool acts as the ultimate heat sink, and is constructed below ground level for maximum strength and security and to minimise the impact of external environmental events. The steam conditions delivered from the NSSS module can be coupled to a standard steam turbine generator set or any other steam cycle application. During normal operation, the space between the reactor vessel and the containment is maintained under vacuum to reduce the heat loss from the reactor vessel and to eliminate the need for any insulation material, thereby removing the industry concern associated with blockages due to lagging material during accident conditions.

The NuScale plant is designed with a reactor thermal power rating of 160 MWt producing a gross electrical output of approximately 50 MWe. The reactor system is designed to operate at a system pressure of 12.7 MPa, an average core inlet temperature of 258.3^{o} C and an average temperature rise in the core of 37.8^{o} C [NuScale, 2018, chapter 1, page 44]. It delivers superheated steam at a pressure of 3.4 MPa and temperature of 306^{o} C, at rated output, with a feed water temperature of 149° C [NuScale, 2018, chapter 10, page 12]. The efficiency of the NuScale plant, due to the primary system operating conditions, will be similar to that of a standard PWR, in the order of 33%.

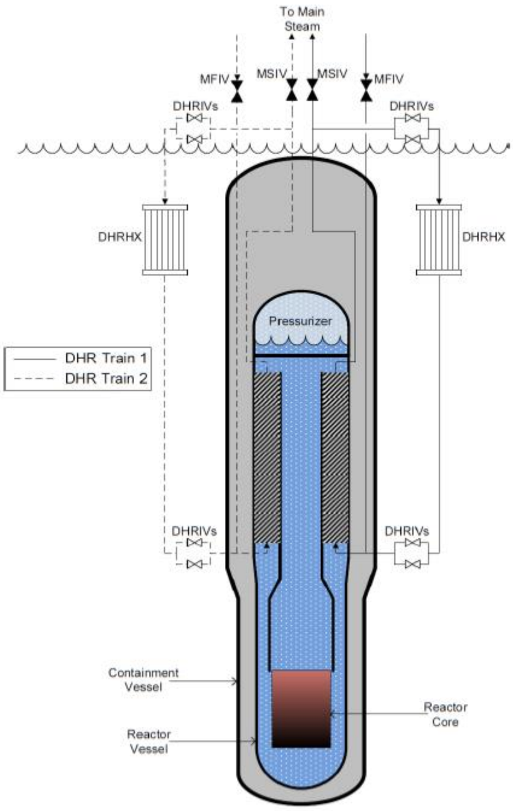


Figure 1 - NuScale reactor vessel and containment vessel schematic
with main steam, feedwater and Diverse Heat Removal (DHR) connections [NuScale, 2011]

The reactor core consists of 37 fuel assemblies similar in design to the 17x17 design fuel assemblies used in PWRs but shorter. The plant uses water as the coolant and moderator, and soluble boron is added to the coolant for reactivity control to compensate for fuel burnup. There are 16 control rods organized in two banks: a shutdown bank used during shutdown and reactor trip events and a regulating bank used during normal plant operation to control reactivity. The fuel cycles are designed for a nominal 2-year length with a fuel enrichment of 4.95% and a maximum fuel burnup of 62 GWd/MTU. The core is surrounded by a neutron reflector which allows fresh fuel assemblies to be placed on the periphery of the core, to reduce the peaking factors, without compromising neutron utilization.

As there is no large bore primary piping outside the reactor vessel, there is no potential initiator that can lead to a LOCA situation. The required emergency core cooling system (ECCS) is designed to initiate a LOCA by distributing the coolant between the reactor vessel and the containment vessel while maintaining the core covered and cooled at all times. The

ECCS consists of three reactor vent valves mounted on the head of the reactor pressure vessel and two reactor recirculation valves mounted on the side of the reactor vessel in the down-comer region at a height above that of the core, see figure 2.

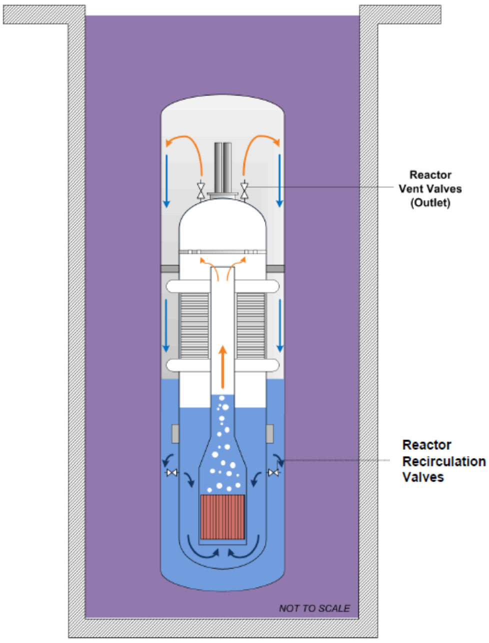


Figure 2 - NuScale emergency core cooling system schematic [NuScale, 2011[

All five ECCS valves are closed during normal operation and open during accident conditions. Core heat is removed through boiling in the core and the water that is vaporized leaves the reactor vessel as steam through the reactor vent valves. The steam is condensed and collected in the containment, and returned to the downcomer region inside the reactor vessel through the reactor recirculation valves. The containment is sized such that the displacement of liquid from the reactor vessel into the containment establishes a liquid level above the reactor recirculation valves and the top of the core, establishing a natural closed circulation loop and keeping the core covered at all times. The natural circulation loop removes decay and residual heat from the core and reactor vessel into the containment. The heat in the containment is then transferred by conduction and convection to the water of the reactor pool.

As a result, during ECCS operation, there is no actual loss of water inventory from the closed natural circulation loop established between the reactor vessel and the containment. The reactor pool water volume provides cooling for over 30 days by which time the decay heat would have reduced to a level that can be removed by convection air cooling of the containment vessel, see figure 3. With the exception of the actuation of the valves, the system is completely passive and does not rely on inventory makeup from an external source or the availability of electric supplies.

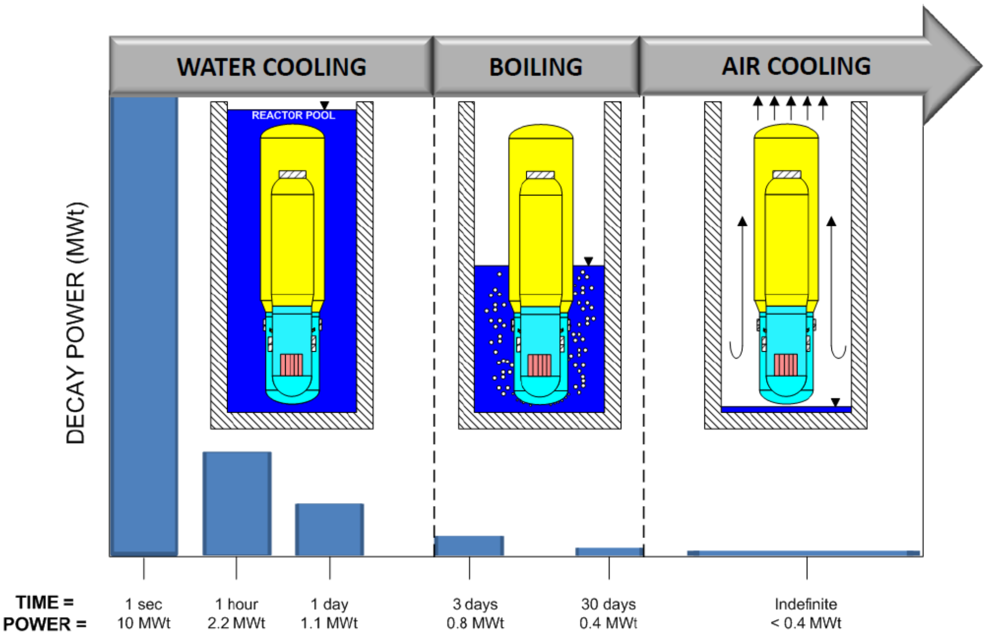


Figure 3 – NuScale long term accident cooling evolution [NuScale, 2011]

The approach taken in the design of the NuScale plant has focused on eliminating initiating events, rather than managing symptoms and demonstrating that the impact on the public is acceptable. The design of the plant does not have a credible LOCA initiating event (large break resulting in a loss of reactor coolant) due to the integrated reactor vessel. The reactor vessel and containment vessel interaction during ECCS operation ensures that the core remains covered with no loss of coolant inventory thereby maintaining an adequate DNBR. As a result, the core temperature can be maintained at an acceptably low level and the decay heat can be removed indefinitely in this passive manner.

The approach of maintaining primary inventory and DNBR (margin to the CHF) ensures that the LOCA limits for PCT, oxidation, and hydrogen production are not violated, as they are temperature driven and require the fuel to be uncovered. This also removes challenges to fuel integrity associated with a LOCA, eliminating changes in core geometry from fuel failure that would prevent the core from being amenable to cooling.

The maximum hypothetical fission product source term used to assess the public impact of a nuclear plant has historically been linked to LOCA as it was the most penalising accident sequence. As the NuScale design does not have a LOCA basis event that results in core damage, alternate postulations to define the source term are required [NuScale, 2015, page 2] the origin for the maximum postulated source term that can be released must focus more on other events, fuel handling type accidents and operational level releases.

In summary, the NuScale design includes features focused on the prevention of severe accidents, preventing fuel damage and reducing the susceptibility of the plant to external events. This must translate into a significant reduction in the core damage frequency (CDF) and the large release fraction per module when compared to the current fleet of nuclear reactors. The NuScale developers are claiming in their design certification application [NuScale, 2018, chapter 19, page136] a CDF per module of 3.0E-10 per year and a mean value large release frequency of 2.3E-11 per year. These values are significantly lower than the typical industry values, but considering the plant design features, they are considered feasible.

2.4 Review of the Safety aspects of the AHTR concept design

The basic concept of the proposed AHTR plant envisaged by Eskom [Eskom, 2017] is based on the experience gained on the PBMR program, advances in HTGR designs, and the continuous improvement in the high temperature performance of materials and TRISO fuel [WNN, 2013]. As early as 2004 a Very High Temperature gas Reactor (VHTR) shown in figure 4 was being proposed as an evolutionary development of high temperature gas reactors [Chapin, 2004], due to the improvement in efficiency at the higher temperatures and the better thermal conditions for process heat applications.

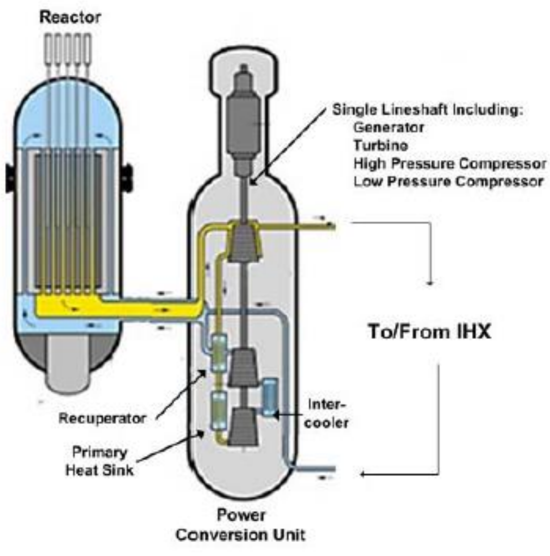


Figure 4 – VHTR concept plant schematic [Chapin, 2004]

The basic form of the proposed AHTR concept design [Eskom, 2017] is a high temperature helium cooled gas reactor driving a gas turbine generator set in a direct Brayton cycle. A variation of the design includes process heat being transferred to an intermediate molten salt circuit via helium-to-molten salt heat exchangers. The use of molten salt in an intermediate cooling loop reduces the probability of water ingress into the helium circuit, and provides the option of thermal storage. The heat from the molten salt is then converted into steam to drive a steam turbo-generator set in a Rankin cycle, see figure 5.

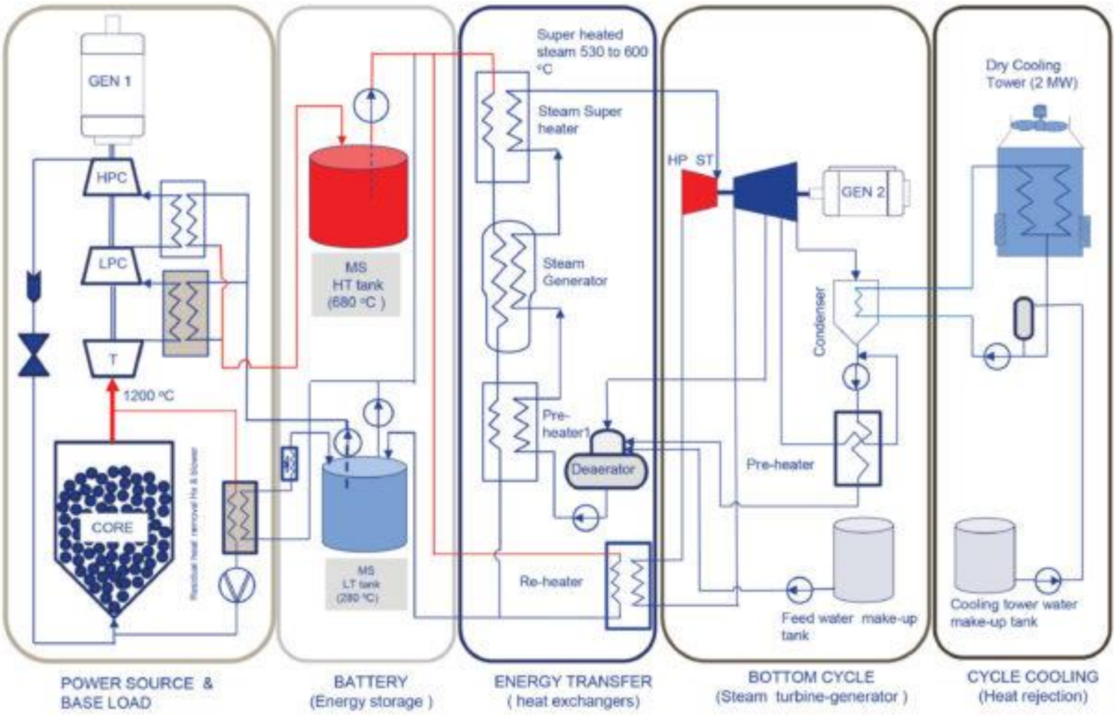


Figure 5 – AHTR concept plant schematic [ESKOM, 2017]

The molten salt circuit of the plant includes thermal storage that can be sized to allow the plant to meet different network demand profiles. By varying the capacity of the molten salt storage, the steam cycle and the turbine generator set, the plant can be designed to support networks with significant variation in the daily load profile, while maintaining a constant load on the reactor. The concept plant design delivers a nominal power of 100 MWe, which is made up of 30 MWe (base load) from the gas turbine-generator and 70 MWe from the steam turbine-generator (equivalent base load power assuming no storage). The thermal efficiency of the Brayton cycle operating above 1000° C will be around 55%. The Rankin cycle efficiency will be higher than that of a PWR due to the high temperature of the salt in the intermediate loop. As the plant only rejects heat form the Rankin cycle, the combined power of the two cycles improves the overall energy efficiency of the plant, which is expected to be around 50%.

