NATIONAL TECHNICAL UNIVERSITY OF ATHENS

SCHOOL OF NAVAL ARCHITECTURE AND MARINE ENGINEERING

MODERN CONCEPTS FOR MARINE NUCLEAR PROPULSION

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Η συγκεκριμένη εργασία αποτελεί το επιστέγασμα μιας πορείας χρόνων, όχι μόνο αυτών του πανεπιστημίου αλλά από την πρώτη μέρα του σχολείου. Μέσα σε αυτό το ταξίδι που ολοκληρώνεται κάπου εδώ, διδάχθηκα τόσο σε εκπαιδευτικό επίπεδο, βίωσα σε εμπειρικό επίπεδο, αλλά πολύ περισσότερο αφομοίωσα σε προσωπικό επίπεδο.

Μέσα σε αυτές τις γραμμές λοιπόν, από τη μία θέλω να ευχαριστήσω όλους τους καθηγητές που βρέθηκαν στο δρόμο μου ώστε να μου διδάξουν. Ένα ιδιαίτερο ευχαριστώ στον κ. Δημόπουλο που δίχως αυτόν, την διάθεση του και τη συνεργατικότητα του η συγκεκριμένη εργασία δεν θα είχε έρθει εις πέρας.

Από την άλλη, νιώθω την ανάγκη να ευχαριστήσω όλους εκείνους που πλέον αποτελούν ανεξίτηλες εμπειρίες και που πάντα θα με συντροφεύουν με νοσταλγία και ένα απέραντο πλατύ χαμόγελο. Αυτοί δεν είναι άλλοι από τους φίλους μου και τους συμφοιτητές μου.

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Μαμά, Μπαμπά και Αντιγόνη σας ευχαριστώ για όλα.

ΠΕΡΙΛΗΨΗ

Η παρούσα εργασία εξετάζει τις πιθανότητες εφαρμογής της πυρηνικής ενέργειας στον Ναυτιλιακό Τομέα. Αρχικά, παρουσιάζεται μια σύντομη σύνοψη με τους πιο σημαντικούς όρους και φυσικούς νόμους που σχετίζονται με την πυρηνική ενέργεια. Παρόλα αυτά η εργασία αυτή επικεντρώνεται κυρίως στην καθιέρωση της πυρηνικής ενέργειας μόνο στο θαλάσσιο περιβάλλον και μιας που δεν είναι η πρώτη φορά που επιχειρείται κάτι παρόμοιο, πραγματοποιήθηκε μια ιστορική αναδρομή. Έτσι παρουσιάστηκαν μερικά από τα σημαντικότερα project που σχετίστηκαν με την πυρηνική ενέργεια στη θάλασσα και μέσα σε αυτά συμπεριλήφθηκαν πυρηνικά υποβρύχια, πυρηνικά παγοθραυστικά αλλά και πυρηνοκίνητα πλοία εμπορικής χρήσης. Ύστερα, αναλύθηκε η εξέλιξη των πυρηνικών αντιδραστήρων όπου και κατηγοριοποιήθηκαν σε ομάδες με βάση τη γενιά τους. Ιδιαίτερη έμφαση δόθηκε στους πυρηνικούς αντιδραστήρες τέταρτης -και τελευταίας- γενιάς, όπου 6 μοντέλα επιλέχθηκαν ως τα πιο ελπιδοφόρα. Κατόπιν, πραγματοποιήθηκε περιγραφή των κυριότερων τεχνικών χαρακτηριστικών και της απόδοσης όλων των πυρηνικών αντιδραστήρων που ήταν υποψήφιοι για χρήση στη θάλασσα. Έτσι, ακολούθησε μία πειραματική έρευνα, όπου επιλέχθηκε ένας από όλους τους αντιδραστήρες ως ο καταλληλότερος για χρήση εν πλω. Για λόγους σύγκρισης, επιλέχθηκε ένα πλοίο μεταφοράς χύδην φορτίου (Bulk Carrier) 400k DWT εξοπλισμένο με ένα συμβατικό ντιζελοκινητήρα. Κατά τη συγκριτική διαδικασία, επιλέχθηκαν διάφορα κριτήρια εξέτασης όπως περιβαλλοντολογικά, οικονομικά, διαστασιολόγησης κλπ όπου το πυρηνοκίνητο Bulk Carrier φάνηκε να υπερτερεί έναντι του συμβατικού.

ABSTRACT

This thesis examines the potential of the implementation of nuclear energy into the Maritimes Industry. To start with, a short summarization of some of the most important definitions, terms and physic laws related to nuclear energy is presented. Nonetheless, this research is mainly focused on the marine environment and since this is not the first time that nuclear power is attempted to be established as a marine concept, a brief overview of the historical background so far is presented. Some of the most significant nuclear naval projects such as nuclear submarines, nuclear icebreakers and nuclear vessels for commercial purpose are included in the portfolio. Afterwards, an inclusive synopsis of the status of the nuclear reactors is followed, along with a categorization depending on the generation of each reactor. Extra emphasis was given on the fourth (IV) -and latest- generation of reactors released, where six designs were promoted as the most promising. Then, the technical specifications and the performance of every nuclear reactor favourable for marine use were analysed. Lastly, a case study was conducted where one of the designs – the Molten Salt Reactor (MSR)- was promoted as the most suitable for a commercial marine application. For comparison reasons in the case study, a Bulk Carrier of 400k DWT with a conventional diesel engine was chosen. For the comparison many criteria were established such as environmental, financial, sizing etc. where the Bulk Carrier equipped with the MSR prevailed.

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INTRODUCTION

Human activity over the past 50 years faces a significant increase across the planet. As a logical consequence, energy demands follow the same exponential trend which entrains carbon dioxide (CO_2) emissions that are related to energy production. The growth in CO_2 emissions which is one of the main contributors to the climate change has become a critical issue internationally and for this reason, is monitored daily. For instance, Ritchie's et al study which is depicted below can easily lead to multiple noticeable and alarming conclusions. However, what can be excluded without any further analysis is that the transportation industry is the second-most aggravating polluter.

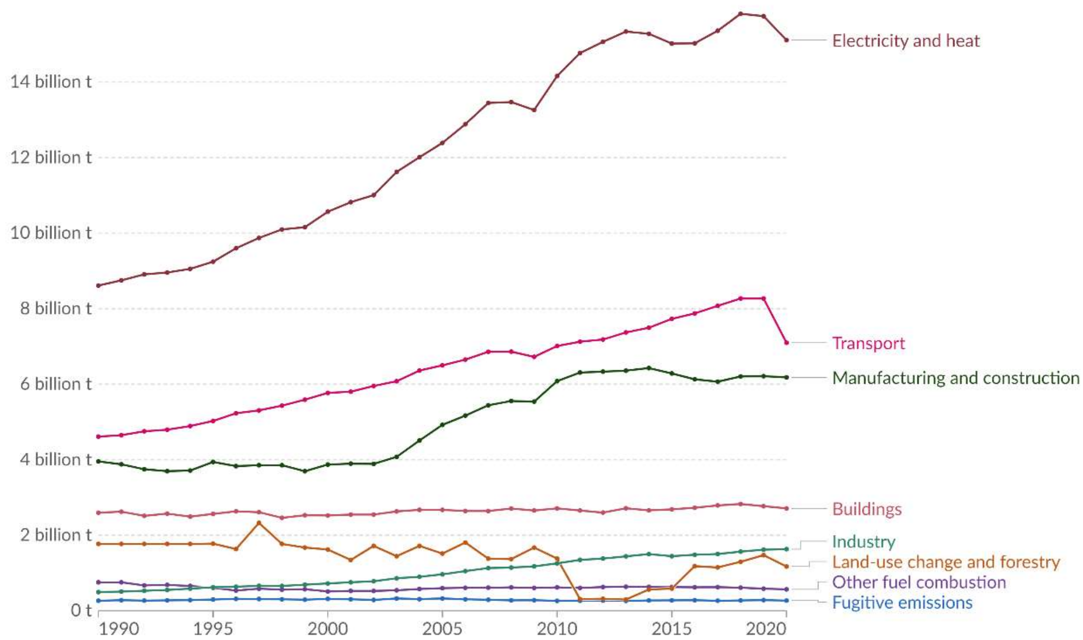

Figure 1: Trend of CO_2 emissions by sector worldwide, Data: Climate Watch (2023)

From a forecasting perspective, there are certain indications that unless meaningful actions are taken, the emissions will be double by 2030 (Argyros , Raucci, & Smith , 2014). Just for quantification purposes also, if international shipping was a country, it would be in the sixth place of emitters right after Germany. On these grounds in 2018, IMO in collaboration with the UN set a strategy for the decarbonization of the global fleet under which:

- CO_2 emissions should be 40% reduced by 2030 and 70% by 2050,
- GHG emissions should be 50% reduced by 2050

All the ratios are concerning 2008 figures.

A more recent study conducted during the COVID-19 period showed that the biggest CO_2 reductions were achieved in the transportation sector (aviation and shipping combined). This notion is also verified by the slopes in the chart above. Hence, this phenomenon certainly ensures the scientific community that transportation should be an industry to focus on. Towards this direction, the challenge was shifted upon the development of alternative fuels since new propulsion technologies lead to higher efficiencies but yet are incapable of achieving the so-called "net-zero" framework.

Among the exploring choices is nuclear energy. Even though nuclear power has been commercially viable since the 1950s it was never truly welcomed in the shipping industry. On the opposite according to Hagen until 2020, 32 countries have operated land-based nuclear power plants while 50 countries have exploited nuclear energy. Thus, there is an existing readiness level that could give an advantage over other appealing technologies. Nowadays and in pure numbers, nuclear power supplies 2% of the primary energy and covers 7% of the generated electricity; quite low figures given its capabilities.

All in all, the urgency for mitigating the human footprint along with the important progress in nuclear reactors has fostered curiosity around nuclear power. It is worth noting that many research centers such as the Joint Research Center (JRC) demonstrate that nuclear could be a sustainable form of energy and what is more, it can play a vital role in energy transition in the shipping industry (Emmen, 2022).

1. NUCLEAR ENERGY

1.1 Definition

Energy is not created or formed instantaneously, nor can it be lost. Energy exists and contrary to the energy production theory, energy is transformed, transferred, released, or even stored (Murray & Holbert, 2015). According to the International Atomic Energy Agency (IAEA), nuclear energy is defined as a form of energy released from the nucleus which is the core of the atom and consists of protons and neutrons.

Even though it is a non-visible form of energy, nuclear energy exists within all the materials of this world. This is the reason that the three principal particles of the atom: protons, neutrons and electrons form bonds and create the elements. In order for these particles to coexist in harmony inside the core of the atom, certain forces restrain them. The key to nuclear energy lies in these forces. Only a small portion of external energy under certain conditions is required to disrupt the balances between the restraining forces, resulting in neutrons and energy release in the form of heat and radiation.

These vast amounts of energy release are transfigured to numerous applications mainly in the industrial field. The primary purpose beyond nuclear exploitation is electricity generation. What makes nuclear energy so attractive is that is a low-carbon source of energy, capable of leading all industries to a CO_2-neutral global economy only with a small quantity of fuel. However, nuclear energy does not resonate only with electricity production since it is a trustworthy solution for seawater desalination, hydrogen production and heating. Additionally, in quite recent years, nuclear techniques have made a significant contribution to the development of many other non-electricity applications such as medicine and agriculture.

Nuclear power is one of the purest and most efficient forms of energy. However certain aspects that accompany this method of power production such as waste management, operational risks, fuel limitations, and the most important one which is the possible malicious uses of nuclear fuels have condemned it to a misconceived form.

1.2 Fission & Fusion

Nowadays, two are the known mechanisms for power generation through nuclear energy: fission and fusion. The substantial distinction is attributed to the method of breaking the balances inside the core. On fission, a larger atom is split into two or more smaller particles, while on fusion two or more lighter atoms join into a heavier one as depicted in Figure 1 below.

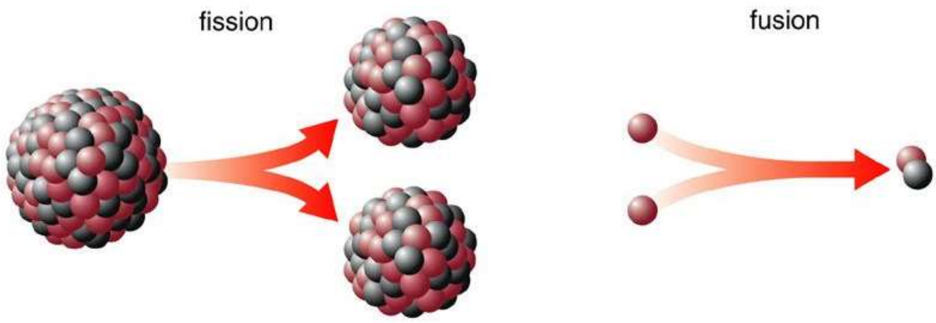

Figure 2: Fission vs Fusion (Duke Energy , 2013)

The process of fission differs fundamentally from chemical reactions like fossil fuel burning since it does not affect the electron cloud that surrounds the atom but directly the nucleus of the atom. Scientists are way more familiar with Uranium (U-235) and Plutonium (Pu-239) as fuels rather than any other element. The procedure initiates with bombarding the nucleus of these elements from another neutron and continues with the splitting of the unstable nucleus into two smaller nuclei known as fission fragments. These fragments are accompanied by two or more nuclei and an immense amount of energy.

However, fission's characteristic that convinced the scientific world of its potential implementation in the industry is that the reaction is self-sustained. In other words, the released neutrons described above due to their kinetic energy and the fertile environment are capable of preserving the reaction since they will trigger another nearby nucleus until they are depleted. This procedure is well known as a "chain reaction" and its mechanism is depicted in Figure 2.

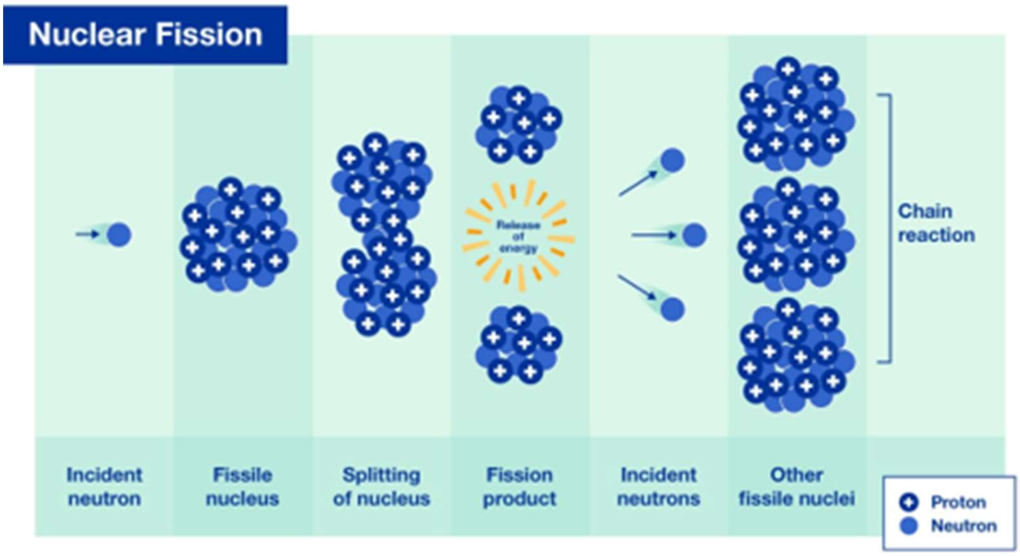

Figure 3: Nuclear Fission Mechanism as explained by IAEA (Galindo, 2022)

Apart from the engineering perspective, fission's development can be attributed mainly to the energy crisis in 1970-1980. Back then, excessive carbon dioxide (CO_2) emissions which were the main reasons for atmosphere pollution, emerged global community and led scientists to the seeking of new resources (Petrescu, et al., 2016). Due to the readiness level from the war period, nuclear fission managed to take over the serious energy deficit in a short time and thus to be established. However, several other concerns -mainly safety- and some accidents have limited its public acceptance. Undeniably, fission comes with some serious disadvantages and yet remains the only feasible application of nuclear energy.

On the other hand, fusion seems to be the challenge of the future. Contrary to fission, in fusion two light atomic nuclei slam together to form a heavier one, releasing great amounts of energy. According to IAEA, fusion's environment takes place in "plasma" which is a hot, charged gas made of positive ions and free-moving electrons. Plasma's properties are not similar to solids, liquids or gases. What is worth noting is that fusion products will be zero radioactive and moreover, there are indications for four times greater energy output than fission (Petrescu, et al., 2016). Furthermore, the reaction will be much easier to control and the fuel cost significantly cheaper (De Ninno, Fratolillo, Rizzo, Del Giudice, & Preparata, 2002). Nonetheless, there is no proof that it can be a self-sustained reaction for long periods due to the high temperatures and pressures (Zadfathollah, Paydar, Balgehshiri, & Zohuri, 2023).

Until now, mankind has not accomplished to implement fusion. The idea of a fusion reaction derives from the natural processes of the stars. For example, fusion in our sun is occurring when hydrogen nuclei fuse to form helium. Most of the concepts that scientists have come up with in order to replicate this reaction involve a mixture of deuterium and tritium which are hydrogen atoms with additional neutrons (increased chance for collision). A detailed analysis from Petrescu et al proves that the materials needed for fusion will be Deuterium and Lithium both found in abundance in nature. It is obvious that the sun's extraordinary conditions offer a fertile ground for fusion to thrive which has not been implemented on an industrial scale yet, but experimental set-ups make the future look more optimistic.

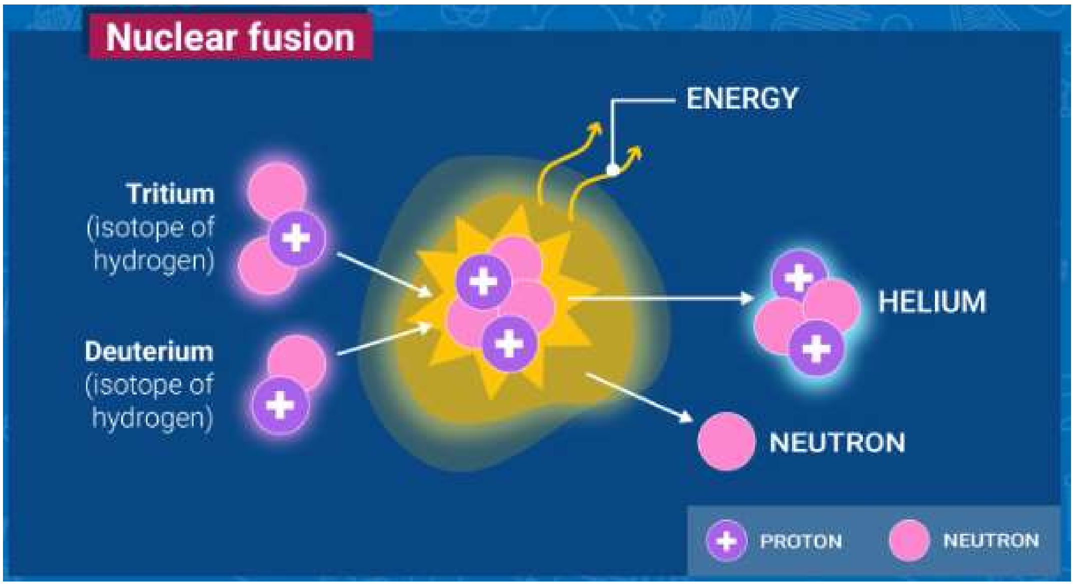

Figure 4: Nuclear Fusion Mechanism as explained from IAEA (Barbarino, 2023)

Fusion's greatest obstacle however which is no other than the harsh conditions required, can turn into the spearhead of this new form of energy. The reason for such a hypothesis is that without the specific extreme conditions (in case of an accident or damage) the plasma will naturally terminate the reaction since it will lose all of its energy instantly. Obviously, nuclear fusion has enormous advantages and once scientific society overcomes the material limitations and simulates the conditions, it can turn to the most powerful energy of mankind. A novel concept of a nuclear fusion reactor is depicted in the next picture, nonetheless, the case of fusion will not be explained any further due to the lack of knowledge on this part.

