

CRANFIELD UNIVERSITY

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ADDENDUM REPORT TITLED: DETAILED REVIEW OF NUCLEAR
ENERGY FOR GENERATION IV NUCLEAR POWER PLANTS

In support of PhD Thesis titled:

TERA OF GAS TURBINE HIGH TEMPERATURE REACTOR FOR
NEXT GENERATION NUCLEAR POWER PLANTS

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ABSTRACT

The report is an addendum to the PhD thesis ***TERA of Gas Turbine High Temperature Reactor for Next Generation Nuclear Power Plants (Nov 2017)***. It provides a detailed review of current literature in the public domain, specifically relating to historic generations of plants as well as the technical factors concerned with operating cycles, working fluids, operational experience and plant control.

The descriptions in this report supported the creation of a high fidelity conceptual tool to bridge the technical, economic and risk gaps in the Generation IV (Gen IV) framework in order to provide a method of evaluating Nuclear Power Plants (NPP).

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Keywords:

Nuclear, Gen IV, Efficiency, Cycle, Performance, Simple, Intercooled, Recuperated, Reheat, Gas Turbine, Economics, Electricity, Reactor

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ABBREVIATIONS

ACRONYMS

AECL	Atomic Energy of Canada Limited
ABWR	Advanced Boiling Water Reactor
AGR	Advanced Gas-cooled Reactor
APWR	Advanced Pressurised Water Reactor
Atm	Atmosphere (pressure)
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium reactor
COT	Core Outlet Temperature
DOE	Department of Energy (US)
DECC	Department for Energy and Climate Change (UK)
ELSY	European Lead-Cooled System
ESBWR	Economic Simplified Boiling Water Reactor
GE	General Electric
GFR	Gas-Cooled Fast Reactors
GIF	Generation IV International Forum
GTHT300	Gas Turbine High Temperature 300
GT-MHR	Gas Turbine-Modular Helium Reactor
GW	Gigawatts
GWe	Gigawatts Electric
HRSG	Heat Recovery Steam Generator
HPC	High Power Compressor
HPM	Hyperion Performance Model
HPT	High Power Turbine

HTGR	High-Temperature Gas-Cooled Reactor
HTTR	High Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
IHX	Intermediate Heat Exchanger
IMechE	Institution of Mechanical Engineers (UK)
JAEC	Japanese Atomic Energy Commission
kg/s	Kilograms per Second
kPa	Kilo Pascal
LBE	Lead-Bismuth Eutectic
LFR	Lead-Cooled Fast Reactor
LMTD	Logarithmic Mean Temperature Difference
LOCA	Loss of Coolant Accident
LPC	Low Power Compressor
LPT	Low Power Turbine
LWR	Light Water Reactor
MIT	Massachusetts Institute of Technology
MPa	Mega Pascal
MSR	Molten Salt Reactor
MW	Megawatts
MWe	Megawatts Electric
MWt	Megawatts Thermal
NGV	Nozzle Guide Vane
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (US)
NSM	Nuclear Safety Measure
NTU	Number of Transfer Units

OGV	Outlet Guide Vanes
PBMR	Pebble-Bed Modular Reactor
PCU	Power Conversion Unit
PCS	Power Conversion System
PWR	Pressurised Water Reactor
RHR	Residual Heat Recover
RPV	Reactor Pressure Vessel
SCWR	Supercritical-Water-Cooled Reactor
SFR	Sodium-Cooled Fast Reactor System
SMR	Small Modular Reactor
SSTAR	Small Secure Transportable Autonomous Reactor
TET	Turbine Entry Temperature
VIGV	Variable Inlet Guide Vanes
VHTR	Very High Temperature Reactor
VVER	Vodo-Vodyanoi Energetichesky Reactors

NOTATIONS

^{238}U	Uranium – 238
A	Area (m^2)
C_p	Specific Heat of gas at constant pressure (kJ/Kg K or $^\circ\text{C}$)
CW	Compressor Work (W)
D	Diameter/Hydraulic Diameter; Distance
He	Helium
H	Total Enthalpy (J/kg)
h	Specific Enthalpy (J/kg)

k	Thermal Conductivity (W/m K)
M	Molecular Weight, Mach number
$\dot{m}$	Massflow (kg/s)
n	Number density or moles
Q	Heat Quantity (J/kg)
q	Specific amount of heat transferred (J/kg)
P	Total (Stagnation) Pressure; Pressure (Pa)
p	Static Pressure (Pa)
PR	Pressure Ratio
R	Gas Constant
Re	Reynolds number
s	Entropy J/K
S-CO ₂	Supercritical CO ₂
SW	Specific Work/Power Output (W)
T	Total (Stagnation) Temperature; Temperature (K or °C)
t	Static Temperature (K or °C)
TR	Temperature Ratio
TW	Turbine Work (W)
W	Work (W); Massflow (Kg/s)
UO ₂	Uranium Dioxide/Uranium (IV) Oxide
UW	Useful Work (W)
u	Velocity (m/s)
V	Volume (m ³)

SUBSCRIPTS

c	Compressor
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c_{in}	Compressor Inlet
c_{out}	Compressor Outlet
f	Fluid (working fluid at the intercooled part of the GT)
he	Helium
he_{min}	Helium with minimum gas conditions
ic	Intercooled Cycle; intercooled coefficient
isc	Isentropic (Compressor)
is_t	Isentropic (Turbine)
ip	Inventory Pressure
log_{mean}	Logarithmic Mean Temperature Difference
MHR	Reactor (Heat Source)
MHR_{in}	Reactor (Heat Source) Inlet
MHR_{loss}	Reactor (Heat Source) Pressure Losses
MHR_{out}	Reactor (Heat Source) Outlet
pc_{in}	Precooler Inlet
pc_{loss}	Precooler Pressure Losses
pc_{out}	Precooler Outlet
$poly_c$	Polytropic (Compressor)
$poly_t$	Polytropic (Turbine)
re	Brayton Reheat Cycle; Recuperator
re_{cold}	Recuperator cold side
re_{hot}	Recuperator hot side
re_{HPloss}	Recuperator High Pressure Losses
re_{LPloss}	Recuperator Low Pressure Losses
re_{real}	Recuperator Real (specific heat transfer)

re_{max}	Recuperator Max (specific heat transfer)
$reth$	Brayton Reheat Cycle Thermal
rev	Reversible
rth	Rankine Thermal
$rrth$	Rankine Reheat Thermal
th	Thermal
t	Turbine
t_{out}	Turbine Outlet
t_{in}	Turbine Inlet

SUPERSCRIPTS

'	Normal Rankine Cycle (fig 4-2); Real Brayton Cycle (fig 4-4); Brayton Cycle with Precooler (fig 4-6); normal temperature drop (fig 4.12); Recuperator inlet conditions
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GREEK SYMBOLS

γ	Ratio of Specific Heats
Δ	Delta, Difference
ε	Effectiveness (Heat Exchanger)
η	Thermal Efficiency
μ	Dynamic Viscosity (kg/m.s)
ν	Kinematic Viscosity (m ² /sec)
ρ	Density (kg/m ³)

1 Introduction

The world's population is set to increase to 10 billion by 2050, as projected by the Generation IV International Forum. This forum is responsible for leading the technological road map for Generation IV nuclear energy systems (Nuclear Research Advisory Committee and Generation IV International Forum, 2002). The demand for energy is also set to increase as a result of enhanced living standards, improved health, the need for economic growth, literacy and longer life expectancies.

2 Nuclear Energy as a Source of Power

In general, technology progresses gradually through the accumulation of knowledge and enhancements in technological developments. As a contrast, nuclear technology was discovered unexpectedly with the detection of neutrons by Chadwick in 1932 and nuclear fission by Hahn, Strassman and Meitner in 1938 (cited by Sime, 1998). These discoveries created a sudden path for advancements in nuclear engineering. Post experimental discoveries, reactors were developed in order to satisfy the conditions for nuclear fission such as an exoergic reaction, sustainability and control of a chain reaction (Sekimoto, 2007). Following the discovery of nuclear fission, several reactors achieved criticality, for example, the American CP-1 reactor in 1942, the Canadian natural uranium heavy water research reactor (ZEEP) in 1945, the American fast reactor (Clementine) and swimming pool reactor (BSR) in 1946 and 1950 respectively (Sekimoto, 2007). However, the reactor that first generated power was the experimental fast breeder reactor (EBR-1) in 1951.

3 Nuclear Power Plants – A look back at Generations and Operational Experience

There are currently 450 nuclear power plants in operation at present in 31 countries with net installed capacity of 392 GWe, whilst 60 plants are currently under construction in 16 different countries, with an installed electrical capacity of 60 GWe (European Nuclear Society, 2016). Figure 3-1 illustrates countries versus number of reactors and Figure 3-2 shows the percentage of nuclear share.

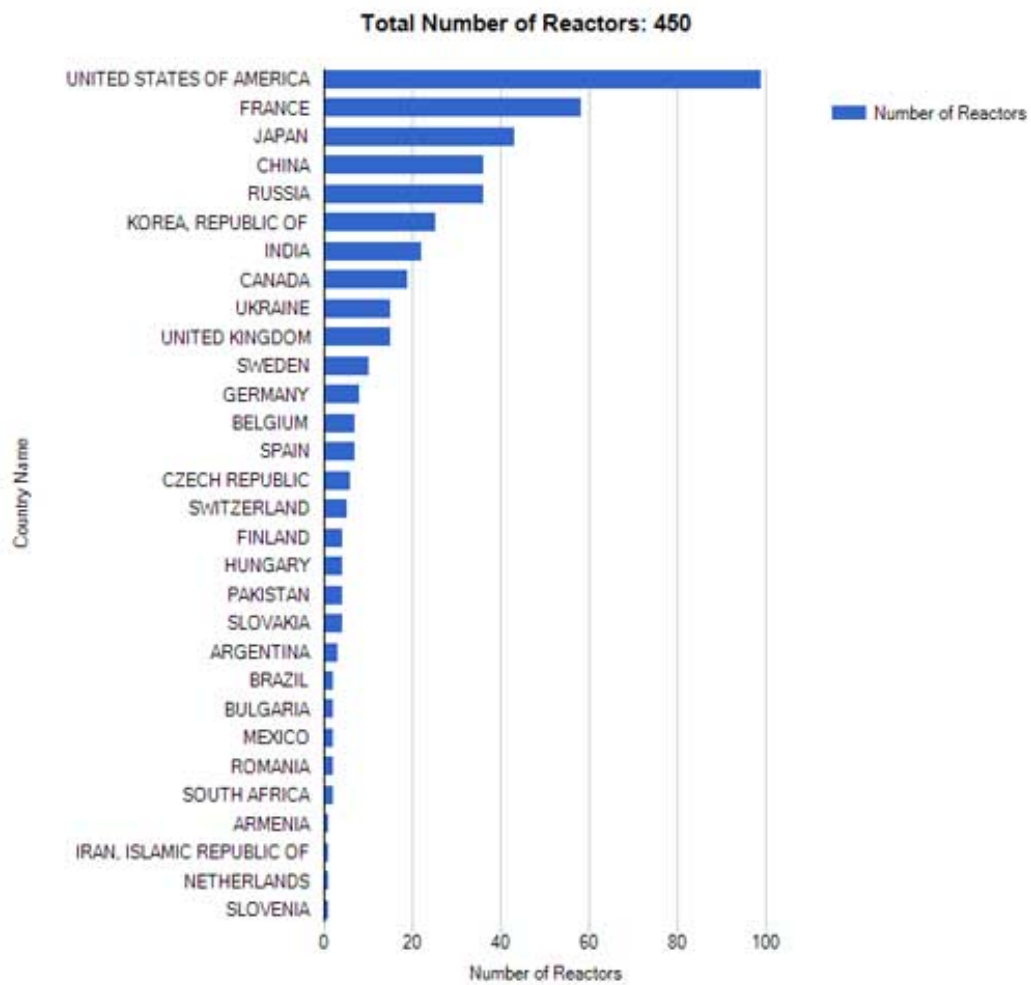


Figure 3-1 – Number of reactors in operation, worldwide, 27/11/2016 (European Nuclear Society, 2016)

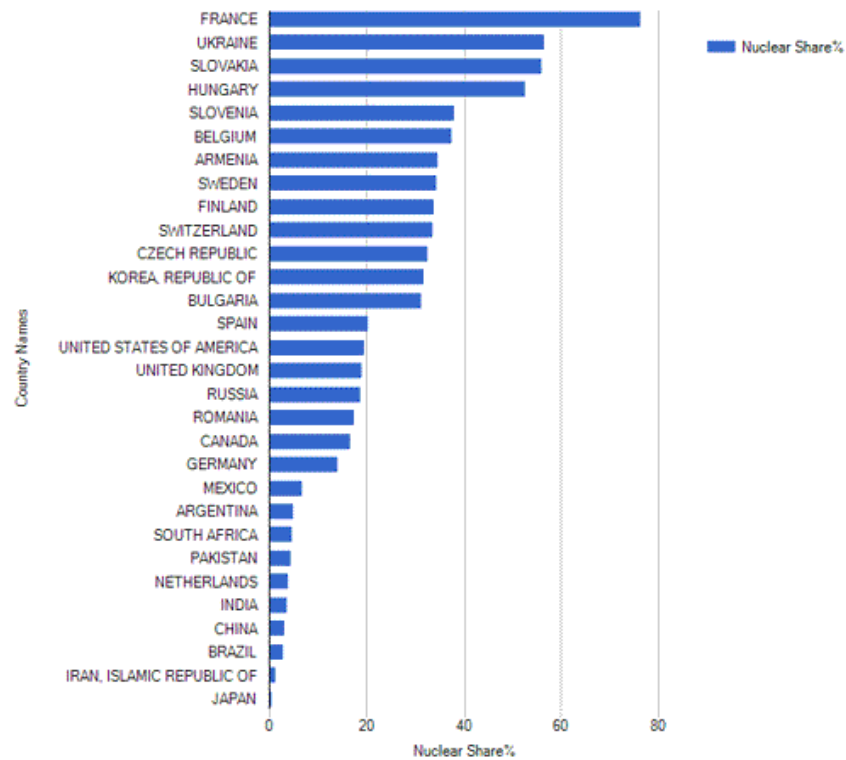


Figure 3-2 Nuclear share in electricity generation, 2015 (European Nuclear Society, 2016)

The foundations for current technological experience comes from generations of prototype, research and fully operational power plants. 3-3 denotes the typical power plant development timeline and their respective generation mark.

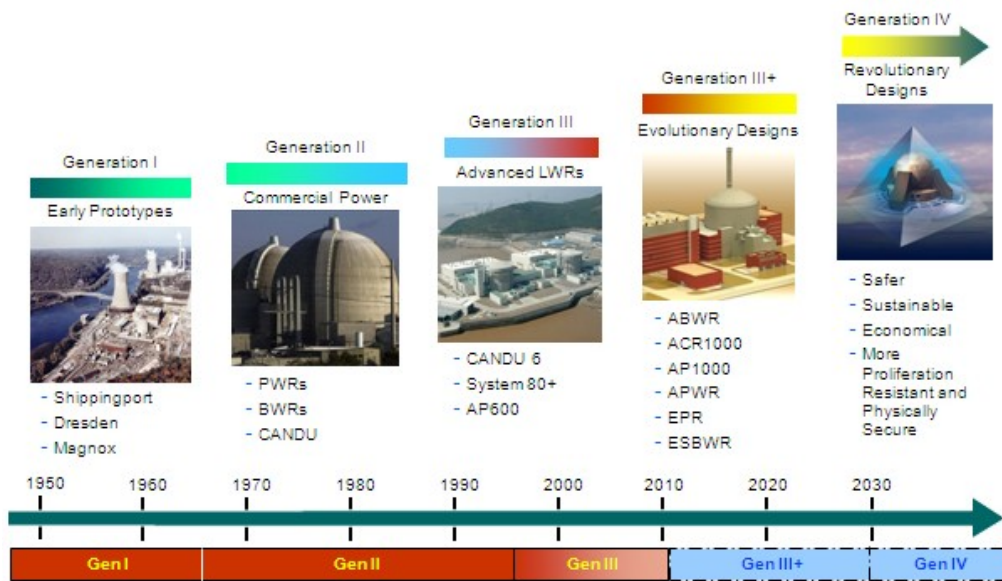


Figure 3-3 Evolution of nuclear power plants (Grochowina, 2011)

3.1 Generation I Power Plants

Generation I reactors are the prototype and operational reactors that were the beginning of nuclear power. The early prototype reactors were developed in the 1950s and 1960s, for example, Magnox Calder Hall -1 in the United Kingdom between 1956 to 2003 (Jensen & Nonbol 1999), Shippingport in Pennsylvania, USA between 1957 to 1982 (DOE Research and Development, 1986; Olson et al., 2002), and Dresden – 1 in Illinois, USA between 1960 to 1978 (U.S. NRC 2014b). The Wylfa nuclear power station in Wales is the only Generation 1 NPP that was still operational as of 2012 (Goldberg & Rosner 2011); the station was eventually shut down in December 2015. With regard to technological enhancements to maintain post millennium operational standards, upgrades were possible because the Magnox reactors were designed and built using the best standards and materials available at the time and operated under modest conditions. Past upgrades were optimised for power production, with upgrades from 46MWe to 60MWe per reactor unit for the Calder Hall station (Jensen & Nonbol 1999).