The reactor vessel is a concrete structure onto which a single vertical shaft, power conversion unit is directly mounted. The power conversion unit includes the gas turbine, the compressors, the generator and heat exchangers in an integral factory manufactured unit, eliminating the need for high temperature gas piping. The concrete reactor vessel is the ultimate heat sink, and is constructed below ground level for maximum strength and security. The helium flow is from bottom to top, entering at the bottom and extracting heat from the reactor fuel as it flows up through the reactor. On exiting the reactor at around 1200^{o} C it enters directly the high speed gas turbine. On exiting the turbine, it is routed through a heat exchanger before continuing upwards through the compressors and pre-coolers. On exiting the last compressor stage, the helium is routed back down to the bottom of the reactor at around 400^{o} C.

The use of helium as a coolant provides excellent neutronic and thermal characteristics in combination with the graphite moderator. Helium is suited to the high temperatures of an HTGR as it does not undergo any phase change and remains chemically inert, even at the postulated accident temperatures. Thus, chemical interactions with the fuel and graphite moderator are avoided. The technical challenges of using helium at the postulated AHTR temperatures are discussed in chapter 5.

The reactor core is a cylindrical cavity formed by the reflector made from graphite blocks. Two independent Reactivity Control Systems are provided, both of which insert absorber materials into channels located in the reflector. Both are designed to independently shut down the reactor if required, while one also manages any reactivity changes required during normal operations. If both the control rod systems fail, the rise in fuel temperature will cause the temperature driven reactivity coefficients to shut down the reactor from any power level even following a loss of forced cooling [INL, 2011, page 18]. The reactor core is filled with TRISO fuel contained in graphite balls in a once through fuel cycle. Used fuel balls are extracted from the bottom of the core, and fresh fuel balls are added to the top of the core while the unit is operating.

The TRISO fuel consists of uranium oxide particles less than a millimetre in diameter, coated with layers of pyrolytic carbon and silicon carbide that provide a containment for fission products which is stable to very high temperatures [IAEA, 1997, page 433]. These particles are then arranged in a billiard ball-sized pebble of graphite encased in silicon carbide, each with about 15 000 fuel particles amounting to about 9 grams of uranium. Due to the fuel structure containing significantly more graphite that uranium by volume, HTGR cores are relatively large for the power they produce, resulting in low core power densities. This large thermal capacity of the core provides thermal stability during normal operations and the ability to passively remove the decay heat by thermal convection, conduction and radiation from the core to the reactor vessel [European Commission, 2017, section 7.2]. In response to a LOCA the temperature of the core will increase to a maximum fuel temperature within a period of days, after which it will slowly decrease [Saragi, 2015] [IAEA, 2010, chapter 9]. The design of the reactor core, structure and vessel must demonstrate that the maximum fuel temperature remains below the temperature at which fuel integrity can be assured.

In 2004, to demonstrate its inherent safety features, the 10 MWt high-temperature gas-cooled demonstration reactor (HTR-10) at Tsinghua University near Beijing was subjected to extreme safety tests when the helium circulation was stopped without the reactor being shut down. The tests confirmed that without any of the control rods being inserted, the heat up of the fuel and the temperature driven reactivity coefficients can shutdown the reactor and bring the reactor power safely to a stable low level. During the tests, the heat generated in the core was passively dissipated without causing unacceptably high temperature in any of the components and fuel temperature was limited to 1250^{o} C [Hu, 2004]. The results of these tests were used as an additional means of benchmarking existing HTGR codes [IAEA, 2013, section 2].

The European Commission report also modelled the VHTR under loss of cooling accident and the results showed that in the worst scenario only a few percent of the fuel elements (less than 5%) reached temperatures between 1500^{o} C and an upper limit of 1550^{o} C. The time span for which these fuel elements experienced these conditions was relatively short, approximately 30 hours [European Commission, 2017, page 134]. In the VHTR core design, the higher temperature increases are experienced in the top part of the core [European Commission, 2017, page 131]. The power is higher at the top because the cold coolant gas enters the core from the top which results in the top of the core having the least negative temperature driven reactivity coefficient relative to the rest of the core.

Performance tests on coated particle fuel have shown particle failure to be a function of temperature, the temperature profile and burnup experienced by the fuel [European Commission, 2017, chapter 4]. From the analysis of the results of the tests performed, no fuel particle failure was observed in fuel with burnup below 11% FIMA (equivalent to about 100 GWd/MTU) at 1600^{o}C, while fuel with higher burnup (>14% FIMA) showed fuel particle failure during the first 300 hours at 1600^{o} C. The report indicates that the allowable fuel temperature limit may be higher than 1600^{o} C if the maximum burnup in the fuel is kept below 11% FIMA. Tests performed at US laboratories found that with the particles they irradiated and tested, most of the medium-lived fission products remain inside the TRISO particles up to 1800^{o} C [WNN, 2013][INL, 2017].

Therefore the expected fuel failure probability and the resultant fission product releases are extremely low as long as the maximum fuel temperature is respected and the quality of the manufacturing process is maintained [IAEA, 2010, section 8.11]. As a result, strict process and quality controls are needed during the production of the coated particles and the fuel spheres. Fuel failure predictions and the quality of the fabrication methods must be validated through testing of irradiated fuel particles at temperatures that envelope the calculated accident temperature profile.

The European Commission report on the safety considerations of the VHTR [European Commission, 2017, page 51] concluded that the German fuel development programme over two decades from 1980 to 1995 constitutes a convincing demonstration of excellence in fuel manufacture. This together with the successful testing of fuel particles at 1800^{o} C [WNN, 2013] demonstrates that extremely low fuel failure rates can be achieved for all operating and accident conditions.

In addition, if the coated fuel particle integrity and its ability to contain the fission products can be assured, then the standard functional requirement of a containment system for a nuclear plant can be relaxed. At the most, a low leakage confinement may be needed rather than a leak tight containment vessel/building. With high fuel reliability, the helium circulating system activity during normal operations will be low and any large sudden release of helium would not require filtering. It is only the potential release from fuel particle failures that might occur during the core heatup following an accident, that constitute the release source term. Due to the low failure probability, the quantity and rate of release of these potential releases could be filtered by a confinement type system.

Although the AHTR is still in the concept phase, it has all the inherent safety features of a HTGRs which preclude the need for any active safety systems. The general HTGR inherent and passive safety features can be summarised as follows [INL, 2014, page 9] [IAEA, 2010, page 12]:

- The beneficial high temperature characteristics of the TRISO coated fuel particles, the graphite moderator and the helium coolant. The advantage of using helium as a coolant is that even in the accident temperature ranges, it has no heat transfer limit and does not change phase.
- The passive heat removal capability from the core, which due to the low power density and large height-to-diameter ratio assures sufficient heat removal, even in the absence of the primary coolant and without the need for active systems.
- The low power density and high heat capacity of the core results in slow and predictable temperature transients.
- The use of fuel that has a high radionuclide retention capability even at high temperatures.
- No corrosion or interaction with the fuel or reactor due to the coolant as the fuel, graphite moderator and helium coolant are chemically compatible under all conditions. This reduces the build-up of radioactive corrosion products within the reactor cooling circuit.
- The inherent ability to limit reactivity and power excursions through the reactor self-shutdown due to a large combined negative reactivity temperature coefficient.
- Safety is not dependent on the presence of the helium coolant.
- No active safety system requiring electrical power, nor is any operator intervention needed to respond to any of the postulated HTGR accident scenarios throughout their licencing history. The response times of the reactor under accident conditions are long (days as opposed to seconds) and the design is insensitive to operator errors.

The final AHTR design will have to demonstrate that the temperature of the fuel does not exceed the maximum allowable fuel temperature during accident conditions. This could pose a challenge to the objective of increasing the reactor coolant outlet temperature to 1200^{o} C. The use of a once through fuel cycle results in the fuel being stratified by burnup. As the maximum temperature limit of the fuel is less for fuel with high burnup, having the high burnup fuel at the bottom of the core where the fuel experiences the lowest normal operating temperature, ensures the greatest margin for the fuel with high burnup. However, the once through cycle also results in a power profile that matches the temperature gradient across the core in normal operations. This temperature and power gradient combined with the higher operating temperature reduces the margin for the fresh fuel at the top of the core to the maximum fuel temperature limit, which may require the operating temperature to be reduced.

The AHTR design once finalised should exceed the required Probabilistic Safety Criteria for Core Damage Event Frequency of less than 1E-5 per year [IAEA, 1999, page 11] by a large margin. A paper on the PSA for next generation nuclear plants did not identify any credible accident scenarios for HTGRs that lead to core damage, even when considering scenarios with a frequency of occurrence as low as 5E-7 per year [INL, 2011, page 19].

2.5 Comparison of the safety aspects of the two SMRs

Irrespective of the design of a nuclear plant, the demonstration of safety can only be accomplished through analysis using qualified codes and appropriate acceptance criteria, of the worst postulated challenges to the fundamental safety functions. Table 1 summarises how the identified safety functions are addressed in the NuScale and AHTR designs.

Safety Function	NuScale	AHTR	Comment
SF 2.1 - Control of reactivity	Two independent methods – control rods and soluble boron. (typical PWR arrangement)	Two independent methods with control systems, and ability to accommodate significant temperature excursion together with the negative temperature reactive coefficients.	Both include the use of control rods designed to regulate power.
SF 2.2 - Control of heat removal	Ability to passively remove decay heat even in LOCA conditions, by maintaining a water level above the core while passively transferring heat away from the core through the containment vessel.	Ability to passively remove decay heat even in LOCA conditions, heat generated in the core can passively dissipate through the reactor structure, ensureing that the fuel does not exceed its temperature limit.	The smaller power ratings and lower core power densities of the SMRs enable the use of passive systems for heat removal.
SF 2.3 - Control / containment of radioactive material	Fuel failure is prevented by keeping the core flooded with adequate heat removal capacity. Additional containment	Fuel failure of the high-quality ceramic coated-particle fuel is avoided as fuel does not exceed “fuel damage” temperatures during accident conditions.	In both cases, the confinement of reactivity is achieved by preventing fuel failure.

	provided by the containment vessel.	Relies on coated fuel particle to perform containment function.	
SF 2.4 - Control of chemical attack	Fuel cladding oxidation and embrittlement is prevented by ensuring fuel remains covered with water.	Use of an inert helium as the coolant and the use of helium to molten-salt heat exchangers to avoid the possibility of water ingress into the helium circuit.	The known forms of chemical attack have either been removed or the conditions under which they occur are prevented.

Table 1 - How the Safety Functions are achieved in the NuScale and AHTR SMRs.

The ability to address the safety functions in a passive manner removes the need for many of the design requirements applicable to standard light water reactors to mitigate condition III and IV occurrences. This significantly reduces the complexity of the plant design. A comparison of the applicability of current-generation plant design requirements to the two SMR designs is reflected in Table 2.

Design Requirement	Current-generation application	SMR application
DR 2.1 - Redundant safety and safeguard systems	Current-generation plant safety-related systems include two independent: • High-pressure safety injection systems. • Low-pressure safety injection systems. • Emergency feedwater systems, storage tanks, and emergency cooling water supplies.	• No active safety injection system required. Core cooling is maintained using passive systems. • Ability to remove core heat without an emergency feedwater system.
DR 2.2 - Containment building	Requires a large containment structure that: • can withstand the energy released during a LOCA and contain any radioactive material released from damaged fuel. • contains sumps with filters to collect water so that the safety-related pumps can continuously circulate cooling water. • has the ability for heat removal and a spray system to manage the containment environment.	Containment function is performed by the fuel as fuel damage is prevented. The NuScale design does include a containment vessel as part of its modular design. Building structure only needs to perform a limited confinement function. • Recirculation and spray systems are not required. • No additional heat removal systems are required because of the passive heat removal.
DR 2.3 - Redundant on-site electrical power sources	Designs require Redundant Emergency diesel generators, redundant electrical distribution systems and in many situations a station blackout supply.	Design does not require any emergency power systems. Core heat is removed by passive heat transfer to the surrounding structures.
DR 2.4 - External hazards	Following the Fukushima incident, there is a requirement to analyse the capability of the plant beyond the design basis envelope and to provide the capability to manage the plant through additional external measures	• Reactor built below ground level. • Resistant to external impact. • Not reliance on any external support.

	should the plant systems be impacted by an external hazard.	
DR 2.5 - Probabilistic risk frequencies	Safety relies on equipment to be able to perform as expected. Probabilistic Risk profile is influenced by: • Complexity of the required systems. • Testing required to verify operability of plant. • Equipment reliability, common mode failures and human error. Goal of not more than 10E–5 severe core damage events per plant operating year.	Simpler plant design and passive safety systems significantly reduces the plant complexity and risk. Safety does not rely on the operation of complex safety systems. NuScale - CDF per module of 3.0E-10 per year. AHTR – No credible CDF scenarios.
DR 2.6 – Assured long term cooling	Besides assured electrical power sources, the long term operability requires support systems like: • Closed water systems for the cooling of safety related components. • The availability of an ultimate heat sink and the associated interfacing systems. As they are active systems, they are subject to failure from external influences such as fouling, oil spills and extreme weather conditions. • Ventilating, cooling, air supply, control systems, etc – needed to support the operation of safety-related systems.	SMR designs are passive and heat is removed by conduction and convection. As there are no large active safety related components, there is no need for closed water cooling systems. Access to a specific external heat sink is not required. The design eliminates the need for safety related support systems.
DR 2.7 – Fuel failure fraction	It is accepted that during severe accidents a percentage of the fuel will fail – releasing their inventory of fission products. It is the function of the containment {DR 2.2} to limit the spread to the environment.	The design of both the NuScale and AHTR plants exclude the possibility of fuel failure: • NuScale - Fuel failure is prevented by keeping the core flooded with adequate heat removal. • AHTR - Fuel failure is avoided as fuel cannot exceed "fuel damage" temperature

Table 2 – Applicability of current generation plant design requirements to the NuScale and AHTR SMRs.