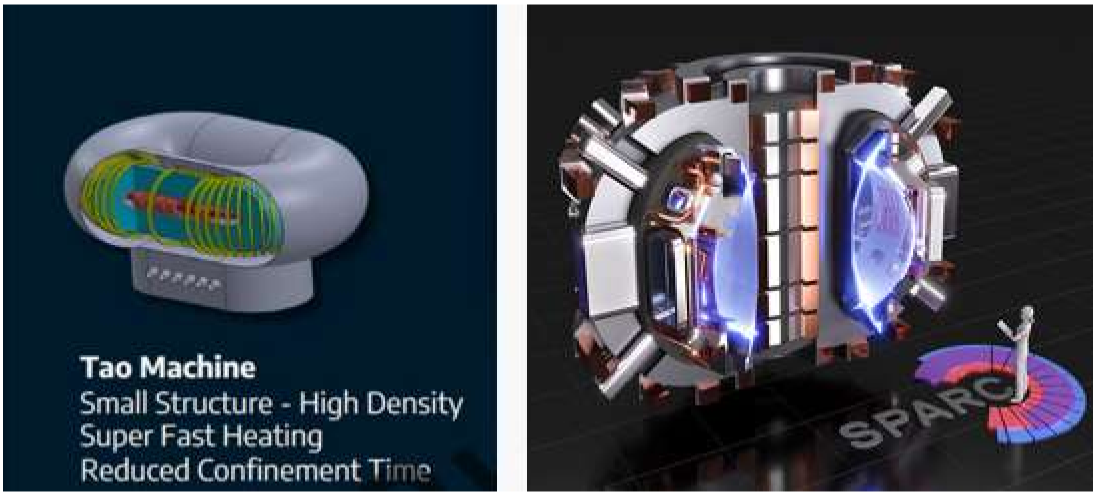

Figure 5: Super Fast Fusion Reactor by MIT (Petrakakos, 2023).

1.3 From Theory to Commerciality

Nuclear power actually is a quite "brand-new" form of energy. Its roots date back to the mid of the 20th century when Enrico Fermi in 1934 came up with the first indications for nuclear fission. Even though the results were not the desirable ones that was only the beginning. Some years later in 1938, Otto Hahn and Fritz Strassmann were surprised to confirm Fermi's thoughts (Energy). Second War II triggered many governments to initialize studies and programs upon fission. Nevertheless, the transition from theory to the commerciality of nuclear power was not a bed of roses and this can be justified as failures and accidents make better news for the public's attention.

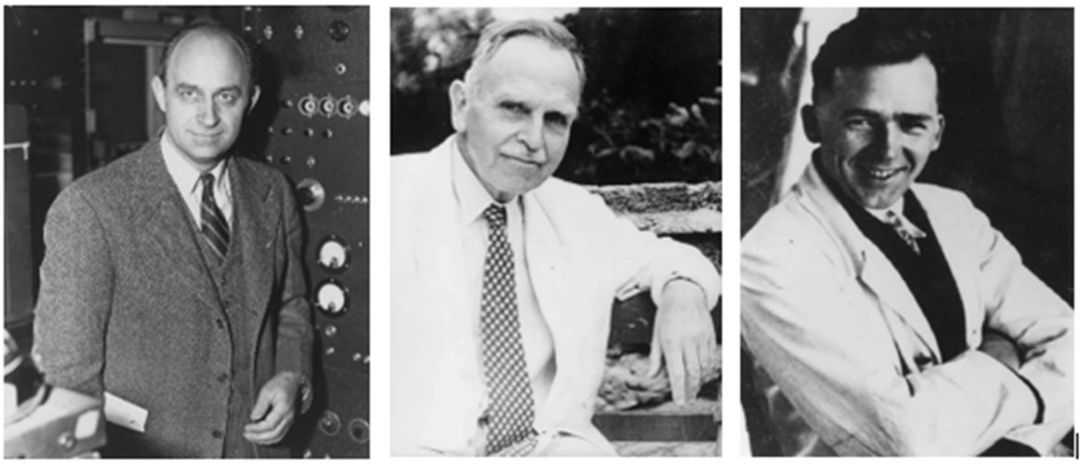

Figure 6: Enrico Fermi, Otto Hahn and Fritz Strassmann

The first-ever nuclear reactor was born in 1942 by Fermi's team in the project "Chicago Pile-I" where under Fermi's guidance a self-sustained reactor was finally a reality. The nuclear era has just started. Unfortunately, after this discovery and in parallel with the war climate, most of the research turned to nuclear weapons. However, some scientists chose to distance themselves from nuclear weapons development and the first non-military nuclear 100kW fast breeder power plant for electricity production was constructed in 1952 in Idaho. That was the turning point for the US government since, from 1953 and after 61 billion USD spent for military purposes (Verbruggen & Wealer, 2021), the research finally focused on peaceful uses.

At the end of this decade, nuclear commercialization was established. Power plant concepts -both land-based and off-shore- started not only to make their appearance but also to gain international acceptance from leading countries such as the USA, USSR, France, UK, China, Japan, etc. According to Aruvian's research, between 1970 and 1980 world's nuclear power plants exceeded 50GW capacity. Nonetheless, the combination of the rising operating costs, the fear due to increasing accidents and the public's unawareness of environmental pollution from fossil fuels restrained nuclear technology expansion.

From a technical perspective, nuclear reactors' evolution counts 4 generations already. All of the 4 Gens are based on fission products while fusion is still in the experimental stage. The majority of the nuclear reactors working as of today belong to the second generation and significantly fewer to the third and the fourth. First-generation reactors were the ancestors of the latter and were mainly used in the first studies.

1.4 Nuclear Obstacles

Before analyzing how questionable is nuclear spreading for the public, it is more than crucial to shed some light on the most interwoven term which is no other than "*radiation"*. According to World Health Organization (WHO), *radiation* by definition is the flow of energy through waves or particles. People are exposed daily to radiation both from cosmic rays and some radioactive materials on Earth. In addition, *radioactivity* is an inherent property of some "unstable" materials that emit radiation spontaneously in order to shift to a more stable state (ACHRE Report).

Radiation can be classified into many categories. For instance, if it is ionizing or not, meaning if it is capable of detaching electrons from the atoms. Other sub-categorizations for ionizing radiation are alpha, beta particles, gamma and X-rays. Apart from the categorization, the most significant question remains whether radiation is harmful to people's health or not. The answer to this question is that nuclear energy riskiness is directly related to the exposure extent. Similarly, the higher the doses, the higher the risk for quantifiable damage. But in general and under the premises of serious protective measures, nuclear energy carries no risk for the people and it can be highly beneficial.

Radiation however is not always controllable and sometimes has proved to be quite relentless with humanity. In the past, several accidents connected to nuclear power plants ended up in true disasters and after that public opinion has been way more reserved and cautious rather than open to commercializing nuclear energy. One simple reference to the major accidents of the past 50 years may not be enough to cover the impact of these three misadventures but is necessary: Three Mile Island in the USA (1979), Chernobyl in Ukraine (1986) and most recently Fukushima in Japan (2011). All of them have different root causes however, their effect both on fatalities and also on the wave of confrontation around nuclear energy were vast.

Due to these serious incidents and some other less significant, different questions have arisen. Nuclear energy opposers nowadays, focus mainly on waste management of the nuclear fuel and equipment, the extent of damages from a sudden accident, the government regulations and port policies and finally the possibility of terrorism and other malicious uses. Besides all of the above matters which have safety as a common core, there is also another countable burden: the lack of expertise and the low readiness level.

2. NUCLEAR APPLICATIONS

As mentioned before, the post-war period marked the beginning of a new era. This was the era of nuclear energy commercialization and the capstone of it was the establishment of the Atomic Energy Commission (AEC) in 1946 in the US. Even though in the first years of its operation AEC was still involved in military tasks, with the passage of time and under the president's Truman blessings, AEC was transferred to civilian hands for the peaceful development of nuclear technology.

The first ever to realize the potentiality of implementing nuclear reactors in a naval application was Captain Rickover in Oak Ridge nearby Tennessee in a laboratory under the authority of AEC. Rickover was persuaded for nuclear energy's superiority as far as submarine propulsion was concerned and in 1951 after 5 years of research, the Nautilus' construction was officially approved. Only 4 years later and in partnership with BOSHIPS, Nautilus was launched and was ready for the first trip (Murray, 2009).

Nautilus could be characterized as the spark that inspired many subsequent developments. Although nuclear reactors were thought to be more appropriate for submarines due to their reduced thermal signature and high fuel range, soon they made their appearance in many surface navy applications and icebreakers and later on even in the commercial marine industry. A more detailed analysis of the implementation of the nuclear reactor follows.

During the following years, the US shared the knowledge and expertise with several other progressive countries such as France, China, UK and India that showed interest in the development of nuclear propulsion systems for their navies. Under the frame of collaboration, these countries coordinated towards the establishment of their fleets with success. Along with these powerful countries, another leader of nuclear power, Russia (former Soviet Union) continues to excel in this field with most of the nuclear civilian use in the maritime sector deriving from their icebreakers.

2.1 Navies

2.1.1 USA

Before expanding on some landmark vessels of the US Navy, it is of paramount importance to decrypt US coding in the reactors. The reactors are briefly described from 2 letters and one number in the form of L-N-L, where L stands for Letter character and N stands for Numerical Character. The first letter indicates the platform and can be either A for Aircraft carrier platforms, C for Cruiser platforms, D for destroyer platforms, or S for submarine platforms. Secondly, the number refers to the generation of the core according to the contractor and its values are integers from 1 to 9. The last letter symbolizes the designer company and it can be B for Bechtel, C for Combustion Engineering, G for General Electric and W for Westinghouse.

Table 1: US Navy reactors decoding

Reactor code	SHP[MW]	Applications
A2W	26.1	USS ENTERPRISE(CVN-65)
A4W	104.4	NIMITZ class aircraft-carriers
C1W	29.8	USS LONG-BEACH(CGN-9)
D2G	26.1	All USS GUIDED MISSILE CRUISERS (except from USS LONG BEACH)
S5W	11.2	USS SPITJACK(SSN-585) HMS DREADNOUGHT(S101-UK)
S5G	12.7	USS NARWHAL(SSN-671)
S6W	26.1	KNOLLS ATOMIC POWER LABORATORY
S8G	26.1	OHIO class submarines (SSGN/SSBN-726)
S9G	29.8	KNOLLS ATOMIC POWER LABORATORY

2.1.1.1 US Nautilus (SSN-571)

As stated, Nautilus is the ancestor of nuclear propulsion. Its history began officially in 1955 with the memorable announcement “Underway on Nuclear Power” by Commander Officer Wilkinson. Nautilus’ principal dimensions were 97.5 m, 8.5 m and 7.9 m in length, breadth and draft respectively. Main propulsion was achieved through an S2W nuclear reactor with a power output of around 1000kW and 23knots service speed. S2W stands for:

- S: Submarine platform
- 2: generation number
- W: Westinghouse for the designer

S2W was a light water moderated with highly enriched U235 as fuel and zirconium-clad fuel plates. Nautilus apart from being a breakthrough itself, accomplished a major venture: the first underwater crossing of the North Pole entirely. Additionally, it covered over 500.000 nm, and is the fundamental reason for US submarines’ future sharpness. Finally, after 2.500 dives and 25 service years, Nautilus was decommissioned in 1980 (Aruvian's Research).

Figure 7: US Nautilus during sea trials.

2.1.1.2 US Seawolf (SSN-575)

Captain Rickover in an effort to increase market competition decided to share his knowledge from Nautilus with General Electric for the production of a second submarine only with the basic distinction in the type of the nuclear reactor. Seawolf was equipped with a liquid metal-cooled (sodium) nuclear reactor and due to this specification, the machinery's room capacity was increased by 40%. Her construction began in 1953 and its commission was dated in 1957.

Figure 8: US Seawolf

However, after just two years, a single reported leakage problem and various concerns about the energy demands for preventing sodium's freezing, Seawolf's reactor was replaced with a PWR. It is very possible that at that age, Seawolf's influence on nuclear reactors' future did not seem so meaningful nevertheless, there has been no other submarine with such a determinant role. From this point on, PWRs were the initial preference over LMRs from many aspects, and in essence, it was the main reason for the latter's abandonment within all industries.

2.1.1.3 US Enterprise (CVN-65)

The transition era from nuclear-powered submarines to nuclear nuclear-powered surface vessels was a matter of time. What has to be noted, is that nuclear technology seemed to be more suitable for ventures that require enormous amounts of power such as aircraft carriers due to the high output power of nuclear reactors. Indeed, in 1961 the longest naval vessel ever built took flesh and bones. US Enterprise was a 342 m aircraft carrier of the US Navy with more than 4,500 people crew. The most impressive fact though remains the propulsion system of Enterprise which was based on 8 A2W nuclear reactors that were pressurized water with highly enriched (up to 93%) Uranium for fuel.

Figure 9: US Enterprise

For the successful propulsion of 94,781 tons of displacement along with the 8 reactors, 4 propellers were necessary. Except for the high-power output compared to the oil-fuelled vessels, nuclear reactors' design also allowed additional space for aviation fuels. Enterprise's novel concept for her age remains the SCANFAR, a way more developed radar with increased capabilities in the tracking of airborne targets. In 2012, after 50 years in service, Enterprise became the first-ever nuclear aircraft carrier to be decommissioned. A few months later, is transformed into a museum for the public.

2.1.1.4 USS Long Beach (CGN-09)

USS Long Beach, a nuclear-powered guided missile cruiser commissioned in 1961, holds a distinguished place in naval history as the first nuclear-powered surface combatant. In commonly accepted terms, USS Long Beach operated as the last cruiser built with cruiser design (all the upcoming vessels were built upon destroyer hulls). It was equipped with two C1W Reactors of 60MW that were used for powering two geared turbines.

Figure 10: USS Long Beach

Equipped with cutting-edge weaponry and radar systems. USS Long Beach played a crucial role during the Cold War era, serving as a powerful deterrent and a symbol of American naval supremacy.

2.1.2 USSR

At the same time with the US Navy's progress in nuclear propulsion, many other countries tried to benefit from the PWR technology whose capabilities were already proven in land. The most noticeable competitor of the US was of course the Soviet Union, particularly during the tense geopolitical climate of the Cold War era. In an effort to outmaneuver the US, the Soviets initiated the modeling and construction both of submarines and icebreakers. What separated the Soviets from all the other navies was the investment in LMRs over PWRs especially with the use of lead-bismuth as coolant.

The spearheads of the Russian navy back then were the Alpha-class submarines, whose innovative and more compact design with the LMRs made them dominants of the seas for years. However, consecutive problems regarding the corrosion of the equipment, the melting point and fuel unloading led to the decommissioning of all of them. An unexpected observation was made at the decommissioning time since the lead provided the necessary shielding for capturing the harmful radiation. The rest of Russia's expertise was under civilian uses of icebreakers and cargo ships and is analyzed further below. From 1950 (USSR Severodvinsk class series) to 2003 USSR/ Russia built 248 nuclear submarines, 5 naval surface vessels and 9 naval icebreakers powered by 468 reactors

2.1.3 UK

First of all, the United Kingdom acquired its first nuclear submarine HMS Dreadnaught in 1862. Rolls Royce became the main constructor for the English Navy and continues to power the UK's underwater defences until today. As of 2021, the UK fleet comprises of 12 nuclear-powered submarines (8 of attack type and 4 of ballistic missile).

2.1.4 FRANCE

French Navy's entry into the nuclear era was not late and in 1962 the first nuclear submarine of its fleet was already ordered. The first designs were called the "Le-Redoutable" class and they were submarines of ballistic missile type. In parallel and after a short period of second thoughts from President de Gaulle another class of submarines similar to the "Le Redoutable", the "Rubis" was approved. The first "Le Redoutable" and the first "Rubis" class submarines were launched in 1970 and 1979 respectively. Even though in the primary years of their lives Rubis faced noise problems French engineers with the development of the Amethyste program found the appropriate solution. As a result, Rubis are thought to be the most compact submarines that have ever been constructed. From a technical perspective, all of the reactors used are PWRs requiring Uranium of Low Enrichment (approximately 5%-7%) which allowed the refueling period to be as high as 7 to 10 years. As of 2021, France possesses 12 nuclear-powered submarines (8 of attack type and 4 of ballistic missile) and has gained significant experience from the Charle de Gaulle aircraft carrier, which is the only aircraft carrier that has ever operated and is not constructed from the US.

2.1.5 CHINA

The People's Liberation Army Navy (PLAN) as it is called involved with the construction of nuclear submarines in the 1970s for the first time. Type 091-attack submarines were the first to be commissioned but soon they were led to scrap due to the noise problems and the inadequate shielding they provided. Information is very limited since most of the files are classified, however, it is believed that PLAN submarines are operating from Russian PWRs and especially with LEU (Hagen, 2022). Until 2021, the Chinese Navy has 6 ballistic missiles and 9 attack submarines.

What's more Chinese have planned to double their submarines by the end of 2030 and also have expressed their interest in the shipbuilding of nuclear-powered icebreakers.

2.1.6 INDIA

Indian Navy is considered to be the newcomer in nuclear propulsion since the "Arihant" class's plan approval began in 2004. The sea trials of the first submarine of this class INS Arihant were completed in 2016 and the second INS Arighat was commissioned in 2022. Two more submarines are expected until the end of 2025. The Arihant class is of ballistic missile type operating with PWRs of HEU.

2.2 Russian icebreakers

Most of the civilian use of nuclear power in merchant shipping has been recorded from the icebreakers that operate in the Russian territory and the Arctic Ocean. The main purpose of the icebreakers was initially to clear the passage from the thick ice for the benefit of the following cargo chips and after 1980 serve also as a cruise ship for wealthy tourists who wish to visit the North Pole. Murmansk Shipping Company (MSC) traced the gap in the market and in collaboration with the Russian State undertook the operation of these vessels having as its service base in Atomflot, a region near Murmansk. A common trip of an icebreaker includes the ports of Dikson, Tiksi, Pevek and the cross of the Barents Sea, the Kara Sea, the Laptev Sea, the Eastern Sea and the Bering Strait. The conditions of this seaway are so harsh -due to the very low temperatures- that ships must break up to 2.5m of thick ice with their bow in order to cross the sea. To achieve this, the vessel's speed must be around 10kns and in the case of ice-free waters the maximum speed can be 21kns approximately. Generally, until 2020 Russia has operated 8 icebreakers under civilian use, 6 of the Arktika class and 2 of the Taymyr class.

2.2.1 USSR Lenin

Icebreaker Lenin is one of the most historical vessels since it was the first nuclear-powered to be available for civilian use. Even though it was launched in 1957, its first ordinary operation happened in 1959. During its lifetime, Lenin faced two accidents related to nuclear reactors in 1965 and 1967. The second one however was insuperable and this led to reactors replacement. Hence the initial three OK-150 reactors were replaced with three brand-new OK-900 units and the vessel was safe to travel again in 1970. The end of Lenin came in 1989 when it was decommissioned. The installed power of the reactors was equal to 270MWTh (90 each reactor from which 32.8 were designed to be absorbed from the shaft axis.

Figure 11: USSR Lenin Icebreaker

2.2.2 Arktika Class

A few years later the second generation of icebreakers was ready to cruise in the cold waters of the Arctic Cycle. The design of Arktika class icebreakers was probably a groundbreaking one with serious Improvements in icebreaking performance and this can be attributed to the reactors' development and the establishment of the turboelectric propulsion.

Figure 12: Arktika Icebreaker

The reactors were of the KLT-40 family, which were PWRs and contained a fuel enrichment of around 30-40% of U235. The power generation was achieved through steam turbines and in more detail for every reactor core, 4 steam turbines were required. The high-temperature steam of the second loop was specially designed to be condensed from the cold seawater of the Arctic Sea. Arktika icebreakers were equipped with 2 OK-900A (KLT-40 family) of 171 MW each and 3 propellers of 75000 hp total output. The first reactor was commonly used for power generation, while the second one was on a stand-by mode. These vessels were constructed with a double hull of 48mm steel thickness in the outer icebreaking layers and 25mm steel thickness in other areas. Maximum speeds were from 18 kns to 22 kns and and up to 3 kns in ice waters of 3m thickness.