3.2 Generation II Power Plants

Generation II denotes the class of commercial reactors designed for reliability and better economics, with a typical operational life of 40 years. These include pressurised water reactors (PWR), boiling water reactors (BWR), Canada deuterium uranium reactors (CANDU), advanced gas-cooled reactors (AGR) and vodo-vodyanoi energetichesky reactors (VVER) (Goldberg &

Rosner 2011). PWR and BWR are typical Generation II reactors. These reactors started operating in the 1960s and represent a large percentage of the world's commercial reactors. They are also referred to as light water reactors (LWRs) and comprise active safety features that are electro and/or mechanically operated through automatic or operator initiation, with some passive systems such as pressure relief valves that function with loss of auxiliary or without operator control power (Goldberg & Rosner, 2011).

BWRs use light water as the reactor moderator and coolant in order to generate electricity. This is achieved by cooling the light water in the reactor core to make steam, which is delivered to the turbine. The main difference between the BWR and other nuclear systems is the steam void formation in the core. The steam-water mixture exits the ceiling of the core and enters the two-stage moisture separation chamber before the steam enters the steam line and makes its way to the turbine. The residual steam is exhausted to the condenser and turned into water. Recirculation pumps are used to return the water back to the reactor vessel (U.S. NRC, 2014). BWRs were originally developed during the Generation I phase, as evident in the Dresden-1 plant developed by General Electric and was known as the first BWR NPP (U.S. NRC 2014b). The design was improved for Generation II with several manufacturers such as Siemens, ABB-Atom, Toshiba and Hitachi supplying many BWRs (U.S. NRC, 2014). The Japanese Fukushima plant involved in the 2011 tsunami disaster was part of an optimised Generation II BWR plant (American Nuclear Society 2012). Figure 3-4 denotes a typical schematic for a BWR:

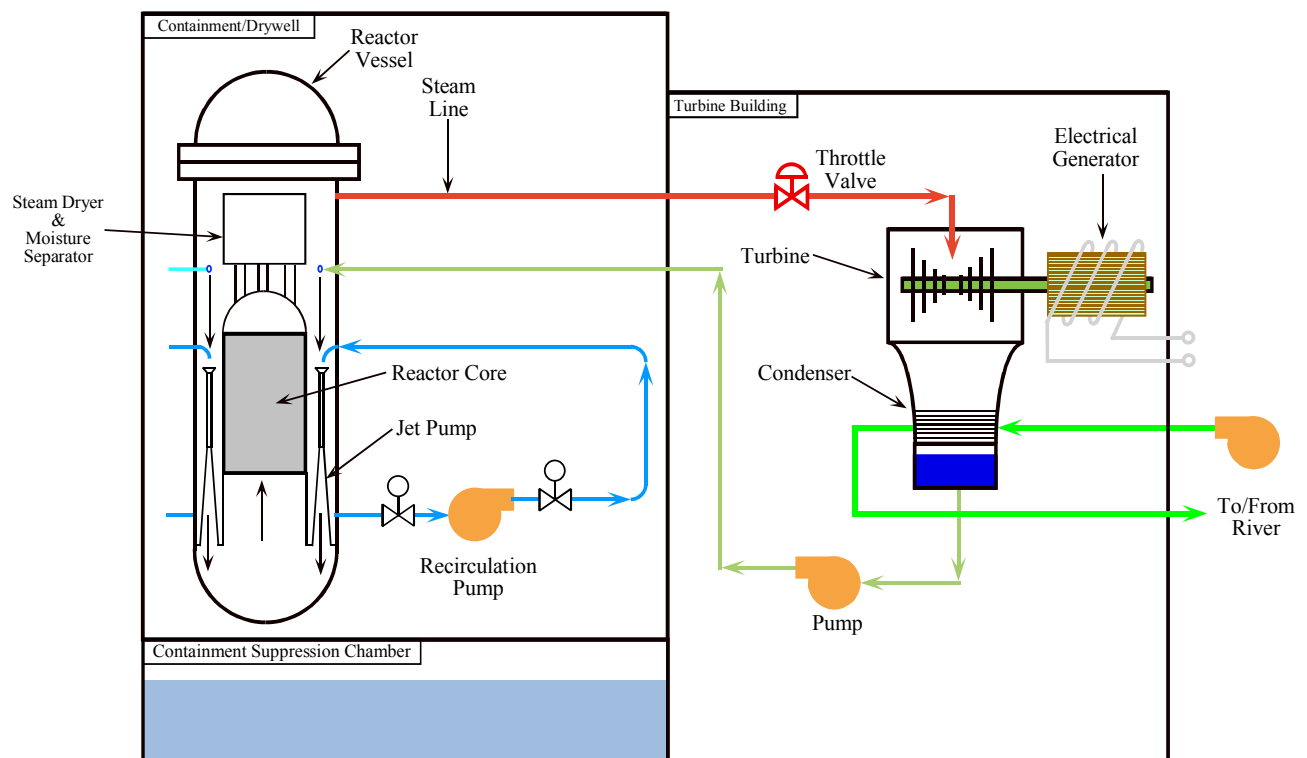


Figure 3-4 Typical BWR plant (U.S. NRC, 2014)

In typical PWRs, the core inside the RPV creates heat, which is transported to the steam generator by the pressurised water in the primary coolant loop. The steam is then directed to the turbine via the steam line, which in turn causes the turbine to spool and turn the generator at the desired frequency. The unused steam is exhausted in the condenser, where it turns back to water and is subsequently removed from the condenser via a series of pumps, heated, and returned back to the steam generator via another pump (U.S. NRC 2013). The core's fuel assemblies are cooled by circulated water using electrical powered pumps. The grid supplies the pumps' electrical sources, including those of other operating and auxiliary systems. In the event of loss of power, the cooling water is supplied by standby pumps, which are powered by onsite generators (U.S. NRC 2013). PWRs were originally developed to serve as nuclear propulsion for nuclear submarines, but were used in the second Shippingport NPP (DOE Research and Development 1986). The American Three Mile Island (TMI) reactors are PWR designs (U.S. NRC 2014c); TMI-2 was involved in the 1979 partial meltdown accident, which

halted construction of NPPs in America for 20 years. Figure 3-5 illustrates a typical PWR design:

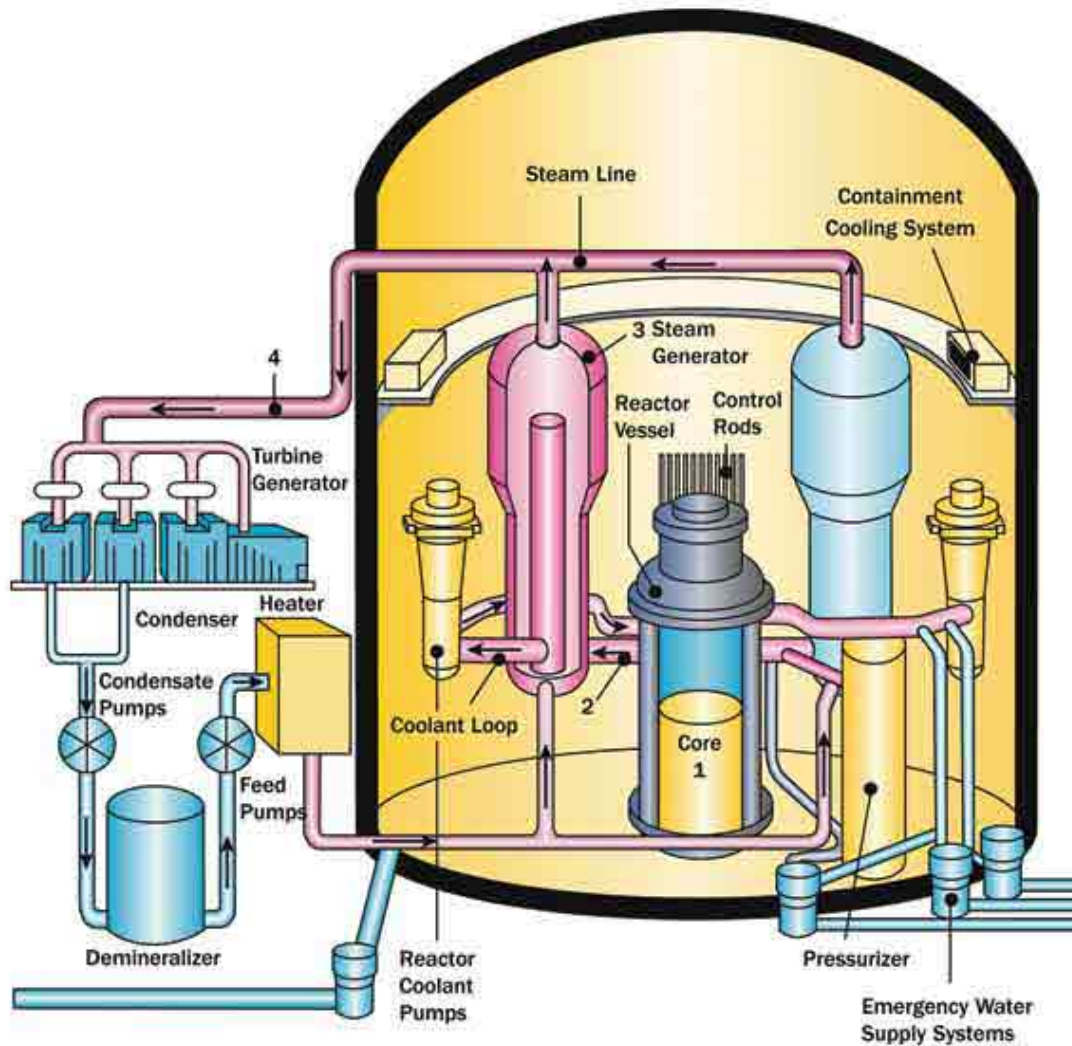


Figure 3-5 Typical PWR design (U.S. NRC 2013)

A third significant player in the commercial operation of NPPs are the CANDU nuclear power systems primarily operated by Atomic Energy of Canada Limited (AECL). They are pressurised heavy water reactors, relying on deuterium oxide moderators and uranium fuel in its original

form. Deuterium is a heavier isotope of hydrogen, with one proton and neutron in its nucleus (as opposed to a proton for hydrogen). This means it absorbs fewer neutrons making it an effective moderator, and allows the use of natural uranium instead of enriched uranium (Waltham 2002). The CANDU reactor functions in a similar manner to PWRs, whereby pressurised coolant is passed through the fuel bundles for cooling purposes, and subsequently the hot pressurised cooling water is carried to a steam generator, whereby the energy is transferred to the separated light water, which is converted to steam. The steam is used to spool the turbines at the required RPM for electricity generation (Waltham 2002). The first Generation II CANDU reactor commenced operation in 1962, and it supplied 20 MWe. This was later replaced in 1967 by the larger prototype called Douglas Point with approximately 200 MWe generation capacity.

3.3 Generation III and III+ Power Plants

Generation III nuclear reactors are essentially Generation II systems with improved robust design modifications. With regards to these improvements, rates of core-sustained damage are lower than the incumbent (Goldberg & Rosner, 2011). Other improvements include improved thermal efficiencies, passively inherent safety systems as opposed to active and modularised construction, with a longer life span of typically about 60 years (Goldberg & Rosner 2011). The Westinghouse house 600 MW advanced pressurised water reactor (APWR) known as the AP-600 was the first Generation III reactor design, with GE producing a similar advanced design for the BWR (ABWR), which came into service in 1996 (Goldberg & Rosner 2011). Examples of Generation III NPPs include the CANDU 6 and the System 80+, which is a combustion engineering design.

Generation III+ reactor designs evolved from Generation II reactors, with significant improvements in safety. Generation III+ development began in the 1990s, with design experience drawn from the LWR NPPs. The biggest improvements in the safety features are the lack of active controls or operator intervention. Instead the systems rely on gravity or natural convection to mitigate abnormal fault cases. Here is a list of Generation III+ designs (based on Goldberg & Rosner (2011)):

- VVER-1200/392M reactor
- AP1000 reactor based on the AP600
- European pressurised reactor
- Economic simplified boiling water reactor (ESBWR)
- Advanced CANDU reactor (ACR-1000)
- APR-1400 based on PWR

Recently Terzi et al. (2016) performed energy and exergy studies on the VVER reactor with 1000MWe. Their analysis indicated a 30% thermal efficiency of the NPP for a reactor producing 3900MWth. Condensers and steam generators add losses of 9% to the steam cycle. This increases the cost of the NPP when generating efficiency is considered. Although this is an improvement on previous generations of NPPs, and presents the typical current technology being considered for developing NPPs (BBC Radio 4, 2017), developing them under more advanced technologies will improve the thermal efficiencies of cycles and improve plant economics.

3.4 Revolutionary Generation IV Designs

According to Locatelli et al (2013), the technological intent is to revolutionise the designs of power plants in a different direction to the incumbent Generation III NPPs. The Generation IV International Forum (GIF) oversees the Generation IV framework, which is illustrated in a roadmap that defines the necessary research and development that will support the Generation IV energy systems comprising the reactor, the conversion systems, support facilities for the overall fuel cycle including ore extraction and final waste disposal (Nuclear Research Advisory Committee and Generation IV International Forum, 2002).

The technological goals defined in the roadmap for Generation IV nuclear energy systems are separated into four areas - sustainability, economics, safety and reliability, and proliferation resistance. These are summarised in detail by Pioro (2016).

The need for sustainable nuclear energy is driven by current generation demands, along with a projected approximation of future energy demands that must also be met. The focus of this roadmap is sustainability, with objectives based on waste and resource management. This will have the following benefits:

- Future generations will be able to recover the necessary energy content from the fuel cycle through recycling used fuel and the conversion of ^{238}U to new fuel.
- It will eliminate the negative impact of energy generation on the environment by displacing polluting energy sources to make way for nuclear electricity generation.
- Geological waste repositories will be better equipped with the capability to cope with waste from future years of plant operations by reducing substantially the waste quantities, and subsequently the decay heat.
- Simplification of the science and analysis required for demonstration of safe and capable repositories with performances to show capability beyond 1000 years. The strategy is to reduce by large amounts the through life and toxicity of residual radioactive wastes transported to repositories for geological disposal.

The need for competitive costs and an understanding of the financial risks is considered under the overall economics that underpin one of the main areas of the framework. If the costs benefit the technological advancements and financial risks are understood, then steps can be taken to achieve a better value for investment. The benefits of meeting this goal are:

- Better life cycle and energy production costs can be envisaged if plant design is simplified and initial manufacture, assembly, test and operation is efficient.
- Manufacturing costs can be reduced by using improved fabrication techniques and modularising the assemblies for ease of manufacture and in-service maintenance.
- The financial risks must not be greater than those of other well-known established energy sources.

The safe and reliable operation of a NPP is critical to the development of Generation IV systems. Enhancements in safety measures and maintainability for improved reliability will play a big part in improving accident management, consequence minimisation and

protection of investments, including the need for off-site emergency response. The benefits of such goals are:

- Increasing the inherent safety features of high integrity nuclear safety measure (NSM) systems that are passive in operation.
- This in turn will enhance public confidence in the safety of nuclear energy.

Controlling and securing nuclear material and facilities falls under the headings of proliferation resistance and physical protection. The benefits are:

- Continual provision of effective and efficient proliferation resistance of nuclear energy systems through improved design features and other necessary measures.
- An increase of physical protection against terrorism through improvements in facility design robustness.

The U.S. DOE Nuclear Research Advisory Committee and Generation IV International Forum (2002) identified systems that were judged to meet the challenges as set out in the framework.

These systems are:

- Gas-cooled fast reactor system (GFR)
- Lead-cooled fast reactor system (LFR)
- Molten salt reactor system (MSR)
- Sodium-cooled fast reactor system (SFR)
- Supercritical-water-cooled reactor system (SCWR)
- Very-high-temperature reactor system (VHTR)

3.4.1 Gas-cooled Fast Reactor (GFR)

Gas cooled fast reactors (GFR) are helium-cooled reactors, with the objective of the technology lying in its ability to bring a high temperature reactor and a fast spectrum nuclear core. Some of the main advantages of this technology include a high core outlet temperature (COT) of

approximately 850 degrees Celsius, which is made possible through an efficient Brayton cycle for electricity generation. In addition, actinides management is designed to efficiently exploit the fuel cycle. A direct thermodynamic cycle is easily adopted, but using gas as a coolant may imply poor thermal inertia and limited heat transfer capability. However, helium has the benefits of single phase cooling in all circumstances, chemical inertness and neurotic transparency (Carre et al., 2010), but the specific heat of helium will mean that it will require pressurisation (Locatelli et al. 2013). A high power density is attributed to GFR due to the use of graphite to increase the core's thermal inertia, but requires cutting edge complex technologies and securities to maintain pressurisation of the reactor pressure vessel (RPV) and loss of coolant accidents (LOCAs) (Dumaz et al. 2007).