In summary, based on the literature reviewed the design philosophies of both the NuScale and AHTR concepts are advances on existing technologies and indications are that they both exceed the level of safety envisaged for the next generation of nuclear reactors. However, like most SMR concepts, they both still require regulatory acceptance based on detailed analysis against justifiable design criteria.

CHAPTER THREE:

NUCLEAR IMPLICATIONS OF LOAD FOLLOW OPERATION

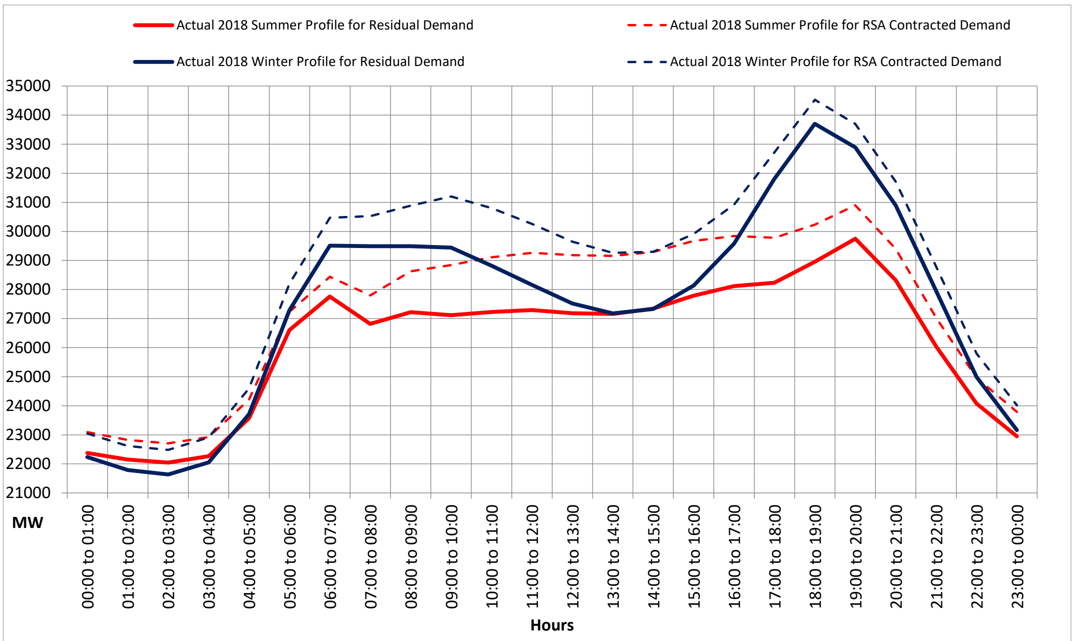
This chapter reviews the requirements for power plants to be able to load follow, the impact that load following has on the reactor of a nuclear power plant and the expected flexibility of SMRs.

3.1 Load Following Requirements

The top priority of an operator of an electrical power system is keeping the grid stable. While the electricity demand can vary for many reasons, there are a number of common factors that influence the demand profile for all grid systems [IAEA, 2018(b), Page 11]. The dominant drivers for the variations in electricity demand are:

- Time of day: Electrical demand is typically lower at night than during the day. No two networks will have the same demand profile but they will typically reflect the minimum electricity demand at night, which can be in the order of 60 - 80% of the demand during the day. The daily demand profile may reflect routine peaks at specific periods of the day determined by the consumption of the different consumer groups.
- Day of week: Electrical demand also depends on the day of the week. Demand is generally lower on weekends and public holidays.
- Season of year: The daily and weekly electrical demand patterns reflect a seasonal influence and depending on the climate can vary significantly between summer and winter. In the case of South Africa, the seasonal influence on the demand profile is in the size and duration of the morning and evening peaks. The typical summer and winter daily demand profiles experienced in South Africa during 2018 are shown in figure 6.
- Weather conditions: Weather conditions such as the ambient temperature or rain can cause a significant deviation to the expected electricity demand. In general, the average daily demand tends to be higher on days where the temperature is either hotter or colder than average.

As the variation in the electricity demand due to time and forecasted weather conditions can be predicted, system operators are able to forecast the long-term demand and the demand for a day ahead with relative accuracy. In response to the forecasted load variations, generating plants are required to be able to operate at different power levels in order to meet the predicted variations in electricity demand. Renewable sources (wind and solar) generate whenever they can and are unable to assist the system operator in managing the variable demand. This can only be done by dispatchable generating sources, typically coal, nuclear and gas, which can be operated either predominantly at constant full output (baseload), or at different power levels determined by the system operator to match the demand (load following). Load following would normally entail operating at a high output during the day and a low power level at night. As these requirements are forecasted ahead of time, generating units can plan and schedule the required load following changes in advance. Some generating units are only called upon to operate occasionally during short periods of excessively high demand (peaking).



Residual demand - demand met by dispatchable plant under the control of the System Operator. (i.e. the Eskom fleet)
RSA contracted demand - total demand supplied by System Operator, includes all plant with renewable IPP contracts.

Figure 6 - Variation in electricity demand for a typical summer and winter day in South Africa in 2018 (Source: Eskom National Control).

The load factor of a power plant is a measure of the actual energy produced by a power plant over a period of time compared to the maximum energy it could have produced in the same time period.. A plant operating as a baseload station would therefore have a load factor close to 100%. Based on the demand profile reflected in figure 6, the average load factor for all the dispatchable generators needed to meet the system demand on an average winter's day in 2018 was around 80%. If the generators needed to meet half of this demand were operating as baseload stations with load factors close to 100%, the system operator would have required the remaining generators to be operate with load factors of only 60%.

In addition to the predictable, time dependent variations in electricity demand, there are other variations in demand that continuously influence the balance between production and demand. These random variations, which occur in timescales of minutes and seconds, manifest in changes to the system frequency. Maintaining the balance between demand and supply requires the availability of reserves that can be called upon through primary and secondary frequency control systems and grid operator action (tertiary control) [Bruynooghe et al, 2010, page 8].

- Primary frequency control is performed automatically (within seconds) by the speed governor, which regulates the speed of the prime mover, of each generating unit connected to the power system that is capable of performing primary frequency control. The governor settings ensure that each generator responds to a disturbance in a proportional way and prevents the governors competing against each other. The primary regulation maintains the balance between the power demanded and power produced by changing the speed (frequency) in relation to the referenced speed.
- Secondary frequency control is triggered within tens of seconds, and is sometimes referred to as Automatic Generation Control. After a change in load and the consequent primary frequency control response, the resultant system frequency will differ from the nominal value [Undrill, 2018, page 14]. The secondary frequency control method entails a control action developed automatically at a central level by the system operator to increase or decrease the load set point of the generating units performing the secondary frequency control function, to return the frequency to within the desired nominal value within a time frame of minutes.
- Tertiary frequency control is triggered within a few minutes by a grid operator if a frequency deviation is not automatically correct through primary or secondary frequency control. It typically involves the grid operator manually dispatching some generating units.

There will also be the occasional large change in system frequency caused by a sudden imbalance between supply and demand such as a trip of a large generating unit, the disconnecting of a large load or a system fault that challenges the stability of the grid. These large random disturbances cannot be predicted in advance, which requires the system to be operated with sufficient reserves. In addition, each generator connected to the grid must respond in a predicted manner under upset conditions to allow the system operator to ensure the stability in grid frequency as well as grid voltage.

In order for the system operator to control the grid system voltage, it is necessary for generating units to be able to assist in the control of reactive power. It is the design of the electrical generator, the exciter and the connection to the transmission system determines the range of reactive power that a plant can provide. For a nuclear unit it is therefore totally independent of the design of the nuclear portion of the plant. The maximum amount of reactive power that a generator can provide can generally be increased by reducing the active power from full load. In extreme situations, a system operator may require a plant to reduce power to provide a greater range of reactive power for voltage control. This can therefore be considered an additional form of flexible operation [IAEA, 2018(b), page 9].

It is therefore necessary for all generating plants feeding electricity into a grid, to comply with clearly defined requirements contained in the relevant grid codes for the network to which it is connected [Modern Power Systems, 2017]. These rules define the minimum requirements based on the requirements of the grid operator and they evolve as the grid dynamics change. For example, in the past, the general rules in the EU specified for small generators that under upset conditions they should trip within a very short time. Today however, due to the significant contribution of renewables and distributed power producers to the total supply, tripping them too early could result in a partial or even complete system collapse. As a result, manufacturers of power plants and generators should keep careful track of national regulations as they progress, in order to ensure their products comply with the latest requirements.

The European Utility Requirements for LWR nuclear power plants was developed as a common accepted specification for use in the EU: it requires modern reactors to have significant manoeuvrability and be able to operate in a load following mode. The European Utility Requirements specified load following capabilities for a unit can be summarised as follows [Lokhov, 2012]:

- The unit must be capable of continuous operation between 50% and 100% of rated power (mandatory). Capability down to as low as 20% can be provided (optional).

- Primary frequency control range of 2% of the rated power is mandatory. The unit must be capable of achieving the full range of control within 30 seconds and be able to maintain it for at least 15 minutes. Higher control ranges up to 5% may be agreed with the system operator.
- Secondary frequency control is not mandatory. If provided, it should include a range of 10% of the rated power and a rate of change of 1% Pn/min. (higher rates can be agreed with the system operator up to a max of 5% Pn/min)
- The unit must be able to perform load following for 90% of the fuel cycle. It is accepted that load following is restricted due to fuel conditions at the end of the cycle. The unit shall be able to load follow between 100% Pn and the minimum load of the unit at a rate of change of 3% Pn/min.
- A unit is expected to be able to perform the following scheduled load following transients from full power to minimum load and back to full power: 2 times per day, 5 times per week, cumulatively 200 times per year.
- Although not a requirement, a unit may be requested to be able to withstand large load rejection at a rate of change of 20% Pn/min. Many nuclear plant designs can accommodate a full load rejection.

Eurelectric and VGB PowerTech assessed whether power plants across Europe are technically flexible enough to provide the required power ramps. The report concluded [Eurelectric, 2011 page 19] that nuclear power plants, based on those in Europe, are technically suitable to perform load following operations (average 5% ramp rate and power range between 100% and 50%). Their limitation is that they cannot be brought online from shutdown or standby conditions in time frames similar to those of the other technologies.

3.2 Power changes and the reactor

Most nuclear power plant reactor designs have strong manoeuvring capabilities. This is evident in France and in Germany where Nuclear power plants operate in load following mode, i.e. some units performing primary and secondary frequency control, while others follow a variable load schedule involving one or two large power changes per day [NEA, 2011, page 49]. In France nuclear load following is needed since nuclear power constitutes a large share in the national mix. While in Germany, load following became necessary as the percentage of renewable generating sources within the national mix increased.

Load following in a nuclear plant requires the reactor power to increase or decrease to match the thermal power being extracted from the coolant by the turbine. The thermal power produced in the reactor core can only be changed by influencing a parameter that results in a change in the overall nuclear reaction rate. The following features are the main drivers that change the reactivity of the core:

- Position of the control rods. Insertion or withdrawal of the control rods involves the addition or removal of neutron absorbing material in the core. Inserting control rods adds negative reactivity that results in a reduction in power and the withdrawal of control rods adds positive reactivity causing power to increase. In PWRs and HTGRs the control rods are inserted from the top of the core. Due to their limited range of influence, they do not have an equal influence throughout the core and their insertion disturbs the neutron flux profile in the reactor core.
- Chemical shim (not applicable to HTGRs). A change to the concentration of boron, which acts as a neutron absorber, in the primary coolant of a PWR adds or removes neutron absorbing material into the core. An increase in the concentration reduces power and a decrease in concentration increases power. As the boric acid is dissolved in the coolant which circulates throughout the entire reactor core, it has a homogenous effect on the core with little impact on the neutron-flux profile. This method is slow in controlling reactivity as it takes several minutes to change the concentration of the boric acid in the primary loop. For rapid changes of reactivity control rods must be used, boron concentration changes are used to compensate for the associated reactivity changes that occur at a slower rate (hours rather than minutes).
- Changes in temperature. Changes that influence the coolant, moderator and fuel temperatures have an impact on the reactivity of the core. As it is difficult to change any operating parameter and not affect every other property of the core, they are collectively referred to as the power coefficient (defect). The value of the power coefficient is required to be negative throughout the core life. The power being delivered by the core has an influence on the overall reactivity of the core through the following temperature feedback mechanisms [NEA, 2011, page 25]:
    - Doppler coefficient: A change in fuel temperature involves the phenomenon usually referred to as the Doppler broadening of the neutron absorption spectrum of Uranium-238 in the fuel. When the temperature of the fuel rises, the absorption of neutrons increases leading to a prompt negative reactivity insertion. The

Doppler coefficient responds instantaneously to a change in power level as the change in energy occurs internal to the fuel. The response of the other temperature coefficients is slower as the heat must to be transferred to the moderator and/or coolant before they experience a change in temperature. The Doppler coefficient is one of the most important feedback mechanisms ensuring reactor safety in reactors.

  - Moderator coefficient: Both PWR and HTGR design, (although for different reasons) have a negative moderator temperature coefficient as the neutron moderation is less efficient at higher temperatures resulting in a decrease in reactivity. This negative moderator temperature coefficient is a stabilising effect for the reactor as any temperature rises will insert negative reactivity, reducing the power being produced. In a PWR, as the coolant is also the moderator and it contains boric acid, the decreasing of the coolant density due to a temperature increase also leads to a decrease in the boric acid concentration and an associated increase in reactivity. The overall impact remains negative as the decrease due to less efficient neutron moderation has a greater effect than the increase due to reduced neutron absorption from the change in boron concentration. It becomes more negative towards the end of the cycle due to the lower boron concentration.

  The power defect acts against both a power increase and a power decrease. When reactor power is increased quickly, the power defect causes a negative reactivity insertion. In a similar manner, when reactor power is decreased quickly the power defect causes a positive reactivity insertion. The power defect is about 2500 pcm for PWRs, and about 800 pcm for graphite-moderated reactors [Nuclear Power, 2019].