Arktika class was a quite successful series of icebreakers and due to this fact, their production lasted for over 30 years. Thus, many of the icebreakers within the same class might slightly differ in some sectors however the general specifications were the below:

Table 2: Arktika Class Specifications

General Characteristics	Arktika Class
Length (m)	148-159
Beam (m)	30
Draft (m)	11.08
Displacement (t)	23000-25000
Fuel's endurance (years)	4
Crew	130-200

The exact fleet of Arktika class icebreakers is presented below:

Table 3: Arktika Class Icebreakers

Vessel	Laid Down	Launched	Commissioned	Decommissioned
Arktika	1971	1972	1975	2008
Sibir	1974	1976	1977	1992
Rossiya	1981	1983	1985	2013
Sovetskiy Soyuz	1983	1986	1989	2014
Yamal	1986	1989	1992	Up to present
50 Let Probedy	1989	1993	2007	Up to present

2.2.3 Taymyr Class

Taymyr class differs from the Arktika class due to the special design that allows them to travel in shallow waters. They were specially adapted for the route from Yenisei River to Dikson port to force through the ice for the benefit of the cargo vessels that used to carry ore, metals and lumber. Two icebreakers belong to this class: NS Taymyr and NS Vaigach which were constructed in Helsinki Shipyards in Finland and were launched in 1989 and 1990 respectively. Their general characteristics are shown in the table:

Table 4: Taymyr Class Specifications

General Characteristics	NS Taymyr	NS Vaigach
Length (m)	150.2	151.8
Beam (m)	29.2	
Draft (m)	8	
Displacement (t)	20000	
Cruising speed (kns)	18.5	
Crew	120-138	

Figure 13: NS Taymyr Icebreaker

Figure 14: NS Vaigach Icebreaker

2.2.4 LK-60Ya Class

In 2020 the Russian government financed the development of *Project 22220*. This plan included the shipbuilding of 7 icebreakers of LK-60Ya type, from which 3 are already in operation in 2024, one has been launched, two have been laid down and the last one is on order. Their principal dimensions are shown in the table below:

Table 5: LK-60Ya Class Specifications

General Characteristics	LK-60Ya
Length (m)	173.3
Beam (m)	34
Draft (m)	10.5
Displacement (t)	33530
Cruising speed (kns)	22(on ice-free waters)/ 2(on 2.8 ice-waters)
Crew	75

The installed power in the new icebreakers derives from two RITM-200 nuclear reactors of 175MWt each, assisted by two turbochargers of 36Mwe each. The total output of 60 Mwe is equally apportioned in three turbo-electric shafts that are responsible for the vessel's propulsion. RITM-200 reactors are PWRs of Generation III+ that operate in the thermal energy spectrum of the neutrons. The main advantage of RITM-200 is the low fuel enrichment in U-235 compared to the KLT-40s of the Arktika class.

Figure 15: Arktika LK-60Ya

2.3 Commercial applications

At the height of nuclear energy's flourishing, many countries attempted to embed this promising form of energy into their commercial shipping. In total, 4 countries succeeded in this venture: the USA, Germany, Japan and Russia.

2.3.1 NS Savannah (NSS)

There could be no other pathfinder rather than the USA. In the late 50s following the tremendous impact of US Nautilus and taking advantage of the meticulous scientific background that the US Government had founded for nuclear development, NS Savannah was already launched. Though it was never officially stated, it is well known that Savannah's purpose was to demonstrate the peaceful use of nuclear power under Eisenhower's desire.

Table 6: NS Savannah Specifications

General Characteristics	NS Savannah
Length (m)	181.66
Beam (m)	23.77
Draft (m)	8.99
Displacement (t)	21800
Cruising speed (kns)	21
Crew	110

NNS Savannah was a general cargo ship with a cargo capacity of 13.500t. The reactor's room was located in the middle of the vessel's length and it was surrounded by the steam generators and steam drums for more compactness. In more detail, the reactor was a PWR of low enrichment fuel (<4.5%) of 74MW of 15m height and 4.3 m diameter. This reactor was powering two steam turbines that supplied the shaft axis with 15500kW power.

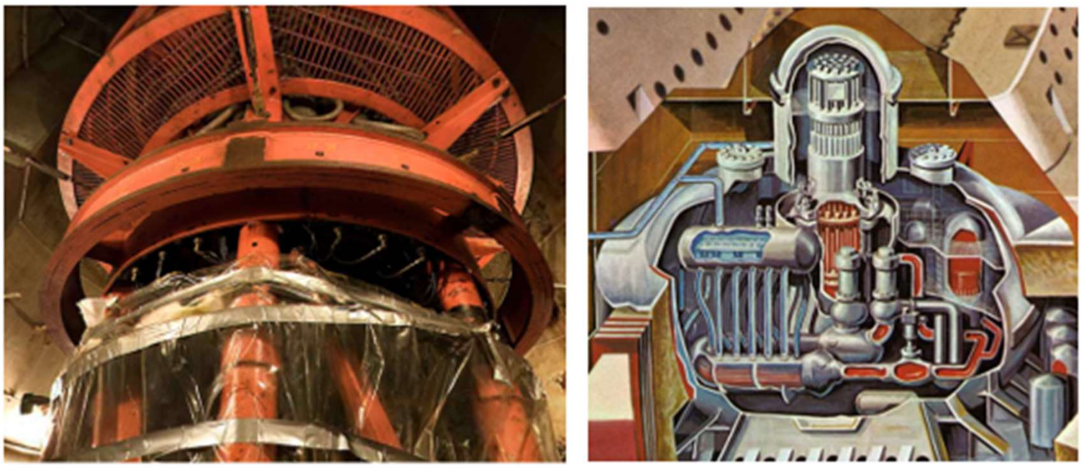

Figure 16: Savannah's rector on the left and concept of the reactor's room on the right

NS Savannah's lifetime ended in 1972 when it was officially taken out of service. This vessel will be always commemorated as the first nuclear-powered merchant ship, however during its operation the economic burden was proved too heavy to be overwhelmed.

2.3.2 NS Otto Hahn

Another government that tried to test the feasibility of nuclear power in the maritime sector was Germany. At the same time as NS Savannah, Germans laid down NS Otto Hahn, a vessel very similar to the latter. In 1968 after 6 years under the construction level, Germany added to its fleet its first nuclear-powered bulk carrier. The first port call was in 1970 at Safi Morocco. Its records are quite impressive since despite its short lifetime (only 10 years), it covered more than 250000nm, visiting 33 ports in 22 countries and its total footprint was equal to 22kg of Uranium.

Table 7: NS Otto Hahn Specifications

General Characteristics	NS Otto Hahn
Length (m)	172.05
Beam (m)	23.4
Draft (m)	9.22
Displacement (t)	25790
Cruising speed (kns)	17
Crew	110

Figure 17: NS Otto Hahn

Otto Hahn's reactor used enriched Uranium of 3.5%-6.6% and was of PWR type. Its propulsion was achieved with a four-bladed propeller and its single axis. The installed power of the reactor was 38MWt. Otto Hahn apart from its commercial applications also served as a research boat. The continuous prohibitions it received from different ports had as a consequence to limit the vessel's operational profile.

2.3.3 NS Mutsu

In an effort to imitate the last 2 governments, Japan tried to expand its knowledge in the field of nuclear shipping. However, this attempt was unsuccessful due to the major problems that the ship faced regarding the reactor's shielding. Mutsu was originally designed to be a general cargo ship nonetheless, it never carried cargo.

Figure 18: NS Mutsu

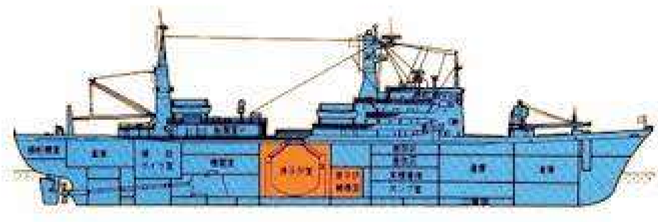

Figure 19: Mutsu Longitudinal Layout

Generally, Japanese people opposed to a significant extent to Mutsu's operation. For instance, it was the public's demand that made the ship be tested 400nm from the Japanese coasts in 1971. As mentioned before, the inadequate shielding of the reactor during the testing led to serious leakages of neutrons and gamma rays. The consequences were catastrophic for the vessel since it never operated for commercial reasons. From 1978 to 1982 the vessel was under repair and from 1982 until 1995 it served as a research vessel. At this point Mutsu's reactor was replaced with a conventional diesel engine, the name was changed to Mirai and the ship still operates as an oceanographic.

Table 8: NS MUTSU Specifications

General Characteristics	NS Mutsu
Length (m)	130
Beam (m)	19
Draft (m)	6.9
Cargo capacity (t)	8240
Cruising speed (kns)	17.2
Crew	80

Mutsu's history nonetheless -despite being short- can be highly beneficial for future attempts at nuclear applications within the shipping industry. Mutsu's reactor was designed by the Westinghouse Electric Company, while the rest of the ship's design- including the shielding- involved many other parties. So even though the shielding was reviewed by the WEC, it is very possible that during the collaboration between these parties, a minor mistake was made and this led to the leakages.

2.3.4 Sevmorput

The Soviet Union could not stay detached from this series of events and in addition to its strong nuclear background in 1982 initiated the project of Sevmorput. Sevmorput was the last of the four in total nuclear commercial vessels that have ever been built and is the only one that remains in service today. In the Russian language, Sevmorput means "Northern Passage" and it was intended to carry various types of cargo such as containers, bulk goods and heavy machinery in the most remote regions of the Russian Arctic Cycle. Sevmorput's versatility in combination with its increased cargo spaces, made it look very attractive since it was capable of traveling both in iced and iced-free water without the need for an icebreaker ahead.

Table 9: Sevmorput Specifications

General Characteristics	Sevmorput
Length (m)	260
Beam (m)	32
Draft (m)	11
Cargo capacity (t)	40000
Cruising speed (kns)	20
Crew	80

For the power generation, a KLT-40 reactor as the ones used in the Taymyr and Arktika class. Sevmorput had an installed power of 135MW and used fuel of 30-40% enrichment. The main difference with the icebreakers is the propulsion system, where in this case Sevmorput used a single four-bladed ducted controllable-pitch propeller. For electricity production, the vessel was equipped with three turbogenerators of 170kW capacity and three diesel generators of 2000kW for more valuable redundancy in case of a nuclear reactor malfunction.

Figure 20: Sevmorput Icebreaker

Despite the collapse of the Soviet Union in the early 1990s, Sevmorput continued its operations, albeit with various challenges and adaptations in the changing geopolitical landscape. Today, Sevmorput remains a symbol of innovation and resilience in the maritime industry but according to different sources, 2024 is expected to be the last year of its operation sealing in that way the most successful marine application of nuclear science.

2.4 Future prospects

Unlike the naval implementations, it is widely accepted that nuclear propulsion has not attained its standing within commercial shipping. The only exceptions so far have been the Russian icebreakers, however the rest of the attempts were limited to prototype vessels. This is a multifactorial phenomenon that can be attributed to many causes nonetheless what is undisputable is that public safety concerns never left the foreground.

A comprehensive analysis would reveal another aspect of why nuclear reactors left so importantly behind diesel engines: the exploitation of most nuclear cargo vessels happened between 1950 and 1990 when global warming was absent from people's consideration (Houtkoop, 2022). The latter reason in combination with the increased maintenance costs that every novel concept brings, condemned nuclear propulsion. The idea nevertheless was not abandoned, many researchers continued to investigate the implementation of nuclear power into commercial shipping, new technologies are entering the market and with the GHG crisis approaching, many companies have foreseen the prospect of nuclear energy.

Towards this direction and mainly after 2020 many projects are rumored within the shipping community. Vard Group, which is commanded by Håvard Lien a strong supporter of nuclear energy over alternative fuels- participates in the NuProShip program that aims to develop a Gen IV reactor specially for marine vessels by the end of 2024.

Another strong evidence of shipping's industry target to implement nuclear energy commercially is the case of the Italian Shipbuilding giants Fincantieri who are willing to join forces with Newcleo and Rina on a feasibility analysis for a 30MW reactor (RINA, 2023). Japanese on the other hand with Imabari Shipbuilding on the front have already invested more than $80 million in CorePower a British start-up commanded by Mikael Boe to explore SMR technology.

What's more, is that apart from RINA also ABS has shown interest in the prospects of nuclear energy within the maritime sector and for this reason, a part of its RnD department has focused on this topic. ABS has signed a federal contract with the Department of Energy for further research on the opportunities for advanced nuclear technology in the maritime sector. In parallel ABS is also supporting separate research into molten salt reactors. DNV from its side has already suggested nuclear energy as a feasible solution for oceangoing vessels on the published forecast Maritime 2050 (DNV, 2023)

According to World Nuclear Association findings, nuclear energy seems promising not only for navy purposes but also for large bulk carriers with fixed lengthy routes, and huge cruise liners whose demands could be compared to the demands of a small town(60MW-70MW).

In addition, several world-famous shipping companies along with Korea Atomic Energy Research Institute and Samsung Heavy Industries are trying to implement the idea of a compact molten salt reactor from 2021 Q3. Actually, this collaboration has already been successful since it has already provided some conceptual designs of a floating nuclear power plant called *CMSR Power Barge,* which will produce electricity from 200MW to 800MW for approximately 24 years.

Figure 21: CMSR Power Barge (World Nuclear News, 2023)

However, no research and study so far can be compared to the KUN-24AP project. This project is the brainchild of China Shipbuilding Corporation’s Jiangnan Shipyards and until this moment is the most prosperous idea. In technical terms, it concerns a 24000 TEU containership with a Gen-IV thorium-based reactor which is way safer than uranium-based reactors. As it can be easily understood, with the regulations becoming stricter and with the option of alternative fuels proving not so feasible, more research and programs concerning the implementation of nuclear power will arise. As most scientists foresee, it is a matter of time until shipping society recognizes the real chances of nuclear commercialization within the industry.

3. NUCLEAR REACTORS

Nuclear energy would never have been spread to this extent if it were not for the reactors. In other words, nuclear reactors are the “vehicle” of nuclear energy to the common world and our daily life. Many improvements have been noticed during the last 70 years since nuclear technology was first introduced in scientific society. These technologies share not only many differences but also several commonalities and for this reason, a primary classification among the reactors has been established. Primarily, the most substantive categorization refers to the generation of the reactors and until today, nuclear society has already introduced Gen I, Gen II, Gen III Gen III+, and Gen IV. Apart from the generation of each reactor, there are several other criteria such as the spectrum of the neutrons, the modularity and the size of the reactor or even the converting abilities of the reactor. In this chapter, the key features and characteristics of every major reactor design are presented along with some promising models whose development has been announced or they have already been established. All the existing proposals concern fission reactors only while fusion reactor design is under research.

3.1 Categorization of reactors

3.1.1 Generations

When it comes to generations’ categorization, the classification criteria refer primarily to the release date of the reactors and secondly to the baseline technology they share. According to a survey from the American Bureau of Shipping (ABS) these are the results of reactor classification:

- GEN I, 1950-1960: Early prototype reactors. GEN I reactors were the first implementation of nuclear energy to nuclear power. Idaho’s project is a typical example of this category, and its purpose was to demonstrate the potential applications that nuclear reactors could support (Farkas, 2010). Today, no one reactor of Gen I has remained in operation as the last one -Wylfa 1 in the UK- shut down at the end of 2015.
- GEN II, 1960-1990: This generation was the signal for an uncontrolled expansion of nuclear energy under a commercial character. A huge emphasis of this generation was given to safety, economic feasibility and their prolonged lifetime which varies from 30 to 40 years. The most successful GEN II reactors that dominated the market are without doubt LWR (Light Water Reactors) including PWR (Pressurized Water Reactors) and BWR (Boiling Water Reactors) with more than 400 units in use. Other noticeable designs of the second generation are HWR (Heavy Water Reactors), VVER (Vodo-Vodyanoi Energetichesky Reactors) and AGR (Advanced Gas-Cooled Reactors).
- GEN III, 1990-2016: The third generation is the generation of the advanced Light Water Reactors. As the World Nuclear Association underlines there is no typical distinction from the second generation but there were specific improvements in different sectors such as fuel technology, thermal efficiency and the modularity of the construction. In addition, extra notice was given to the improvement of passive safety. The target for the life

expectancy of these reactors was set to 60 years. The third generation is interwoven with ABWR (Advanced Boiling Water Reactors).

- GEN III$^+$, 2016-currently. Gen III$^+$ reactors were in general an evolutionary development of the third generation with a major difference in safety. Emerging from the Fukushima accident, this generation is designed with enhanced passive cooling capabilities for addressing melting problems. Typical examples of this category are: VVER-1200 and EPR (European Pressurized Reactors)
- GEN IV, is the latest version of reactors but apart from a 200 MW HTR (High-Temperature Reactor) from the Chinese Government all other designs are still under investigation. It has been suggested that Gen IV reactors will have improved safety, reduced nuclear wastes, competitive economics, good performance and simplicity in construction and maintenance compared to the previous generations (Generation IV International Forum, 2001). According to this forum, six designs were promoted for further research: the gas-cooled fast reactor (GFR), the molten salt reactor (MSR), the sodium-cooled fast reactor (SFR), the lead-cooled fast reactor (LFR), the supercritical-water-cooled reactor (SCWR) and the very high-temperature reactor (VHTR). Commercial deployment is set to start in 2030.

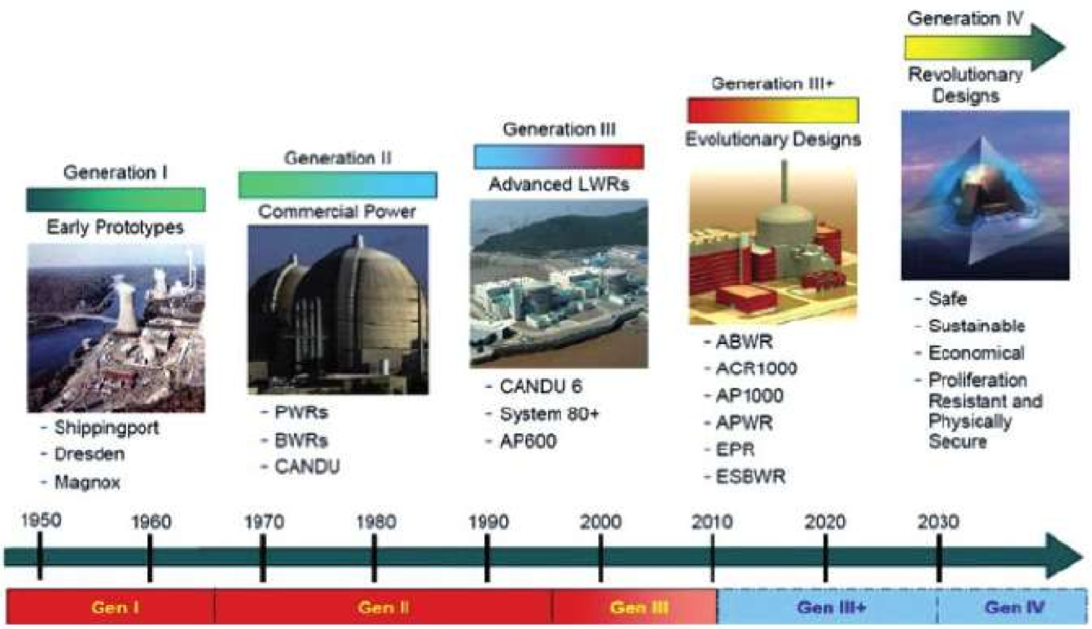

Figure 22: The evolution of reactors per generation (Xenofontos, 2018)

3.1.2 Spectrum

Besides the chronological classification, as explained in the previous chapter another distinction among the reactors is made due to the spectrum of the involved neutrons in the reaction. The two prevailing spectrum families for reactors are thermal and fast reactors, characterized in this way by the kinetic energy of the participating neutrons. As ABS recommends:

Table 10: Reactors' Categorization based on the working spectrum

Thermal Neutrons	Fast Neutrons
Pressurized Water Reactors (PWR)	Traveling Wave Reactors (TWR)
Boiling Water Reactors (BWR)	Molten Chloride Fast Reactors (MCFR)
Heavy Water Reactors (HWR)	Stable Salt Fast Reactors (SSFR)
Advanced Gas Cooled Reactors (AGR)	Molten Chloride Salt Fast Reactor (MCSFR)
Molten Salt Reactors (MSR)	Sodium Cooled Fast Reactor (SFR)
Thorium Molten Salt Reactors (TMSR)	

3.1.3 Modularity

Another method for classifying the nuclear reactors is by their modularity. Generally, as the IAEA has proposed, there are 3 distinct categories of nuclear reactors:

- Large Conventional Reactors with more than 700 MW(e) power capacity
- Small Modular Reactors, from 10MW(e) up to 300MW(e) power capacity
- Microreactor, up to 10MW(e) power capacity.

From the above three, the most promising selection for any naval propulsion system is undeniably SMR, as the reactors of this design possess the capacity to meet the requirements of the vessel while also maintaining a compact size, which is critically important within the confines of an engine room. What should not be ignored, is the unique characteristic of modularity which makes possible the replacement of any defective component easily and without risk. All the GEN-IV reactors that are under consideration are thought to be SMR.

3.1.4 Converting/ Breeding

Another feature that sometimes separates the reactors into two major groups is the ratio of fissile material consumption over production. Typically, most reactors consume more fissile material than they produce and based on this notion they are called “burners”. On the opposite, some reactors generate more fissile material (deriving from the fertile) and for this reason, they are called “breeders”.

3.2 Reactor key terms

Burnup: This is a measure of the amount of energy extracted per mass of the initial fuel. In other words, burnup quantifies the amount of consumed fuel during the reactor's operation. It is measured in Megawatt-days per metric kg of Heavy Metal (MWd/kgHM) and generally, the rule is that the higher the burnup ratios are, the more efficiently the fuel is used. As far as the waste is concerned, the volume of high-level waste declines since less unburned fuel is left, and the recycling of spent fuel becomes more meaningful. Nevertheless, the radioactivity and heat are increased since more fissile products are turned to spent fuel.