The first reported Generation IV design was the 600MWt core coupled to a direct helium Brayton cycle, which was revised in 2007 to a 2400MWt core (Locatelli et al., 2013; Dumaz et al., 2007) due to concerns over economics and the high qualities demanded by the nuclear fuel cycle. The 2400MWt proposal is intended to take advantage of economies of scale to reduce costs. This will be achieved through a GT-MHR RPV and a combined helium and nitrogen indirect option, which will utilise a Brayton and a cogenerative Rankine cycle. Dumaz et al. (2007) argue that this solution simplifies the primary circuitry and security systems, but it will increase complexities in geometry, which is a deviation from the Generation IV framework.

The lower thermal efficiency of 45% for the combined cycle when compared to the Brayton cycle (48.2%) is also noted in published material . It is argued that Brayton cycle efficiency is dependent on component efficiencies, which according to Dumaz et al. (2007) will be challenging to meet. However, the authors also acknowledge that the economic relationship between efficiency and investment costs needs to be analysed in more detail. A demonstrative reactor (Allegro) is planned in Europe to evaluate the commercial feasibility of the GFR. Several studies have commenced to support the feasibility assessment of Allegro, which is due to be commissioned in 2026 (Berthoux & Cadiou 2010), which also include the high temperature helium loop facility for the research of advanced gas-cooled reactors (Berka et al. 2015).

Figure 3-6 shows the schematic representation of the indirect-combined cycle. Table 3-1 shows a summary of the GFR projects (taken from Locatelli et al. (2013)):

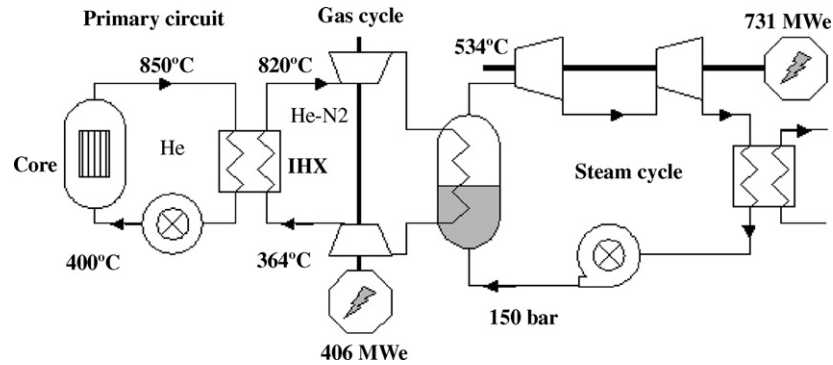


Figure 3-6 Schematic of the indirect 2400MWt combined cycle (Dumaz et al., 2007)

Table 3-1 A summary of GFR projects (Locatelli et al., 2013)

	GFR-600	GFR-2400
Thermal/electrical power (MW)	600/288	2400/1120
Core pressure (MPa)	7	7
COT (°C)	850	850
Thermodynamic cycle	Direct Brayton cycle	Indirect Brayton cycle
Working fluid	Helium	Helium/nitrogen mixture
Net electrical efficiency (%)	48	45
Economics	Medium	

3.4.2 Lead-cooled Fast Reactor

LFR design comprises a liquid metal reactor with lead/lead-bismuth eutectic (LBE) cooling, used for electricity production and actinides management. According to Locatelli et al. (2013), the benefits include excellent neutron and thermofluid dynamic properties, which enables natural convection in the RPV to be established, thereby promoting better heat transfer. It does not react with air and water, and has a high boiling point with no limitations on maximum cycle achievable in the thermodynamic cycle. Lead or LBE has several challenges in that they both corrode structural material through precipitation at modest low temperatures. At high temperatures, dissolution is more prevalent and movement in the fluid can also cause erosion. Due to these challenges, advanced alloys are required (Rivai & Takahashi 2008).

Current LFR programmes include the European lead-cooled system (ELSY) (Bandini et al., 2011; Cinotti et al., 2011), which is based on a sub-critical Rankine steam cycle (Alemberti et al. 2011). In addition to ELSY, there is the small secure transportable autonomous reactor (SSTAR), which is concerned with the design of a modularised SMR with core power of 45 MWt with potential demand (Smith et al. 2008). With a higher COT (560 degrees Celsius) than ELSY, a S-CO₂ Brayton cycle could be adopted for a compact conventional solution. Another prototype called the SUPERSTAR (sustainable proliferation-resistance) is being proposed to replace SSTAR (Locatelli et al. 2013). SUPERSTAR adopts a subcritical steam Rankine cycle for electricity production, and is intended to address some technical issues, which prevented the near deployment of SSTAR (Bortot et al., 2011). The Russians have also developed a SMR called the SVBR 75/100, which is also based on a Rankine cycle, but with modest efficiencies of 34%. The NPP is set for commissioning in 2020 (Locatelli et al. 2013). Table 3-2, which is taken from Locatelli et al. (2013) summarises the notable aspects of LFRs.

Table 3-2 A summary of LFR projects (Locatelli et al. 2013)

	LFR target	ELSY	SSTAR	SVBR 75/100
Thermal electrical power (MW)	125–3600/60–1620	1400/600	45/19.8	280/75
Coolant	LBE–pure lead	Pure lead	Pure lead	LBE
Convection	Forced–natural	Forced	Natural	Forced
Fuel	Nitride fuel	MOX	Nitride fuel	MOX–nitride fuel
Fuel cladding	Ceramic	T91	Coated HT9	Depend by fuel
Breeder ratio	1.0	1.0	1.0	1.0
Outlet core temperature (°C)	800	480	560	307
Thermodynamic cycle	S-CO ₂ Brayton	Rankine	S-CO ₂ Brayton	Rankine
Net electrical efficiency (%)	45	42	43.8	34
By-products	Production of hydrogen and desalinized water	/	Production of desalinized water	District heating
Economics	Medium	Similar to LWR	Similar to LWR	Similar to LWR

3.4.3 Molten Salt Reactor (MSR)

MSRs are fast thermal reactors, incorporating graphite as moderator and cooled by molten salts, when in its liquid phase (Locatelli et al. 2013). The main benefit MSRs is that there is no need for fuel fabrication during the processing phase, thus eliminating these costs. In addition, the increased heat transfer capability due to appropriate material selection and thermal stability at high temperatures for molten salt means the high working temperature can be sustained with consistency, avoiding hot spots. It also has inherent safety aspects with regards to reactivity and lower mass of fissile material (Locatelli et al. 2013). It requires an intermediate cycle for coupling to the gas turbine, but the composition of the coolant requires optimisation for

neutronic and economic purposes (Locatelli et al. 2013). This is difficult to achieve as the thermofluid dynamic behaviour of irradiated molten salts are partially unknown. The operational process for Generation IV is still in concept phase, although early reports indicate a higher level of complexity than foreseen, it is still the objective to design a simple and economical process due to its fundamental criticality for demonstrating feasibility (Beneš & Konings 2009; Le Brun 2007).

The other aspect is the corrosiveness of the environment, which will lead to impurities and the formation of protective oxide layers. High temperature nickel alloys can be used in MSR design, but auxiliary equipment does remain a concern. Current developments of MSRs include the molten salt fast reactor (MSFR); a fast breeder reactor with an output capability of 1300 MWe from a COT of 700 degrees Celsius (Merle-lucotte, et al., 2008). Although cycles are not defined yet, the design intent is for a high net electrical efficiency multiple reheat helium Brayton cycle and a supercritical Rankine cycle (Locatelli et al. 2013). This will make the design challenging due to the additional complexities in geometry, and the increase in initial and maintenance costs, although thermal efficiencies of 45%-55% are estimated. The program predicts a constructed prototype by 2020.

The second reactor is the pebble bed-advanced high temperature reactor (PB-AHTR). The proposed fuel type removes the aforementioned advantages, but means that the environment will be less corrosive due to the absence of fissile material in the fluid. As with the previously discussed advantages, power density is increased and consistent due to the high specific heat of molten salt. The core is 900 MWt with a COT of over 700 degrees Celsius, but an innovative design of the pebble fuel is required (Forsberg 2008). The cycle is based on a reheat helium Brayton cycle (Fei et al. 2008), with the operational intent to couple the cycle to a desalination plant without reduction in net efficiency (Peterson & Zhao 2006), but consideration of losses due to complexities in pipe work, ducting and geometry will need to be taken into account. A prototype is expected for 2025 (Holcomb et al., 2009). Table 3-3, which is taken from Locatelli et al. (2013) summarises the notable aspects of MSRs.

Table 3-3 A summary of MSR projects (Locatelli et al. 2013)

	MSR target	MSFR	PB-AHTR
Thermal/electrical (MW)	2000/1000	3000/1300	900/410
Spectrum	Thermal (possible breeder)	Fast	Thermal
Moderator	Graphite	–	Graphite
Fuel	Liquid UF ₄ –ThF ₄	Liquid UF ₄ –ThF ₄	Solid TRISO pebble
Primary molten salt	NaF–ZrF ₄	⁷ LiF	⁷ LiF–BeF ₂
Salt treatment plant	On-line	Offline	–
Outlet core temperature (°C)	700 (850 for hydrogen production)	700	704
Thermodynamic cycle	Multiple reheat helium Brayton cycle	Multiple reheat helium Brayton cycle	Multiple reheat helium Brayton cycle
Net electrical efficiency (%)	44–50	45–55	46
By-products	Production of hydrogen and desalinized water	Low temperature cogeneration	Low and medium temperature cogeneration
Economics	Low	–	Superior to LWR

3.4.4 Sodium-cooled Fast Reactor (SFR)

SFRs are cooled by sodium in liquid phase. According to (Locatelli et al. 2013), the SFR is the most researched fast reactor. The main technological advantages are the ability to convert fertile material into fissile products, increase fuel usage efficiency by a factor of 50, adopt different fuel cycle strategies i.e. a plain burner, and convert the actinides, thereby reducing high-level waste production and flexibility in the breeder ratio (Locatelli et al. 2013). Two processes (aqueous and pyro-metallurgical) are currently under investigation, both require a closed fuel cycle. Sodium is a good coolant with high specific heat (without pressurisation of the RPV) and is less corrosive than the lead alternative. Its boiling point of 883 degrees Celsius constrains the upper temperature limit of the thermodynamic cycle. Locatelli et al. (2013) state that the economics of SFR are not competitive due to the construction costs of the intermediate circuit.

Current research and development aims to simplify the primary and intermediate circuits by adopting high-grade performance steel including more analysis to assess economies of scale. The Japan Atomic Energy Agency sodium fast reactor (JSFR) and the Korea advanced liquid metal reactor (KALIMER) are two main SFRs, which are of medium to large sizes and employ the Rankine cycle. The JSFR is a 3570 MWt SFR with a closed fuel cycle based on the advanced aqueous process (Konomura & Ichimiya 2007) and a breeder reactor. The predicted output is 1500 MWe when it is commercialised in 2050, with a demonstration reactor with an output of 500-700 MWe operational around 2025 (Aoto et al. 2011).

One of the primary solutions for reducing construction costs is to integrate a primary pump and heat exchanger into a single module (Handa et al. 2011), but it is acknowledged that heat exchanger effectiveness is critical to the efficiency of the plant. To this end, high conductive heat resistant materials and shorter more compact pipework are being investigated (Konomura & Ichimiya 2007; Aoto et al. 2011). Cost reduction solutions being considered for the KALIMER plant also include shorter pipework and a smaller RPV (Hahn et al. 2007). A prototype is scheduled for 2028 with wide adoption in 2040 (Hahn 2009).

In addition to the JSFR and the KALIMER, the BN series of NPPs by the Russians are currently being investigated (Saraev et al. 2010), but face similar challenges to the SFR NPPs. A cost effectiveness strategy will be dependent on achievable economies of scale. Another project is the SMFR (a SMR with 125MWt) with a S-CO₂ Brayton cycle. It is considered to be more compact, more efficient and less expensive than the Rankine cycle. However, Locatelli et al. (2013) claim it is less proven due to the need for an intermediate cycle to deploy non-radioactive sodium, as the reaction between carbon dioxide and sodium could lead to a LOCA (Chang et al., 2007). Nonetheless, Chang et al. (2005) point out the benefit of coupling the plant to a desalination plant without loss of net efficiency.

A Toshiba design called 4S (super-safe, small and simple) denotes the need for small compact solutions. The 4S is a 30MWt Rankine cycle with a net output of 10MWe, a 30 year refuelling cycle and low maintenance requirements due to its choice in components. It is intended for construction underground (Locatelli et al. 2013). PRISM (power reactor innovative small module) is the final project reported in this SFR section. PRISM is being developed by GE Hitachi, which is being considered in the UK for reprocessing of actinides from LWRs and for producing electricity (Triplett et al., 2012). Table 3-4, which is taken from Locatelli et al. (2013) summarises the notable aspects of the main SFR projects:

Table 3-4 A summary of main SFR projects (Locatelli et al. 2013)

	SFR target	JSFR	KALIMER	SMFR
Thermal/electrical power (MW)	1000–5000/400–2000	3570/1500	1523/600	125/50
Fuel	Oxide–metal alloy	TRU–MOX	U–TRU–10%Zr	U–TRU–10%Zr
Fuel cycle	Aqueous–pyro-metallurgical	Aqueous	Pyro-metallurgical	Pyro-metallurgical
Breeder ratio	0.5–1.3	1.03–1.2	1.0	1.005
RPV layout	Loop–pool	Loop	Pool	Pool
Outlet core temperature (°C)	530–550	550	545	510
Thermodynamic cycle	Brayton	Rankine	Rankine	Brayton
Working fluid	S–CO ₂	Steam	Steam	S–CO ₂
Net electrical efficiency (%)	> 40	42	39.4	38
Economics	Medium	Less than LWR	Less than LWR	Less than LWR

3.4.5 Super-Critical Water-cooled Reactor (SCWR)

SCWRs are thermal fast reactors with SC water used as the cooling medium. They owe their design and development to the BWR and are considered an evolution from the Generation III BWRs. The main differences between the Generation III and SCWRs are reduced flow rate water for reactor cooling, which allows the pipe and pumps to be smaller. In addition, plant layout can be simplified because the single phase coolant removes the need for steam dryers and recirculation assemblies and systems. Most critical is the increased net efficiency for electricity generation, which is estimated to be 10% higher than Generation III BWRs (Locatelli et al. 2013). As a direct comparison, the construction, operation and maintenance costs are predicted to be lower than current LWRs, but SC water poses corrosion problems and will be critical to the material section process of the core (Sun et al., 2009; Was et al., 2007).

Several SCWR developments are currently being worked on, such as the super LWR (SLWR) and high performance LWR (HPLWR), which are tailored for electricity production, with outputs of 1000 MWe for both NPPs (Locatelli et al. 2013). There are similarities in the PCS layout and cores, which adopt water rods and UO₂ as fuel. The difference is in the core layout - the HPLWR uses a three-pass core, whilst the SLWR uses a two-pass simpler core (Ishiwatari, et al., 2007; Yamaji, et al., 2005). The SCWR programme envisages an experimental design demonstration by 2020 (Ishiwatari et al. 2007). The HPLWR aims to develop a commercial NPP before 2035 through technological advancements using previous LWR (Schulenberg et al. 2011; Starflinger 2010), but with Germany leading the research consortium and their position post the Fukushima incident in 2011, the progression of this project is uncertain.

Other projects include the super fast LWR (SFLWR), which is similar to the SLWR but with a fast spectrum fuelled core. According to Oka et al (2011), the main advantage of this design is a higher power density and the improved management of actinides (Locatelli et al. 2013). The CANDU programme also has a CANDU-SCWR, which has an improved net overall thermal efficiency with a supercritical Rankine cycle, a COT of 625 degrees Celsius and 2540 MWt (Locatelli et al. 2013), with ability for additional reheat in the Rankine cycle (Chow & Khartabil 2007). However, the thermodynamic cycle is not fully understood in terms of the economic evaluation (Naidin et al. 2009), emphasising the need for this assessment to be completed. The CANDU-SCWR deployment timeframe is 2025-2080 but there is no timeframe for a demonstration reactor (Torgerson et al., 2006). Table 3-5, which is taken from Locatelli et al. (2013) summarises the notable aspects of the main SCWR projects:

Table 3-5 A summary of main SCWR projects (Locatelli et al. 2013)

	SCWR target	HPLWR	Super LWR	Super-Fast LWR	CANDU-SCWR
Thermal/electrical power (MW)	3860/1700	2300/1000	2300/1000	2358/1000	2540/1220
Spectrum	Thermal/fast	Thermal	Thermal	Fast (breeder ratio $\approx$ 1)	Thermal
Fuel	UO ₂ /MOX	UO ₂	UO ₂	MOX	UO ₂ /Th
Moderator	Light water	Light water	Light water	Light water	Heavy water
Reactor layout	RPV	RPV	RPV	RPV	Pressure tubes
Outlet core temperature (°C)	550	500	500	500	625
Core pressure (MPa)	25	25	25	25	25
Net electrical efficiency (%)	44	43.5	43.8	43.8	48
Economics	High	Higher than LWR	Higher than LWR	Higher than LWR	Higher than LWR

3.4.6 Very-High Temperature Reactor (VHTR)

VHTRs are high temperature thermal reactors, cooled by helium in gaseous phase and moderated by graphite in the solid state. The core has a COT of 750-850 degrees Celsius and can be capable of 1000 degrees Celsius (subject to material capability). Their main advantage is potential increases in overall plant efficiency, attributed to the high thermal capability with a helium Brayton cycle. Helium as a chemically and radiologically inert gas is stable at all conditions in the thermodynamic spectrum (Giacobbe 1998), with graphite providing the necessary thermal conductive and intensified specific heat capacity. This is appropriate for accidental events as it can slow down transitions (Locatelli et al. 2013).