- Changes in the mass flow rate of the coolant. A change in the pressure or flow rate of the coolant through the reactor influences the mass flow rate and therefore the rate of heat transfer from the fuel to the coolant. An increase in the heat being removed from the fuel will result in an increase in the power being produced, while a decrease in heat removal will result in less power being produced. In both cases feedback from the power defect will drive the change in power in response to the temperature changes caused by the change in heat transfer. In a PWR this is not a viable option as the coolant is virtually incompressible at operating temperatures, and the flow cannot be changed as it is determined by the speed of the primary pumps which is fixed as they are driven by induction motors directly connected to the grid.

If the thermal load (turbine load) remains the same, any reactivity changes due to rod movement or boron concentration changes will only result in a change in the average coolant temperature. For example, an insertion of the control rods will initially cause a reduction in reactor power. However, as the thermal load being drawn from the primary coolant remains the same, the coolant temperature returning to the core will decrease adding positive reactivity from the power defect. The coolant temperature will continue to decrease until the reactivity addition from the power defect matches the negative reactively of the initial control rod insertion. As a result, the reactor returns to the same power level, matching the turbine load, but at a lower average coolant temperature. This natural ability of a reactor to match the turbine load makes it ideally suited for load following.

3.3 Load following with a reactor and its limitations

Due to the feedback mechanism of the power defect, the reactor power can be adjusted by changing the thermal load being extracted from the reactor coolant (turbine load). An increase in turbine load will cause the coolant return temperature to drop until the power defect has caused a reactivity change and a reduction in core power such that the thermal power being extracted from the core matches the thermal power being drawn by the turbine. Similarly, a decrease in turbine load will result in an increase in the coolant return temperature and a reduction in power being produced by the core. The challenge with load following is maintaining the designed relationship between the coolant temperature and reactor power without exceeding any Operational Requirements.

In response to an increase in turbine load, the control rods would automatically respond to the drop in coolant temperature and step out in an attempt to add the additional reactivity needed to maintain the average coolant temperature within its targeted range. This will continue until the target average coolant temperature is reached. If the control rods cannot step out fast enough, or they reach a point where they are fully withdrawn from the core, the average coolant temperature will continue to drop. Without any intervention, this will continue until either the turbine loading is stopped or by the reactor protection system reaching a low coolant temperature trip setpoint. Due to this ability of the turbine to influence the reactor power, the turbine control system contains limits that prevent a primary or secondary frequency control response from increasing power beyond the thermal limit of the reactor.

For a decrease in turbine load, the control rods would automatically respond to the increase in coolant temperature and step in, to achieve the required average coolant temperature. This will continue until the target coolant temperature is reached. If the control rods cannot step in fast enough the coolant temperature will continue to increase until either the decrease in the load ends, manual intervention by an operator or automatically by the reactor protection system upon reaching an overpower related temperature trip setpoint.

In a PWR, the axial flux distribution changes as a result of reactor power level changes. During normal operation the axial temperature gradient that exists in the core results in axial variations in the density of the coolant and its effectiveness as a moderator. As the moderation in the upper (hottest) part of the core is less efficient than in the lower (coolest) part, during a power increase, the power (flux) distribution is naturally pushed to the lower part of the fuel [NEA, 2011, page 25]. Inserting control rods compounds this by reducing the power at the top of the core, further shifting power to the lower part of the core.

Any change in reactor power level disturbs the equilibrium concentration of Xenon (Xe-135) within the core. Xe-135 is an extremely strong neutron absorber, with a half-life of around 9 hours. Xe-135 is produced from the decay of Iodine-135 (I-135) which has a half-life of around 6.5 hours. The production rate of I-135 in the core is proportional to the reactor power, while the production rate of Xe-135 depends predominantly on the concentration of the I-135 [Lilley, 2001, page 285]. As a result, if the power of the reactor is increased, the concentration of Xe-135 and the associated negative reactivity decreases to a minimum after a period of time (in the order of 4 to 5 hours) before returning to a new equilibrium, level higher than before, after about 40 hours. Conversely, following a reduction in reactor power, the concentration of Xe-135 and the associated negative reactivity increases initially to a maximum before returning to a new equilibrium level lower than before, in similar time scales. The magnitude of the xenon reactivity changes and the initial rate of change of the xenon concentration depend on the reactor power levels before and after the change. As a result, following a change in reactor power, ongoing reactivity changes are needed, to compensate for the changes in Xe-135 concentration, until the new equilibrium concentration is reached which can challenge the manoeuvrability of the plant.

The NuScale design, like most PWRs, uses both the control rods and boron concentration adjustments to manage the reactivity changes associated with the change in reactor power level and the influence of xenon poisoning that occur during load following [NuScale, 2018, chapter 4 page 4.3-4]. HTGRs only have the use of control rods to compensate for all forms of reactivity changes associated with load following. Like what was done for the PBMR, the AHTR design analysis will have to demonstrate that the rod control system is capable of managing the reactivity requirements associated with the plant defined load following capability [Rietsma, 2004].

The use of “regulating control rods” (also referred to as “grey” rods) with different characteristics to “shutdown control rods” is a routine feature in new plant designs as they improve the load following capability of a nuclear plant. The design of the regulating control rods is optimised to cause smaller depressions in the neutron flux in the vicinity of the rods allowing them to be inserted deeper into the core. This enables them to compensate for large power changes with limited impact on the axial flux distribution [IAEA, 2018(b), page 71]. As a result, on a PWR, the use of regulating control rods reduces the need for the large, relatively fast, changes in boron concentration that would typically be needed during a 100%-75%-100% load following sequence [NEA, 2011, page 32, 34].

As the safety analysis must cover all possible initiating conditions, it has to include incident initiation from the intermediate power levels allowed during load follow operation. In some incident sequences, the most penalising initial plant conditions could be during load following operation as the redistribution of power in the core due to the power change can result in peaking factors higher than those experienced during baseload operating conditions. Therefore, as described in chapter 2, operational limitations may be imposed to limit the amplitude of such perturbations and ensure the plant is always operated within the design and the safety analysis [IAEA, 2018(b) page 48]. Although the design studies must demonstrate the plant load following capability, the following are three common examples of physical operational limitations that must be respected which could constrain the flexibility of a plant if not managed appropriately:

- Control rod insertion limits. During operation insertion limits restrict the amount by which the control rods can be inserted for a given power level. The instantaneous negative relativity insertion that would occur should the regulating control rods be inserted from above these limits would be sufficient to shut down the reactor as per the safety analysis. During load following operation boron concentration changes may be required to allow the control rods to be withdrawn back above the power dependant rod insertion limit [NuScale, 2018, chapter 4, page 4.3-5].

- Axial flux deviations. During operation, variations in the axial flux profile are limited to ensure the core power peaking factors used in the safety analysis are respected and to prevent inadvertent xenon oscillations (axial or radial) that the reactor cannot naturally suppress. These xenon oscillations can go unnoticed as they do not impact the overall power level of the reactor yet they can cause local power peaks. In a PWR, the measure of axial offset (often referred to as delta I) is derived from the difference between the flux in the top and bottom sections of the reactor as measured by the neutron detectors positioned outside the reactor vessel. A constant axial offset control scheme is commonly used to control the axial flux distribution. It involves maintaining the axial offset within a tolerance band around a target value for each specific power level [NuScale, 2018, chapter 4, page 4.3-9]. As a result, during power changes, rod position changes together with boron concentration adjustments may be required to return or maintain the axial flux difference within the target band. In addition, as the concentration and distribution of xenon within the core are both flux and time dependent, core operation outside the defined acceptable range can increase the probability of initiating a xenon oscillation within the core [Nuclear Power, 2018].
- Power ramp rates: Light water reactor fuel is manufactured with a gap between the fuel pellet and the cladding. During operation in the reactor, the fuel pellets expand due to thermal expansion and irradiation induced swelling at different rates to the cladding. This reduces and may even close the gap resulting in contact between the fuel pellet and the cladding. An increase in local power density may, due to this different expansion rates, produce sufficient increase in cladding stress to cause cladding failure. Recent operating history and the prevailing operating conditions (eg current and recent power levels and the rate of power changes) influence the magnitude of this stress. Regulators may require power manoeuvring limits to reduce the risk of cladding failure due to pellet-cladding interaction. This could include limitations on the maximum ramp rate for unconditioned fuel early in the fuel cycle until the fuel has been properly conditioned. As conditioned fuel can be deconditioned through extended low power operations (in the order of weeks), additional restrictions may also be required on power ramp rates following periods of extended low power operation [Bruynooghe, 2010, page 17]. In the NuScale design, no additional operational limits are specified to prevent pellet-clad interaction as the maximum

transient-induced cladding strain remains below the stress intensity limits [NuScale, 2018, Chapter 4, page 4.2-18]

The operating flexibility of the NuScale design, which through analysis has been demonstrated to respect all safety, design and operational limits, can accommodate a daily load following profile starting at 100%, ramping down to 50% over two hours, remaining at 50% for two to ten hours, and then ramping back up to 100% in two hours for the remainder of the 24-hour cycle. In addition it can operate in an automatic mode in response to grid frequency changes through peak-to-peak power changes of 10% Pn at 2% Pn/min, perform 20% Pn step demand increase or decrease within ten minutes, perform an increase or decrease of 10% Pn in 60 seconds without trip while operating between 50 and 100 percent power and has the capability to remain online following a sudden load reduction down to the minimum operating load [NuScale, 2018, chapter 10 page 10.2-6]. This level of flexibility exceeds the EUR requirements for continuous operation, primary and secondary frequency control capability, and the ability to tolerate emergency load variations. The illustrative daily load following profile specified in the NuScale design certification application does not meet the EUR required ramp rate of 3% Pn/min, as it states 2 hours to change power between 100% and 50%. Due to the stated ramp rates that the plant can achieve in response to frequency changes and step load changes and the turbine bypass system, it is expected that the plant could achieve the required ramp rate.

3.4 Impact of Load Following

In theory load following can result in the need for additional maintenance and possible replacement of components due to the increased wear or ageing. As nuclear power plant designs are required to include manoeuvrability capabilities and operation at different power levels there is no impact on the large static components as a result of load following. There could however be some influence from load following on active components (although very few in the SMRs) and thus one can expect a slight increase in the maintenance costs if compared to baseload operation [NEA, 2011, page 49]. A study to determine the cost of cycling and varied load operations in fossil plants could not find any (statistically) significant impact of starts, regardless of type, on unit reliability, and the results suggest that utilities, in the period of the study, performed the appropriate maintenance to address the additional startups rather than suffer deterioration in reliability. More importantly, the study concluded that it is more important how the maintenance is conducted, as the amount of damage and the cost of repairing damage from a forced outage is orders of magnitude greater than the damages and costs associated with cycling. [EPRI, 2002, page 7-7].

The economic consequence of load following is mainly associated with the reduction in the load factor. [NEA, 2011, page 49]. Although the fuel costs represent only a small fraction of the electricity generating cost, nuclear plants that operate with a defined cycle length, like the NuScale design, load fuel for a cycle assuming a certain load factor. Operation at lower load factor than that assumed would impact on fuel costs as the fuel will be used in a non-optimal manner. Operation at a higher load factor over the cycle is not possible unless the reactor is shut down earlier for refuelling. Optimising the fuel utilisation and being able to offset the fixed operating and maintenance costs through a higher output, results in lower generation costs per kilowatt-hour. High load factors are therefore needed to pay back the high investment costs typically associated with nuclear plants [IAEA, 2018(b), page 96].

A study by the Nuclear Energy Agency on the impact of the deployment of renewables on the load factors and profitability of dispatchable technologies in the OECD countries concludes that for a renewable penetration of 10% and 30% the impact on the load factors for nuclear generators would be in the order of -4% and -20% while the impact on their profitability would be in the order of -23% and -39% [NEA, 2012, page 9].

3.5 Flexibility of SMRs

The conclusions drawn from this research when considering SMR flexibly can be summarised as follows:

- Future SMRs need to have the capability to provide flexible operation to meet modern grid system requirements [IAEA, 2018(b), page 101]. The increase in renewable capacity and the variability in their output has to be backed up by flexible sources.

- Modern reactor designs can provide large operational flexibility, including the possibility of planned and unplanned load following in a wide power range and with ramps of 5% Pn/min, and extremely fast power changes in the frequency control mode with ramps of several percent of the rated power per second, in a narrow band around the power level. [NEA, 2011, page 49] However, they cannot be brought online from cold and warm conditions in timeframes similar to those of the other technologies, and only storage facilities such as pumped storage and hydro storage schemes with peak generation capabilities are able to provide the very high power ramps needed for large generation-driven fluctuations (that challenge the available reserve margin) [Eurelectric 2011, page 19].

- The operating limitations described in Chapter 2 {OR 2.1 to 2.6} are enforced through the plant Technical Operation Specifications which defines the limits that cannot be overstepped under normal operating conditions. The limits (such as: control rod positioning, flux profile deviations, maximum ramp rates, etc) ensure that the reactor is always operated within the analysed envelope, while providing adequate margin and flexibility for load following [Bruynooghe, 2010, page 14].

- Baseload operation of nuclear power plants is financially the preferred mode of operation as it provides the highest return on the capital investment. Economic research conducted indicates that nuclear generating units with their high capital costs and low fuel costs could be significantly impacted if the financial impact of providing flexible services are not addressed within the energy system [NEA, 2012, page 9] [IAEA 2018(b), page 101].

- The load factor and financial viability of new generation plants could be improved if the excess thermal energy (e.g. in forms of heat) available when operating at reduced electrical output could be used in other industrial applications. This could include district heating, desalination of sea water or supplying process heat (e.g. providing steam for energy intensive processes such as coal liquefaction/gasification, hydrogen production, etc) [IAEA 2018(b), page 23].

CHAPTER FOUR

COGENERATION OPTIONS FOR SMRS

This chapter explores the principal of cogeneration, the potential industrial processes and the potential to improve the viability of SMRs through cogeneration. In addition, the AHTR concept of utilising energy storage to achieve a high load factor and increased flexibility in generating capability is reviewed.

4.1 Cogeneration

Cogeneration is the simultaneous use of energy, from a single fuel source, in two or more applications, usually for electricity generation and another application requiring process heat. Matched to a suitable alternate process heat application, cogeneration with a nuclear plant can enable the reactor to be operated in a baseload mode while still providing the required flexibility in the generation of electricity. Cogeneration is also a way to maximise the use of the energy contained in the fuel through the efficient utilization of "waste heat" instead of generating the thermal energy separately for each application. The benefit of this approach is a possible additional source of revenue for power plants [Van Ballegooyen, 1999].