Poisoning: This defines the phenomenon of neutron-capturing by other materials of nuclear reactors without undergoing fission (Houtkoop, 2022). As a result, less fissile material is available for energy production unintentionally and this reduces the reactor's efficiency. However, there are some occasions where neutron poisons are intended like the control rods.

Fuel Density: This term expresses the concentration of fissile material over fertile material within the nuclear fuel. Of course, a rise in fuel density is equivalent to more efficient use of fuel, downsizing of the fuel core and an associated increase in the power output of the plant. Nonetheless, balances must be kept with respect to several safety concerns.

Proliferation: Proliferation is a term that describes the risk of weapon development from fuel cycles that are related to both civilian and military use. Acceptable solutions for mitigating this risk are the continuous reprocessing of spent fuel and secondly the strict monitoring of the nuclear facilities.

LOCA: LOCA stands for the Loss-of-coolant-accident and describes a very common danger for nuclear reactors which is no other than a possible leakage of coolant. In such an event, temperature could reach extreme levels and the results then can be catastrophic.

3.3 Reactors basic concept

Even though every generation brings new improvements and features, the concept of reactors' operation remains identical. As implied in the first chapter, the primary target of a nuclear reactor is to transmit the kinetic energy of some unstable elements that serve as fuel into useful power.

The majority of reactors comprises of essential components such as the core, control rods, fuel, and coolant. Moreover, in the outer layers of reactor designs, additional features such as shielding, and occasionally a moderator and/or blanket, are incorporated. A basic notion of how all these systems cooperate is represented in the scheme below.

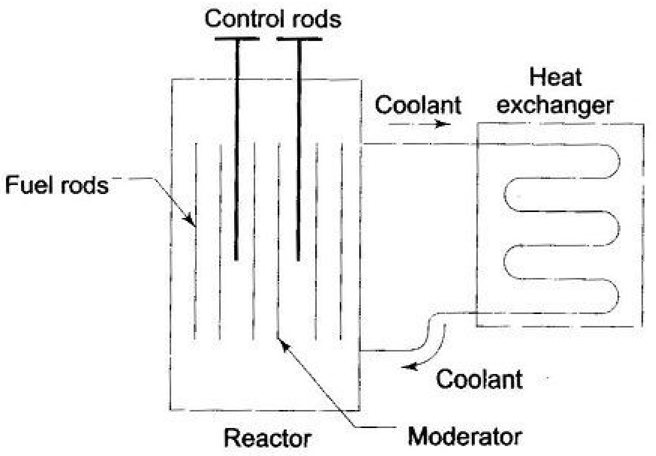

Figure 23: Reactor Compartments

3.3.1 Core

At the heart of every nuclear reactor lies the core, where the nuclear fission reactions take place. Typically contains nuclear fuel, such as uranium or plutonium, arranged in fuel assemblies, the core serves as the primary site for generating heat through controlled chain reactions. Apart from the fuel, which is a fissile nuclide, usually a fertile material co-exists.

3.3.2 Control rods

3.3.2.1 Reactivity

A measure to monitor the pace of the reaction is the reactivity *k,* which is the fraction of the number of fissions over the number of fissions at the previous generation n. When this fraction is exactly 1 only then is the reaction self-sustained. In any other case, meaning that is lower or higher than 1, then the reactor will produce reduced or increased power respectively. The appropriate condition of any reaction is to maintain a stable condition for the reaction so that it can be self-sustained.

3.3.2.2 Target

The functionality of the control rods is to intervene whenever there is a dispute in the reactor's demands. The rods operate by absorbing the free neutrons and according to the rate of absorption, the number of fissions and consequently the output power are affected. From a mechanical perspective, the rods are lowering and raising between the fuel. Thus, the available surface for collision is differentiated. It is also unambiguous that the selection of a suitable material for the construction of the control rods is of high priority. The most significant part of the control rods according to Houtkoop is the *high neutron cross-section* and the materials that are mostly used are Cadmium and Boron. Except for the need to maintain the criticality of the reaction what is undeniably more urgent sometimes is the need for a shutdown and control rods are equally responsible.

3.3.3 Fuel

3.3.3.1 Sources

Power production in nuclear reactors requires a different method from the conventional combustion that most of the machines are designed for. Nuclear fuels are not numerous however some categorizations within this limited number are made. Apart from the differences in order for nuclear fuels to be effective, the coexistence of both a fissile and a fertile element (typically heavy actinides) is more than vital. In a more simplistic approach, fissile materials are responsible for starting the reactions by releasing neutrons with increased energy ready to collide, while fertile material is the absorbent mechanism that will turn to fissile. The absence even of one of the fertile or the fissile materials, signals the discontinuation of the chain reaction leading to a fuel renewal.

The options among nuclear fuels are not limitless especially when referring to naturally created fuels. Nowadays, the only naturally occurring fissile fuel is U_{235} which is found as an ore and it can be used in various forms (Glasstone & Sesonske, 2012). Apart from the natural Uranium fuel, some other options such as Th_{232} and U_{238} which could be characterized as natural fertile materials under certain procedures can turn into fissile U_{233} and Pu_{239} respectively. However, these options require human intervention since both options need external neutrons to become fissile under decaying, as illustrated below.

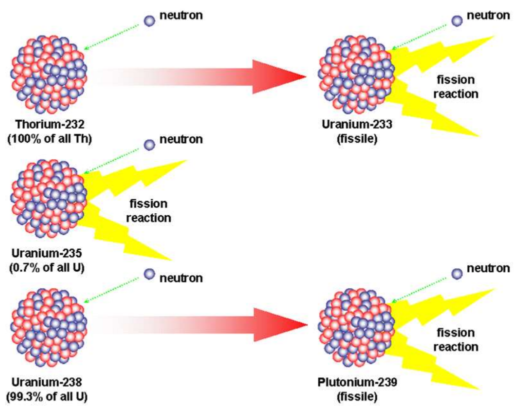

Figure 24: Transformation of fertile materials to fissile[1]

U_{235} even though it is the outcome of a natural process, it does not contain an adequate portion of fissile material over fertile. In more detail, natural Uranium's concentration is about 99.267% of fertile U_{238} and only 0.711% of fissile U_{233}. At these levels, the reactor will not be so effective and what is more, it will require more frequent refueling. For this reason, most of the reactors use nuclear fuels that have undergone a typical procedure called *enrichment*. Enrichment is a typical and fabricated process under which the percentage of fissile material is controllably increased

[1] https://steemit.com/chemistry/@pinkspectre/the-mighty-thorium

compared to the fertile material. Based on the composition at the final stage there are these types of enriched Uranium fuels:

Table 11: Uranium Classification according to enrichment ratio (%)

U_{235} Classification			
SEU	HALEU	HEU	HEU Weapon Grade
<2%	2%-5%	20%-80%	>80%

3.3.3.2 Fuel Forms

Natural Uranium on its path to become nuclear fuel, goes through several procedures and fabrications. The upper goal is the fuel's optimization and, on these grounds, different fuel configurations have been established and tested so far. Among them, the two prevailing categories are Oxide and Metal fuels. As most of the researches have shown, Oxide forms serve their purpose on a better grade since due to their higher melting point, they cannot burn.

The most popular Oxide used in nuclear reactors is Uranium dioxide, with a 1:2 ratio of Uranium over Oxygen. Generally, Oxides are preferred due to the increased structural stability in high temperatures that Oxygen offers. Another verified solution is the MOX fuels, which are oxides of mixed Uranium and Plutonium. A certain benefit of MOX is undoubtedly Plutonium recycling, which ensures a lower volume of high-level waste and in addition can guarantee a more effective utilization of the spent Plutonium contributing in that way to an extended fuel lifetime. A noticeable aspect is that with less Plutonium available which is the main component of nuclear weapons, the proliferation risk is mitigated too. UCO also is a combination of Uranium oxide and Uranium carbide which can provide a good outcome for the reactor.

Metal form of nuclear fuels on the other side is an acceptable solution due to the increased heat conductivity that metals can offer. Although there are certain cases where pure metal Uranium has been used, metal alloys are proven to be the rule rather than the exception. Most commonly seen nuclear alloy fuels include uranium aluminum uranium zirconium and uranium silicon. What's more, fast neutron reactors can employ an alloy of all the aforementioned metals in combination with minor actinides which are produced by neutrons captured by U and Pu.

Different forms than metal and oxides can also be deployed efficiently. Non-oxide ceramic fuels constitute a potential, with Uranium Nitrides and Uranium Carbides being the most appealing options. Lastly, the most challenging fuel of the future since they seem to fit with the new generation of reactors are liquid fuels, especially molten salt in which actinides are dissolved.

Table 12: Various forms of nuclear fuels

Potential Forms of Nuclear Fuels						
Oxides		Metal		Non-Oxide		Liquid
UO_2	MOX	Pure U(ore)	Alloys	U Nitrides	U Carbides	Molten Salt

3.3.3.3 Fuel Cycle

The Fuel Cycle of nuclear fuels refers to the several stages that the fuel will undergo during its whole lifetime. From a general perspective, the Fuel Cycle could be decomposed into three periods:

I. Before utilization: This is the part where Uranium is mined, collected, enriched and converted to suitable forms ready to enter the reactor
II. During reactor operation: This is the part where the fuel is capable of producing power and effective utilization.
III. Reprocess and waste management: This is the stage that is closer to the end of the useful lifetime of the fuel and for this reason, fuel assemblies are extracted from the reactor in order to be cooled and stored as waste. Some countries nonetheless have developed methods for extracting the active Uranium and Plutonium which has remained in the cells (closed fuel cycle). In this way, both the need for new Uranium and the risk of weapon development are significantly reduced.

3.3.3.4 Fuel Assemblies

There are several designs and modifications before the fuel entrance into the reactor, but the two prevailing ones are prismatic assembly and pebble bed assembly.

The first option is the most traditional type of fuel assembly and consists of fuel pellets that are vertically arranged in tubes (fuel rods). In the past, the suggested material for these tubes was stainless steel, however nowadays zirconium alloy tends to equip the majority of the reactors. These rods are tied up together in a lattice structure with many shapes such as hexagonal, triangular or even annular forming in that way bundles. All the bundles are cladded with a corrosion-resistant material together to compose the nuclear reactor.

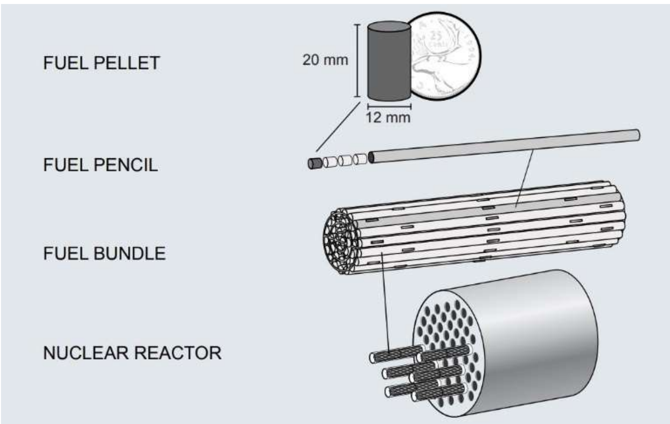

Figure 25: Forms of nuclear fuel

In the second option, the fuel follows the TRISO (TRIstructural ISOtropic) model in which the fuel is placed in the center of a spherical kernel under the form of an oxide, carbide or oxycarbide. The fuel is surrounded by three layers with protective roles: an inner pyrolytic carbon (IPyC) layer, a silicon carbide (SiC) layer and an outer pyrolytic carbon (OPyC) (Powers & Wirth, 2010). In contrast to prismatic assemblies, in this version pebbles flow freely inside the core during the operation.

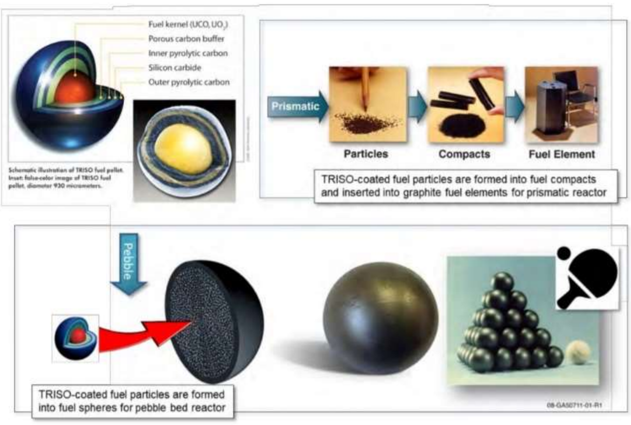

Figure 26: The TRISO design (Laranjo, 2017)

The third option which has not been tested in so many applications but seems more prosperous than any other is the liquid fuel in the Molten Salt Reactors (MSRs) where the nuclear fuel is dissolved in a circulating molten salt mixture.

3.3.4 Moderator

3.3.4.1 Neutrons Spectrum

The spectrum of neutrons describes certain energy ranges at which the neutrons of a reactor operate. Neutrons in most cases work in the low energy spectrum which is characterized as the *thermal* spectrum. However, there are several exceptions where neutrons are designed to operate in higher energy conditions called *fast* and the range is MeV. The main distinction between these two is the likelihood of neutron collision. In the fast spectrum, the kinetic energy of neutrons is incomparably higher than in the thermal spectrum and therefore the likelihood of interaction with the atomic nuclei is diminished.

3.3.4.2 Operation

The role of the moderator is similar to the one of the control rods. The moderator's primary purpose is to slow down the moving neutrons by causing their collision with its materials. This procedure of energy loss in scientific terms is characterized as *scattering.* The most used materials for moderators are water (H_2O), heavy water(D_2O) and graphite (C). Apart from the increased possibilities for interaction, moderators offer an extra tool for power control since they can affect the reactivity rate. Even though it seems more logical that moderators apply to the fast spectrum reactors for restraining the highly kinetic neutrons, moderators are more suitable for thermal spectrum reactors.

3.3.5 Reflector, blanket & shielding

A nuclear reactor might have on the outer layers of its core a reflector, a blanker and of course shielding. All these means, have protective and preventive character since they act as deterrents for leaking problems. The reflector uses its surface to change direction to the escaping neutrons Another satisfying solution is the blanket which is mostly found in breeder reactors (Houtkoop, 2022) since it is made from fertile material ready to interact with the neutrons and turn them into fissile. Shielding of course is the most important part of a reactor as far as the safety is concerned. It is made from very dense material that will weaken the gamma and the X-rays that are emitted from the fuel. The coexistence of all these in combination with a careful preparatory study provides a more reliable environment for the involved parts.

3.3.6 Coolant

Without the use of the coolant, it is more than obvious the nuclear reactor would not operate successfully. Coolant is responsible for absorbing the generated heat, transferring the produced energy from the core to the generator and at the same time preventing materials from overheating. Most of the known applications use water as a coolant, nonetheless, new advances in this field have unlocked multiple prosperous paths. These options could be either liquid or gas and more information about them is given afterwards. What remains important no matter what the coolant is, is the output power of the reactor over the output amount of heat. In other words, the coolant's properties play a dominant role in the whole cycle's efficiency, while typical values for the current state are between 30% and 40%.

3.4 Reactor designs

This analysis focuses on certain reactor concepts that are either of proven technology or their prospects seem very promising for the short term. The summarization for each model contains information regarding the basic operation, the internal conditions (temperature and pressure), the fuel technology and certain dimensions whenever such data were accessible. In the case of Gen-II, Gen-III, and Gen-III+ reactors where there has been massive production, average values extracted are certainly more trustworthy, whereas for the Gen-IV models the figures refer to demonstration models that have not been implemented yet.

3.4.1 Generation II, III, III+

It is evident from the historical data from Statistica.com that only specific designs have found broad application and more particularly the most dominant ones are only three: PWRs, BWRs and PHWRs. Based on this figure, it is comprehensible that the PWR readiness level cannot be compared to any other, nonetheless, significant knowledge can be gained from PHWRs and BWRs too.

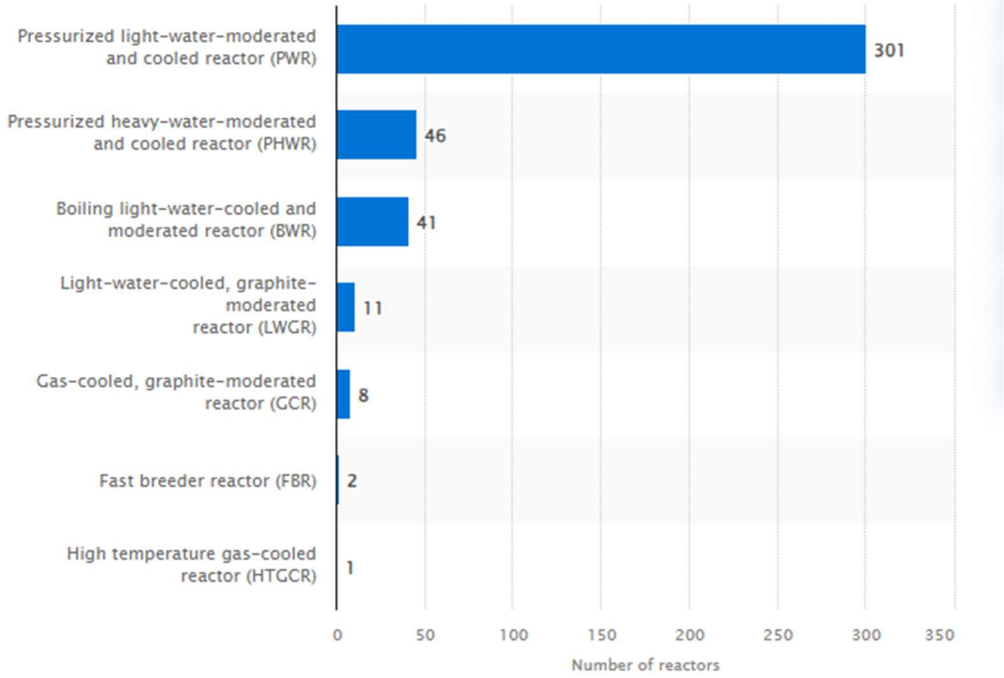

Figure 27: Population of nuclear reactors by type by Statista (Garside, 2024)

3.4.1.1 Pressurized Water Reactors (PWRs)

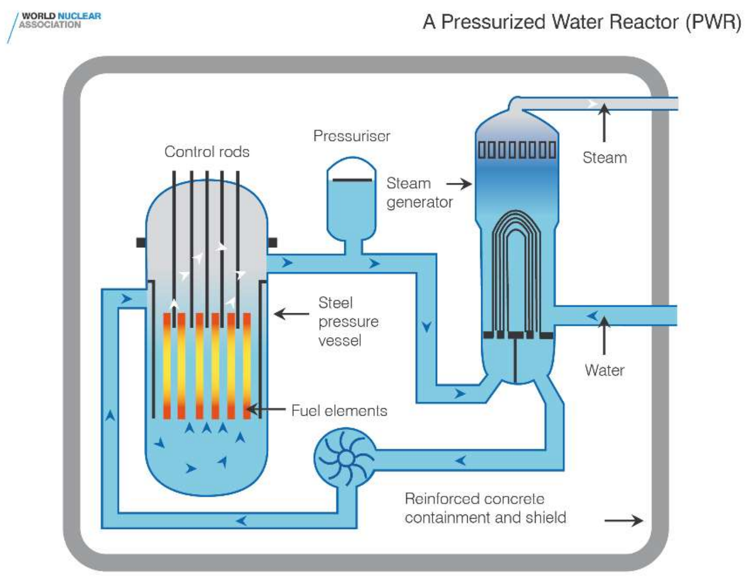

Figure 28: PWR design (World Nuclear Association, 2024)

PWRs are probably the most widespread solution for nuclear applications as of today. In pure numbers, PWRs were found to cover more than 65% of the world's nuclear reactors (Breeze, 2019) with 300 operable reactors for power generation and about 100 especially for naval propulsion. PWRs belong to the second Generation (GEN-II) and their neutrons belong to the thermal spectrum. Their signature characteristic is the use of pressurized (non-boiling) water as a coolant in the primary loop which at the same time acts as a moderator too. After the heat extraction from the nuclear fuel, the pressurized water transfers the heat through a heat exchanger into the secondary loop where steam is recycled. The two loops are isolated and thus the steam is not in direct contact with radioactive material. The power range for PWRs varies from 60 MW_e to 1650 MW_e, but generally high outputs concern land-based applications.

Logical pressures in the primary loop are within 15-16 MPa, while in the secondary loop steam works at pressures less than 6 MPa. From the temperature aspect, PWRs cannot compete with newer designs with higher temperatures -and thus higher efficiencies- due to water constraints. As a result, 330°C is the highest temperature that can be recorded in the core, and for this reason, efficiency is mitigated within 30% to 35% always in direct connection with a Rankine cycle.