Pebble bed and prismatic cores are the two main options for VHTR according to Locatelli et al. (2013). VHTR is considered a SMR, and according to the GIF (2014), it is the only concept out

of the six proposed Generation IV technologies that is suitable for optimum efficiency hydrogen production, but this research and development into its viability is still in its early phase. Shin et al. (2013) claim that the layout and technical aspects of the hydrogen facility have to take into account the development of heat exchangers and coolant gas ducts and valves, which need to be isolated.

The safety aspects are addressed in Bertrand et al. (2011), and a comprehensive study on the economics of hydrogen production using HTGR with high temperature steam electrolysis is provided by Patterson, et al. (2010), who adopt methods that can be incorporated into a techno-economical assessment of the cycle. Nonetheless, the economics need to be proved on a holistic scale (electric and hydrogen production). Carelli et al. (2010) claim that the economies of SMR are comparable with larger plants, so there may be some partial compensation if a VHTR is also used for industrial chemical production.

Current projects include the high temperature gas-cooled reactor-pebble bed module (HTR-PM) and next generation nuclear plant (NGNP), which are two of the lowest operating COTs (750 degrees Celsius) for the VHTR. They adopt an indirect subcritical Rankine cycle. The HTR-PM of which the demonstrator is the HTR-10 reactor (Scari et al. 2015), is a Chinese project that utilises two cores with 250 MWt capability to produce 210 MWe (Zhang et al. 2009). An economic assessment is presented in Boarin et al. (2012), which covers economies of multiples and the learning from replication and co existing economies.

The other project (NGNP) is an American VHTR programme with a target thermal power of 600 MWt, intended for use as electric NPP and industrial cogeneration (McKellar et al., 2010; Mills et al., 2007). Further programmes include the gas turbine-modular helium reactor (GT-MHR), the HTTR, the PBMR and the gas turbine high temperature 300 (GTHTR300). The GT-MHR is an American/Russian project, with a COT of 850 degrees Celsius, and employs a direct helium Brayton cycle, thereby increasing net efficiency and decreasing construction costs as a result of its modular approach. The GT-MHR has the highest net electrical efficiency amongst the VHTR with dry cooling implementation in the cycle (Baxi et al., 2008; Labar, et al., 2003).

The high temperature engineering test reactor (HTTR) is a Japanese project constructed by the Japan Atomic Energy Research Institute (JAERI). The project is a 30 MWt reactor which aims to develop HTR technology, hydrogen production technology and integration with the actual reactor (Takizuka 2005). The project has improved the efficiency of components from the initial development phase, especially heat exchanger performance (Goto et al. 2012). This will help with business case justification of future NPPs if there is a technology transfer. The PBMR was a South African program for the development of the PB core for the VHTR. It had a reported thermal power of 400 MWt and a COT of 900 degrees Celsius, with a direct helium Brayton cycle for electricity production. A prototype was due for commissioning in 2010, but the programme was cancelled in 2010, citing technical issues and an unproven technology with the helium gas turbine (Locatelli et al. 2013), and an increase in reactor thermal power from 267 MWt without any significant modification in the engineering design (Thomas 2011).

Earlier research on a 115MW, three shaft helium Brayton cycle for the PBMR was completed by Pradeepkumar (2002) to assess flow characteristics and the impact on plant performance during load control. Pradeepkumar's study (2002), including publications from Pradeepkumar et al (2001a), Pradeepkumar et al (2001c), Pradeepkumar et al (2001b), Pradeepkumar et al (2002a) and Pradeepkumar et al (2002b), provide useful bases for the research. The GTHTR300 is explained further in the performance modelling and analysis work carried out for this assessment period as part of the overall study. Table 3-6, which is taken from Locatelli et al. (2013) summarises the notable aspects of the main VHTR projects:

Table 3-6 A summary of main VHTR projects (Locatelli et al. 2013)

	VHTR target	HTR-PM	NGNP	GT-MHR	GTHTR300C
Thermal/electrical power (MW)	600/300	2 × 250/210	600/240	600/286	600/274
Core layout	Prismatic block/pebble bed	Pebble bed	Prismatic block/pebble bed	Prismatic block	Prismatic block
Fuel	TRISO clad with ZrC	TRISO	TRISO	TRISO	TRISO
Outlet core temperature (°C)	850	750	750	850	850
Outlet core pressure (MPa)	9	7	–	7	7
Thermodynamic cycle	Brayton	Rankine	Rankine	Brayton	Brayton
Net electrical efficiency (%)	50	42	40	48	46
By-products	Hydrogen production	–	Industrial cogenerative application	Production of hydrogen and desalinized water	Production of hydrogen
Economics	High	Similar to LWR	Similar to LWR	Higher than LWR	Higher than LWR

Generation IV systems currently in development aim to bring about benefits that will provide a cost effective solution for power generation. However, the cycle costs and the relationship between plant capacity and the efficiency of the cycle and plant are not fully understood in the majority of cases. In addition, some of the solutions appear to introduce complexities to the design, and these will be detrimental to the initial capital and start-up costs of the plants. Although the stockpile for natural resources such as uranium is not of global concern (Harper et al. 2015), the ability to extract more energy from the fuel is definitely a critical feature in the choice of reactors, thus an indication of the design burn-up is key.

György & Czifrus (2015), recently performed burn-up calculations on all Generation IV reactor designs. Their study indicated that the decision on the choice of technology is heavily reliant on the type of reactor configuration and coolant. Small modular reactors are seen as a way to solve the issues of fuel use, thereby reducing the amount of waste from the fuel cycle. Recent reports by the Institution of Mechanical Engineers (IMechE) (Hibbert 2014) suggest that government policies, and indeed the power companies, may favour small modular reactors (SMR). SMRs have also been developed by several countries such as Japan and France with a thermal capacity range of 50 MWt to 125 MWt (Yan et al. 2012; Chang et al. 2005; Chang et al. 2007). Their development aims to exploit the characteristics of inherently fast reactors because they satisfy some near term commercial niche, whilst providing the possibility that a truly advanced reactor technology can be scaled and made available for long term cogenerating applications.

Boarin et al. (2012) also echo the attractiveness, but propose small to medium size reactors. These have an output capability of 300 MWe to 700 MWe (Carelli et al., 2010). Locatelli & Mancini (2010) suggest that the attractiveness lies in the easier plant to grid matching. This is because small grids and old infrastructures are more suited to small-medium size NPPs, and will not be compatible with large concentrated NPPs. Furthermore, the modularity of the assembly will mean shorter construction times for a fraction of the price of a large plant, with more flexibility in the capacity, and increased adaptability to market changes (Locatelli & Mancini 2010).

Nian's (2017) study for the Association of South East Asian Nations (ASEAN) suggests that SMRs are a strategic option in ASEAN's long-term energy planning, but emphasises the need for competency development. With a lack of visibility on the current technological capabilities of SMRs, Rowinski et al. (2015) provide an in-depth review of numerous designs currently under development with a focus on the technology, safety features, and ability to mitigate proliferation. SMRs pose significant challenges when it comes to the issue of non-proliferation and safeguards. Prasad et al. (2015) discuss several aspects to assess non-proliferation, such as fissile inventory, core life and refuelling, high burn-up, material inventory accountancy, reactor location (over or underground), sealing design, enrichment levels, excess reactivity and fuel element size. SMRs must undergo thorough scrutiny under the aspects discussed before they are viewed as credible, and will need to demonstrate their capabilities before they are used in NPPs.

4 Nuclear Operating Cycles

The focus of this research within the grand scope of nuclear power plant Generation IV design and development is performance design, development and operation of the cycles. Pioro et al. (2016) suggest that cycles for Generation IV NPP is an area which needs much further research. The choice of performance operating cycle(s) is significantly driven by the thermo-economic analysis of the system because it is important to get the right balance between high thermal efficiency and capital costs (Pooneh 2012).

Input into the thermo-economic analysis requires an understanding of the thermodynamic performance of the system, which encompasses the gas turbine, the reactor and the heat exchangers. This is determined by the use of mass conservation and energy equations. System calculations in most cases adopt the classical thermodynamics approach, which requires properties such as pressure, temperature, enthalpy and entropy to calculate the properties at various states. The first law of thermodynamics provides the concept of balance of energy within a typical plant. The second law provides a framework for calculating energy inefficiencies and quantification of the losses within the system. The fundamentals of thermodynamics and energy conversion within a system is extensively covered by (Decher 1994). The theory of gas

turbine performance and the thermodynamics of the system in different applications is covered extensively in Saravanamuttu et al. (2009), Walsh and Fletcher (2004) and Palmer & Pilidis (2010).

Gas turbines used in industrial applications can be classed as open or closed cycle. In an open cycle gas turbine, the working fluid is atmospheric air which flows constantly through the compressor, where work is performed on the fluid to raise the pressure and temperature beyond that of the atmosphere. The combustor burns a fuel of a given type to a given stoichiometric quantity, to heat up the air, thereby further raising the pressure and temperature prior to the working fluid being expanded and the power extracted in the turbine. The wasted heat can be expanded at atmospheric pressure if used on a compact industrial rig, or can be accelerated in a nozzle if utilised in an aerospace propulsive application. It can also be incorporated as part of a combined cycle, where the waste heat can be recovered and used for other purposes via various methods.

In a closed application, the working fluid is typically water, air, nitrogen or chemically inert gases such as helium or argon. The working fluid is cycled through the components, as is the case with an open cycle. However, heating is provided by an external source such as a nuclear reactor. The working fluid which is now the hot gas is not released into the atmosphere but is sent through additional components, such as a heat exchanger, to transfer heat to the cold working fluid and raise the temperature of the flow aft of the compressor and fore of the heat source, thereby improving the efficiency of the cycle. Commercially, the continuing interest in closed cycle gas turbines as a prime mover for the generation of electricity has always been apparent (Pradeepkumar 2002).

With regards to nuclear applications and this study, the focus will be on closed cycles. Closed cycles can be based on a Brayton or Rankine cycle, and each cycle can be directly or indirectly coupled.

4.1 Rankine Cycles

The majority of plants currently being operated are based on the Rankine or steam cycle. Such plants employ the LWR configuration explained earlier in this report. With reference to BWR or

PWR plants, the primary flow circuit is filled with water, which undergoes a phase transformation into steam in the steam generator and expands within the turbine. Additional components include a condenser that enables a phase change back into water, and a pump to return the water back to the start of the cycle is included. The ideal Rankine cycle is based on the Carnot cycle, and is denoted in the T-s diagram in figure 4-1:

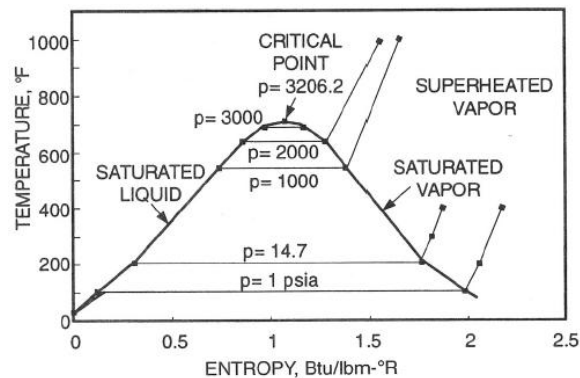


Figure 4-1 Typical steam ideal cycle (Black and Veatch, 1996)

There are four processes involved in the Rankine cycle, as shown in the amended T-s diagram in figure 4-2. Firstly, water is pumped (1-2) prior to heat being added at constant pressure in the steam generator (2-3'). The resultant steam is expanded through the turbine (3'-4'), prior to the vapour at the exhaust being cooled via the condenser at constant pressure and temperature to revert the phase (4'-1). A super reheated Rankine cycle can be adopted, which is denoted as 1-2-3-4 in the process description in figure 4-2. The superheated cycle involves heating the saturated vapour at the turbine exhaust. Gains are made by ensuring the superheated temperature is as high as possible to increase efficiency. Wark et al. (1998) argue that the temperatures should be limited to 600 degrees Celsius, but developments in materials can improve resistance to fatigue thereby bringing about improvements of the cycle in currently operated LWR plants.

Another benefit is the superheat results in a drier steam vapour at the turbine exhaust, thus reducing the corrosive environment within the turbine. The deviation from the typical steam curve illustrates inefficiencies in the cycle due to components.

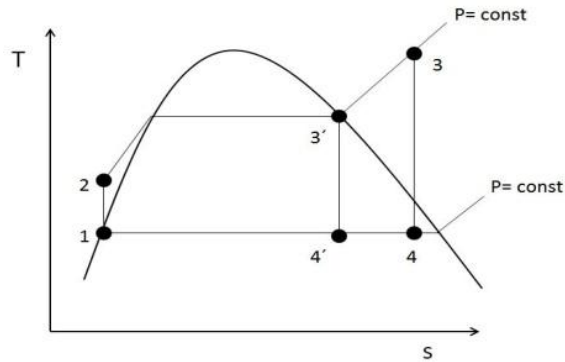


Figure 4-2 Ideal vs real Rankine saturated and super heated cycle (Black and Veatch, 1996)

Assuming unit mass steady flow and ignoring changes in kinetic and potential energies, the first law of the Rankine cycle is:

Heat added in the boiler:

$$q_{inlet} = q_{23} = h_3 - h_2 \dots \dots \dots eq 1$$

Turbine work:

$$W_{outlet} = W_{34} = h_3 - h_4 \dots \dots \dots eq 2$$

Rejected heat in the condenser:

$$q_{outlet} = q_{41} = h_1 - h_4 \dots \dots \dots eq 3$$

Compressor/pump work:

$$W_{inlet} = W_{12} = h_2 - h_1 \dots \dots \dots eq 4$$

The cycle thermal efficiency of the Rankine cycle can be expressed as:

$$\eta_{rth} = 1 - \frac{h_4 - h_1}{h_3 - h_2} \dots \dots \dots eq \ 5$$

Several other modifications can be made to the cycle to increase thermal efficiency, such as including an indirect reheat cycle across a HPT and LPT. The steam is expanded partially in the HPT prior to entering a second boiler downstream, where the steam is reheated to its fore HPT temperature. The reheated steam is expanded in the LPT to the pressure conditions suited for the condenser (Hussain 2010). The increase in efficiency makes the reheat cycle attractive, but the geometry especially for the pipework can become complex, thereby increasing plant capital and maintenance costs. Figure 4-3, which is taken from Hussain (2010) but attributed to the University of Oklahoma (2004), shows a typical reheat Rankine cycle:

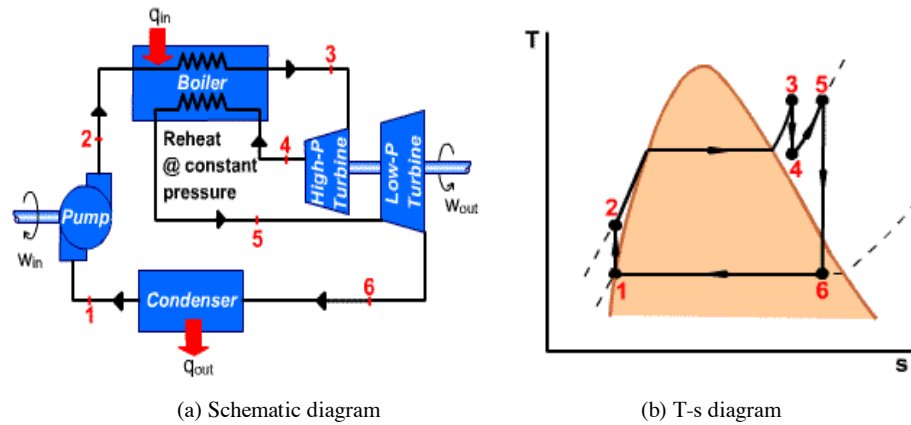


Figure 4-3 Typical reheat Rankine cycle (Hussain, 2010, taken from University of Oklahoma, 2004)

When the superheat cycle is considered, the cycle efficiency of the Rankine reheat steam cycle becomes:

$$\eta_{rrth} = 1 - \frac{h_6 - h_1}{(h_3 - h_2) + (h_5 - h_4)} \dots \dots \dots eq 6$$

A further modification to the indirect Rankine cycle is to adopt regenerative heating, which enhances thermal efficiency by about 3% (Ishiyama et al. 2008). Finite temperature deltas during heat transfer play a dominant effect in irreversibility, and hence inefficiency. The regenerative cycle resolves this by bleeding off steam from the turbine to heat the water prior to entering the boiler. This is achieved using heaters (internal heat exchangers) (Hussain 2010). The resulting improvements in thermal efficiency reduce the specific power due to a drop in enthalpy for a given plant size (Abaad 2012). If the plant with internal heaters can be compacted, then the costs of the plant could be reduced. However, in some cases, the task includes complexities and the overall cost benefits to the plant are not necessarily realised because of factors that affect effectiveness, including the choice and specific heat of the cooling fluid.