An assessment of the feasibility of coupling Small Modular Reactors with cogeneration technologies suggests that performing load following with a combination of multiple SMRs is technically and economically feasible [Locatelli, 2014]. The research focused on sites with multiple SMRs because they offer the possibility to split the total power of individual units within the power station. The main technical conclusion was that if a cogeneration process must be operated continuously without some form of of flexibility or "buffer" (storage) then it is not suitable for cogeneration using only the excess thermal energy during flexible load following operation. The economic results showed that a desalination plant can be a viable investment in several scenarios, primarily because the end product is not consumed immediately and can be stored in some manner. From the Locatelli study, the feasibility and viability of cogeneration depends on not only the existence of a market for both products, but also on the ability to meet the variability in the demand of each, the impact of the variability of each product on the production of the other, and the possible need for backup sources.

According to the IAEA [IAEA, 2017(b)] nuclear based cogeneration for electricity and process heat is gaining increasing importance because of the growing awareness of the environmental impact of energy production. Figure 7 reflects a few examples of heat application processes that could be considered for cogeneration with nuclear power. The operating temperature ranges of the majority of the reactors in service, has limited the existing nuclear based cogeneration to applications whose process operating temperatures are under 150° C, predominantly district heating and desalination. Many potential industrial heat applications exist in the medium and high temperature ranges where cogeneration with SMRs could yield major benefits, examples of such applications include petrochemical processes, steel and hydrogen production.

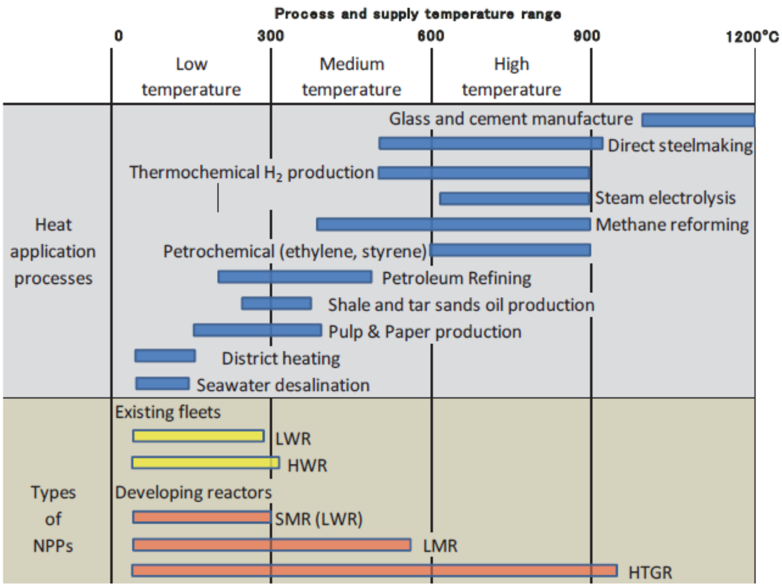


Figure 7 - Temperature ranges of heat application processes and nuclear power plant. [IAEA, 2017(b)]

The IAEA analysis [IAEA, 2017(b)] shows that through cogeneration, the performance of a nuclear power plant can be increased due to the more efficient use of fuel and the resultant reduction in carbon dioxide emissions can reduce the environmental impact by up to 35%. It highlights that an economic evaluation of the benefits of generating a second product in addition to electricity has to take into account the market value of the additional product, the volume of both products, and then be compared to the costs associated with only the flexible generation of electricity. The report concludes that in many cases, cogeneration appears competitive when compared to the cost of fossil fuel alternatives. The benefit would be greater and more options could be economically viable if the associated reduction in carbon dioxide emissions become a financial driver.

An analysis [Beccera et al, 2005] concluded it would be technically feasible and economical to use a nuclear reactor to provide the energy needed for an oil-from-sands extraction facility using Steam-Assisted Gravity Drainage technology. Nuclear energy was shown to be two to three times cheaper than the use of natural gas for each of the scenarios analysed. In addition, a plant producing 100,000 barrels of bitumen per day would eliminate the release of up to 100 megatons of carbon dioxide per year into the atmosphere by using nuclear energy.

The interface between the nuclear heat source and the cogeneration production plant can be achieved by heat transfer directly from a heat-exchanging component in the primary circuit or by the heat being transferred indirectly via an intermediate circuit. The use of an intermediate circuit to decouple the primary circuit from the cogeneration process may be needed for any of the following reasons [Verfondern, 2007, page 76]:

- For physical separation between the primary circuit and the process application plant for safety reasons.
- To prevent any chance of radioactive contamination of the process application plant or the actual product;
- To prevent any possible introduction of corrosive media into the primary circuit;
- Ease of maintenance and repair of the heat utilization system;

The heat exchanger and isolation valves that provide the interface with the primary circuit have to be included in the safety analysis for the nuclear portion of the facility and are subject to the relevant nuclear codes and standards. The cogeneration production plant itself, being a typical industrial facility, only needs to comply with the regulations applicable to that industry. The operating modes between the two plants will depend on the required flexibility of the grid and the cogenerated product, but all possible modes and transients will have to be included in the plant safety analysis.

4.2 Cogeneration with SMRs

As a generating option, the small size and the flexibility of the NuScale SMR modular design enables individual or multiple modules to facilitate the deployment of intermittent renewables, displacing the need for backup power from fossil-fuelled sources. The main means of changing the power output from a NuScale SMR facility are:

- Load following (site level): shutting down one or more reactors if needed.
- Frequency control (individual unit level): through the ability of each unit to load follow.
- Load rejection or Turbine Bypass (individual unit level): the ability to bypass the turbine and dump steam directly to the condenser, although affecting efficiency, enables the turbine to change load without impacting the reactor.

In addition, as a NuScale SMR module provides process heat (steam) which can be used for any industrial application, it provides the flexibility for a single or multi-module site to integrate with a wide range of cogeneration applications. Depending on the demand, modules can be switched between electricity production and providing process heat to support an industrial process. This flexibility enables a multi-module site to efficiently integrate the outputs from the different modules with the collective demands (grid and process heat), optimizing the primary fuel utilization. Another attractive feature of a multi-module site is the ability to stagger the scheduled refuelling of each module. Due to the high level of independence between modules, when one module is off-line for fuelling the other modules can continue to supply the demand (electricity or steam).

A study [Ingersoll et al, 2014] evaluated options for coupling a NuScale SMR plant to a various desalination technologies. The study included an economic comparison of the different technologies coupled to a site containing eight NuScale SMR modules. The study assumed a site sized to provide electricity and water (190 000 cubic meters per day) to support a city of 300 000 people. The analysis demonstrated that a NuScale SMR plant can be coupled to a desalination facility, and can economically cogenerate electricity and water, with a payback period as low as 17 years (excluding financing costs).

In addition, a technical evaluation [Ingersoll et al, 2014(b)] assessed the viability of cogeneration supporting the large energy demands of the oil industry. As the output temperature of a NuScale module is at the bottom end of the process temperature range used in the oil industry, a variety of options were considered to boost the steam temperature. The economic assessment was performed for the case of a representative refinery sized to process 250,000 barrels/day of crude oil. The analysis showed that it is economically viable for a 10-module NuScale plant to cogenerate steam in support of large refinery applications at gas prices below the price of natural gas in many countries, without the benefit of any credit for emission reduction.

Due to its operating temperatures, a HTGR has the ability to supply cogenerated process heat at higher temperatures, thereby supporting a wider range of industrial applications. A report on the “Sustainable Nuclear Energy Technology Platform” [European Commission, 2007] made recommendations regarding the direction of nuclear research in the European Union that included the application of very high temperature reactors for providing cogeneration capacity to industrial processes. The IAEA publication on opportunities for cogeneration with nuclear energy [IAEA, 2017(b) page 28] provides a summary of some of the results of this research and some of the conceptual HTGR cogeneration options. The European Commission issued a report [European Commission, 2017] on the safety considerations of the VHTR Module developed by Interatom/Siemens which is one of the most advanced HTGR concepts in Europe for cogenerating electricity and high temperature process heat.

Despite the potential benefits of nuclear cogeneration, there are numerous challenges that will constrain its deployment [Futterer et al, 2014]. The most prevalent being the ability to co-site the nuclear and industrial facilities, the compatibility of the life spans of both facilities, the conservative nature of industry that prefers business as usual unless forced to change, and the need for stable long-term energy policies together with predictable nuclear licencing processes.

An analysis of the economic viability of small nuclear reactors in cogeneration markets [Carlsson et al, 2012] could only derive target costs, based on competing energy costs in a range of markets, as detailed cost estimates are not publically available. More importantly the report highlighted that the successful operation of nuclear a nuclear plant demonstrating the ability to cogeneration on a production scale, would be the most convincing argument for potentially interested investors.

4.3 AHTR storage option

The AHTR concept, as depicted in figure 5, uses energy storage to produce electricity in a flexible manner through a direct helium Brayton cycle and a steam Rankine cycle. The Brayton cycle consists of a gas turbine-compressor-generator unit with helium-to-molten salt heat exchangers. The generator, due to the high speed, is connected to the grid via a static frequency converter. The objective is for the Brayton cycle to provide a constant power to the grid and to transfer a constant amount on thermal energy to the molten salt circuit. This enables the reactor to operate constantly at full power, base-load operation, while providing flexibility through the steam turbine in the Rankine cycle.

The energy transferred to the molten salt is stored in a hot tank at a relatively high temperature. As needed, the energy stored in the molten salt is converted into steam and the cooler molten salt is stored in a separate tank for cold molten salt. Cold molten salt is pumped from this cold tank to the heat exchangers in the reactor circuit to be reheated. The steam generated in this cycle is routed to a standard steam turbine. The complete two tank storage system, steam generator, turbine, condenser, feed water pumps and de-aerator concept is the same as that used successfully in existing Concentrated Solar Power Plants up to 200 MWe which have been available since 2012 [IRENA, 2012, page 10].

The power capacities for the concept plant is for the reactor to operate at a fixed power level providing a combined output of 100 MWe, which consists of 30 MWe from the generator in the gas cycle and a thermal energy transfer to the molten salt that enables the constant generation of 70 MWe from the steam turbine-generator. The ability to store the energy imparted to the molten salt enables the steam turbine-generator to be designed to deliver higher peak values. The net result is the potential for the customer to specify the required peak capacity and storage time needed to assist in meeting specific grid requirements. This provides the capability to follow a variable daily demand profile as shown in figure 8 while the reactor operates in a baseload mode.

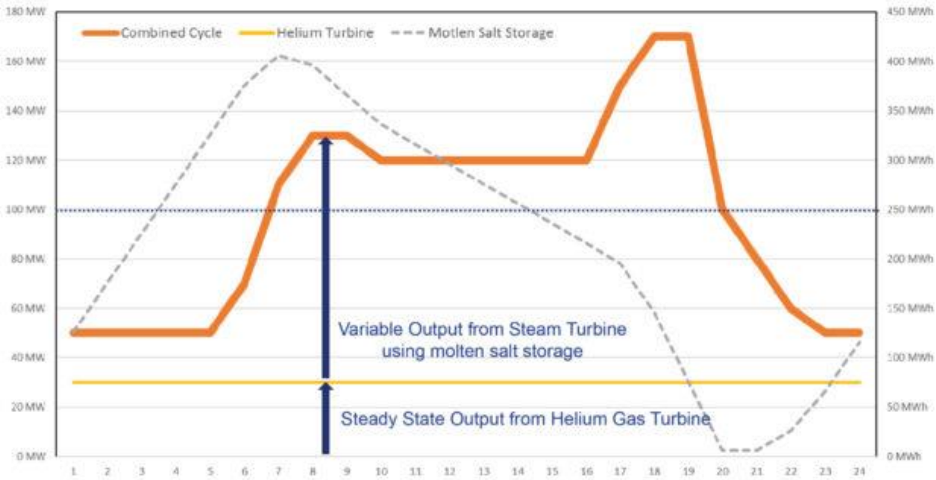


Figure 8 – AHTR: demand profile that could be met [Eskom, 2017]

The profile in figure 8 could be achieved with six hours of molten salt storage capability and a steam turbine generator set with a capacity of 140 MWe. It represents a load factor of

100% for the reactor and the gas turbine-generator, but a load factor of only 50% for the steam turbine-generator (based on the generator design capacity of 140 MWe).

The AHTR concept enables the reactor to operate at a load factor of 100% maximizing the possible return on the financial investment needed for the nuclear portion of the plant. The advantages associated with the concept are a high load factor on the reactor, the small high speed gas turbine generator set, the flexibility in the load profile that can be provided and the ability to provide a peaking capacity larger than that of the reactor. The disadvantage is the lower overall thermal efficiency. The feasibility of this option depends on the comparative cost of simply increasing the Brayton cycle capacity and operating the reactor at a load factor of 70%. To achieve this, the reactor thermal capacity would have to be increased from around 200 MWt to around 280 MWt and the gas turbine-generator from 30 MWe to 140 MWe.

CHAPTER FIVE

TECHNICAL UNCERTAINTIES AND ECONOMIC VIABILITY

This chapter reviews the technical readiness and technical uncertainties associated with the NuScale and AHTR concepts. The predications associated with the potential market and economic competitiveness of SMRs are also reviewed.

5.1 Technical uncertainties

The advanced designs of SMRs do not come without challenges. The technical readiness is a key factor for attracting prospective customers and can be demonstrated through the use of proven technology, physical demonstration, or the certification of the design.

The NuScale design is based on the proven light water technology and the natural convection cooling has been verified in an integral test facility. This reduces the technological and regulatory risks and the design certification review being performed by the US Nuclear Regulatory Commission is expected to be completed in September 2020.

In contrast, the AHTR concept is designed outside the current HTGR experience base due to the temperature at which the system is being designed to operate. The feasibility of the concept will depend on the materials being capable of withstanding the high temperatures. Some of the specific technical challenges that need to be addressed are outlined below:

- The analysis of the behaviour of the graphite structures in the core needs to take into account the AVR experience. During the decommissioning of the AVR, an inspection of the internal graphite structures showed that pieces of graphite had come loose and cracks had developed in some of the blocks, resulting in around 170 pebbles remaining stuck inside the core after it was unloaded [IRSN, 2014, page 70].
- The safety of the AHTR design relies on the robustness of TRISO fuel. This robustness has been demonstrated through testing programs simulating HTGR operating conditions. The current accepted fuel qualification limits are 1250° C during normal operation with a maximum burn-up of 100 GWd/MTU, and around 1600° C for the maximum fuel temperature during accident situations [IRSN, 2014, page 72]. This range may be insufficient for the AHTR due to the higher temperature objectives, and is currently the subject of additional fuel testing. The design analysis will need to demonstrate that the maximum fuel temperature during accident

conditions does not exceed the temperature to which the fuel is qualified. Other HTGR projects are still targeting operating temperatures below 850° C.