Typically, PWRs use enriched Uranium of 3%-5% in order to balance the neutrons' losses from the water and the materials (Ho, Obbard, Burr, & Yeoh, 2018) in the shape of Uranium Dioxide pellets. However, in most of the naval applications, due to restrictions in the volume, the selected Uranium is highly enriched (>20%) and is fabricated in the fuel rods as a metal alloy (U-Zirc or U-Al alloys). The refueling intervals can exceed 7 to 10 years, overhauling a fuel range from 1 to 6 years as the evidence derived from the Navies imply. The same sources support that overhauling jobs are scheduled every 20 years and that reactors' lifetime is estimated to exceed 60 years (WNA, 2021).

Every 200 rods from the ones described above are bundled up together to form a fuel assembly. PWRs are equipped with numerous assemblies depending on the desirable power output starting from 35 to even 315 per unit. What should be highlighted is the transition of PWRs from Large Conventional Reactors into SMR types. If it were not for this transition, then submarines and the rest of naval applications could not be equipped with PWRs. Existing figures of PWRs employed in the shipping industry and at this power range vary from 3.8m to 23.1 m in height and from 0.2m to 6.5 m in diameter.

As far as safety is concerned, the shielding of the reactor probably plays the most crucial role. As Hirdaris et al notice, a typical PWR reactor would require more than 100 tons of lead shielding and for this reason, the total weight of the reactor would exceed a diesel engine with a similar power range.

Table 13: PWRs technical characteristics

PWRs technical characteristics					
	Number of investigated reactors:	22	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	60	1650	866
	Plant Efficiency	[°C]	23.3	36	33.6
	Thermodynamic Cycle	Rankine			
Primary-Secondary Loop specs	Coolant Inlet Temp	[°C]	80	299	281.2
	Coolant Outlet Temp	[°C]	120	330	315.1
	Coolant Operating Temp	[MPa]	0.6	16.3	14.9
	Steam Max Temp	[°C]	272.7	300	287.4
	Steam Max Pressure	[MPa]	4.5	7.2	6.76
Fuel specs	Fuel Material	UO_2, MOX, Cermet			
	Cladding Material	Zirconium Alloy			
	Fuel Cycle Length	Months	12	72	23.9
	Fuel Enrichment	[%]	2.76	13	4.9
	No. Fuel Assemblies		37	312	182.9

	Lattice geometry	Square, Hexagonal, Spherical, Triangular			
	Fuel Power Density	[kW/kgU]	21.3	117.8	41.3
	Discharge Burnup	[MWd/kg]	15.3	65	51.5
Dimensions	Core Height	[m]	1.9	4.2	3.5
	Core Diameter	[m]	1.5	3.9	3
	Reactor Height	[m]	3.8	23.1	10.9
	Reactor Inner Diameter	[m]	0.2	6.4	4.1
	Reactor Shell Thickness	[m]	0.015	0.25	0.2
	Weight	[tons]	5	795	337.6

3.4.1.2 Boiling Water Reactors (BWRs)

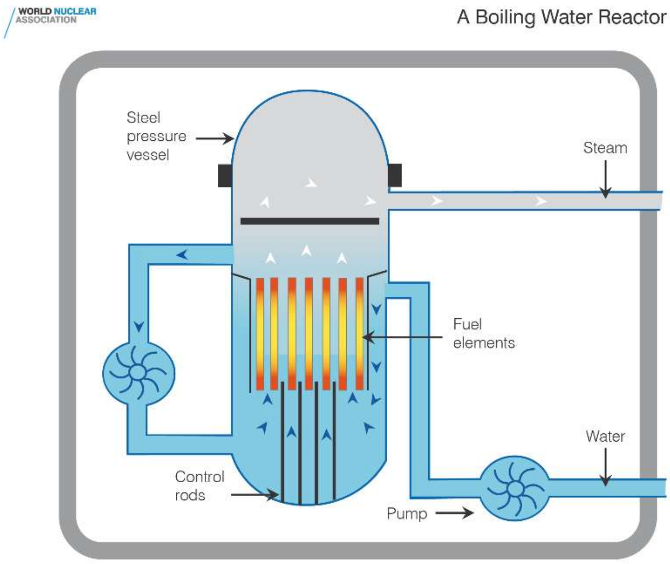

Figure 29: BWR design (World Nuclear Association, 2024)

Boiling Water Reactors (BWRs) are the second most ubiquitous solution worldwide, with more than 20% of the world's power plants using this technology. Initial BWR concepts were developed in US Laboratories in collaboration with General Electric and in the mid-60s BWRs went commercial as GEN-II products.

Their design is very similar to the previous one of PWRs -they belong in the same family of LWRs either way- with a fundamental differentiator: there is a unique loop instead of two (primary and secondary) and the working fluid is boiling water. However, since the fluid is in direct contact with the fuel, water becomes radioactive and thus leakages are strictly forbidden. To achieve this state of water (boiling), lower pressure is required within the loop. When the water passes through the reactor core and is boiled, is directly driven to the turbine for electricity and from there to a condenser for the final step of heat exchange. When the water comes out from the condenser is ready to record another cycle. Power output levels for BWRs vary from 290 MW_e to 1638 MW_e.

According to the ABS study, pressure values that favor water's boiling should not overcome 7MPa. The boiling temperature at these conditions is around 285°C and the maximum theoretical efficiency (Carnot Cycle) of a BWR ever recorded was 42%. Nonetheless, achievable efficiencies are less than 40% but in general higher than PWRs (Nave, 2015).

Since there have not been SMR types of BWR, all the fuel specifications and dimensions are not comparable because they concern land-based applications with higher power capabilities. The appropriate fuels for a BWR are rods of UO_2 or MOX with Zirconium being suggested as the optimum cladding material (Kok, 2009). The enrichment levels are similar to the PWR levels always kept under 15%. However, due to the existence of water which acts as a moderator and hence more neutrons than normal are absorbed, enrichment of 3.4%-4.95% will be certainly necessary (Lamarsh & Baratta, 2001). The average discharge Burnup is estimated at 50 MWd/kg while fuel autonomy might be from 1.5 to 2 years.

The fuel assemblies created from the tied rods range to a wide spectrum from 200 to 1150 per reactor. BWRs are avowedly bigger in comparison with PWRs and less compact however their design is less complicated and requires less instrumentation According to the ARIS database, BWRs on average cover 23.5 m in height and 7 m in diameter ending up to 165 m^3 in volume. Typical weights for these reactors are around 900 tons.

Significant projects: Until today, only GE Hitachi has announced the construction of BWRX-300 -a 10th generation BWR capable of producing 300 MWe- within 2024 and has set a time frame for the commercial launch until 2028. Even though, the generated power is more than sufficient for marine use all the hypotheses for BWR will be based on this design

Table 14: BWRs technical characteristics

BWRs technical characteristics					
	Number of investigated reactors:	6	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	290	1638	1224.5
	Plant Efficiency	[°C]	33	40	35.2
	Thermodynamic Cycle	Rankine			
Primary-Secondary Loop specs	Coolant Inlet Temp	[°C]	270	283	277.7
	Coolant Outlet Temp	[°C]	287	290	288.1
	Coolant Operating Temp	[MPa]	7	7.5	7.2
	Steam Max Temp	[°C]	211	290	275.4
	Steam Max Pressure	[MPa]	7	7.5	7.21
Fuel specs	Fuel Material	UO_2, MOX			
	Cladding Material	Zircaloy			
	Fuel Cycle Length	Months	15	24	21.5
	Fuel Enrichment	[%]	3.4	11.4	5.79
	No. Fuel Assemblies		208	1132	700
	Lattice geometry	Square, Triangular			
	Fuel Power Density	[kW/kgU]	24.7	27.3	26
	Discharge Burnup	[MWd/kg]	49.5	60	53.9
Dimensions	Core Height	[mm]	1.2	3.81	3.1
	Core Diameter	[mm]	2.5	7.2	5.28
	Reactor Height	[mm]	19.4	27.6	23.35
	Reactor Inner Diameter	[mm]	4	8.9	6.9
	Reactor Shell Thickness	[mm]	0.13	0.19	0.17
	Weight	[tons]	485	1246	899.3

3.4.1.3 Pressurized Heavy Water Reactors (PHWRs)

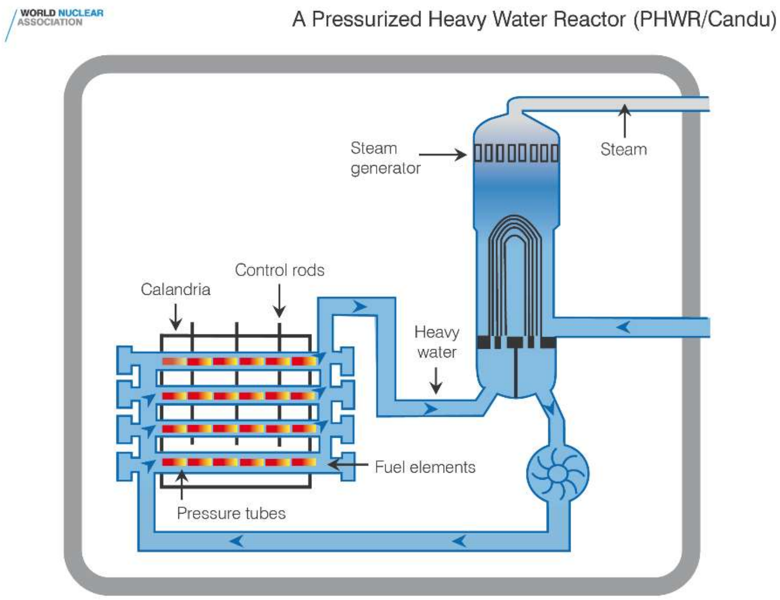

Figure 30: PHWR design (World Nuclear Association, 2024)

This category of reactors is identical to the first one -the PWRs. The only significant difference is found in the working fluid of the primary loop which in these cases is not Light water, but Heavy Water and plays the role of the moderator too (neutron capture is 600 times less likely to happen than in light water). Heavy water in scientific terms is called Deuterium Oxide(D_2O) and contains Deuterium which is an isotope of Hydrogen. PHWRs are subconsciously related to a specific brand of the market named CANDU (Canada Deuterium Uranium) since they were first developed by the Canadian government in partnership with Canadian General Electric. Due to the needs of the Heavy Industry for which they were destined, CANDU's electrical production covers a wide spectrum with the minimum power generated being around 210 GW_e and the maximum reaching 1082 GW_e.

To prevent the boiling of Deuterium, the maximum allowed pressure is around 11MPa and the operating temperature levels of the fluid are kept within 260-310^{o}C (INTERNATIONAL ATOMIC ENERGY AGENCY, 2002). CANDU’s efficiency is comparable with PWRs converting about 28% to 36% of the initial thermal power to electricity.

The basic property of Deuterium that makes it suitable for every nuclear reactor is its low neutron absorption. Consequently, fuel enrichment stops being a necessity and natural Uranium without any fabrication can be used directly. What’s more, heavy water is intended to last for the whole life of the reactor and besides all, it is also reusable. As a result, comparably to LWRs, fuel demands in mined Uranium are reduced to a rate of 20%-30%. PWHRs are usually fueled with rods made up of UO_2 packed in zirconium alloy tubes. The discharge Burnup is significantly lower than on the other occasions ending up only on a 16% on average.

The elements inside the tubes are assembled in a cylindrical form and are in a circular array. The whole core might include from 452 to even 6420 assemblies indicating this way their high producing ability in net power but also their reduced power density (similar power output with more assemblies). The sizes that support such structures are shorter than 6m and 7m in diameter.

The advantage of PWHR however, is that during the refueling of the core, shutdown is not required. The new fuel pushes the old from edge to edge at each tube separately so the reactor can maintain its operation The financial benefit from the enrichment's absence nonetheless is balanced due to the significantly high price of Deuterium. Nonetheless and despite the high level of readiness that exceeds 150 reactor years, typical CANDUs do not meet SMR standards. Thus, without the modification of a PWHR in SMR type, there are no prospects in a marine application. Apart from the size, proliferation is another parameter to consider since the faculty of unloading fuel at any time and the tritium production from neutron absorption by heavy water could encourage weapon development.

Thus, all the data stem from CANDU models and more specifically from CANDU-6 and CANDU-9.

Table 15: PHWRs technical characteristics

PWHRs technical characteristics					
	Number of investigated reactors:	5	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	210	1082	579.2
	Plant Efficiency	[°C]	28	36.5	31.9
	Thermodynamic Cycle	Rankine, Modified Rankine			
Primary-Secondary Loop specs	Coolant Inlet Temp	[°C]	249	275	263.2
	Coolant Outlet Temp	[°C]	285	319	303.5
	Coolant Operating Temp	[MPa]	7	11.6	9.4
	Steam Max Temp	[°C]	251	285	265.5
	Steam Max Pressure	[MPa]	4	7	5.2
Fuel specs	Fuel Material	UO_2,			
	Cladding Material	Zircaloy			
	Fuel Cycle Length	Months			
	Fuel Enrichment	[%]	0.7	3.25	1.55
	No. Fuel Assemblies		452	6240	3925.6
	Lattice geometry	Circular			
	Fuel Power Density	[kW/kgU]	9.24	33	21.6
	Discharge Burnup	[MWd/kg]	7	38	16.3
Dimensions	Core Height	[mm]	3.5	6.28	5.1
	Core Diameter	[mm]	4.5	6.3	5.4
	Reactor Height	[mm]	5	6	5.5
	Reactor Inner Diameter	[mm]	5.9	7.8	7.1
	Reactor Shell Thickness	[mm]	0.025	0.032	0.029
	Weight	[tons]	21.3	265	94

3.4.2 Generation IV

Until this point of the study, there have been presented reactors only from the second and third generation. This sub-chapter investigates 6 concept designs that 100 experts from GIV Forum stand out from 130 potential reactors. What is set as a priority is the modularity of all the selected reactors, having as a target the establishment of these brand-new reactors in many industries apart from electricity generation. The six designs mentioned before are Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), Supercritical Water-cooled Reactor (SCWR), Sodium-cooled Fast Reactor (SFR) and Very High-Temperature Reactor (VHTR).

3.4.2.1 Very High-Temperature Reactors (VHTRs)

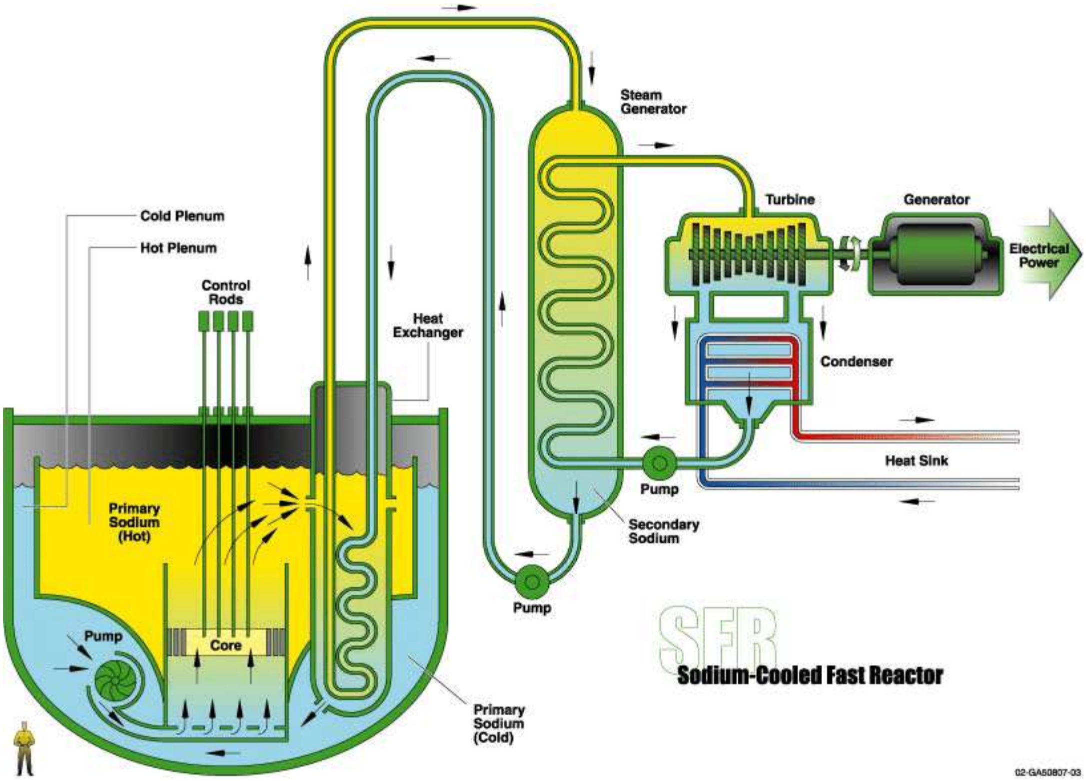

Figure 31: VHTR design (GenIV International Forum, 2020)

Very High-Temperature Reactors are thought to be an evolution of High Gas-Cooled Temperature Reactors (HGTR) which even though they count more than 60 years in commerce, were overshaded by LWRs (Macedo, et al., 2024). It is important to highlight that until 2024, VHTRs are the only IV reactors under operation. Their key features are that they are graphite-moderated, helium-cooled and of course as their name implies, their design is specifically developed to endure high temperatures.

VHTRs are thermal neutron spectrum reactors and their basic distinction from their predecessors is the increased outlet temperature of the coolant that typically approaches 750 °C -850°C while the efficiency at these levels reaches 47%. The long-term target in the outlet temperature however is set beyond 1000 °C (Sahin & Sahin, 2018) and this rise is generally desirable due to the potential increase in the reactor's efficiency. However, such high temperatures would require the use of new structural materials not available today (Guidez, 2023). The pressure levels of the coolant (helium basically) are set at 7MPa at maximum while the steam pressure could exceed 17 MPa. Another parameter to take into consideration is the potential implementation of Brayton Cycle with such temperatures instead of Rankine.

As for the fuel options, VHTRs usually are equipped with enriched uranium of less than 20% (typically between 10%-15%) within a graphite matrix which acts as a moderator. The fuel configuration however follows the two models analysed in advance: pebble bed and prismatic block. In both options nonetheless and due to the high temperatures of the reactor, TRISO form is superior and safer to use than the fuel pellets. TRISO coating could bear extreme temperatures such as 1600°C without failing. Alternative fuels could be Plutonium, MOX, UCO or even Thorium with Uranium. The fuel inside the core will need a renewal every 1 or 2 years.

VHTRs operate under multiple fuel cycles but is more than common to deploy the open fuel cycle thus after the combustion becomes waste. In addition, due to the high temperatures, the burnup index of VHTRs is significantly high (approx. 115 MWd/kg and as a result, most of the studies foresee an increase in long-lived waste which makes it a less appealing option (Emmen, 2022).

VHTRs size are expected to be a challenge since the height of the whole reactor will exceed 20m and the diameter will be 6 m on average. Although these figures seem competitive, VHTRs have impressively low power density which leads to larger units for the same power output compared to other reactors and bigger tanks for increased waste. This theory is proved in the Chinese occasion where the waste tanks are enormous (the 100MWe prototype needs the same tank with a 1600Mwe PWR).