Improved modern plants such as typical LWRs with Generation III+ technologies operate an indirect supercritical Rankine cycle. The feedwater is pressurised beyond the critical pressure of 22.12 MPa (Ishiyama et al., 2008). When heat is added, the volume and temperature of the working fluid rises, but the two-phase region is not reached. Instead a supercritical fluid is realised, thus there is an indistinguishable scenario between the liquid and gas phase. The supercritical cycle has a better heat capacity at the maximum permissible temperatures than a subcritical cycle operating at typically 7-16 MPa (Naidin et al., 2009), which can improve thermal efficiency by about 2-3% (Zhang et al., 2009).

With regards to the scope of this research study, Generation IV technology concepts that utilise the Rankine cycle within the primary flow circuit are not considered. This is because the thermodynamic performance requirements suggest that the Rankine cycle does not provide high enough thermal efficiency for a compact cost effective solution. The focus is on the Brayton cycle because of the thermal efficiency benefits, and the possibility of reducing the cycle inventory and improving part load performance.

4.2 Brayton Cycles

The Brayton cycle has been well developed for use with various NPP components and considerable efforts were devoted to enhancing the Brayton cycle in the 1960s (Bhargava et al., 2007). The use of gas turbines in Brayton cycles is possible due to their ability to handle large mass flow with a reasonable operating efficiency (Decher 1989). There are notable differences between the Brayton and Rankine cycles. The Brayton comprises a compressor and precooler, with gas as the working fluid. The Rankine cycle includes a pump, condenser and water as the working fluid.

The possibility of heat rejection at a suited average temperature gives closed Brayton gas cycles an advantage over typical Rankine steam power cycles. Typically, condensers in Rankine cycles operate at 4.3 kPa, corresponding to a steam saturation temperature of 30 degrees Celsius (Peterson & Zhao 2006). High coolant flow rates are required to minimise coolant temperature increases in addition to the kink-point temperature to ensure that heat is rejected from the condensing steam.

On the other hand, the intercoolers and precoolers of closed cycles have inlet temperatures of approximately 85-165 degrees Celsius, and outlet temperatures of approximately 55-95 degrees Celsius (Peterson & Zhao 2006), which are higher than Rankine cycles. The temperature difference between the coolant and the gas offers an attractive option to use dry coolant, where heat rejection is offered by cooling recirculated water. This is particularly suited for areas with water resource scarcity. The biggest advantage of the Brayton cycle over the Rankine cycle employed within LWRs is the efficiency attainable. Typical efficiencies for the direct Brayton cycle are approximately 47% - 51%, in comparison with 32% for conventional LWRs (Baxi et al. 2008).

The Brayton cycle comprises four stages – compression (1-2 or 2'), heating (2 or 2' – 3 or 3'), expansion (3 or 3' – 4 or 4'), and cooling (4 or 4' - 1) (Ablay 2013). Figure 4-4 denotes a typical T-s diagram of the Brayton cycle for real (denoted by ') and ideal cycles:

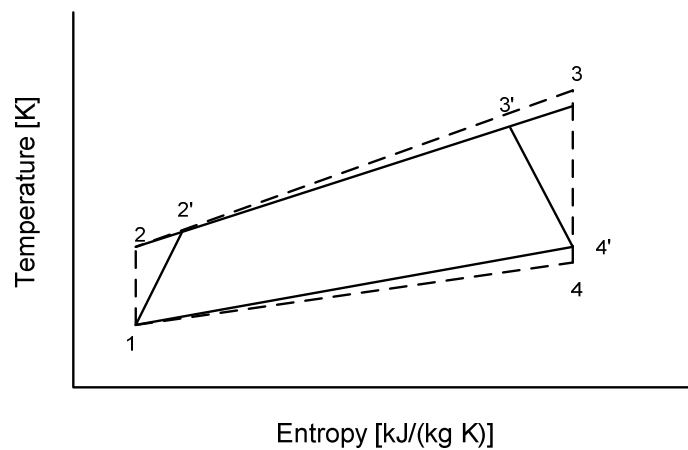


Figure 4-4 T-s diagram of the real and ideal Brayton cycles (Verdiere 2010)

Compression work is lower than turbine work, thus positive work from the cycle can be used to drive a load, but component efficiencies mean that the compression and expansion phases are not isentropic. As a result, heating and cooling of the cycle is not achieved at constant pressure, hence losses are observed in the cycle. The losses translate into more work input required for the compression process due to increase in temperature, resulting in a higher exit temperature. In addition, heat addition into the cycle is not isobaric (i.e. at constant pressure). This reduces total gas exit pressure, thus the total power extraction possible is reduced due to reduced gas exit pressure and reduced component efficiency. Exhaust heat is typically hotter than expected, which makes the compression inlet temperature hotter than ideal. The process described above is the direct simple Brayton cycle. A typical plant layout is depicted in figure 4-5:

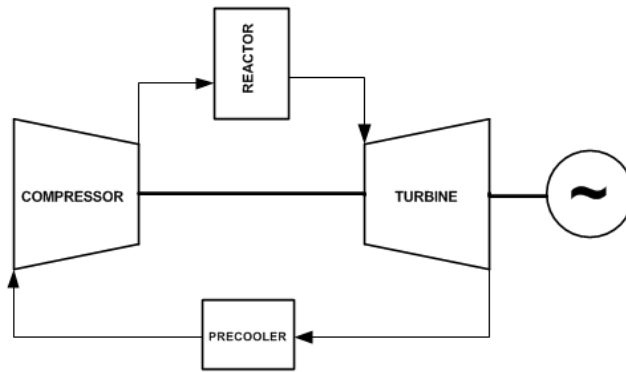


Figure 4-5 Typical simple Brayton cycle (Grochowina 2011)

The energy exchange within the cycle, which is known as enthalpy, is expressed as:

$$\Delta H = W \cdot C_p \cdot \Delta T \dots \dots \dots eq \ 7$$

Temperature rise is proportional to the heat flow within the cycle. If the quantity of heat undergoes no change when the state is altered from one point of the cycle and returned to the same point through a reversible path, then this heat quantity must be a property that can be expressed as having values that are dependent on only state variables (Decher, 1994). The property of the system defined through the reversible heat transfer process (entropy) is expressed as:

$$dS \equiv \frac{dQ_{rev}}{T} \dots \dots \dots eq \ 8$$

whereby for a process with end states, the change in entropy is given by:

$$\int_1^2 dS = \int_1^2 \frac{dQ_{rev}}{T} = S_2 - S_1 \dots \dots \dots eq \ 9$$

The existence of entropy in the system is a useful parameter to quantify reversibility within the Brayton cycle. If we consider an ideal gas, $S_2 - S_1$ becomes:

$$S_2 - S_1 = Cp \cdot \ln\left(\frac{T_2}{T_1}\right) - R \cdot \ln\left(\frac{P_2}{P_1}\right) \dots \dots \dots eq 10$$

The addition of the precooler ensures that the gas can be cooled using a heat sink medium or dry air as studied by Conboy et al. (2014). Cooling at the compressor entry achieves the temperature or lower for point 1 (1') as denoted in Figure 4-6. This reduces compressor work, but also reduces compressor exit temperature, which will increase the input thermal power in the reactor. The reactor thermal power is fixed for a given exit COT. A precooled Brayton simple cycle T-s diagram is shown in figure 4-6:

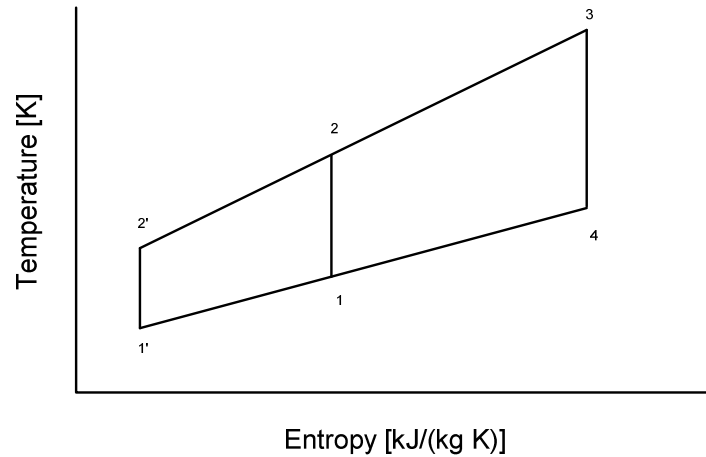


Figure 4-6 T-s Brayton cycle with precooler (Verdiere, 2010)

As reactor thermal power is fixed for a given exit COT, the precooler cycle will not yield the specific work required for the NPP. This devalues the economics of the plant, as more fuel will be required to raise the COT. The problem is resolved by introducing a recuperator, as shown in figure 4-7:

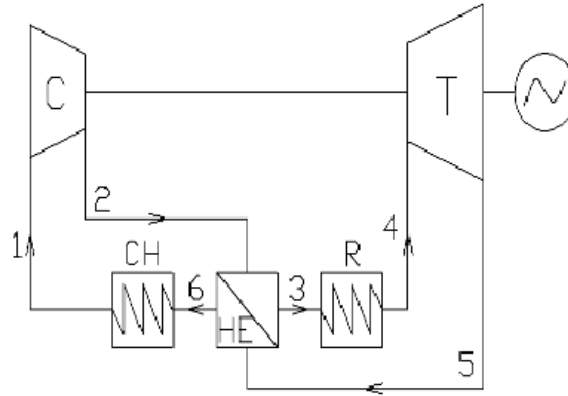


Figure 4-7 Typical simple cycle with recuperator (Grochowina, 2011, taken from Kulhánek, 2007)

Typical NPP plants that incorporate a direct simple Brayton cycle include a recuperator. In the recuperated Brayton cycle, heat energy from the turbine outlet gas is used to preheat the working fluid, thus raising the temperature to reduce the amount of thermal heat input into the cycle from the reactor. The recuperated cycle introduces pressure drops in the cycle. This impacts cycle thermal efficiency in addition to the costs of the recuperator. This is mitigated by ensuring a high effectiveness, because a reduction in recuperator effectiveness has a greater impact on overall plant efficiency (Bhargava et al. 2007). Figure 4-8 illustrates the T-s diagram for the recuperated cycle:

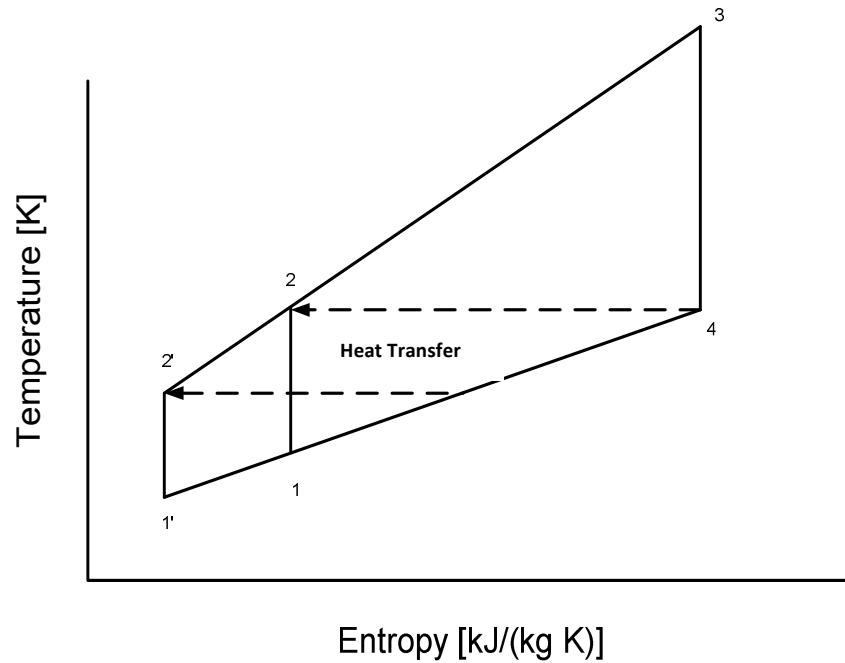


Figure 4-8 T-s Diagram of the Brayton cycle with recuperator (Verdiere, 2010)

The useful work can be further improved for a given plant capacity by introducing intercooling within the cycle, as shown in figure 4-9:

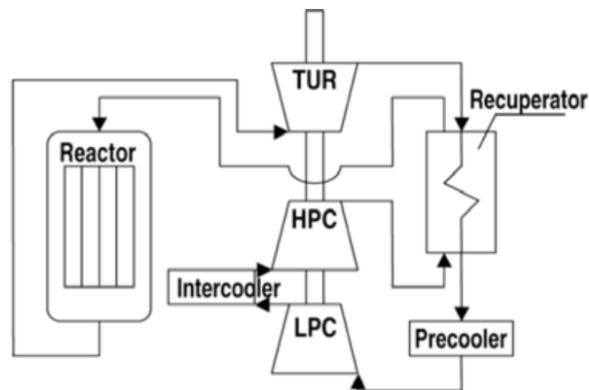


Figure 4-9 Typical simple intercooled Brayton cycle with pre-cooler, recuperator (Baxi et al. 2008)

This requires an intercooler and a second compressor (HPC) for the same turbine work. Improving the useful work requires a reduction in compressor work. The useful work is:

$$UW = TW - CW \dots \dots \dots eq 11$$

and the compressor work is the product of the working fluid massflow, the specific heat and the delta temperature:

$$CW = W_f \cdot Cp \cdot \Delta T_f \dots \dots \dots eq 12$$

For a given pressure ratio, the delta temperature (the difference between the inlet and exit compressor temperature) is determined by finding the exit temperature:

$$T_2 = T_1 \left[1 + \frac{\left(\frac{P_2}{P_1} \right)^{\frac{\gamma-1}{\gamma}} - 1}{\eta_{isc}} \right] \dots \dots \dots eq 13$$

With $\Delta T_f = T_2 - T_1$ and with all other parameters fixed, ΔT_f is reduced to a lower temperature in the intercooler prior to entry into the second compressor, with negligible reduction in pressure observed in T_2 . With the addition of an intercooled cycle, the pressure ratio for both compressors is determined as:

$$PR_{ic} = \sqrt[ic]{PR} \dots \dots \dots eq 14$$

Whereby the ic coefficient denotes the number of intercoolers in the cycle +1. This leads to a reduction in the pressure ratio per compressor. Figure 2.16 shows the T-s diagram of an intercooled cycle. It is noted from studies of simple versus intercooled cycles that the intercooled cycle increases thermal efficiency by approximately 2%. However, the additional plant components increase plant complexity, and this was not necessarily favoured for the development of the GTHTR300 design (Yan et al., 2003).

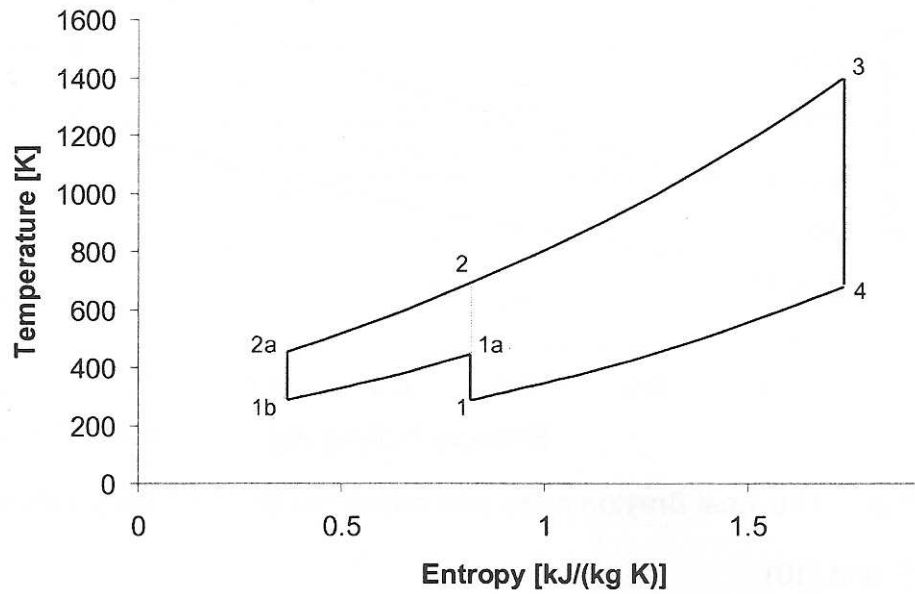


Figure 4-10 T-s Diagram for the intercooled Brayton cycle (Verdiere, 2010)

A reheat cycle can be included to increase the specific work of the plant. This is achieved by separating the expansion process and reheating the gas between the HPT and LPT (Saravanamuttoo et al., 2009). The reheat cycle takes advantage of the fact that regarding the thermodynamics, constant pressure lines diverge when plotted on a T-s diagram (Bhargava et al., 2007). This results in a work extraction process that is considerably higher if compared to a single expansion between the two pressure lines, but the net power output of the overall cycle is dependent on the intermediate pressure of the expansion. A particular advantage of using a reheat cycle is the implementation of a combined power cycle to reheat a HRSG with higher enthalpy conditions.