- The feasibility of the AHTR design is also based on the equipment in the hot leg of the primary circuit being able to withstand the high-temperatures. For the power conversion unit it is possible to develop a configuration where the pressure boundary operates at lower temperatures by being exposed to the lower compressor inlet and the reactor return helium temperatures. This approach is common design practice for gas cooled reactors and can be achieved by arranging the heat exchanges in the annular space around the turbine-compressor shaft with the hot helium flowing from the centre outwards. To fit the high temperature heat exchangers in the annular space it is necessary to use a compact heat exchangers design with a high power density. Printed circuit heat exchangers (PCHE) are a possible choice for these high temperature helium to molten salt heat exchange [Steven et al, 2013, page 21].
- PCHE technology is well established in the hydrocarbon processing, petrochemical and refining industries. The benefits of this type of heat exchanger are the high heat transfer area density, the high operating temperatures, their high heat transfer efficiency and the low pressure drop. PCHEs are constructed by chemically etching metal plates and then diffusion bonding the plates together by applying high pressures and temperatures to the layers until the plates are welded through diffusion. This process eliminates weaknesses associated with traditional joints or welds [Aris, 2014, page 12].

  As part of the European Commission HTGR program, PCHEs have been manufactured and tested. A paper focused on the experimental investigation of the PCHE mock-up tested on an air test loop [Pra et al, 2008] confirmed the ability of this technology to reach High Temperature Reactor heat exchanger requirements in terms of thermal duty and pressure loss.

  Efficient heat transfer by the heat exchangers of the AHTR is essential to achieving high overall plant thermal efficiency. Although helium has a high thermal conductivity relative to other gases, it is still significantly lower than that of water. A significant improvement in the effectiveness of helium heat exchangers was achieved using a secondary cooling fluid with a higher thermal conductivity and lower specific heat such as water or a molten salt [Figley, 2009, page 101]. The choice of molten salt as the cooling medium, not only removes the risk of water ingress into the primary system, but also enables the use of existing technology related to the storage and ultimate extraction of the thermal energy from the molten salt.

The challenge for the AHTR is to physically accommodate the heat exchangers needed to provide the required heat transfer capacity within a compact power conversion unit.

- Based on the number of ongoing HTGR SMR projects being developed in different countries [IAEA, 2018(a). page 2] the high temperature helium gas turbine, although posing design challenges due to the physical properties of helium and the operating conditions, is not perceived by industry to be a significant technical risk. The current single crystal turbine blades are considered adequate for traditional HTGR temperatures. The AHTR however, due to its desired operating temperature and the turbine receiving the helium directly from the reactor, will push the boundaries of this research and may require the development of ceramic materials for the turbine.

The biggest technical uncertainty in the deployment of any new nuclear technology is associated with the regulatory and licencing processes. An assessment performed in 2010 on the US process, estimated the time from start (first application) to finish (licence to operate) for the first SMR could be anywhere between 11 to 17 years. For subsequent plants based on a certified design, the time could reduce to around 9 years, as the regulatory uncertainty would be limited to issues associated with siting [Coyne, 2010, page 34]. In contrast to current generation plants, SMRs are designed with passive safety systems and features that prevent fuel failure, they therefor require a different approach and methodology in their analysis. These methodologies, supporting analysis, and all the technological aspects of the plant need to be reviewed independently and accepted by the regulator. Without applicable regulatory guidance, industry codes and standards, the risk is on the licence applicant to demonstrate acceptance of the methodology, criteria and material properties. When applying for a licence in a different country, although the lessons learnt from the reviews performed by other regulators can be considered, the process has to be repeated.

The view from the European utilities perspective is that the long timescales involved in licencing and permitting power stations makes it difficult to plan for appropriate long term back-up capacity. This would also preclude technologies that are still under development.

Therefore, it was recommended that the licencing procedures for erecting and operating power stations need to be speeded up and simplified [Eurelectric, 2011, page 42].

Issues specific to SMRs that require ongoing regulatory attention include things like the common control room environment for multi-unit sites, how to define the source term, the associated emergency planning requirements and where necessary developing new codes and standards that are applicable for the applied technologies [IAEA. 2018(a), page 1].

5.2 Economic Considerations and Opportunities

The main potential of SMRs, based on a summary by the Nuclear Energy Agency [NEA, 2016, page 65] and the NNR [NNR, 2017], lie in the following factors:

- The smaller size and the limited need for safety systems should translate to lower overnight capital costs, due to less raw material being required.
- Their small size and the modular design enables them to be built almost completely in controlled factory environments, improving the overall level of quality and maximising the efficiency through the rapid learning curves associated with series production of standardised modules.
- Modular factory fabrication reduces the scope of the on-site construction activities and minimises the time to achieve commercial operation.
- The standardisation of a module design improves the safety and efficiency through standard manufacturing processes and standard operating and maintenance procedures.
- Their small size and scalability makes them suitable for different markets due to the variety of deployment options they offer. Their size and passive safety features make them an option for regions with smaller grids and limited transmission infrastructure. Additional SMR units could be added to a site in the future to accommodate an increase in the demand.
- Multi-unit SMRs, due to their minimal siting requirements, can be built on the same site as aging fossil fuel plants that need to be replaced. SMRs are well suited to replace aging generating plant due to their flexibility and the improved energy security achievable from multiple units in balancing load and intermittent renewable generation.

- The low requirement for access to cooling water makes them suitable for remote locations and other specific co-generation applications such as desalination or district heating.
- SMRs, like all nuclear reactors, offer a clean, carbon-free thermal energy source. The postulated reliability and flexibility offered by SMRs address both energy security and climate change objectives.

The benefits associated with the modularity of construction, the smaller size of each unit, the fewer components and a short construction period will reduce the overnight capital costs required per unit. This together with the ability to stagger the financial impact of additional units is expected to make SMRs easier to finance than large nuclear units. This does not necessarily mean that SMRs can directly compete with large nuclear units as the fixed operating, maintenance and fuel costs as a function of power level are predicted to be higher for SMRs than for large nuclear plants [NEA, 2016, page 21].

The economic benefits associated with series factory production of the modules are important for SMRs to be competitive. The costs to build the modules for the first plant, as a once off manufacturing task, will be higher than those for subsequent repeat orders. The economic benefit of this learning process in a production factory environment will enable the costs to reduce to an optimum NOAK level after a number of plants. This learning process that includes possible proof of concept LEAD plants and the predicted reduction in the FOAK costs over time to NOAK levels as described by Rosner, Goldberg and Hezir, is reflected in Figure 9 [Rosner et al, 2011, page 17]. Although LEAD plants are unlikely to be constructed for SMRs, the concept emphasises the significant resource required to finalise the standardised design used for a FOAK plant. For example, the NuScale designers built a one-third scale, electrically-heated prototype test facility, at Oregon State University in Corvallis, Oregon. The information obtained from this and other test facilities has been used to validate the standard plant design that was been submitted to the US NRC for certification.

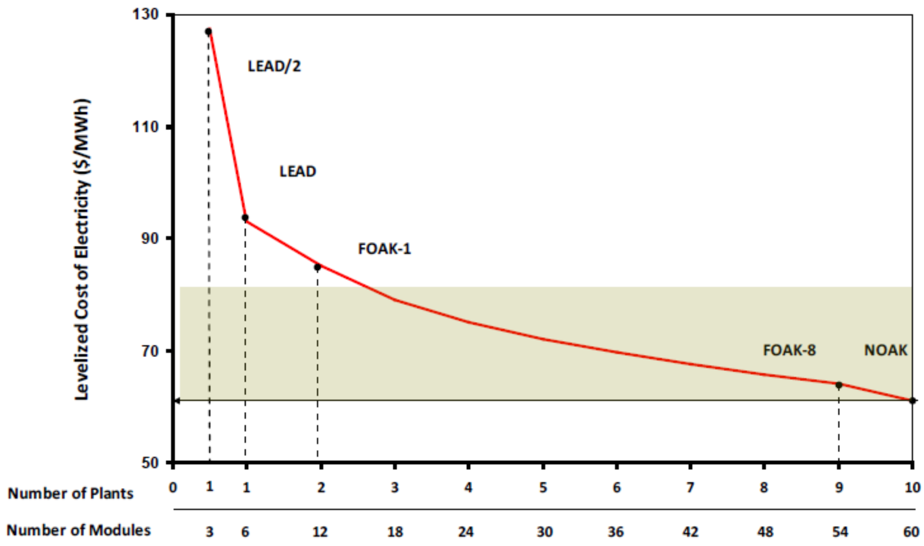


Figure 9 - Levelized cost of learning plants [Rosner et al, 2011]

An economic analysis of SMRs performed by Boldon and Sabharwall suggests a greater reduction between the FOAK and NOAK costs and that the future of small modular reactors will ultimately be determined by market competition [Boldon et al, 2014]. The report presents scenarios in which SMRs may be made more attractive and competitive compared to other load following generation technologies and demonstrates how both storage and cogeneration may add to the overall profit margin of an SMR. The analysis estimated the LCOE from a SMR to be in the range of 66 to 84 $/MWh, which is similar to the conclusions of Rosner and Goldberg. According to the NEA report on the near term deployment of SMRs, if the their competitive advantages are achieved, depending on the cost of fossil fuels and carbon dioxide emissions costs, SMRs are expected to have a LCOE between those of coal and large nuclear plants [NEA, 2016, page 10]. A recent comparison of the LCOE for different technologies puts coal in the range of 60 to 103 $/MWh and nuclear between 112 and 189 $/MWh [Lazard, 2018].

Clearly, SMR development takes time and requires significant financial investment to achieve FOAK deployment. The financial capability of the vendor and the ability to secure a buyer for the FOAK plant is critical for the successful development of any SMR. Many SMR initiatives, like the Eskom PBMR, fail to make it past the design phase due to the inability to secure additional investment or a buyer for a FOAK plant. The AHTR initiative is likely to follow this same route, as Eskom does not have the resources to develop beyond the concept phase without government or investor backing.

The developers of the NuScale design (NuScale Power) have already “secured” an investor for the FOAK plant and are planning the deployment of a twelve module plant (720 MWe) for Utah Associated Municipal Power Systems. The plant will be built on the US Department of Energy’s Idaho National Laboratories site near Idaho Falls. The first module is expected to be in operation in 2026 and the full plant would be operational by 2027.

The study by the Nuclear Energy Agency on the market potential for near term deployment of SMR concludes that there is a large potential for SMRs. The high-case scenario could result in up to 21 GWe of SMR capacity being added globally by 2035. The actual SMR market development will strongly depend on the successful deployment of prototypes and FOAK plants [NEA, 2016, page 13]. The study also recommends that SMR vendors, Governments willing to develop nuclear power, the nuclear industry, manufacturers and potential customers need to work together to realise the potential of SMR technology.

The international media is continuously reporting new initiatives and collaborative agreements between different parties and countries associated with SMR technology, research initiatives and potential deployment. This demonstrates a continuing global interest in the technology and its potential. NuScale Power as an example has signed MOUs with energy companies to explore the development, licencing and construction of SMRs in Romania, Jordan and Canada. They have also reached agreements with manufacturers, the latest being the signing of a MOU with Doosan Heavy Industries and Construction Company on 29 April 2019 for cooperation to support the deployment of the NuScale SMR worldwide [WNN, 2019].

However, most parties are cautious about being the first to invest in a first-of-a-kind technology, and want them to be built and tested elsewhere. Until a SMR design has been successfully licenced, constructed and operated, the uncertainty associated with the licencing of a new technology and the potential for long delays is likely to prevent any large-scale deployment in the near future.

CHAPTER SIX

CONCLUSION

6.1 Discussion

The following discussion integrates the pertinent facts established in the previous chapters to answer the key research question on the viability of SMRs posed in chapter one.

The general safety approach of the two SMRs evaluated is to prevent fuel failure by ensuring the fuel never exceeds temperatures that could result in fuel failure. This is achieved by ensuring a passive means to remove the decay heat from the fuel during all postulated conditions. This approach significantly reduces the plant complexity as it eliminates the need for the traditional safety systems designed to minimise the amount of fuel failure and to prevent the spread of fission products following fuel failure. It is therefore credible for the overall safety of modern SMRs to be markedly higher than that of current reactor designs.

The passive heat transfer capability of an SMR is design dependant but is practical due to the lower power density in the reactor core. The NuScale design requires the operation of valves linking the reactor and containment vessels which allows the evaporation and condensation of the coolant to transfer the heat from the fuel to the containment vessel while keeping the fuel covered with water. In the AHTR design, the capability to remove the heat by thermal convection, conduction and radiation from the core, following a depressurisation of the helium cooling circuit, prevents the fuel from exceeding its maximum design temperature without any intervention. The large fuel temperature increase required in the AHTR to drive the transfer of heat also provides an inherent ability to cope with a failure of the control rod system due to the associated large negative reactivity insertion.

As the two designs are based on PWR and HTGR principles, they benefit from years of operating experience and previous regulatory involvement and licencing of the technology. Based on this, these SMR should have fewer regulatory hurdles and the design certification process should be quicker than for the other SMR types. The choice of technology does come with design constraints. For example, the NuScale design has to operate at high pressures to prevent the coolant from changing phase and the AHTR has to withstand a loss of the coolant as a credible incident. Other SMR technologies that are not yet at the same level of maturity eliminate both these design constraints by using coolants such as lead or molten salt. SMRs using these coolants could be simpler and safer as they can operate at the required high temperatures but at pressures that are at or near atmospheric pressure. Inherently safe plants that require no active intervention and can passively shut down from any situation are more likely to gain public acceptance.

The passive cooling capability of a SMR removes all reliance on off-site resources (off site electrical supplies, water supplies, etc) during accidents. In addition, the positioning of the reactor below ground level makes them significantly less vulnerable to external hazards. The improvement in the overall safety through the prevention of fuel failure without the reliance on off-site resources significantly reduces the emergency planning requirements. These factors make them suitable for deployment in almost any location.

The modular design enables them to be built almost completely in controlled factory environments ensuring a high level of quality. This reduces the scope of the on-site construction activities and the time to achieve commercial operation. Design certification and efficiency improvements due to the learning curves associated with series production of standardised modules, will optimise the overall project duration and reduce the costs for subsequent units. Maintaining a standard, certified design reduces the required licencing burden for new plants as it would focus primarily on siting issues and provides potential investors with confidence that a license for a new plant could be obtained in a reasonable time frame.