Table 16: VHTRs technical characteristics

VHTRs technical characteristics					
	Number of investigated reactors:	5	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	165	272	212
	Plant Efficiency	[°C]	40	47	43.2
	Thermodynamic Cycle	Brayton, Rankine			
Primary-Secondary Loop specs	Coolant Inlet Temp	[°C]	250	587	346.8
	Coolant Outlet Temp	[°C]	750	850	770
	Coolant Operating Temp	[MPa]	6	7	6.5
	Steam Max Temp	[°C]	540	566	549
	Steam Max Pressure	[MPa]	12	17.3	15.1
Fuel specs	Fuel Material	UO_2, MOX, UCO			
	Cladding Material	TRISO			
	Fuel Cycle Length	Months	10	24	17.5
	Fuel Enrichment	[%]	8.5	15.5	13.45
	No. Fuel Assemblies		90	1020	555
	Lattice geometry	Spherical, Hexagonal			
	Fuel Power Density	[kW/kgU]	85.7	85.7	85.7

	Discharge Burnup	[MWd/kg]	92	165	117.2
Dimensions	Core Height	[mm]	7.9	11	8.7
	Core Diameter	[mm]	0.4	4.8	2.9
	Reactor Height	[mm]	22	24	23
	Reactor Inner Diameter	[mm]	3	8.5	6.1
	Reactor Shell Thickness	[mm]			
	Weight	[tons]	880	880	880

3.4.2.2 Sodium-Cooled Fast Reactors (SFRs)

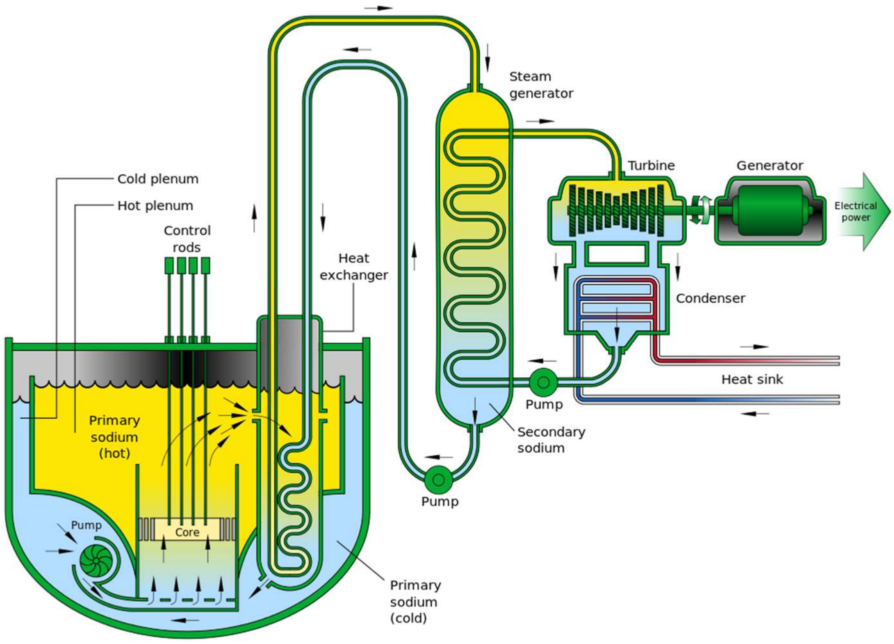

Figure 32: SFR design (GenIV International Forum, 2020)

In the dawning of nuclear reactors, Uranium sources were thought to be limited (Abram & Ion, 2008), so the need for Uranium's optimum allocation was eager. This purpose could be served to a satisfying extent by fast neutron reactors where the efficient fission of other Uranium isotopes seemed a sustainable process. Nonetheless, the extreme temperature conditions in combination with the increased heat amounts requiring transfer, shed some light on other coolants with improved properties. Among many choices, sodium was believed to be the most suitable fluid since it has a low melting point(98ºC) and performs to a sufficient extent with the other reactor materials. This property of sodium favored the development of SFRs.

Material properties allow outlet temperatures around 500ºC-550ºC and efficiency can be up to 38% which is reasonably lower than other IV designs (Emmen, 2022). Except for the low temperatures, SFRs work at extremely low pressures and as a result, the required volume of the coolant is minor. As with VHTRs, the Brayton Cycle could be another pathway for the power conversion problem.

In general, two types as far as the position of the first heat exchanger have been developed. They are usually mentioned as loop type and pool type and their main difference lies in how the coolant is circulated. Loop-type reactors use separate primary coolant loops to circulate coolant through the reactor core, while pool-type reactors submerge the entire core within a single pool of primary coolant. The output of the SFRs depends on the above configurations and ranges from (Koizumi, Okawa, & Mori, 2021):

- 600MWe-1500MWe for loop type with MOX fuel (U-Pu), Large size
- 300MWe-1500MWe for loop type with oxide or fuel metal, Intermediate size
- **300MWe-1500MWe**, SMR size

Fuel options for SFRs are the MOX and the metal alloys (Lineberry & Allen, 2002). In addition, SFRs could potentially employ a closed fuel cycle since they can be classed as *breeders,* meaning that they can produce fissile material from fertile material on their own on the hypothesis that a reprocessing procedure will be established. The burnup ratios of SFRs are estimated to vary between 40 MWd/kg and 100 MWd/kg. Therefore, a rise in high-level waste is expected. And on these grounds the implementation of a closed fuel cycle is imperative.

Reactor cores on SFR cases are smaller than in Water-cooled occasions. In more detail, core height and diameter will be around 1.3m and 1.7m accordingly ending up with a more compact construction. These cores consist of the fuel assemblies whose array can be either hexagonal or triangular. The total weight is anticipated to be meaningfully low with less than 100 tons for the whole unit.

Table 17: SFRs technical characteristics

SFRs technical characteristics					
	Number of investigated reactors:	10	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	10	1140	507.8
	Plant Efficiency	[ºC]	33.3	41.7	38.9
	Thermodynamic Cycle	Brayton, Rankine			
Primary-Secondary Loop specs	Coolant Inlet Temp	[ºC]	330	410	374
	Coolant Outlet Temp	[ºC]	475	550	522.4
	Coolant Operating Temp	[MPa]	0.05	0.6	0.23
	Steam Max Temp	[ºC]	450	510	471
	Steam Max Pressure	[MPa]	10.8	18	15.3
Fuel specs	Fuel Material	UO_2, MOX, Nitride, U-Pu, u-Zr			
	Cladding Material	HT-9			
	Fuel Cycle Length	Months	12	360	88
	Fuel Enrichment	[%]	17	26	21
	No. Fuel Assemblies		18	177	97.5
	Lattice geometry	Hexagonal			
	Fuel Power Density	[kW/kgU]	3.4	3.4	3.4
	Discharge Burnup	[MWd/kg]	34	100	67
Dimensions	Core Height	[mm]	0.55	2.5	1.3
	Core Diameter	[mm]	0.88	3.4	1.74
	Reactor Height	[mm]			
	Reactor Inner Diameter	[mm]	3.4	3.4	3.4
	Reactor Shell Thickness	[mm]	0.03	0.03	0.03
	Weight	[tons]	86	86	86

3.4.2.3 Gas-Cooled Fast Reactors (GFRs)

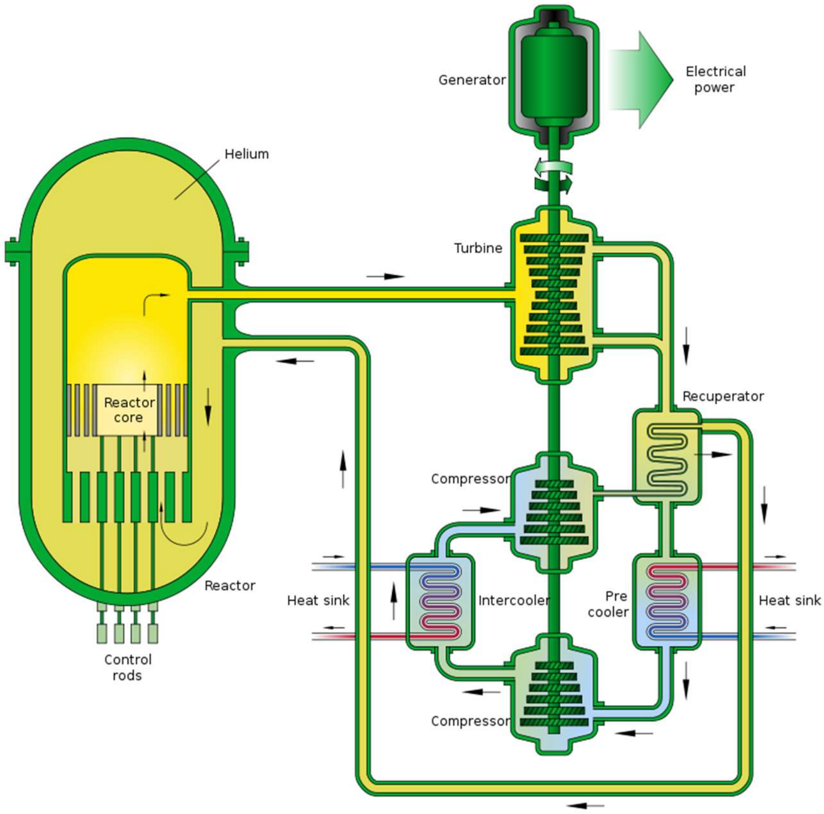

Figure 33: GFR design (GEN IV Forum, 2022)

Gas-cooled fast reactors belong to the fourth generation too and could be characterized as a hybrid of the SFRs and the VHTRs designs. This aspect could be attributed to their fuel recycling processes which are very similar to SFRs' operation (helium-cooled loop under a closed fuel cycle-reprocessing of the fuel) but at the same time these units allow quite high temperatures leading to higher thermal efficiencies just like VHTRs. GFRs operate in the fast neutron spectrum as the rest of the Gen-IV reactors and thus there is no need for a graphite moderator. As stated above, in the first cycle helium in gas form is circulated from the core to the heat exchanger while in the second loop the coolant which can be either helium or even sCO_2 is in direct connection with a gas turbine. The core will be of breeding technology meaning it will be capable of converting fissile material from fertile on its own.

GFRs are designed to maintain coolant outlet temperatures within 400°C and 850 °C, while the fuel temperatures inside the core are estimated to exceed 1000°C in normal operation. According to the GIF Forum again, technologies considering the materials and the structures of these reactors should rely on the VHTRs developments. The pressure levels due to the use of gas instead of water are thought to be low comparable to the LWR designs, approximately at 7MPa (GEN IV Forum, 2022). The power conversion could be achieved either by the indirect Rankine cycle or on the Brayton cycle and in this case, there are indications for efficiency rates over 50%.

Candidates for the fuel selection vary and enriched Uranium, Plutonium, MOX and thorium-based fuels are all acceptable solutions. The fuel will be in the traditional form of pellets/pins while in the future there are thoughts for more radical concepts with ceramic coating. The fuel elements then will be embedded in a ceramic/metallic matrix (Stainsby, et al., 2010). The burn-up value will be 123MWd/kg and the design life for such reactors is estimated at 60 years.

The dimensions of GFRs remain undefined part from the EM_2 model. The vertical dimension of EM_2 will be 11.5m and its diameter close to 5m. GFR is the only concept from the Gen-IV reactors that has never been implemented before and as a result, the readiness level is zero.

Table 18: GFRs technical characteristics

GFRs technical characteristics					
	Number of investigated reactors:	3	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	265	1000	632.5
	Plant Efficiency	[ºC]	33	53	43.1
	Thermodynamic Cycle	Rankine, Combined			
Primary-Secondary Loop specs	Coolant Inlet Temp	[ºC]	200	550	336
	Coolant Outlet Temp	[ºC]	400	850	593
	Coolant Operating Temp	[MPa]	13	15	14
	Steam Max Temp	[ºC]			
	Steam Max Pressure	[MPa]			
Fuel specs	Fuel Material	UO_2, MOX, UC			
	Cladding Material	SiC, Stainless steel			
	Fuel Cycle Length	Months	360	360	360
	Fuel Enrichment	[%]	7.7	18	12.8
	No. Fuel Assemblies		85	1073	579
	Lattice geometry	Hexagonal			
	Fuel Power Density	[kW/kgU]	48	48	48
	Discharge Burnup	[MWd/kg]	100	143	121.5
Dimensions	Core Height	[mm]	3.7	3.7	3.7
	Core Diameter	[mm]	4.6	4.6	4.6
	Reactor Height	[mm]	11.5	11.5	11.5
	Reactor Inner Diameter	[mm]	4.8	4.8	4.8
	Reactor Shell Thickness	[mm]			
	Weight	[tons]	301	301	301

3.4.2.4 Lead-cooled Fast Reactors (LFRs)

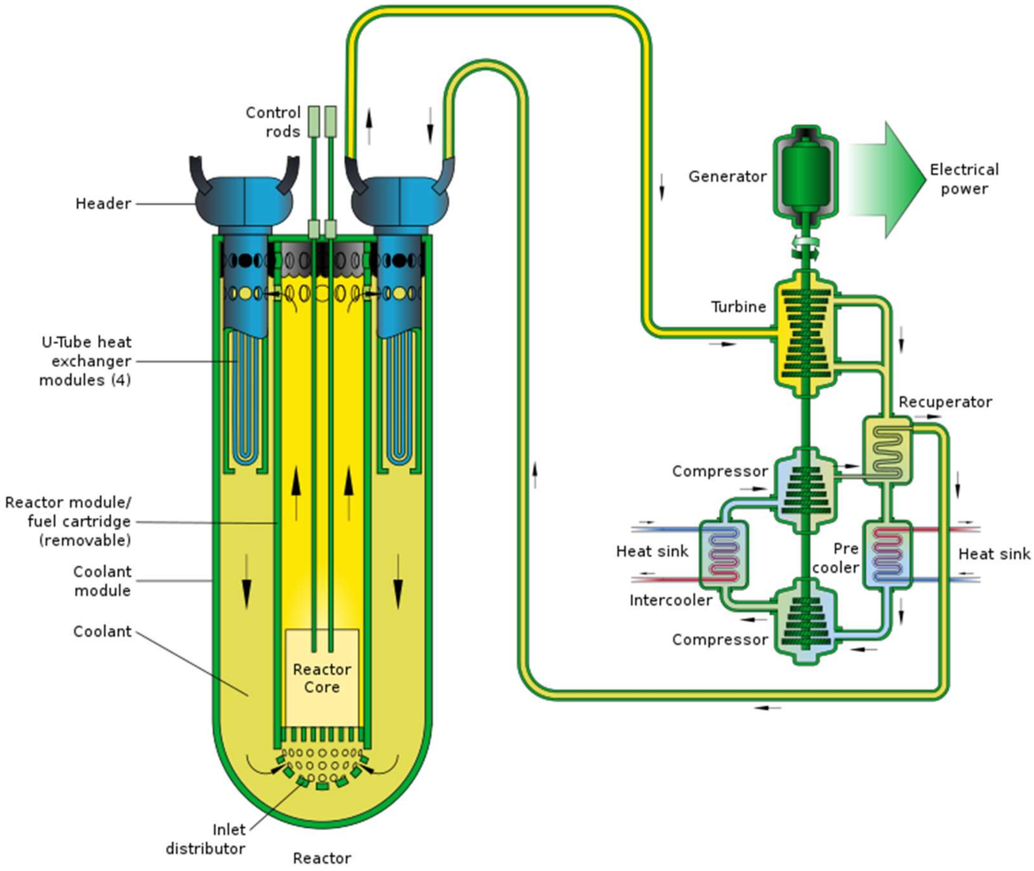

Figure 34: LFR design (GenIV International Forum, 2020)

Lead-cooled Fast Reactors are yet another solution under development with many similarities with Sodium-cooled Fast Reactors. They both operate on the fast neutron spectrum and are cooled from liquid metals under identical conditions (temperature and pressure). The coolant options that are currently under investigation are molten Lead and Lead-Bismuth Eutectic (LBE). Lead displays certain benefits over Sodium such as the high boiling point (1743°C) which makes it a safer element to use and its comparable inertia with air and water. A major drawback of the two coolants however is their corrosive nature which mandates significant progress in material engineering. Unit sizes differ and cover a wide spectrum of power outputs from 25MWe to 630MWe.

According to WNA, coolant outlet temperatures vary between 500^0C-800^0C, while liquid metal cooling happens at atmospheric pressure by natural convection. Admittedly, this working environment in combination with the high boiling point of Lead eliminates the chances of an accident due to overpressure and creates substantial potential in terms of safety and complexity. Conversely, a disadvantage that both coolants share is their high density which disturbs the smooth flow within the pumps and as a result more power and more frequent maintenance tasks are required (Tanju, 2019). On the one hand, Lead can be found in abundance, has great properties as far as its neutron absorbance is concerned and extremely safe, but its increased melting point (330°C) complicates the required anti-freezing systems and procedures. On the other hand, Bismuth's abundance on the planet does not seem promising for sufficient coverage hence it cannot be qualified as a design for further investigation (Abram & Ion, 2008).

LFRs can utilize a variety of fuel types, including enriched uranium, plutonium, or thorium-based fuels. Generally, LFRs are known for their ability to support a closed fuel cycle by burning reused fuel and especially in the case of thorium LFRs can operate also as breeders. The forms of the fuel can be either oxide or nitride or even metallic alloy for higher burn-ups-about 86 MWd/kg on average. The enrichment of the fuel will be preserved within the reasonable values of 10% to 20%.

Fuel is stored in fuel assemblies that are placed in hexagonal lattices. The number of fuel assemblies per core is estimated within 60 and 170. The fuel rods inside the assemblies use HT-9(steel) for cladding purposes. LFRs are believed to sustain their operation for a long time, perhaps for more than 30 years. The final dimensions of LFRs are thought to be very adjustable with the first designs indicating a reactor height of less than 8m. The maximum diameter will be between 4 and 6 meters and the total weight of the construction will not go above 50 tons.

LFRs were mostly deployed from the Russian Navy for their submarine fleet with the LBE as coolant counting more than 80 reactor years. Certain failures regarding coolant freezing affected their expansion however the superiority of Lead as a coolant makes LFRs one of the most appealing options.

Table 19: LFRs technical characteristics

LFRs technical characteristics					
	Number of investigated reactors:	12	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	25	630	244
	Plant Efficiency	[%]	39.3	48.4	42.4
	Thermodynamic Cycle	Rankine, sCO_2 Brayton			
Primary-Secondary Loop specs	Coolant Inlet Temp	[ºC]	360	420	368
	Coolant Outlet Temp	[ºC]	390	650	493
	Coolant Operating Temp	[MPa]	0.01	6.7	0.8
	Steam Max Temp	[ºC]	230	530	427.5
	Steam Max Pressure	[MPa]	4	18	13.7
Fuel specs	Fuel Material	UO_2, MOX, U Nitride, U-TRU-Zr, Pu,N			
	Cladding Material	HT-9,			
	Fuel Cycle Length	Months	16	360	123
	Fuel Enrichment	[%]	11.8	19.7	16.1
	No. Fuel Assemblies		61	171	105.6
	Lattice geometry	Hexagonal			
	Fuel Power Density	[kW/kgU]	42.5	42.5	42.5
	Discharge Burnup	[MWd/kg]	60	100	86.6
Dimensions	Core Height	[mm]	0.85	1.3	1.1
	Core Diameter	[mm]	1.9	2.3	2.1
	Reactor Height	[mm]	6	7.5	6.7
	Reactor Inner Diameter	[mm]	3.9	6	4.9
	Reactor Shell Thickness	[mm]			
	Weight	[tons]	35	50	42.5

3.4.2.5 Supercritical Water-cooled Reactors (SCWRs)

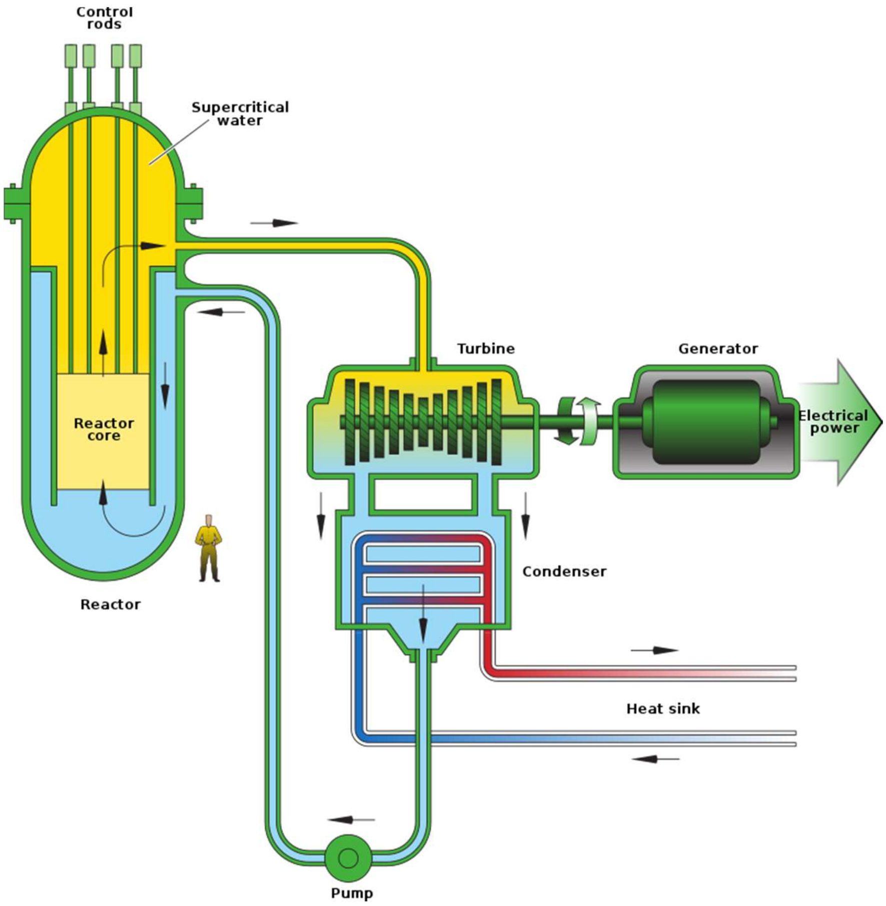

Figure 35: SCWR design (GenIV International Forum, 2020)

Supercritical Water-Cooled Reactors are high-temperature reactors that use supercritical water (light or heavy) as coolant above its critical point (374°C, 22.1MPa). When a fluid is defined as supercritical, it means that liquid and gas phases coexist at the same time enhancing in this way the heat transfer efficiency. Of all the Gen-IV reactors under research, SCWR is the unique design that can be developed from the existing water-cooled reactors (LWRs). SCWRs can operate both in the thermal and fast neutron spectrum based on the core design which can be either a pressure vessel or a pressure tube. SCWR's main advantage is the system's simplicity since the supercritical water is converted directly to supercritical steam within the same cycle and thus there is no need for a secondary loop. The power output of SCWRs will be approximately from 1000 GW_e to 1620 GW_e

The supercritical water is typically maintained at 25MPa pressure and has an outlet temperature of 500°C-550°C. There are indications that the efficiency at SCWRs could reach 44% instead of the 30%-35% that typical LWRs record. On account of a single loop in SCWR design, coolant pumps and heat exchangers are no longer necessary, offering that way reduced capital costs for the plants. In addition, light or heavy water acts as a moderator.