A parametric study by Bhargava et al. (2007), showed that as much as 25% of specific work increase can be realised with reheat. Figure 4-17 shows a schematic of a three horizontal shaft distribution of multiple reheat loops using 3 turbomachineries. Each module is connected to a compressor, turbine, generator, cooler, heater and recuperator. The arrangement pattern in figure 4.17 is part of a cycle being investigated for a sodium cooled fast reactor, utilising a Brayton cycle (Zhao & Peterson 2008). The pattern is arranged to maximise the total length of

hot ducts, which they suggest are more expensive than cold ducts and generate heat losses (Zhao & Peterson 2008).

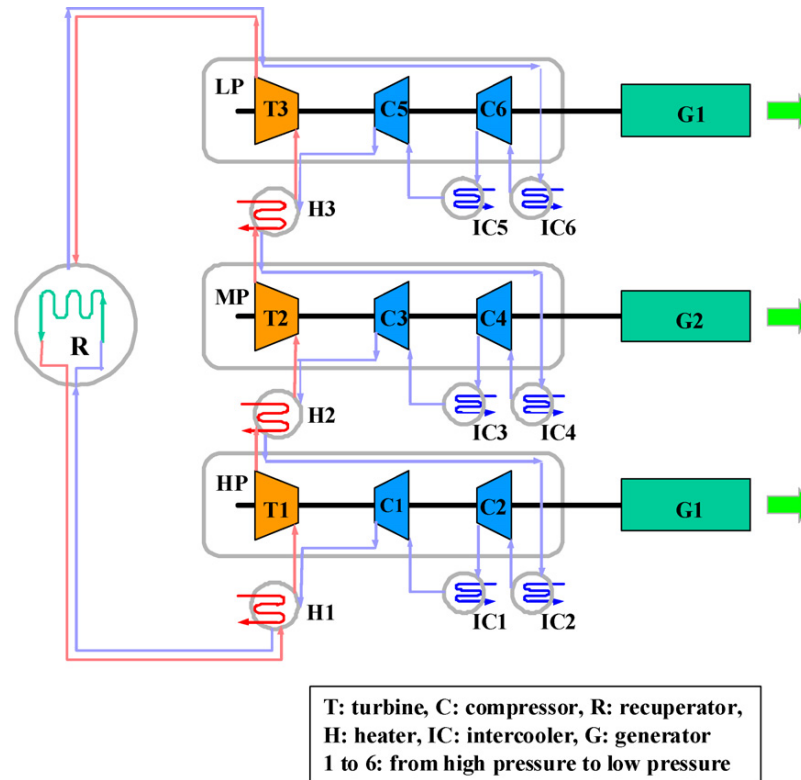


Figure 4-11 Schematic flow diagram for the distributed horizontal shaft multiple reheat cycle using 3 turbomachineries (Zhao & Peterson 2008)[1]

There is an increase in constant pressure lines as entropy increases, $(T_3 - T_4) + (T_5 - T_6) > (T_3 - T'_4)$ (see figure 4-12). If it is assumed that the gas is heated to the same temperature as T_3 , the expression for specific work dictates that the optimum period in the expansion for which reheat should occur is when the expansion pressure ratio and the accompanying temperature drop during work extraction for the HPT and LPT are equal (Saravanamuttoo et al., 2009). This is known as the intermediate expansion process. The expression for the reheat cycle for specific work and efficiency becomes:

$$SW_{re} = \frac{W}{CpT_1} = \left\{ \left[2 \left(\frac{T_3}{T_1} \right) \right] - \left[\left(\frac{P_3}{P_4} \right)^{\frac{\gamma-1}{\gamma}} \right] + \left[1 - \frac{2 \left(\frac{T_3}{T_1} \right)}{\sqrt{\left(\frac{P_3}{P_4} \right)^{\frac{\gamma-1}{\gamma}}}} \right] \right\} \dots \dots \dots eq \ 15$$

$$\eta_{reth} = \left\{ \frac{\left[2 \left(\frac{T_3}{T_1} \right) \right] - \left[\left(\frac{P_3}{P_4} \right)^{\frac{\gamma-1}{\gamma}} \right] + \left[1 - \frac{2 \left(\frac{T_3}{T_1} \right)}{\sqrt{\left(\frac{P_3}{P_4} \right)^{\frac{\gamma-1}{\gamma}}}} \right]}{\left[2 \left(\frac{T_3}{T_1} \right) \right] - \left[\left(\frac{P_3}{P_4} \right)^{\frac{\gamma-1}{\gamma}} \right] - \frac{\left(\frac{T_3}{T_1} \right)}{\sqrt{\left(\frac{P_3}{P_4} \right)^{\frac{\gamma-1}{\gamma}}}}} \right\} \dots \dots \dots eq \ 16$$

Studies of typical specific work outputs for closed cycle gas turbines indicate that the reheat cycle remarkably increases the specific output, but at the expense of efficiency if it is operated over a smaller temperature range (Saravanamuttoo et al. 2009). If the cycle is operated at its maximum temperature, then the reduction in efficiency can be minimised. Adding a recuperator to the cycle improves initial cycle efficiency without compromising the specific work; in fact, efficiency is higher with a reheat and recuperator than without a reheat. The economics of the cycle versus the complexities of NPP plant construction require further understanding and research to quantify the benefits and limitations.

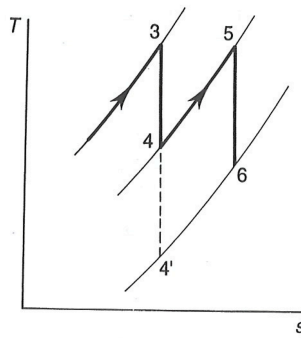


Figure 4-12 Reheat cycle (Saravanamuttu et al., 2009)

With regards to coolants and working fluid, the Brayton cycle enables the working fluid to be used as a coolant for the reactor, but the loop is separated from the primary loop. Reactor and turbine cooling are important aspects of plant and cycle design, but the decision to use the working fluid as a coolant depends on the chemical stability of the fluid. The introduction of cooling into the direct Brayton cycle means that secondary loops have to be created, therefore the cycle becomes an indirect cycle. Indirect Brayton cycles are employed as part of a combined cycle plant, whereby the Brayton cycle primary loop is supported by a secondary loop that uses another Brayton cycle, or a Rankine cycle employing a HRSG to produce superheated steam for other plant processes (Bhargava et al., 2007).

A typical indirect and combined cycle in addition to a direct Brayton cycle (48.2% efficiency) are being considered for gas-cooled fast reactors (GFRs) as part of a study by the French Alternative Energies and Atomic Energy Commission (CEA). Dumaz et al. (2007) list the two cycles. The first is a nitrogen Brayton cycle of 46.8% efficiency, with a primary loop/circuit temperature inlet and outlet of 480 – 850 degrees Celsius at 7 MPa pressure, and a secondary loop circuit of 444 – 820 degrees Celsius at 6.5MPa pressure. The second is an indirect combined cycle with helium in the primary loop, helium or nitrogen in the second loop and steam in the tertiary loop with 45% efficiency. Temperatures are 400-850 degrees Celsius at 7 MPa in the primary, 364 – 820 degrees Celsius at 6.5 MPa in the secondary, with a subcritical steam pressure of 15 MPa. A case was made for 2400 MWt using the indirect combined cycle.

Dumaz et al. (2007) argue that a significant improvement in efficiency (2-3%) is expected using a supercritical Rankine cycle of 15-30 MPa. However, operational experiences at those pressures require appropriate materials at the secondary and tertiary interface (Hough 2009). This would add to the expansive and complexity of the layout, whilst the cost effectiveness of cycle efficiency and plant costs is not fully understood.

A study by McKellar et al. (2010) investigated the effect of COT on electric power generation, hydrogen production and process heat. They produced a chart (see figure 4-13), which showed various power cycle thermal efficiencies as a function of COT including the Carnot cycle at 66% of its conversion efficiency. The conclusion with regards to thermal efficiencies as a function of COT was that as the COT increases, power cycle efficiency should increase. Power cycles that operate well at low COTs do not operate well at high COTs. This is well known for Rankine cycles because they operate well at 600 degrees Celsius in comparison to Brayton cycles, which is ideal for a COT of 900 to 950 degrees Celsius.

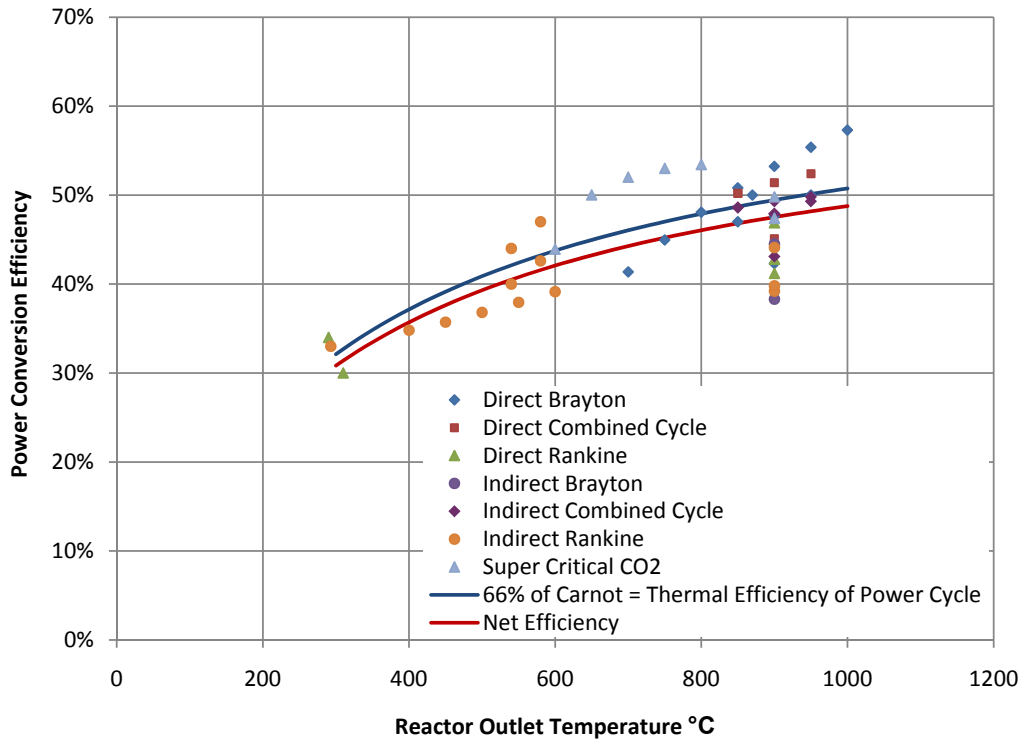


Figure 4-13 The net efficiency and 66% of Carnot conversion efficiency as a function of the ROT compared with various power cycles (McKellar et al. 2010)

The fundamental characteristics as described for closed direct and indirect Brayton cycles including combined cycles, represent characteristics of typical gas cooled fast high temperature reactors due to the benefits of high temperature heat supply and accompanying high thermal efficiency (Pradeepkumar 2002). All Brayton cycles used in NPP employ a method which recovers the waste heat back into the cycle. This reduces the compressor pressure ratio, and essentially the compressor work, thereby ensuring efficiencies in excess of 45% (Qu et al. 2017). In some cases, a heat recovery steam generator (HRSG) is employed in the secondary loop using the closed Rankine cycle (Qu et al. 2017). One cycle configuration yet to be investigated for NPP is the intercooled cycle without any recuperator. This has the benefit of simplifying the plant. The IC has experienced technological advancements with gas turbines such as the General Electric LMS100 which gives efficiencies in excess of 46% using air (Bhargava et al. 2007).

5 Investigating Working Fluids

Demand for plants with higher power levels post deployment of fossil-fired gas turbines with air as the working fluid instigated studies to evaluate other gases in a closed cycle loop. Performance analysis and component assessments were performed using gases such as helium, nitrogen, carbon dioxide, dinitrogen tetroxide and various other gaseous mixtures (McDonald 2012).

5.1 Dissociating Gaseous Mixtures as Working Fluids

Typical dissociating gaseous mixtures such as dinitrogen tetroxide were first investigated in the early 1970s in studies by Nesterenko (1972) and Krasin (1973) as future alternative coolants for nuclear reactors. Dissociating gases dissociate when heated and recombine when cooled. The benefit for NPPs is that the weight of heat exchange can be reduced within the turbomachinery (Ragheb 2014). To consider dinitrogen tetroxide:



The doubling of the number of working fluid gas molecules from n to $2n$, doubles the total work per unit mass in the ideal equation:

$$PV = 2nRT \dots \dots \dots eq\ 18$$

This doubling of the work done per unit mass of the working fluid allows the use of smaller sized turbomachineries and heat exchangers. Ragheb (2014) has proposed this for use in nuclear submarines to increase the attainable speed by 30% for the same reactor power. It also makes it applicable for space propulsion installations. Table 5-1 gives proposed dissociate gases, which include aluminium chloride and aluminium bromide. Table 5-2 shows characteristics of different turbines using steam and dissociating gases. Tables 5-1 and 5-2 were both taken from Ragheb (2014).

Table 5-1 Proposed dissociating gases (Ragheb, 2014)

Dissociating gas	Increase factor in gas constant	Thermal release from reaction [Kcal/mole]	Temperature Range °C
$N_2O_4 \rightleftharpoons 2NO_2$	2	13.7	25-170
$2NO_2 \rightleftharpoons 2NO + O_2$	1.5	27.0	140-850
$Al_2Br_6 \rightleftharpoons 2AlBr_3$	2	30.0	300-1,400
$Al_2Cl_6 \rightleftharpoons 2AlCl_3$	2	29.8	200-1,100
$Al_2I_6 \rightleftharpoons 2AlI_3$	2	26.4	230-1,200
$2NOBr \rightleftharpoons 2NO + Br_2$	1.5	-	25-500
$2NOCl \rightleftharpoons 2NO + Cl_2$	1.5	-	25-900
$Al_2Cl_6 + 4Al(liquid) \rightleftharpoons 6AlCl$	6	263.8	670-1,200
$Al_2Br_6 + 4Al(liquid) \rightleftharpoons 6AlBr$	6	282.4	670-1,400

$Al_2I_6 + 4Al(liquid) \rightleftharpoons 6AlI$	6	196.4	670-1,300
$HgCl_2 + Hg(liquid) \rightleftharpoons 2HgCl$	2	70.4	280-700
$HgBr_2 + Hg(liquid) \rightleftharpoons 2HgBr$	2	63.7	250-700
$SnCl_4 + Sn(liquid) \rightleftharpoons 2SnCl_2$	2	38.6	-
$SnBr_4 + Sn(liquid) \rightleftharpoons 2SnBr_2$	2	65.3	-
$Ga_2Cl_6 \rightleftharpoons 2GaCl_3$	2	20.0	10-1,000
$Ga_2Br_6 \rightleftharpoons 2GaBr_3$	2	18.5	150-1,200
$Ga_2I_6 \rightleftharpoons 2GaI_3$	2	11.0	250-1,300
$Ga_2Cl_6 + 4Ga(liquid) \rightleftharpoons 6GaCl$	6	58.8	100-1,000

Table 5-2 Characteristics of different turbines, cycles and working fluids

	Working Fluid			
	H ₂ O Steam Turbine	H ₂ O Steam Turbine	Al ₂ Cl ₆ Gas Turbine	Al ₂ Br ₆ Gas Turbine
Output, MWe	500	300	555	340
Pressure, turbine inlet, ata	240	240	80	80
Temperature, turbine inlet, °C	580	580	600	750
Pressure, turbine exhaust, ata	0.035	0.035	5	5
Mass flow rate, metric tonne/hr	1,495	880	17,900	21,900
Turbine revolutions, rpm	3,000	3,000	3,000	3,000
Number of exhausts	4	3	2	4
Total number of turbine stages	42	39	6	12
Mean diameter of last stage, m	2.550	2.480	1.338	0.915
Height of last stage blade, m	1.050	0.960	0.495	0.250
Internal efficiency				
High pressure cylinder	-	80.0	89.9	90.0
Intermediate pressure cylinder	-	89.5	-	-
Low pressure cylinder	-	82.0	-	-
Number of turbine shafts	1	1	1	1
Turbine length, m	29.1	21.3	9.0	7.6
Weight of turbine, metric tonnes	964	690	55	90
Power to weight ratio, [MWe/Metric tonne]	0.52	0.43	10.09	3.78

The table above shows the power to weight ratios achievable, with promising results for dissociating gases. However, their use with a commercially scalable NPP requires further research in the areas of part performance, tip losses within the turbomachinery, and stability within the nuclear reactor (perhaps as a secondary loop) in order to demonstrate their viability. Validation will also require test facilities, which will require funding.

5.2 Nitrogen as a Working Fluid

Nitrogen as a secondary loop working fluid has been investigated within a gas turbine module coupled to the HTR-10 test reactor using an indirect cycle. Nitrogen as a real gas is advantageous as much of the experience comes from historic projects, such as the closed gas turbine nuclear reactor developed to meet the needs of the U.S. Army in the late 1950s (McDonald 2012), and aero-engine technology. It is also less challenging on the power extraction and conversion process (IAEA 2001).