Compared to current large nuclear plants, the design of SMRs is simplified by the passive safety capability as it eliminates the need for complex safety systems, significantly reducing the total number of components. The small unit size and a simple design should translate to lower overnight capital costs compared to large nuclear plants, as the costs associated with manufacturing and construction are related to the amount of equipment and raw material that is required.

The overall output of a site can be increased as and when needed by adding additional units. For a multi-unit SMR site, the staggered commissioning of the individual units would result in the earlier generation of revenue stream compared to the construction of a single large nuclear plant. A multi-unit plant has the additional benefit over a single large unit of avoiding periods of site unavailability through the total independence of each unit and the staggering of maintenance and refuelling interventions.

SMRs, like all nuclear plants, are capable of load following and can be designed to comply with the applicable grid system requirements on flexibility. Reactor cores are capable of providing large operational flexibility especially if the plant design includes regulating control rods. The safety analysis of a plant ultimately defines the load following capability and the operational limitations, as the analysis has to demonstrate that all the safety criteria are respected throughout the complete designed range of operational flexibility. SMRs are therefor well suited to offer flexibility for utilities operating in markets with a large share of variable renewable generating resources, or operating in small grids with fluctuations in the demand.

Providing operational flexibility comes at a price as the energy a plant produces determines the revenue it earns. If a plant does not receive appropriate compensation for the flexibility it provides, there will be a load factor below which it is not financially viable. Cogeneration can improve the utilisation of an SMR in situations where a high level of flexibility is required.

SMRs have the potential to cogenerate electricity and thermal energy for use in a wide variety of industrial applications. The higher the reactor coolant operating temperature, the greater the number of potential industrial applications. This was one of the drivers for the VHTR and the AHTR concepts as they were initially targeting the production of hydrogen that requires temperatures up to 900°C. Even at the lower operating temperatures of the NuScale design, studies have demonstrated its viability to support applications such as desalination and the oil industry. However, despite the potential benefits of nuclear cogeneration, there are challenges that will constrain its deployment. These include the existence of a market for both products, the ability to co-site the nuclear and industrial facilities, the compatibility of the life spans of both facilities and the impact of the variability of each product on the production of the other.

The storage of thermal energy for future electricity generation, as proposed by the AHTR, is an alternate way of improving the load factor. It enables the reactor to operate continuously at full power while being able to supply a variable electrical output that matches the daily demand profile. The proposed two-tank storage system is similar to that successfully employed at Concentrated Solar Plants with thermal storage capability. Due to their operating temperatures, this storage concept could be applied to liquid metal, molten salt or high temperature gas SMRs. The feasibility of this option depends on the cost of available alternate energy sources and the cost of increasing the capacity of the SMR and operating it at a lower load factor.

Due to the current established technology base for light water and graphite cooled gas reactors, the research that has gone into liquid metal and molten salt reactors, and the ongoing development in the different SMR technologies, it would appear that there is no technical obstacle preventing any of them from becoming viable, inherently safe, SMR designs. The biggest uncertainty in the deployment of SMRs is the time needed for the regulatory and licencing processes, as it can only start once the plant design is complete. For example, the concept for the NuScale project started in 2003, the design certification application was submitted during 2017, final design certification is expected in 2022 and the first unit is targeted to be operational in 2026.

SMRs designs based on an established technology, like the NuScale design that is based on PWR technology, require the plant design and the associated safety analysis to be justified using existing codes and standards and where necessary, the justification of revised methodologies and requirements. The main licencing issues are expected to focus on things like how to define the source term, the associated emergency planning requirements and the common control room environment for multi-unit sites. This obviously takes time, and until the plant design has some level of regulatory approval, it is unlikely to attract investor commitment.

For SMR designs like the AHTR, that push the boundary of the established technology, there are additional technical challenges that need to be resolved during the design phase. For the AHTR the critical issues revolve around the high operating temperature that is being targeted due to the associated efficiency improvement. Examples include the justification of the material selection for the turbine blades and the reactor exit structure that will be exposed to the proposed high core exit temperatures. From a safety perspective, the analysis must demonstrate that there is enough margin between the proposed operating temperature and the maximum allowable fuel temperature to accommodate the heat up following a potential loss of coolant accident. If, during the design phase, it is concluded that the aspiration for the high operating temperature is unachievable, the operating temperature could be reduced to within the range of the existing HTGR experience base and materials research.

The smaller unit size, the significantly fewer components, the modular design and the shorter construction period of SMRs will reduce the overnight capital costs required per unit if compared with large conventional nuclear units. This lower capital cost per unit, the shorter period until a revenue stream is established and the ability to stagger the financial impact of additional units is expected to make both individual SMRs and multi-unit SMR sites easier to finance than large nuclear units. There is therefore a large potential market for SMRs, but the actual SMR deployment will strongly depend on the successful licencing, construction and operation of actual plants.

Due to the costs associated with the development of a FOAK plant, it can be expected that some form of government, research or state owned company funding or support, will be needed to stimulate the construction of the initial plants. Once successfully demonstrated on an operational basis the future of any SMR design will be determined by its unique features and market competition. A large factor in the competitiveness of an SMR design will be the number of orders placed, as a minimum number will be needed to achieve the economic benefit associated with the factory production of the modules.

The small size and the scalability of SMRs definitely make them viable options for many different markets. Their size and safety features make them ideally suited for regions with smaller grids and limited transmission infrastructure. In these applications, they have the added advantage that over time additional units can be added to the site as demand grows. Multi-unit SMRs, due to their flexibility and the improved energy security achievable from multiple units, can support grids that have a high percentage of intermittent renewable generation. In addition, as a clean source of energy SMRs are well suited to support strategies to reduce greenhouse gas emissions and can be used as on-site replacement for ageing fossil fuel based energy sources.

Besides the potential benefits associated with SMR designs, the main disadvantages are associated with the licencing uncertainty and the lack of manufacturing, construction and operating experience.

6.2 Summary

SMRs that have safety performance levels far exceeding those of current reactor designs are definitely viable and this goal is the driver for the numerous ongoing SMR initiatives. The ability to prevent fuel failure through passive cooling simplifies the design by eliminating the need for complex safety systems and reduces the constraints associated with siting, opening up energy markets where previously nuclear reactors would not have been viable. Their flexibility and the ability to add additional units over time enable them to integrate into any size electrical network and a variety of energy markets. As a clean energy source, SMRs are also well suited to support strategies to reduce greenhouse gas emissions.

With these benefits, this preliminary review indicates that SMRs should have an important role to play in the future of nuclear energy and further assessment is warranted. The deployment of SMRs will be influenced by local market competition, the ability to predictably obtain construction and operating licenses in a reasonable time frame from the relevant Regulatory Authority and the level of public and government support. Until a SMR design has been successfully licenced, constructed and operated, the uncertainty associated with the licencing of a new technology and the potential for long delays during construction are likely to prevent any large-scale deployment in the near future. As a result, the most convincing argument for deployment of SMRs remains the successful demonstration of SMR technology on a production scale.

REFERENCES

ANSI, 1973, *N18.2: Nuclear Criteria for the Design of Stationary Pressurized Water Reactors*, Published by the American National Standards Institute Inc, Available from Koeberg Library.

Aris V, 2014, *Heat exchanger analysis for innovative molten salt fast reactor*, Masters Thesis submitted at the Laboratory for Reactor Physics and System Behavior, Paul Scherrer Institute, Villigen in August 2014, available online at: https://www.psi.ch/sites/default/files/import/fast/PublicationsEN/FB-DOC-14-020.pdf

Baker J, 2018, *Solar Leader China Is Slashing Its Subsidies On Solar Power - What You Need To Know*, article published on the Forbes website dated 18 June 2018, Available online at: https://www.forbes.com/sites/jillbaker/2018/06/18/solar-leader-china-is-slashing-its-subsidies-on-solar-power-what-you-need-to-know/#213a2ca02f9a, last accessed on 19 February 2019.

Becerra G, Esparza A, Finan A, Helvenston E, Hembrador S, Hohnholt K, Khan T, Legault D, Lyttle M, Miu K, Murray C, Parmar N, Sheppard S, Sizer C, Zalszewski E, Zeller K, 2005, *Nuclear Technology and Canadian Oils Sands: Integration of nuclear power with in-situ oil extraction*, Project report by the MIT Class of 2006 Nuclear Design Team in December 2005, available at http://web.mit.edu/kadak/www/oilsands_report.pdf.

Boldon L, Sabharwall P, 2014, *Small modular reactor: first-of-a-kind and nth-of-a-kind economic analysis*, published by the Idaho National Laboratory, Idaho in August 2014, INL/EXT-14-32616 available at https://www.osti.gov/biblio/1167545-small-modular-reactor-first-kind-foak-nth-kind-noak-economic-analysis

Bruynooghe C, Eriksson A, Fulli G, 2010, *Load-following operating mode at Nuclear Power Plants and incidence on Operation and Maintenance costs. Compatibility with wind power variability*, published by the Publications Office of the European Union, Luxembourg in 2010, ISBN 978-92-79-17534-3 available at http://publications.jrc.ec.europa.eu/repository/bitstream/111111111/15308/1/reqno_jrc60700_ldna24583enc.pdf%5B1%5D.pdf

Carlsson J, Shropshire D, Van Heek A, Futterer M, 2012, *Economic viability of small nuclear reactors infuture European cogeneration markets*, published by Elsevier in the Energy Policy journal volume 43 in April 2012, available at https://www.sciencedirect.com/journal/energy-policy/vol/43/suppl/C

Chapin D, Kiffer S, Nestell J, 2004, *The very high temperature reactor: a technical summary,* published by MPR Associates Inc, Virginia in June 2004, available at https://pdfs.semanticscholar.org/6259/d27420e0e9c1ce252872b23ac06e58a51744.pdf?_ga=2.114894727.1089383006.1553444625-1363840385.1553444625

Coyne PD, 2010, *Addressing how light water small modular reactors should be licensed*, published by the Washington Internship for students in Engineering in July 2010, available at https://www.wise-intern.org/journal/2010/PhilipCoyneWISE2010.pdf

EPRI, 2002, *Determining the cost of cycling and varied load operations: methodology*, published in Nov 2002 by the Electric Power Research Institute, available at https://www.epri.com/#/pages/product/1004412/

Eskom, 2017, *Eskom considers relaunch of pebble bed modular reactor programme*, Article by Mmeli Fipaza, David Nicholls, Saneshan Govender, Marius Fox, and Frikkie

Ellis, Published August 14th, 2017 on the EE publishers website, http://www.ee.co.za/article/eskom-considers-relaunch-pebble-bed-modular-reactor-programme.html, accessed 1 November 2018

Eurelectric, 2011, *Flexible generation: Backing up renewables*, Report published by Eurelectric in October 2011 ref D/2011/12.105/47, available at https://www.gasnaturally.eu/uploads/Modules/Publications/flexibility_report_final-2011-102-0003-01-e[1]-2.pdf

European Commission, 2007, *The Sustainable Nuclear Energy Technology Platform Special Report: A Vision Report EUR 22842*, published by the office of the European Union, Luxemburg in 2007, ISBN 978-92-79-05591-1, available at https://www.gppq.fct.pt/_7pq/_docs/brochuras/online/sne%20tp%20-%20vision.pdf.

European Commission, 2017, *HTGR Safety considerations of the HTR-Modul*, published by the office of the European Union, Luxemburg, authored by Kugeler, K., Nabielek, H., Buckthorpe, D, ISBN 978-92-79-71311-8, available at http://publications.jrc.ec.europa.eu/repository/handle/JRC107642, accessed on 20 August 2018.

Figley JT, 2009, *Numerical modeling and performance analysis of printed circuit heat exchanger for very high-temperature reactors,* thesis submitted for a Master of Science degree in Nuclear Engineering at the Ohio State University in May 2009, Reference OSU1244036854, available at https://etd.ohiolink.edu/pg_10?0::NO:10:P10_ACCESSION_NUM:osu1244036854

Futterer M, Carlsson J, de Groot S, Deffrennes M, Bredimas A, 2014, *European energy policy and the potential impact of HTR and nuclear cogeneration,* published by Elsevier in the Nuclear Engineering and Design journal in May 2014, available at https://www.sciencedirect.com/science/article/pii/S0029549313005979

GTM (Green Tech Media), 2018, *Inside Ontarios Clean Energy Contract Cancellations*, article by staff writer EF Merchant dated 26 July 2018, Available online at: https://www.greentechmedia.com/articles/read/inside-ontarios-clean-energy-contract-cancellations-uncertainty-and-optimis#gs.xeHWDknv, accessed on 19 February 2019.

Hu S, Wang R, Gao Z, 2004, *Safety demonstration tests on HTR-10*, Paper presented at 2nd International Topical Meeting on High Temperature Reactor Technology Beijing, CHINA, September 2004 available at http://www.uxc.com/smr/Library%5CDesign%20Specific/HTR-10/Papers/2004%20-%20Safety%20Demonstration%20Tests%20on%20HTR-10.pdf, accessed on 1 November 2018.