SCWRs can employ both open and closed fuel cycles by using uranium, plutonium, or thorium-based fuels again. In the case of the open cycle, the fuel is enriched (6%-9%) instead of the closed cycle where recycled fuel is utilized. Apart from oxides, several other fuel forms such as carbides, nitrides and dicarbides are thoroughly assessed (Naidin, et al., 2000). The average discharge burnup of SCWRs is expected to be around 50 MWd/kg.

Fuel assemblies are expected to be numerous from 200 until 1400 and they are designed to last between 1 and 2 years. Their geometry inside the core will be on a square lattice and the cladding inside an assembly will be made from Stainless Steel. This type of reactor is going to be similar to PWRs as far as the size is concerned. The height and the diameter of the reactor vessel will approach 15m and 5m on average respectively.

Three SCWR concepts are under construction for demonstration purposes but all of them have an extreme power range (more than 100MW):

- CSR1000: 1000Mwe-China
- HP-LWR: 1046Mwe-EU
- JSCWR: 1700Mwe-Japan

Table 20: SCWRs technical characteristics

SWCRs technical characteristics					
	Number of investigated reactors:	3	Min	Max	Avg
Key specs	Power Output, net	$[GW_e]$	1000	1620	1310
	Plant Efficiency	[°C]	43.5	44	43.6
	Thermodynamic Cycle	Rankine			
Primary-Secondary Loop specs	Coolant Inlet Temp	[°C]	280	310	293
	Coolant Outlet Temp	[°C]	500	560	520
	Coolant Operating Temp	[MPa]	25	25	25
	Steam Max Temp	[°C]	500	560	520
	Steam Max Pressure	[MPa]	24	25	24.7
Fuel specs	Fuel Material	UO_2			
	Cladding Material	Stainless steel			
	Fuel Cycle Length	Months	10.2	18	13
	Fuel Enrichment	[%]	6.2	9	7.4
	No. Fuel Assemblies		192	1404	798
	Lattice geometry	Square			
	Fuel Power Density	[kW/kgU]	26.3	48.1	37.2
	Discharge Burnup	[MWd/kg]	45	60	50
Dimensions	Core Height	[mm]	4.2	4.2	4.2
	Core Diameter	[mm]	3.3	3.5	3.4
	Reactor Height	[mm]	13.6	15.9	14.9
	Reactor Inner Diameter	[mm]	4.5	4.8	4.7
	Reactor Shell Thickness	[mm]	0.44	0.44	0.44
	Weight	[tons]	656	656	656

3.4.2.6 Molten Salt Reactors (MSRs)

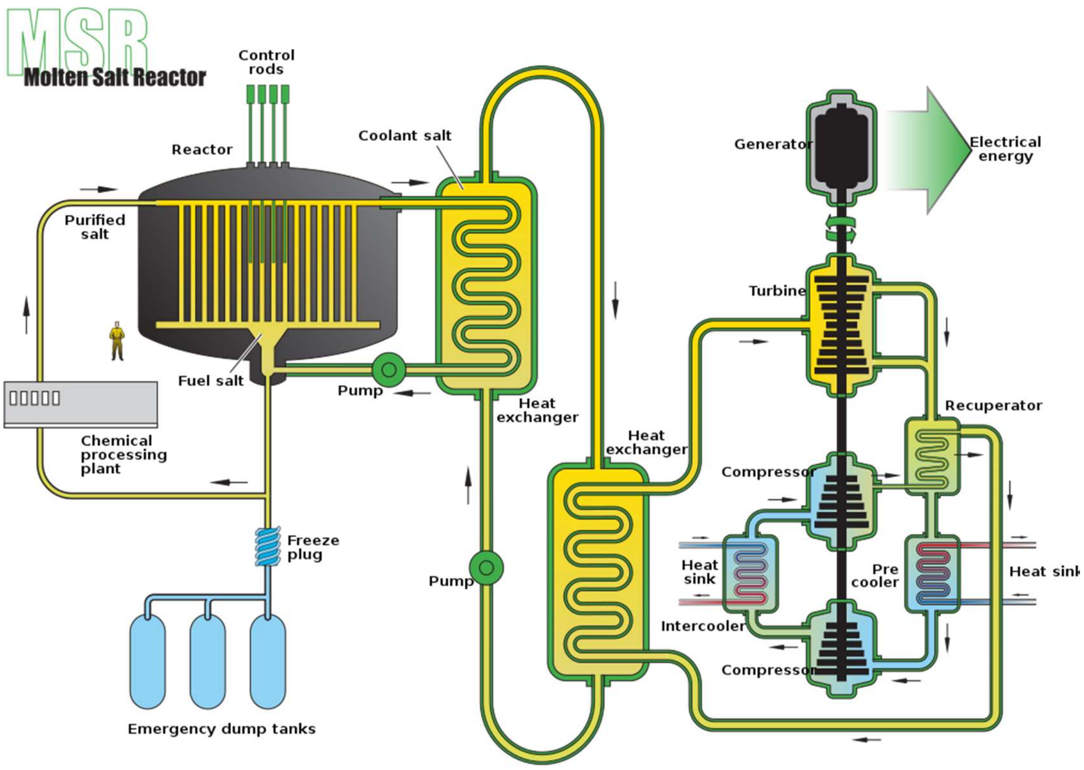

Figure 36: MSR design (GenIV International Forum, 2020)

Molten Salt Reactor is a broad term that includes several categories of reactors. Although they might have some discrepancies, all the reactors that are described as MSR use salt either as a coolant or as fuel, or even for both (Arostegui & Holt, 2019). The liquid salt is typically fluoride or chloride of lithium, beryllium, sodium, or potassium. The origins of the MSR concept are thought to be around 1950, when Bettis and Briant attempted unsuccessfully to develop a nuclear-powered (Emblemsvåg, 2021). With a good premonition for their concept however, they persisted in their efforts and in the mid-60s they managed to operate a 7MWth Thorium Molten Salt Reactor for more than 4 years. These demonstration MSR concepts were designed on the thermal neutron spectrum with the use of graphite as a moderator. Nonetheless, the modern interest is focused on fast spectrum MSRs where downsizing of the reactor both in output and in size is believed to be essential.

According to the Canadian Roadmap on SMRs, there are 3 distinct MSR categories:

- Liquid fuel where the fissile fuel is dissolved in salt and acts as a coolant too
- Liquid fuel in tubes separately from liquid coolant
- Solid fuel, separately from liquid coolant.

It has to be noted that the short-term applications are oriented onto the solid fuel option which will act as a training stage for the rest two designs. Despite the low readiness level, MSR is thought to be the most ambitious design from the GEN-IV concepts.

Outlet temperatures of the salt vary from 500ºC to 670ºC (1000ºC is the target) and still due to the nature of the fluid, the pressure is kept near to atmospheric (less than 1MPa). In this way, the idea of an explosive environment no longer exists, and the low pressure is one of the reasons that MSRs are believed to be impressively safe. The heat generated from the first cycle at these temperatures will be transferred to the secondary loop where it will have great synergy with the Brayton cycle. Although the efficiency will be reduced due to the existence of a secondary loop, radiotoxic Thorium will be isolated from the turbine and even with these conditions, there are indications for achievable efficiencies of more than 45%.

MSRs theoretically can use a variety of fuels such as Uranium, Plutonium, and MOX. The most outstanding fact however is that they can operate as breeders with a Thorium salt blanket only with a small portion of fissile Uranium or Plutonium for initiating the reaction. Moreover, MSRs are considered suitable for deploying a closed fuel cycle where the fuel will be reprocessed on-site minimizing in this way both the radiotoxic waste (95% of the initial fuel) and the proliferation risk. It is worth noting that even used fuels from existing PWRs will be reusable. Burnup is roughly estimated between a wide range of 30 to 500 MWd/kg (Houtkoop, 2022). The most astonishing factor nevertheless is the length of the fuel cycle which can get as high as 60 years.

What has to be pointed out on the MSRs is the idea of the Freeze Plug which adds another obstacle to a potential fuel leakage. The Freeze plug is a mechanism made from solid fuel salt and if the liquid salt reaches an unusually high temperature, then this melt providing passive safety by cooling the mixture into dump tanks which are sealed afterwards.

MSRs as implied before will not use the common fuel rods under assembly. The fuel will be formed under the TRISO standards for extra protection.

One of the challenges that engineers have to address before MSRs find commercial application is undeniably the material's performance. Molten salts are extremely corrosive, radioactive and in combination with the increased temperatures, constraining them inside the loop is of first priority. As with the SFRs, MSRs have another drawback during the start-ups and the shutdowns, where external cooling will be imperative for maintaining the salt in liquid phase.

Table 21: MSRs technical characteristics

MSRs technical characteristics					
	Number of investigated reactors:	8	Min	Max	Avg
Key specs	Power Output, net	[GW_e]	185	1500	533.7
	Plant Efficiency	[ºC]	42.5	46.4	44.4
	Thermodynamic Cycle	Rankine, Brayton			
Primary-Secondary Loop specs	Coolant Inlet Temp	[ºC]	500	670	596
	Coolant Outlet Temp	[ºC]	650	750	697
	Coolant Operating Temp	[MPa]	0.1	1	0.47
	Steam Max Temp	[ºC]	538	585	148.5
	Steam Max Pressure	[MPa]	19	25.5	22.2
Fuel specs	Fuel Material	UF_4, UCO molten salt, ThF_4, LiF-(U,Pu)-F_3			
	Cladding Material	TRISO			
	Fuel Cycle Length	Months	6	720	214.5
	Fuel Enrichment	[%]	2	19.8	12.3

	No. Fuel Assemblies		-	-	-
	Lattice geometry	Spherical, Plate			
	Fuel Power Density	[kW/kgU]	370	370	370
	Discharge Burnup	[MWd/kg]	29	509	207.4
Dimensions	Core Height	[mm]	2.5	5.5	3.9
	Core Diameter	[mm]	2.1	4.9	3.8
	Reactor Height	[mm]	5	11	7.9
	Reactor Inner Diameter	[mm]	3.3	7.8	4.5
	Reactor Shell Thickness	[mm]	0.05	0.06	0.05
	Weight	[tons]	22.5	343	131.5

4. POWER CONVERSION

Until this point of the paper, most of the research focuses more on the characteristics of nuclear reactors and less on the method of converting the produced energy to a desired form. Nonetheless, the key point for the successful implementation of this idea is undoubtedly the mechanics beyond the conversion of the generated energy. It could be argued that the produced heat from the reactor must be transformed either to mechanical energy or to electricity for the adequate coverage of vessels' needs.

In terms of power conversion systems for nuclear-powered vessels, several options are available. All of the available options are based on thermodynamic cycles with Rankine Cycle and Brayton Cycle being the most developed and guaranteed solutions so far. In the case of the Brayton cycle nevertheless, two variations will be examined since both the Closed-loop Brayton cycle (CBC) and the Open-loop Brayton cycle (OBC) seem promising.

All in all, the choice for the most suitable power conversion system should not be approached as an unifactorial problem since the decision parameters vary. Under no circumstances should the power conversion method be decided by considering only the criterion of the cycle efficiency. Instead of this monocular approach, several other factors such as safety, economic, maintenance, compatibility and complexity issues must be taken into serious consideration.

4.1 Rankine cycle

The Rankine cycle is a thermodynamic cycle that converts heat to power with the use of water. The primary source of energy will stem from the nuclear reactor in thermal form and it will be transformed into electricity. Responsible for the transfer of heat throughout the cycle is the water that during the cycle’s operation is converted to steam. What should also be noted is that in comparison with the Brayton cycle, most of the existing nuclear plants are operating under the Rankine cycle (Ahlgren, 1994) and hence the technology readiness level is way more advanced than any other.

4.1.1 Thermodynamic background

Rankine cycle consists of four processes and the working fluid is the water. Specifically, the processes are listed below:

- During the first process (1→2), an isentropic compression takes place that pumps the working fluid into the boiler. The working fluid during this process, remains in its liquid phase and hence the input energy is significantly lower.
- During the second process (2→3), heat is added under constant pressure to the high-pressure working fluid in the boiler. As a result, the liquid thanks to the external heat source is transformed into dry saturated vapor.

- During the third process (3→4), the produced dry saturated vapor is driven from the boiler into the turbine where it expands to the stationary and rotating blades converting part of the initial thermal energy to mechanical power.
- During the last process (4→1), the low-pressure steam is cooled at the condenser under constant pressure transforming it to low-pressure liquid ready to restart the cycle.

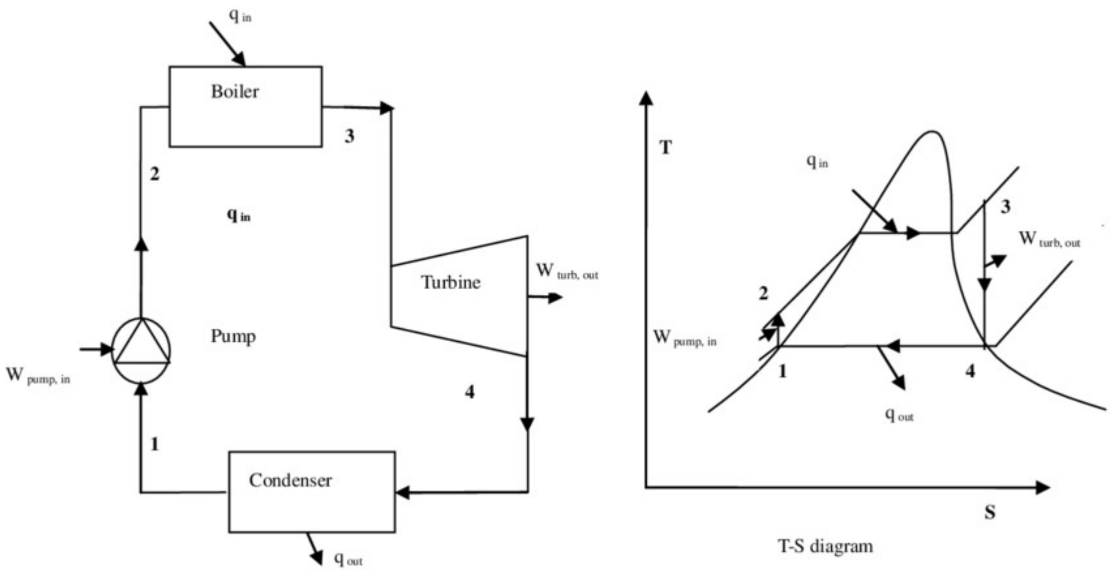

Figure 37: Rankine Cycle

It is imperative to clarify that power production and consumption happen under constant entropy, while heat transfer and rejections take place under constant pressure.

4.1.2 Working fluid

It could be argued from the above analysis, that the whole cycle's operation is based on the phase and the properties of the working fluid. The choice of the working fluid for the Rankine cycle is dependent on a variety of factors: operating temperature range, fluid costs, environmental impact etc. However, in most of the cases water is the recommended fluid because it meets most of the terms described before. Despite the dominance of the water there are several other fluids with quite a few applications such as: butane, pentane, hexane, silicon oil and ammonia (Dumont & Lemort, 2022). Especially in the field of aerospace where there are many cases of nuclear plants working according to the Rankine cycle, water is replaced by metal elements like Potassium and Mercury.

What is crucial to point out is that during the operation of the Rankine cycle, the fluid works within a great range of temperatures and pressures. Actually, the changes in the working fluid's phase are responsible for the successful power conversion and in the case of water, the possible phases under different external conditions are depicted below:

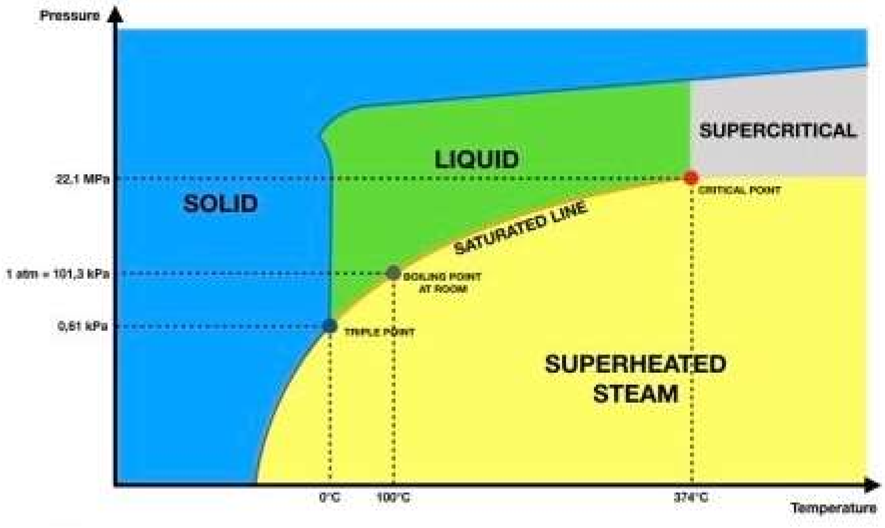

Figure 38: Phases of water (Palo, 2018)

Nevertheless, the essential requirement for the proper operation of the cycle is the elimination of water vapor which can be proved extremely harmful for the moving parts of the mechanism during their violent rehydration. Thus, an appropriate solution suggested for the upper phenomenon is the maximization of the inlet pressure and temperature which will increase the cycle's efficiency in parallel.

In general, some important properties of the working fluid in the Rankine cycle refer to:

- Density: mass/unit volume
- Viscosity: resistance to flow
- Specific heat: amount of heat energy required in order to raise T by one degree
- Heat of vaporization
- Thermal conductivity: ability of fluid to transfer heat

4.1.3 Rankine efficiency

Some normal values of Rankine's cycle efficiency for low temperatures (300^{o}C to 450^{o}C) range between 30-38% according to Fleming et al. These cases refer to Light-Water-Reactors (LWR) which are the most widespread nuclear reactors in the industry, in comparison with coal-burning plants which can achieve efficiency up to 45% due to higher boiler temperatures. Also, the efficiency rate of the Rankine cycle can be increased with some advanced configurations such as using reheating and regeneration as it was presented earlier. Assuming that Generation IV reactors work under the thermodynamic analysis of a regenerative Rankine Cycle with a single reheat, a hypothesis of an efficiency rate of around 35%-45% could be achieved. These rates are verified from the ARIS platform where the specifications of all the Gen-IV reactors are gathered.

4.2 Brayton cycle

The Brayton Cycle is another power conversion method under which a nuclear reactor can deliver the energy of the fuel into desirable forms. The substantive difference from the Rankine Cycle is the working fluid which in this occasion -contrary to steam- is typically a gas that remains at the same phase during the whole cycle. What's more about the Brayton Cycle, is that there are two prevailing concepts: the closed Brayton Cycle and the Open Brayton Cycle. An outstanding fact about Gas Turbines working under the Brayton Cycle is the perceptible decrease in the plant size compared to the Steam Cycle.

4.2.1 Closed Brayton cycle (CBC)

The closed Brayton Cycle appears to be one of the most promising options to replace the Rankine cycle as far as nuclear engineering is concerned. Among a variety of different criteria, the most significant advantage of CBC is the higher thermal efficiency that can admittedly be achieved. This privilege stems from the working fluid, its higher heat capacity and the higher inlet temperatures that can handle, which consequently is translated to more efficient heat transfer than the steam.

4.2.1.1 Thermodynamic background

Even though Rankine and Closed Brayton Cycles differ at crucial points such as the working fluid, they share some commons behind the thermodynamics processes. As exactly in the Rankine cycle, the Closed Brayton Cycle is a multistage procedure of 4 procedures at which:

- During the first process (1→2), an isentropic compression takes place that pumps the working fluid into the heat exchanger instead of the boiler.
- During the second process (2→3), heat is added under constant pressure to the working fluid from the heat exchanger.
- During the third process (3→4), the fluid is driven from the heat exchanger into the turbine where it expands to the stationary and rotating blades converting part of the initial thermal energy to mechanical power.
- During the last process (4→1), the expanded fluid passes through another heat exchanger in order to cool down.