The primary helium temperature at the reactor outlet is 950 degrees Celsius in HTR-10 reactors. The helium is cooled down in an heat exchanger (IHx) by the secondary loop nitrogen to 486 degrees Celsius and proceeds into the steam generator, where it is further cooled by the water/steam content to approximately 287 degrees Celsius. The primary helium enters the reactor at 300 degree Celsius. Benefits as stated by Sun et al. (2001) taken from IAEA, (2001) is that a residual heat recovery (RHR) system takes the place of a recuperator and precooler, with no intercooler requirements foreseen due to the moderate heat transport properties of nitrogen, a relatively low mass flow rate and pressure in comparison to other inert gases (Wang & Gu 2005). Nitrogen can also be used for cooling of the turbine components.

5.3 Carbon Dioxide as a Working Fluid

CO₂ enables the improvement of cycle efficiency at its optimum critical range due to characteristics such as low heat of vapourisation, low surface tension, and low compressibility when compared to air. If the compressibility achieved is close to the critical point range, the compression work will reduce, thus leading to improved efficiencies. Table 5-3 shows a list of working fluids at their critical temperature and pressure. The critical values for CO₂ are close to typical compression inlet conditions for the turbomachinery utilised in a NPP. Figure 5-1 shows results of calculations in a study by Grochowina (2011) which compares compression work (%) of CO₂ and air for a set of given inlet conditions, TET and pressure ratio. The bar chart shows close to a 100% increase in compression work when utilising CO₂ in the cycle:

Table 5-3 Critical Data of Typical Fluids (Kato et al., 2004; taken from Grochowina, 2011)

No.	Fluid	Critical temperature	Critical pressure
0	-	°C	MPa
1	CO ₂	31	7.38
2	H ₂ O	374	22.06
3	N ₂	147	3.40
4	NH ₃	132.2	11.13
5	Helium	-268	2.275

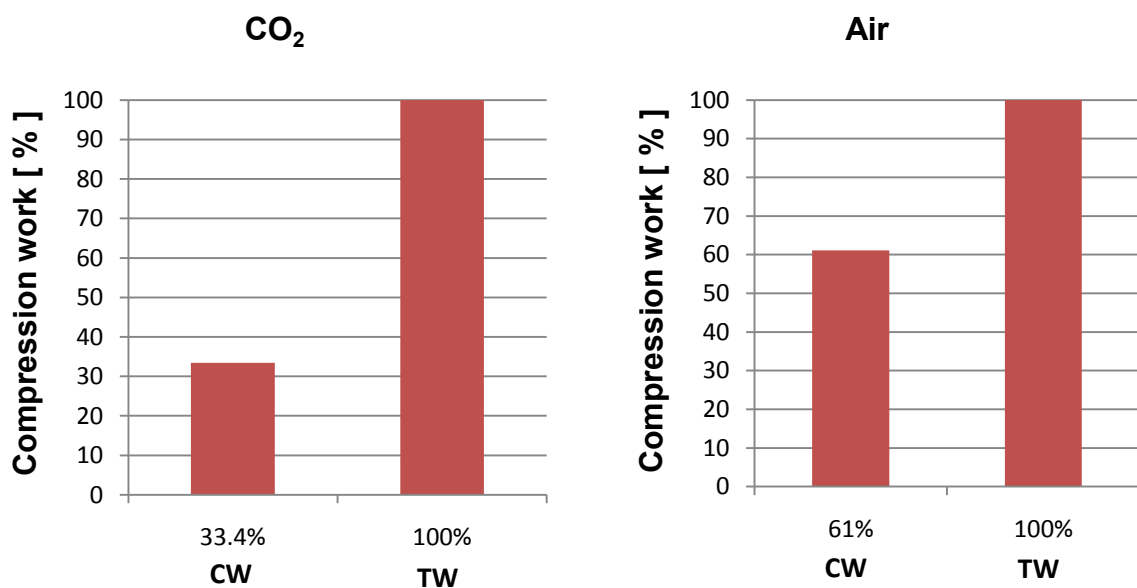


Figure 5-1 Compressor work between CO₂ and air (Grochowina, 2011)

Grochowina (2011) also showed that compressor and turbine work is dependent upon pressure ratio and inlet pressure when CO₂ is used as a working fluid due to the critical point

(supercritical phase), which defines its single-phase existence. Thus, the supercritical phase temperature (32 degrees Celsius for CO₂) and pressure is important and desirable for the compressor at the inlet in order for the cycle to benefit from low compressor work.

The supercritical CO₂ (S-CO₂) cycle, i.e. pressure being above the critical point has undergone several developments since it was first patented as a partial condensed CO₂ cycle by Suzler Broz in 1948 (Dostal et al., 2004). S-CO₂ as a working fluid has several benefits such as relative inertness, stability, non-toxicity, abundance, low cost and a wealth of knowledge on the thermodynamic properties. Feasibility studies of S-CO₂ as a working fluid in closed cycle NPPs have been undertaken by Dostal et al. (2004), Hejzlar et al. (2006), Ishiyama et al. (2008), Kato et al. (2004) and Yari & Sirousazar (2010). A study by Yari (2010) suggests 39% thermal efficiency is achievable with a modest TET of 550 degrees Celsius, although Figure 4.13 McKellar et al. (2010) shows a theoretical achievable thermal efficiency of over 50% for a TET of 800 degrees Celsius.

Other studies have investigated the thermodynamic effects of super-critical working fluids in comparison with transcritical CO₂ (T-CO₂) working fluids. S-CO₂ is limited in its heat exchange effectiveness due to the properties of CO₂ near the critical zone (Cao et al. 2017). This issue of operation near the critical point is also linked to other components such as the compressor. Lee et al. (2016) conducted a compression experiment near the critical point in a purpose built facility, which was set up to solve the issues of compressor design and operation at the near critical point of CO₂. Their observations suggest that there is a high uncertainty in low pressure ratio compressor design performance. This uncertainty is due to the peak region in thermodynamic property variation of CO₂ near the critical point, combined with low pressure operation. Lee et al. (2016) suggest that this could be mitigated by using a high pressure ratio compressor, but the number of stage increases would mean an increase in compressor cost, and may not be optimal for the cycle in terms of maintaining efficiencies. The potential for phase change due to significant decrease in static temperature and static pressure in the local area of flow acceleration means that inlet conditions should be chosen at the lowest temperature (Pham et al. 2016). This adds restrictions to the conditions under which the coolant can operate and is still an area where a lot of research into CO₂ is channelled.

5.4 Helium as a Working Fluid

Helium is regarded as an ideal fluid for closed cycle gas turbines because of its chemical inertness and passive neutronic state (i.e. it does not absorb fissile products during a chain

reaction). In addition it does not cause corrosion in the components within the loop. Helium obeys the ideal gas state equation and has beneficial thermal properties suited to a high temperature reactor, with simplistic performance and mathematical representation. Helium can also be used to cool the reactor (Decher 1994). With regards to the properties of gases which are used as coolants within a reactor, Table 5-4 gives a comparative summary of typical gases, some of which have been reviewed earlier within this report.

Table 5-4 Typical reactor coolant gases and properties (Thring, 1960 taken from Pradeepkumar, 2002)

Gas	Chemical compatibility with reactor and engine materials	Induced radio-activity and dissociation	C_P KJ/Kg. K γ , M	General Remarks
H ₂	Generally good, but may embrittle some materials	None	14.68 1.4 2.016	Very good heat transport properties. Difficult to contain at high pressure
He	Very Good	None	5.2 1.66 4.006	Good heat transport properties, but expensive
CO ₂	Satisfactory	Negligible radioactivity. Some dissociation at high temperatures	1.09 1.21 44.07	Good heat transport properties. May preclude use of graphite as moderator material at high temperatures (unless both moderator and fuel elements are canned)
N ₂	Probably satisfactory	Some induced radioactivity, not of significant level or type	1.09 1.4 28.02	Moderate heat transport properties
Air	Poor, due oxidation of metals at high temperature	Some induced radioactivity in argon and nitrogen	1.006 1.4 28.9	Moderate heat transport properties
Ne	Good	None	1.048 1.64 20.18	Poor heat transport properties

According to Thring (1960), the desirable properties of a selected gas are:

- It should not react adversely with the materials within the reactor, turbomachinery and peripheral units.

- It should have a low induced radioactivity with the capability to dissociate during irradiation and at higher temperatures.
- Its heat transport capabilities should be considerably good, and its thermodynamic properties should enable a simple compressor and turbine design.

The suitability of a gas for a simple gas turbine design is determined largely by the C_p rather than γ thus helium poses a challenge for turbomachinery mechanical design due to its large specific heat in comparison with nitrogen and carbon dioxide (see comparisons in Table 5-4). The number of stages required for the compressor and turbine is approximately proportional to the C_p value, meaning a high number of stages are required for the compressor and turbine to achieve the necessary compression and expansion for the cycle (Pradeepkumar, 2002).

5.4.1 Consequential Thermodynamic Effect of reverting to Helium from Air

Decher (1994) provides comparisons of properties between air and helium at 300K and 1000K for respective pressures of 1atm and 30atm. The values in Table 5-5 show that the kinematic viscosity and thermal conductivity deviate with temperature for both gases.

Table 5-5 – Characteristics of air and helium at various pressures and temperatures (Decher, 1994)

Property	p(atm)	T(K)	Air	Helium
Mol. Wt.			28.9	4
γ	1	300	1.4	1.67
	30	1000	1.36	1.67
C_p	1	300	1.005	5.20 kJ/kg K
	30	1000	1.142	5.20
k	1	300	0.028	0.15 w/m-K
	30	1000	0.068	0.36
ν	1	300	16	$120 \times 10^{-6} \text{ m}^2/\text{sec}$
	30	1000	3.4	28

These effects (see Table 5-6) are further discussed in terms of parameters that influence the turbomachinery design. Jiang et al. (2015) quantify the differences and effects of helium in comparison to air. Their study investigated the cascade flow characteristics of a highly loaded compressor versus an air compressor, and concluded that at low Mach numbers the deviation is small to the point that the design methods and performance maps of air can be used for helium.

At high Mach numbers and increased cascade loading, the differences become apparent, with Reynolds number affecting the aerodynamic performance of a highly loaded cascade (Jiang et al. 2015). All other more obvious differences are summarised below in Table 5-6.

Table 5-6 Cycle characteristics for air and helium using a 1.5 temperature ratio in compression process (Decher, 1994)

	air	helium
pressure ratio (p_2/p_1)	4.4	2.66
volume ratio (v_2/v_1)	0.36	0.55
stage pressure ratio	1.2	1.03
number of compressor stages	8	33
flow velocity	1	2.3
flow area (high pressure side)	1	0.62
flow area (low pressure side)	1	0.37

5.4.2 Volumetric Ratio

Simpler turbomachinery designs constitute smaller volumetric ratios for the compression and expansion geometry. The volumetric ratio is expressed as:

$$\frac{V_2}{V_1} = \left(\frac{T_2}{T_1}\right)^{\frac{1}{\gamma-1}} \dots \dots \dots eq \ 19$$

5.4.3 Pressure Ratio

The pressure ratio is set as a function of the temperature between 2 stations for a set of given conditions and is expressed as:

$$\frac{P_2}{P_1} = \left(\frac{T_2}{T_1}\right)^{\frac{\gamma}{\gamma-1}} \dots \dots \dots eq \ 20$$

with the exponential values for helium (2.5) in comparison to air (3.5) as denoted in equation 20, meaning that a lower pressure ratio is expected for helium, thus a compression process that is easier due to less magnitude in the pressure gradient. Table 5-6 summarises a compression process for both gases for a 1.5 temperature ratio between inlet and outlet.

5.4.4 Pressure Drops in Ducts and Pipework

Fractional pressure drops affect flow velocity and certainly have an impact on overall system design and compactness. For a given pressure loss effect observable (based on comparable flow conditions), Decher (1994) states that the ratio of velocity allowable to helium is expressed as:

$$\frac{u_{He}}{u_{air}} = \sqrt{\frac{\left(\frac{\gamma-1}{\gamma} \text{mol}\right)_{air}}{\left(\frac{\gamma-1}{\gamma} \text{mol}\right)_{He}}} = 2.27 \dots \dots \dots eq 21$$

where mol is the gas molecular weight. Typical velocities of 100m/sec are nominally considered in practice.

5.4.5 Specific Heat

Helium is five times larger than air with regards to its specific heat. Thus, for a given temperature increment, helium carries five times the power per unit mass.

5.4.6 Flow Cross-Sectional Area

The flow cross-sectional area is given by the steady flow continuity equation:

$$A = \frac{\dot{m}}{\rho u} \dots \dots \dots eq 22$$

whereby the density is given by the state equation and the mass flow rate by an enthalpy flux:

$$\dot{m} \sim \frac{\text{power}}{C_p T} \dots \dots \dots eq 23$$

and the velocity is given by the equating pressure loss effect relationship in eq 21, and the pressures on the high pressure side are treated as equal. For the two cycles, one gets the flow area ratio (A(He)/A(air)) for the high and low pressure sides (Decher, 1994). The final conclusion when comparing flow cross-sectional areas is that the use of helium in a closed cycle leads to compact flow devices when compared with air, if the flow area is used as a determining criterion (Decher, 1994).

5.4.7 Effect of Pressure Level of Helium Closed Cycle Gas Turbines

The efficiency of a closed cycle using helium is more dependent on the pressure ratio and relative temperature ratio, and less dependent on the pressure. The pressure comes into play when the flow through the duct of a given diameter is considered. This is dependent on the

density and thus the static pressure for a given static temperature. The overall power output of the engine will be affected by the choice of pressure including the physical characteristics of the pertinent components. Decher (1994) provides methods for consideration of the physical dimensions of the components, the through the flow passages and physical dimensions of heat exchangers.

6 Operational Experience of Helium Closed Cycle Gas Turbines

Since gas turbines were developed they have been used in the nuclear industry, but they have never reached the large scale required for high temperature gas cooled reactors in comparison with the large scale commercialisation that has occurred for nuclear power plants operating steam cycles. This lack of large scale commercialisation is due to the high TET demanded by the Brayton cycle, which was limited by material choices. A unique opportunity exists to reduce the cycle inventory and improve part load performance (Pradeepkumar 2002).

As valuable experience has been gained in the design, fabrication, operation and maintenance of plants with air as the working fluid, the future prime mover using helium as a working fluid, whilst coupled to a high temperature reactor was seen as offering opportunities of high thermal efficiencies. However, operational experience was lacking, leading to the conversion of operational facilities to include helium turbomachineries for test purposes (McDonald 2012). Closed cycle gas turbines posed huge challenges in the 60s and 70s due to the associated complexities. Pradeepkumar (2002) summarised these complexities as problems that manifested during the operational phases:

1. Substantial leakage of helium through seals and gaps due to its small molecular size compared to other gases, which seal development was based on. Tip leakage in the turbomachinery was also experienced, resulting in heavy losses. Cooling of the turbine was also expensive due to the higher flow rates when performed in the conventional way.
2. The welding effect of high temperature helium, which stopped the functioning of guide vanes, mechanical bearings and control valves. Material transfer was evident when helium passed over 2 moving metallic surfaces, which was harmful for these sensitive components.

3. Rotor vibrations were regular due to the slim single rotors that were used with a gear facility at the generator side.

Electromagnetic bearing designs, adjustable sealing systems and multiple rotors were proposed and are considered as part of today's designs due to development work with credible testing and substantiation to ensure that the solutions resolved the issues. The key running experiences of helium plants are described in below.

6.1 La Fleur Helium Turbine (1962)

Earlier developments include the La Fleur helium turbine air separation process plant, which consisted of a compressor, turbine, heater, and recuperator and precooler, which has a bleed offtake after the compressor. A secondary helium loop was used, which was cooled to approximately the condensing temperature of air and expanded in a cold turbine and recirculated back into the compressor inlet to reduce the compressor work (Frutschi 2005). Helium was used as the working medium because it remained gaseous at the relevant low temperature range. This was required to liquefy air. The hot gas temperature was 650 degrees Celsius with natural gas as the fuel and helium flow of 4.8kg/s through the compressor at a base pressure of 1.27 MPa, but the turbine entry pressure was 1.8 MPa.

6.2 Oberhausen II EVO Helium Turbine (1974)

This project was to develop a 50MW helium turbine, with a hot gas temperature of 750 degrees Celsius fuelled by coke oven gas. The base pressure was 1 MPa and the pressure in front of the turbine was 2.7 MPa. The development suffered from changes in technical leadership, which lead to the project being outsourced to a technical University. The original design intent by Escher Wyss (previously responsible for the turbomachinery) was a 50MW terminal output at the turbine, which was never realised. The turbomachinery had a 30MW output as it was designed for a lower rotational speed, resulting in an unfavourable hub to tip ratio for the compressor and turbine and leading to poor component efficiencies. Pressure losses were excessive by a factor of 4, especially the cooling and sealing flows, which resulted in efficiencies of 23% (the design intent was 34.5%) (Frutschi 2005).

7 Control and Operational Performance

According to Frutschi (2005), where there is constant TET and fixed blade geometry there are two control methods for output and/or rotational speed. These are inventory or pressure level control, and bypass control. Power plants using these methods offer the advantage of power

production with a time variability to match that of the load (Decher 1994). It is imperative to produce part power at high efficiencies and in a way that minimises thermal shock leading to stresses of the heat source, especially in a nuclear reactor.