IAEA, 1997, *TECDOC-978 - Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors*, printed by the IAEA in Austria, Nov 1997, ISSN 1011-4289, available at https://www-pub.iaea.org/MTCD/Publications/PDF/te_978_prn.pdf

IAEA, 1999, *NSAG-12 - Basic Safety Principles for Nuclear Power Plants*, printed by the IAEA in Austria, October 1999, ISBN 92-0-102699-4, available at https://www-pub.iaea.org/MTCD/Publications/PDF/P082_scr.pdf

IAEA, 2003, *Safety Report Series No-30 - Accident Analysis for Nuclear Power Plant with Pressurized Water Rectors*, printed by the IAEA in Austria, November 1999, ISBN 92-0-110603-3, available at https://www-pub.iaea.org/mtcd/publications/pdf/pub1162_web.pdf

IAEA, 2005, NS-G-1.12 - *Design of the Reactor Core for Nuclear Power Plants*, printed by the IAEA in Austria, April 2005, ISBN 92-0-116004-6, available at https://www-pub.iaea.org/MTCD/publications/PDF/Pub1221_web.pdf

IAEA, 2009, *SSG-2 - Deterministic Safety Analysis for Nuclear Power Plants*, printed by the IAEA in Austria, December 2009, ISBN 978-92-0-113309-0, available at https://www-pub.iaea.org/MTCD/publications/PDF/Pub1428_web.pdf

IAEA, 2010, Tecdoc-1645 - *High Temperature Gas Cooled Reactor Fuels and Materials*, printed by the IAEA in Austria, April 2013, available at https://www-pub.iaea.org/MTCD/Publications/PDF/TE_1645_CD/PDF/TECDOC_1645.pdf

IAEA, 2013, *TECDOC-1694 Evaluation of High Temperatur Gas Cooled Reactor Performance*, printed by the IAEA in Austria, April 2013, ISBN 978-92-0-137610-7, available at https://www-pub.iaea.org/MTCD/Publications/PDF/TE-1694_web.pdf

IAEA, 2016, *SSR-2/1 - Safety of Nuclear Power Plant Design*, printed by the IAEA in Austria, February 2016, ISBN 978-92-0-109315-8, available at https://www-pub.iaea.org/MTCD/publications/PDF/Pub1715web-46541668.pdf

IAEA, 2017, *Nuclear Power Reactors in the world, reference data series No. 2*, printed by the IAEA in Austria, May 2017, ISBN 978-92-0-104017-6, Available online at: https://www.iaea.org/publications/12237/nuclear-power-reactors-in-the-world

IAEA, 2017(b), *Opportunities for cogeneration with nuclear energy*, printed by the IAEA in Austria, May 2017, ISBN 978-92-0-103616-2, Available online at: https://www-pub.iaea.org/MTCD/Publications/PDF/P1749_web.pdf

IAEA, 2018(a), *Advances in small modular reactor technology developments*, printed by the IAEA in Austria, IAEA, September 2018 edition, available at https://aris.iaea.org/sites/Publications.html

IAEA, 2018(b), *NP-T-3.23 Non-baseload operation in nuclear power plants: load following and frequency control modes of flexible operation*, printed by the IAEA in Austria, April 2018, ISBN 978-92-0-110816-6, available at https://www-pub.iaea.org/MTCD/Publications/PDF/P1756_web.pdf

Ingersoll D, Houghton Z, Bromm R, Desportes C, 2014, *Integration of NuScale SMR with desalination technologies*, Paper presented at the Proceedings of the ASME 2014 Small Modular Reactors Symposium of April 2014 in Washington DC, published by ASME in 2014, SMR2014-3392, available at https://www.researchgate.net/publication/267649351_Integration_of_NuScale_SMR_With_Desalination_Technologies

Ingersoll D, Colbert R, Bromm R, Houghton Z, 2014(b), *NuScale Energy Supply for Oil Recovery and Refining Applications*, Paper presented at the Proceedings of ICAPP 2014 Charlotte, April 2014, Paper 14337, available at https://www.researchgate.net/publication/289665894_NuScale_energy_supply_for_oil_recovery_and_refining_applications

INL, 2011, *Next generation nuclear plant probabilistic risk assessment white paper*, published by Idaho National Laboratory in Idaho, September 2011, INL/EXT-11-21270, available at http://large.stanford.edu/courses/2015/ph241/al-alami1/docs/11-21270.pdf

INL, 2014, *Modular high temperature gas-cooled reactor safety basis and approach*, published by Idaho National Laboratory in Idaho, January 2014, INL/EXT-13-30872, available at https://inldigitallibrary.inl.gov/sites/sti/sti/6101987.pdf

INL, 2017, *Tomorrows Nuclear Fuel*, article by Idaho National Laboratory communications specialist Nora Imholte, posted on ansnuclearcafe, 18 September 2017, at http://ansnuclearcafe.org/2017/09/18/tomorrows-nuclear-fuel/ Accessed on 2 November 2018.

IRENA, 2012, *Renewable energy technologies: cost analysis series on concentrating solar power*, printed by the International Renewable Energy Agency (IRENA) in June 2012, available at https://www.irena.org/documentdownloads/publications/re_technologies_cost_analysis-csp.pdf

IRSN, 2014, *Review of generation IV nuclear energy systems*, printed by the Instut du Radioprotection et la Surete Nucleaire (IRSN) in January 2014, available at https://www.irsn.fr/EN/newsroom/News/Documents/IRSN_Report-GenIV_04-2015.pdf

Lazard, 2018, *Lazards levelized cost of energy analysis - version 12.0*, published by Lazard in November 2018, available at https://www.lazard.com/media/450773/lazards-levelized-cost-of-energy-version-120-vfinal.pdf

Lilley J, 2001, *Nuclear physics - principles and applications*, published by Wiley in June 2001, ISBN: 978-0-471-97936-4, available at http://www.uct.ac.za/sites/default/files/image_tool/images/281/we_frahn_library/Courses/Theory_and_Design_of_Nuclear_Reactors/Course_Reading/lil-np-c10.pdf

Locatelli G, Boarin S, Pellegrino F, Ricotti ME, 2014, *Load following with small modular reactors: A real options analysis*, published by Elsevier in February 2015 in Energy volume 80, pages 41 - 54. ISSN 0360-5442, available online at: https://www.sciencedirect.com/science/article/abs/pii/S036054421401295X

Lokhov A, 2011, *Load following with nuclear power plants*, published in NEA News Volume 29 No. 2 by the Nuclear Energy Agency in 2011, available online at: https://www.oecd-nea.org/nea-news/2011/29-2/nea-news-29-2-e.pdf

Lokhov A, 2012, *Load-following capabilities of NPPs*, Article by Lokhov A, Article published online by Nuclear Engineering International, dated 1 June 2012, available at https://www.neimagazine.com/features/featureload-following-capabilities-of-npps/

Modern Power Systems, 2017, *Meeting Europe new grid code requirements*, Article published on the Modern Power System website on 28 March 2017, available at https://www.modernpowersystems.com/features/featuremeeting-europes-new-grid-code-requirements-5773146/, accessed on 10 October 2018

NDRC (National Development and Reform Commission), 2018, *Notice on matters related to photovoltaic power generation in 2018*, Article No. 823 dated 31 May 2018, Available at http://www.ndrc.gov.cn/gzdt/201806/t20180601_888639.html, accessed on 18 February 2019.

NEA, 2003, *Fuel safety criteria in NEA member countries*, published on 17 July 2003 by the OECD publishers, available at https://www.oecd-nea.org/nsd/docs/2003/csni-r2003-10.pdf

NEA, 2009, *Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions, State-of-the-art Report*, Published by OECD publishers in 2009, NEA#6846, ISBN: 978-92-64-99091-3, Available online at: http://www.oecd-nea.org/nsd/reports/2009/nea6846_LOCA.pdf

NEA, 2011, *Technical and Economic Aspects of Load Following with Nuclear Power Plants*, Published by the OECD publishers in July 2011, Available at https://www.oecd-nea.org/ndd/reports/2011/load-following-npp.pdf

NEA, 2012, *Nuclear energy and renewables, system effects in low-carbon electricity systems,* published by the OECD publishers in 2012, Ref ISBN 978-92-64-18851-8, Available at https://www.oecd-nea.org/ndd/pubs/2012/7056-system-effects.pdf

NEA, 2016, *Small Modular Reactors: Nuclear Energy Market Potential for Near-term Deployment,* published by OECD publishers in 2016, NEA#7213, Available online at: https://www.oecd-nea.org/ndd/pubs/2016/7213-smrs.pdf

NEI (Nuclear Energy Institute), 2018, *US Capacity Auctions Leads To Warnings On Future Of Nuclear*, article on the NEI website by T Swanek dated 31 May 2018, Available online at: https://www.nei.org/news/2018/pjm-auction-need-compensate-nuclear-attributes, accessed on 19 February 2019.

NNR (National Nuclear Regulator), 2017, *Next generation reactors: safe and economical tools for sustainable energy*, article published on the NNR website dated 16 November 2017, Available online at: http://www.nnr.co.za/next-generation-reactors-safe-economical-tools-sustainable-energy

Nuclear Power, 2018, *Load following power plant*, article available on the Nuclear Power web site, https://www.nuclear-power.net/nuclear-power/reactor-physics/reactor-operation/normal-operation-reactor-control/load-following-power-plant/ Accessed on 1 November 2018.

Nuclear Power, 2019, *Power Defect - Criticality of Power Reactor*, article available on the Nuclear Power web site, https://www.nuclear-power.net/nuclear-power/reactor-physics/nuclear-fission-chain-reaction/reactor-criticality/power-defect// accessed on 6 April 2019.

NuScale, 2011, *Overview of NuScale technology,* paper presented by Dr. J. Reyers, the Chief Technical Director of NuScale Power, at the IAEA workshop on SMRs held in Vienna during December 2011.

NuScale, 2015, *Accident source term methodology,* published by NuScale Power in Oregon, December 2015, TR-091 5-1 7565-NP, available at https://www.nrc.gov/docs/ML1600/ML16004A217.pdf

NuScale, 2018, *NuScale standard plant design certification application,* revision 2 submitted by NuScale Power, October 2018, available at https://www.nrc.gov/reactors/new-reactors/design-cert/nuscale/documents.html#dcApp

Olander D, Motta A, 2009, *Light water reactor material - chapter 9,* draft version dated 8/30/2009 used by Pen-state University, published under same title in 2017 reference ISBN: 978-0-89448-461-2. draft version available at https://www.mne.psu.edu/motta/Chapters/Chapter9_Thermal_performance%20aug09.pdf

Pra F, Tochon P, Mauget C, Fokkens J, Willemsen S, 2008, *Promising designs of compact heat exchangers for modular HTRs using the Brayton cycle,* published by Elsevier in Nuclear Engineering and Design volume 238, issue 11 in November 2008, ISSN: 0029-5493, available online at: https://www.sciencedirect.com/journal/nuclear-engineering-and-design/vol/238/issue/11

Rasmussen NC, 1975, *Reactor safety study - an assessment of accident risks in US commercial nuclear power plants,* WASH-1400 report published by the US NRC in

October 1975, available at http://journeesdetudes.org/atomescrochus/recherche/rapport-rasmussen.pdf

Rietsma F, 2004, *The Pebble Bed Modular Reactor layout and neutronic design of the equilibrium cycle*, Paper presented at the physics of fuel cycles and advanced nuclear systems conference held in Chicago, April 25-29, 2004, available at https://www.researchgate.net/publication/265922093_The_Pebble_Bed_Modular_Reactor_Layout_and_Neutronics_Design_of_the_Equilibrium_Cycle

Rosner R, Goldberg S, 2011, *Small modular reactors - key to future nuclear power generation in the US*, Paper published by the Energy Policy Institute at the University of Chicago, November 2011, available at https://www.energy.gov/sites/prod/files/2015/12/f27/ECON-SMRKeytoNuclearPowerDec2011.pdf

Saragi E, Sitorus J, Sumijanto, 2015, A*nalysis of temperature profile of the PBMR 400 MWt during anticipated transient without scram accidents*, Paper presented at the 2nd nuclear energy technology seminar in Bali, October 2015, available at https://www.researchgate.net/publication/289860808, accessed on 12 September 2018.

Scheuermann W, Haneklaus N and Fütterer M, 2017, *The high temperature gas-cooled reactor: safety considerations of the (V)HTRModul*, published by the Publications Office of the European Union, Luxembourg, 2017, EUR 28712 EN, ISBN 978-92-79-71311-8.

SSM (Swedish Radiation Safety Authority), 2011, *Probabilistic safety goals for nuclear power plants*, Report authored by Bengtsson L, Holmberg JE, Rossi J, Knochenhauer M, published in May 2011 by the SSM, number: 2010:35 ISSN: 2000-0456, available at www.stralsakerhetsmyndigheten.se

Steven A. Wright, Milton E. Vernon, Paul S. Pickard, 2013, *Concept design for a high temperature helium brayton cycle with interstage heating and cooling*, Published by Sandia National Laboratories, Albuquerque, New Mexico in December 2013, SAND2006-4147, available at https://prod-ng.sandia.gov/techlib-noauth/access-control.cgi/2006/064147.pdf

Tamma P, 2018, *Germanys green dreams run into climate change reality*, article by P Tamma dated 10/9/18, published on the Politico website, available online at: https://www.politico.eu/article/germanys-green-dreams-run-into-climate-change-reality-nuclear, last accessed on 19 February 2019

Undrill J, 2018, *Primary frequency response and control of power system frequency*, Published by Berkeley National Laboratory, February 2018, Ref LBNL-2001105, available at https://www.ferc.gov/industries/electric/indus-act/reliability/frequency-control-requirements/primary-response.pdf

US NRC, 2019, *Domestic licencing of production and utilization facilities,* Regulation 10CFR Part 50 published by the NRC, available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/full-text.html

Van Ballegooyen H, 1999, *Cogeneration Principles*, published by Gryphon International Engineering Services Inc, Ontario Canada, available at http://www.chacanada.com/chacanada/assets/File/CognPrnc_0900.pdf

Verfondern F, 2007, *Nuclear Energy for Hydrogen Production*, Published by Forschungszentrum Julich GmbH, 2007, ISBN 978-3-89336-468-8, available at http://juser.fz-juelich.de/record/58871/files/Energietechnik_58.pdf

Westinghouse, 1975, *An evaluation of a rod ejection accident in Westinghouse pressurized water reactors*, published by Westinghouse Electric Corporation in Pittsburgh, Pennsylvania, January 1975, WCAP-7588, available at https://www.osti.gov/servlets/purl/4073388

Westinghouse, 1989, *Use of 2700 PCT Acceptance Limit in Non-LOCA Accidents*, issued by Westinghouse Electric Corporation in Pittsburgh, Pennsylvania, October 1989, available at https://www.nrc.gov/docs/ML1610/ML16105A075.pdf

WNA, 2016, *Nuclear power in France*, article by the World Nuclear Association, updated July 2016, available at http://www.world-nuclear.org/information-library/country-profiles/countries-a-f/france.aspx, accessed on 19 February 2019.

WNA, 2019, *Nuclear power in Japan*, Library of country profiles maintained by the World Nuclear Association, updated 19 February 2019 available at http://www.world-nuclear.org/information-library/country-profiles/countries-g-n/japan-nuclear-power.aspx, last accessed on 20 February 2019.

WNN, 2013, *TRISO fuel triumphs at extreme temperatures*, article published by World Nuclear News on 26 September 2013, available at http://www.world-nuclear-news.org/ENF-TRISO_fuel_triumphs_at_extreme_temperatures-2609137 accessed on 10 October 2018.

WNN, 2019, *Doosan to participate in NuScale SMR development*, article published by World Nuclear News on 30 April 2019 available at http://world-nuclear-news.org/Articles/Doosan-to-participate-in-NuScale-SMR-development accessed on 13 June 2019.