7.1 Inventory or Pressure (IP) Level Control

With regards to IP control, the plant is allowed to run at part load with almost unchanged efficiency and specific work. The cycle is connected to a gas accumulator and is used to store some of the working fluid during part load operations. The fluid is released into the gas generator during full load, thus ensuring that pressures within the cycle and the shaft power remain relative to the gas density in the cycle (Frutschi 2005). The unchanged efficiency and specific work is possible as the cycle operates at the same temperature and pressure ratio, regardless of the reduced mass flow rate. When part load operation is demanded, the fluid is ducted from the lowest temperature point in the cycle with appropriate means of cooling in order to minimise heat transfer to the accumulator (Decher, 1994). The unchanged temperature regardless of reduction in mass flow rate indicates that the sound speeds are constant, with the Mach number fixed by the geometry expressed as:

$$M_{ip} = \frac{\sqrt{\left(\frac{p_{static}}{P_{cycle}}\right)^{\frac{\gamma-1}{\gamma}} - 1}}{\left(\frac{\gamma-1}{2}\right)} \dots \dots \dots eq \ 24$$

in addition to the local flow velocities remaining constant to first order. The massflow rate is proportional to the density with velocities constant, which for constant temperature is also proportional to total pressure. Thus this relationship of mass flow, density, velocity and geometry is denoted by:

$$W_{ip} = \rho Au \dots \dots \dots eq \ 25$$

with the gas densities at part load remaining relative to the shaft power demanded due to the static conditions. It is expressed as:

$$\rho_{ip} = \frac{p}{Rt} \dots \dots \dots eq \ 26$$

Figure 7-1 shows the thermodynamic cycle operating at various pressure levels. Variations in cycle performance are based on the working fluid properties, and are not influenced by static pressure when the working fluid is monatomic (single atom). Helium satisfies this condition because it is not influenced by static pressure. For other gases, the specific work and thermal efficiency of the cycle will be affected by changes in static pressure. With reference to Figure 7-1, the relationship between efficiency and a low power setting is relatively flat and part load power can be achieved by operating at low static pressure. However, frictional losses due to a decrease in density are affected, in addition to the Reynolds number. Decher (1994) states that the effect on the fluid's frictional losses increases the importance of viscous losses. The dynamic viscosity (μ) and the density affect the flow conditions, as denoted by the Reynolds number (the ratio of inertia forces and viscous forces), expressed as:

$$Re_{ip} = \frac{\rho u D}{\mu} \dots \dots \dots eq \ 27$$

The effect is to reduce or limit efficiency by a small fraction due to a reduction in the power output because component efficiencies are reduced.

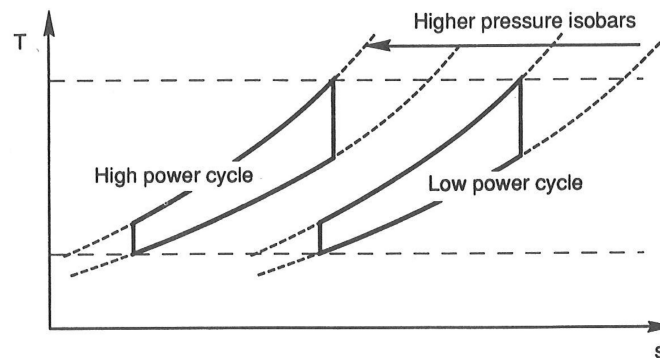


Figure 7-1 T-s Diagram for full and part power for inventory controlled closed Brayton cycle (Decher, 1994)

7.2 Bypass Control

Bypass (BP) control or compressor back flow permits rapid unloading and reloading of the shaft power (Frutschi 2005). Flow capacity is dictated by a valve that is added prior to the turbine. The purpose is to bleed off high pressure flow back into the compressor, thereby reducing mass

flow to the turbine and altering the suction characteristics. The directed flow is circulated into the compressor with a gas inventory, which is unchanged. This increases the pressure level, thus increasing compressor work resulting in further reductions in turbine power and a subsequent shaft power reduction. This throttling is a source of irreversibility and leads to part power efficiency. Cycle temperatures can be held unchanged. The thermal input power matches the deficit to maintain the cycle temperature in the reactor at the reduced mass flow, thus minimising associated stresses due to temperature excursions. Consequentially, performance impact can be calculated as the cycle can be assumed to be fixed. The process of calculating references the schematic in Figure 7-2.

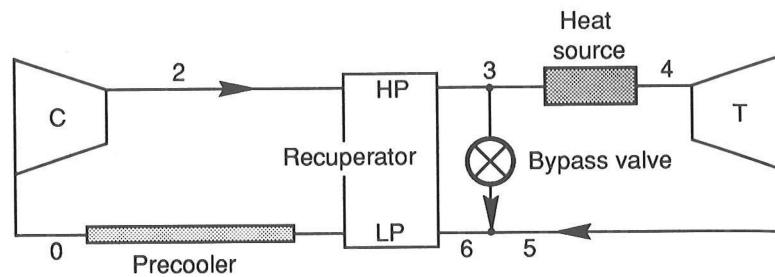


Figure 7-2 Bypass power level control of Brayton cycle (Decher, 1994)

With reference to Figure 7-2, the cycle can be considered as process equal masses on either side, thus the ideal scenario of $T_3 = T_6$ is always retained at part power. This holds true because with a constant T_4 at unchanged compressor pressure ratio, $T_3 = T_5$. A mixer is incorporated to combine the cold flow with the hot gas, bringing about an enthalpy balance, thus yielding $T_3 = T_6$. The effect of this is considered trivial for an ideal and perfect gas such as helium. Using the specific work for both the normal and bypass configuration, the equation is used to determine the efficiency of the cycle:

$$\eta_{th} = \left\{ 1 + \left[\frac{SW}{\left(\frac{T_4/T_0}{T_2/T_0} - 1 \right) (T_2/T_0 - 1)} \right] \left[\frac{T_4/T_0}{T_2/T_0} - 1 \right] \right\}^{-1} \dots \dots \dots eq \ 28$$

whereby

$$SW_{max} = \left(\frac{T_4/T_0}{T_2/T_0} - 1 \right) (T_2/T_0 - 1) \dots \dots \dots eq \ 29$$

$$TR_{total \ to \ static} = T_4/T_0 \quad \text{and} \quad T_{stagnation \ compr} = T_2/T_0$$

Figure 7-3 shows the plot of η_{th} versus the power fraction for $T_4/T_0=4$ and $T_2/T_0=1.5$ and gives efficiency results that are identical for actual bypass control. The processes of bypass and peak temperature reduction have the same thermodynamic merit. However, part power performance must be evaluated, giving significant consideration to the irreversibilities in the cycle, rather than being considered on a theoretical basis (Decher 1994). Figure 7-4 compares the bypass and inventory pressure control methods using thermal efficiency and thermal power as a function of load:

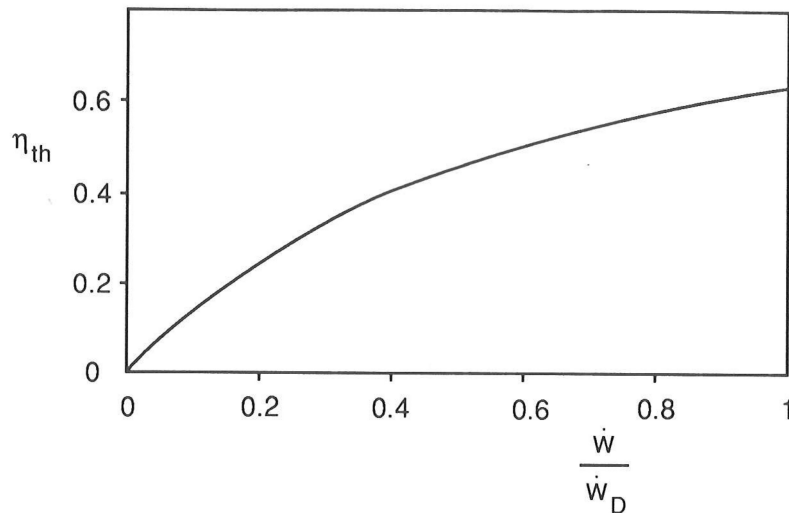


Figure 7-3 Thermal efficiency vs power fraction for a regenerated cycle with ideal components (Decher 1994)

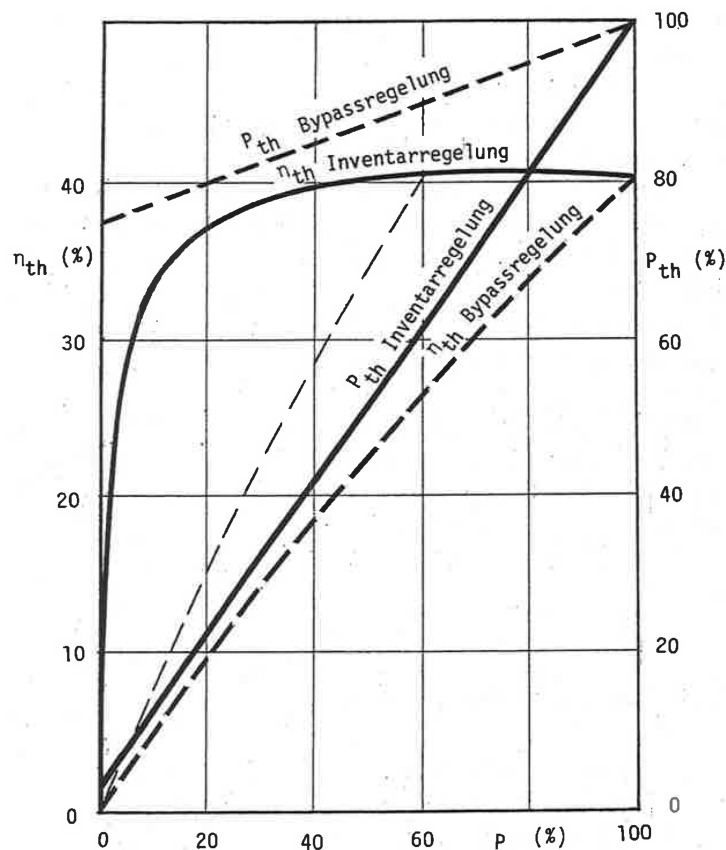


Figure 7-4 Part load behaviour with bypass and inventory pressure control (Frutschi, 2005)

This review of the literature on part power performance for nuclear reactors has indicated that current NPPs conduct part load operations by movement of the control rods to alternate the reactor power (Locatelli et al. 2015). This approach involves inserting the control rods deeply to control the power, which induces distortion of the neutron flux in the core, leading to hot spots. This control method would be limited to prevent damage to the internal core components. Full reactor power without interfering with the control rods would be favourable from an economic viewpoint since nuclear power productions are capital intensive, but there is the potential for low operational costs if the design, operation and control processes are optimised.

Finalised designs for the control system capable of part power operation of Generation IV reactors are yet to be realised for the various concepts. Current literature is based on proposed

strategies which are yet to derive sound power control solutions that can be validated. One such case is the study conducted by Sun & Zhang (2017) for the SCWR concept. The study proposes three approaches for load following based on reactor, feedwater and turbine leading approaches. However, results indicated undesired outputs and fluctuations in power because it is difficult to guarantee to control stability and performance.

Another method that is being proposed is to operate plants as co-generating plants. Locatelli et al. (2015, 2017) [2] suggest that part power operation should involve diverting the power output for co-generation tasks as demanded in a processing or desalination plant. This is something that can be reviewed as a strategy for plants that have already been built. In the case of this research work, this idea can only be considered as future work due to the complexity and size of typical electrical power plants with co-generation. This is because the plant has to be designed for the best economics and minimum complexity as a starting point.

For concepts such as the GFR and VHTR that utilise helium, extensive research was undertaken in the 1970s and 1980s in Germany, France and the USA. This involved the inventory and bypass control methods explained earlier in this section. The majority of the work in the 1970s revolved around the Obserhausen plant in Germany (Bammert et al. 1974b; Bammert & Krey 1971a; Bammert 1971; Bammert & Deuster 1974; Bammert et al. 1974a; Bammert & Krey 1971b; Bammert & Rehbach 1971). The project looked at the viability of helium as a working fluid and employed the inventory control method on the plant. Bitsch & Chaboseau (1970) had made initial advancements by conducting a parametric study which looked at the number volume versus the load variation achievable. The General Atomic Company had conducted studies on the close loop and control of the HTGR-GT in the 1980s, which went further to provide results on transient operations (Chan et al. 1980; Openshaw & Chan 1980).

However, the investigation by the General Atomic Company had begun in the late 1970s (Openshaw et al. 1976), after the advancements made by the Europeans. Massachusetts Institute for Technology had pioneered research into close loop control of helium MGR-GT in the late 1980s and early 1990s (Staudt 1987; Yan 1990), whereby improvements were made to

the control systems which utilised inventory control. In all cases, the strategy for control of the plant is key to designing the control system. Sato et al. (2012) defined the control strategy for the GTHTR300C as follows:

- Maintain constant reactor thermal power
- Reduce thermal stress generation of reactor internal components
- Reduce thermal stress generation of the turbine blades
- Keep the high efficient power generation

Sato et al. (2012) provide an automated control system based on a systems simulation code for the GTHTR300C cogeneration plant, which incorporated inventory control, turbine inlet temperature control, recuperator heat rate control, turbine speed control, reactor outlet temperature control and recuperator differential pressure control. The results were encouraging in terms of demonstrating efficacy of the control methods. Sato et al. (2012) conclude that part load performance with constant reactor power and thermal efficiency is feasible, although no data was presented to show the effect on thermal efficiency and specific work. One key observation is that control without affecting the output of the reactor only seems possible with concepts and cycles that use helium as the working fluid.

Part power operation requires similar strategies to those developed by Sato et al. (2012), but more importantly, the Generation IV concepts of interest will need to be assessed against the TERA requirements for part load performance. This is an area that the thesis attempts to address. Finally, as the demand for energy on the grid is prioritised, there may be a significant reduction in the availability of NPPs. If this were the case, it would not be economically viable to run inventory pressure control for long periods such as weeks or months because the fuel consumption costs are not justified when there is a drop in plant capacity for such a long time. This presents a big gap in current operation because there is no evidence that suggests that NPPs operate to specific off design points, which give the best economics. An attempt is made by the author to resolve this gap in knowledge in the PhD thesis.

8 Conclusion

This report documents a review of the current status with regards to nuclear power plants particularly the incumbent and researched technologies and aspects that influence them. A summary of the literature review is provided as follows:

With the world's population set to increase to 10 million by 2050, the demand for energy is expected to increase accordingly. To meet the deficit in energy supply, nuclear energy as a source of clean energy will have to be adopted. There are currently 450 nuclear power plants in operation in 31 countries, with a net installed capacity of 392 GWe, and there are 60 plants under construction. Such is the scale of investment in nuclear energy, especially in the research and development of Generation IV concepts, that it drives the need for a framework to assess the technical, economic and risks to enable optimal designs to be conceived with sound economics. The framework will therefore add value to investment decisions.

Generation IV concepts present a revolutionary step change from current solutions. A key feature of these concepts is high efficiency cycles. Although not explicitly acknowledged, cycles are the blueprint for a plant's design and configuration. A review of cycles clearly demonstrated that the closed Brayton cycle should be the choice for this study due to its high thermal efficiency capabilities. This means that the Generation IV concepts of interest in the project are the GFR and VHTR. Furthermore, the Brayton cycle will allow for simple plant configuration if the working fluid can be used as a coolant to make it a direct Brayton cycle, which produced the highest efficiency.

A review of currently adopted nuclear cycles also indicated that there was a gap which had not been explored for nuclear. The intercooled cycle had experienced technological advancements when used with air as the working fluid. The investigation of working fluids revealed that disquieting gases still require further research to assess their merit in a gas turbine setting. Nitrogen is proposed as a secondary circuit because of its moderate heat transport properties compared to CO₂ and helium. Helium was chosen as the coolant due to its superiority in terms of heat transportation and because CO₂ presented uncertainties in phase changes due to the significant decrease in static temperature and pressure.

With regards to operational historic experience of helium closed cycle gas turbines, several projects have provided a lot of operating experience in design to prevent leakage of helium and minimising root vibrations. These resulted in the HTR-PM and GT-HTR300 projects, which have showed promising signs. However, beyond design point high efficiency operation, high efficient off-design operation is also necessary. Currently, operators move the control rods to maintain power, which disturbs the neutron flux in the core. One of the methods that was discussed was inventory pressure control, which could ensure the plant maintains close to design point efficient levels with helium. This would be favourable for economics but not sustainable over long periods. A gap in knowledge exists in the off-design point operation, whereby the specific optimal points are derived to enable the plant operate at off-design settings for efficiency purposes.

Future research will aim to close this gap; more so, it will aim to provide tools that would support the decision-making process by comprehensively demonstrating an understanding of the technical issues, which are linked to the economics and risks. It is acknowledged that due to the timeline for development of Gen IV applications, such tools will be conceptual and will be far from verification and validation. However, the advancement in knowledge in the absence of public data will depend on these tools and intellect as a platform for future development.

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