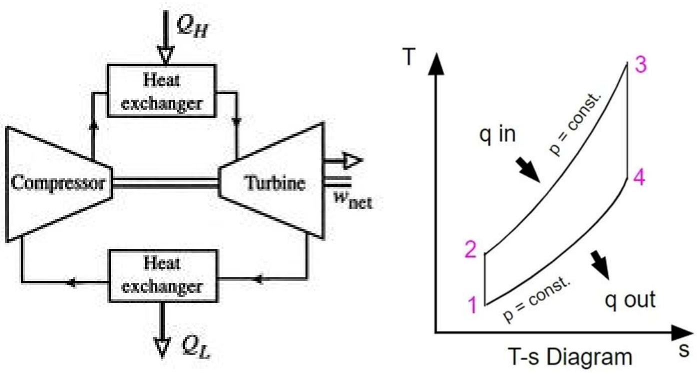

Figure 39: Closed Brayton Cycle

4.2.1.2 Working fluid

In the case of CBC, the selection of the working fluid is possibly the most substantial question of the design. Typically, the working fluid of the CBC is gas and the two popular options are Helium (He) and supercritical carbon dioxide (sCO_2). There is no clear answer to the dilemma of what is the best option since the data are not sufficient.

Helium is an already proven solution for nuclear reactors and its certain properties have made it a prosperous option. First of all, Helium is an inert element which means that even in its gaseous form it cannot be corrosive and damaging for the materials. What separates Helium however is its superior behavior under hot temperature environments where the conditions might exceed 800^{o}C.

Supercritical CO_2 is thought to be an innovative idea in power conversion systems and has gathered the interest of many researchers over the past few years. Its performance is identical to Helium and there are certain indications that at the same temperatures as Helium and despite the cost of higher pressure needed (8MPa to 20MPa) it has possibly greater efficiency (Bae, Lee, Ahn, & Lee, 2013). Another factor that makes sCO_2 a serious competitor of Helium is the downsizing of the whole plant since the size of such turbine will be multiple times smaller from Helium and Steam Turbines. (Gang Zhao, 2018).

4.2.2 Open Brayton cycle (OBC)

4.2.2.1 Thermodynamic background

Contrary to the previous systems, the Open Brayton Cycle which is the second most widespread variation of the Brayton Cycle is an open loop cycle. The characteristic of the open loop described the circulation of the working medium which in this case does not actually circulates. The process of this cycle follows the same steps with the closed Brayton cycle, with the last step of the condensing process to be omitted. The configuration of the Open Brayton Cycle is illustrated in the graph below:

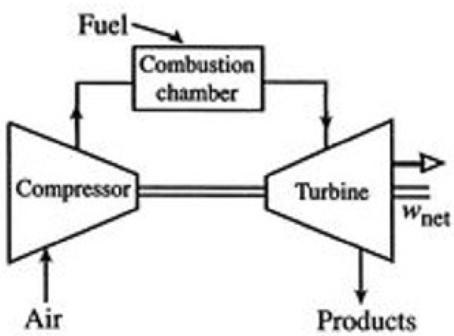

Figure 40: Open Brayton Cycle

4.2.2.2 Working fluid

As the name implies the medium is continuously renewed because the medium is the atmospheric air itself. The air is drawn from the environment near the turbine and exhausted with the combustion products. The rejection of the used air along with the combustion products makes the condenser unnecessary, reducing in this way both the size, the weight and the complexity of the plant (Bahman, 2014).

However, due to the working environment of the vessels, there is no need to surpass the condensing step due to a potential lack of a cooling medium. In addition, the atmospheric air above salt waters is not appropriate due to the high portion of moisture and salt inside the air's composition. To overcome possible problems from air-prohibitive properties, extra modifications in the inlet of the system are imperative leading to a further decrease in the cycle's efficiency (Houtkoop, 2022). For these reasons, open Brayton cycle turns to be less favorable for offshore applications.

4.3 Combined cycle gas turbine (CCGT)

4.3.1 Thermodynamic background

An innovative idea for Power Conversion Systems mixes the two fundamental thermodynamic cycles that were described above aiming to efficiency increase. The Combined Cycle Gas Turbine deploys both the principles of a Gas Turbine and a Steam Turbine within the same cycle. The gas turbine operates under the open Brayton cycle to drive an electric generator while the exhaust products are funneled to a heat exchanger for producing steam which is responsible for additional power generation. The mechanism that includes the concept of the heat exchanger is called Heat Recovery Steam Generator (HRSG).

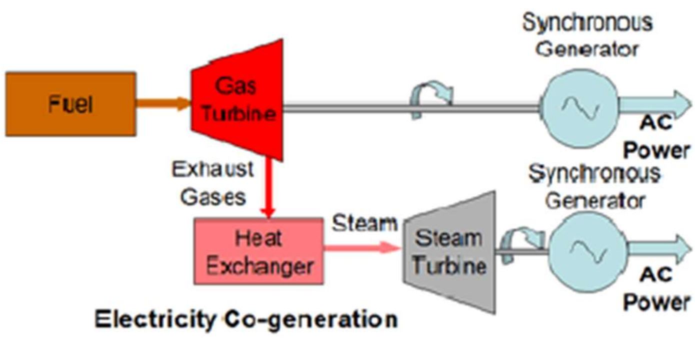

Figure 41: Combined Cycle Gas Turbine

The calculation of the cycle's efficiency is approximately estimated with this formula:

$$\eta_T = \eta_B + \eta_R - \eta_B * \eta_R$$

Where:

-η_T: total efficiency

-η_B: Brayton's efficiency

-η_R: Rankine's efficiency

These assumptions lead to efficiencies greater than the efficiencies of simplified cycles around 60%. However, for the same reasons as with the Open Brayton Cycle, this impressive option in terms of efficiency has to be abandoned.

4.4 Modifications of thermodynamic cycles

The operation of the cycles that were analysed before is focused on the most simplified version of each cycle. In reality however, some configurations that improve the efficiency of the cycles have been developed and have gained significant acceptance. This research expands on three of them:

- Regeneration
- Intercooling
- Reheating

4.4.1 Regeneration

The main operation of regeneration is based on the increase in the temperature of the working fluid before its entrance to the boiler. The increase of the temperature however is extracted from a portion of steam that was heading towards the turbine. Even though this kind of steam disenchantment seems as a negative impact on the efficiency rate, the reduction in the temperature difference between the hot and the cold reservoirs of the cycle proves that this method ends up with a positive impact on the thermal efficiency. Generally, the thermal efficiency of the regenerative cycle could be argued to increase at a decreasing rate as the number of feed waters increases (S.O. Oyedepo, 2020). It has been found that the optimum number of feed waters on a regenerative cycle is around 6 to 7 for pressures from 17 to 35MPa (P.U. Akpan, 2018). Nonetheless, the implementation of one regeneration stage has been estimated to improve the thermal efficiency by 4%-5%. (Yarong Wang, 2021).

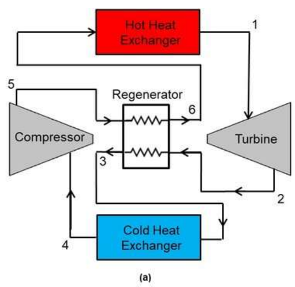

Figure 42: Implementation of regeneration

4.4.2 Intercooling

This method aims to relieve the compressor by cooling the flow of the medium and decrease in such a way the required power to realize the compression task. This is achieved with the assistance of a Heat Exchanger that uses seawater as a cooling fluid, however a certain downside of this solution is the contribution to the increase of system complexity. A single stage intercooling is proved to enhance the cycle's output by 2% on average.

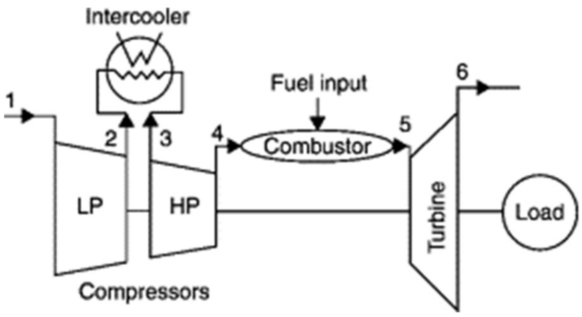

Figure 43: Implementation of intercooling

4.4.3 Reheating

The main improvement with reheating is that the expansion of the medium takes place not only within one stage but in two or more. Similarly, as the medium exits the boiler and enters the high-pressure turbine, it goes back to the boiler again in order to pass through a second turbine for a consecutive expansion and work production. Thus, the temperature of reheat before the low-pressure turbine is more or less equal to the inlet temperature of the high-pressure turbine, while the maximum improvement of the cycle efficiency is accomplished when reheat pressures are about 19% of the fluid's generator pressure (Habib, Said, & Al-Zaxarna, 1995). The goal behind this process is to reheat the medium to such multiple stages so that the latter is fully expanded and cannot 'give' more energy and that the moisture is removed. It is more than obvious, that the higher the number of reheating stages, the more improved the cycle efficiency will be. According to Dincer and Zamfirescu a single reheat increases cycle's efficiency by about 1%-3%, whereas a double stage can improve the rate up to 8%. However, the increase of reheating stages is associated with some increased fixed costs and system-enhanced complexity.

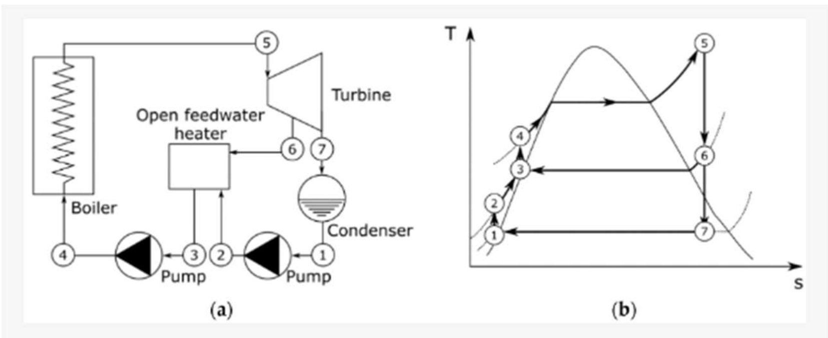

Figure 44: Implementation of reheating

4.4.4 Regenerative cycle with single-stage reheat

The optimization of a cycle refers to many factors and thermal efficiency is not the only determinant of the problem. What is more than imperative when a cycle is designed for transportable applications such as cars, trains, vessels and aircraft is the minimization of the complexity of the system. For this purpose, a one-stage regenerative Cycle with a single reheat is suggested. According to Cangel and Boles (1989), an increase of the order of 10% should be considered when these two configurations are combined at the initial Cycle.

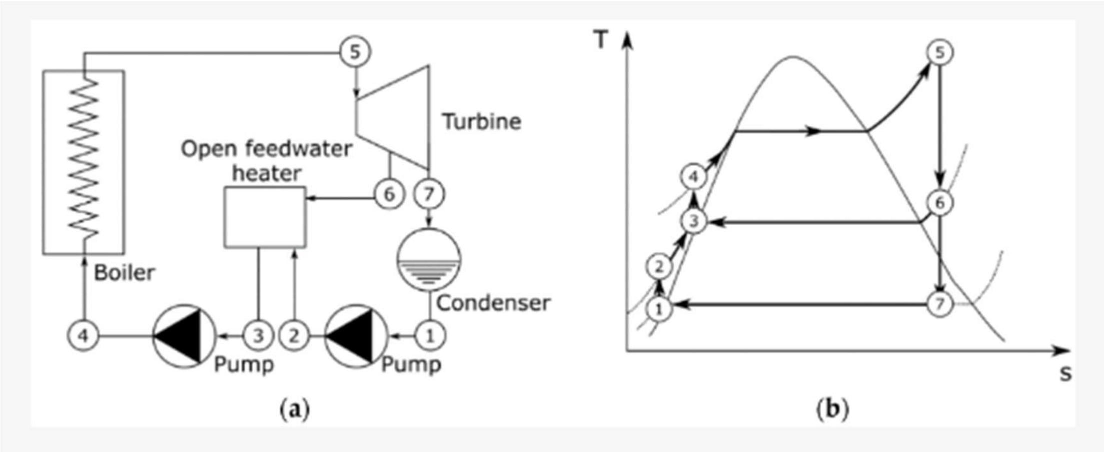

Figure 45: Implementation of regeneration with a single-stage reheating

Table 22: Cases of efficiency increase (%)/modification

Efficiency Increase %		
Reheating	Regeneration	Intercooling
1%-3%	4%-5%	2%
7%-8%		-

4.5 Efficiencies

4.5.1 Rankine efficiency

Some normal values of Rankine's cycle efficiency for low temperatures (300ºC to 450ºC) range between 30-38% according to Fleming et al. These cases refer to Light-Water-Reactors (LWR) which are the most widespread nuclear reactors in the industry, in comparison with coal-burning plants which can achieve efficiency up to 45% due to higher boiler temperatures. Also, the efficiency rate of the Rankine cycle can be increased with some advanced configurations such as using reheating and regeneration as it was presented earlier. Assuming that Generation IV reactors work under the thermodynamic analysis of a regenerative Rankine Cycle with a single reheat, a hypothesis of an efficiency rate of around 35%-45% could be achieved. These rates are verified from the ARIS platform where the specifications of all the Gen-IV reactors are gathered.

4.5.2 Closed Brayton efficiency

It is evident from the data of ARIS platform that conversions islands that obey to the Closed Brayton Cycle have improved efficiency over the Rankine Cycle. Most of the achieved efficiencies may begin from 35% and could get up to even 50%. On top of that, a Closed Brayton with regeneration can exceed the previous value.

4.6 Heat exchangers

No matter what the preferred power conversion system will be, there are some components that their role is mandatory for the successful operation of the plant. These are the Heat Exchangers and their purpose is to transfer heat from one medium to another without bringing them into direct contact. Although in most cases there is one heat exchanger within the reactor that transfers the heat that flows in the coolant of the primary loop to the working fluids (steam, He, sCO_2, etc) of the secondary loop, there are occasions where more than one heat exchangers are deployed. The most widespread variation of multiple applications of Heat Exchangers is the use of Indirect Cycles. These cycles involve an Intermediate Heat Exchanger (IHE) additionally, which even though results in an efficiency reduction of 2% to 4% and makes the system more complex, has the noticeable asset of easier maintainability (Bahman, 2014).

Heat exchangers come in various types, numerous sizes and mostly for a wide range of operating conditions with different materials. Heavy Industries nowadays have developed many designs with the ones listed below being the most recognized:

- Shell and Tube Heat Exchangers
- Plate Heat Exchangers
- Double-Pipe Heat Exchangers
- Helical Coil Heat Exchangers
- Printed Circuit Heat Exchangers

Each type has certain advantages and some limitations. A major factor to take into consideration should be the desirable pressure conditions, where the shell and tube type is suggested for low-pressure applications while the plate type seems more promising for high-pressure applications (Houtkoop, 2022).

4.7 Load response

In the case of the ship propulsion, for which the nuclear plant is designed, the changes in the vessel's speed are necessary. The speed differentiation however can be achieved only with fluctuations in the power output of the plant. The term that reflects the responsiveness and flexibility of the plant is the load response. Typical measurement of load response is the ramping rate which is expressed in MW/min or in load percentage/min.

According to research published by the Nuclear Energy Agency Organisation for Economic Co-Operation And Development, the power ramps ranged from 1%-5%/min. More specifically, if it is about a light load following between 60%-100% P_{out} the maximum speed is 5%/min, whereas if it is about a deep load following between 25%-60% P_{out} the maximum speed is 2.5%. The minimum achievable load of a steam turbine is estimated at around 50% of the maximum P_{out} (Salazar, 2017). BWRs seem capable of attaining a ramping rate within 5% to 10% however there are no reliable indexes for GenIV reactors yet.

4.8 Conclusions

When examining existing designs and assessing the primary technology choices, it becomes evident that creating an optimized design involves an intricate balance of various factors. These factors include cost, efficiency, development time, ease of maintenance, and the technological trajectory, all of which must be taken into account within the framework of an integrated power conversion system before making a final assessment.

The ultimate benchmark for assessing the worth of power conversion alternatives is the cost of the electricity generated. This cost is determined by factors such as the recovery of capital and operating expenses, as well as the efficiency and reliability of the system. While evaluating costs can be challenging without detailed integrated designs, it is feasible to identify the elements influencing the performance, cost, and technical risks of components and systems.

5. SECONDARY CONCERNS

As it is common knowledge, nuclear energy is not easily accepted by public opinion. This negative perception of nuclear energy is mainly attributed to specific accidents in nuclear plants and secondly to the deficient engagement with nuclear energy. However, if someone is not constrained to the resonant articles and news that ensued the accidents but is willing to investigate the evidence in pure numbers, a different view will undoubtedly be unveiled. According to WNA once again, only 3 incidents resulted in greater radiation doses than from natural exposures in more than 18500 cumulative reactor years.

Nonetheless, public acceptance should never be underestimated and in order for nuclear reactors to expand to a worthy commercial level the safety not only of the reactor but also of the whole procedure from the beginning (fuel production) until the end (waste management) is a top concern. Along with the safety of the reactor and the fuel management two more areas must be explored. The one should be dedicated to the parameter of health including all the aspects that could potentially increase the risk both for the workers but also for the consumers of this form of energy. Lastly, policies and regulations that consequently affect the security measures is another field where more efforts are required.

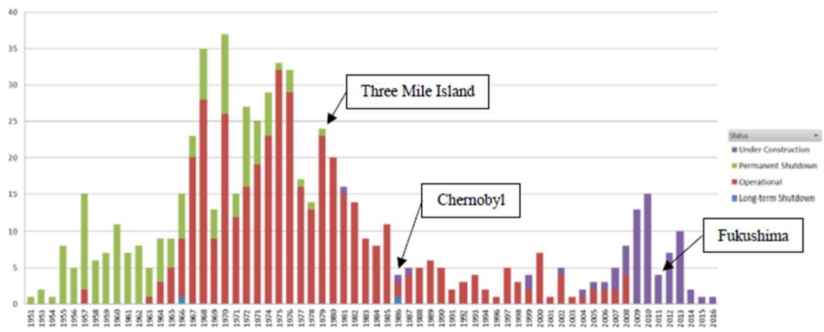

Figure 46: Status of Nuclear Reactor by Construction Start Year (Stones, 2014)

5.1 Safety of reactors

The status of the reactors' safety is already on a satisfying level. Only the figures connected to deaths and important side effects from fossil fuels use can justify nuclear superiority. Immune from accidents, nonetheless, is impossible for all industries and thus most of the measures already developed are based on mistakes of the past. But before analysing the mechanisms for preventing such incidents it is important to clarify the reasons that could potentially escalate to tragedy. These reasons can be divided into two main categories: the mechanical failures and actions associated with terrorism. What must be prevented to any cost under any circumstances, however, is the dispersal of radioactive products that can pose a direct radiation hazard.

In an effort to mitigate the risk of a large-scale accident due to mechanical failure, safety features are continuously updated and reassessed. Over the years and after several previous accident analyses, engineers have focused on three core pillars.

The first one is associated with fuel technology, the second with barrier development and the third with passive safety. The last updates regarding fuel technology indicate that molten fuels are safer to use since new studies show that less radioactive material escapes the core. In parallel, accident-tolerant fuels are a new field to research. Barriers' development on the other hand is the most applied solution nowadays and their significant role is to maintain the radioactive content within the reactors' limits. The term "barriers" describes all the structures that act as an obstacle to radiation's expansion, from the control rods inside the vessel to the enhanced containment structures rounding the whole plant. These structures are so advanced that can suffer the consequences of natural phenomena such as earthquakes, flooding, tsunamis etc.

As for the term "passive safety", it includes many different systems with all of them sharing the same principle under which the operator's intervention is not required. The passive safety systems' operation is achieved through natural processes such as cooling circulation with gravity or natural heat decay.

What is more, the electronic systems provide the operators with all the significant information from the sensors that monitor the operational profile of the plant. Hence whenever an unusual situtation is tracked, a series of actions are automatically running on the background. Inherent safety is attained by the elimination of hazards through decisions made during the conceptual design phase. With thorough understanding of the physics 4 phenomenon governing the operation of a particular reactor concept, nuclear reactors can be designed to preclude the possibility of certain accident scenarios. For example, a reactor designed with such that when the fuel heats up, the reactivity of the reactor decreases (via selection of material or core configuration) is inherently safe with respect to increasing temperature. As it can be drawn, the factor of safety is constantly enriched aiming to operational conditions fully discharged from risks and hazards.

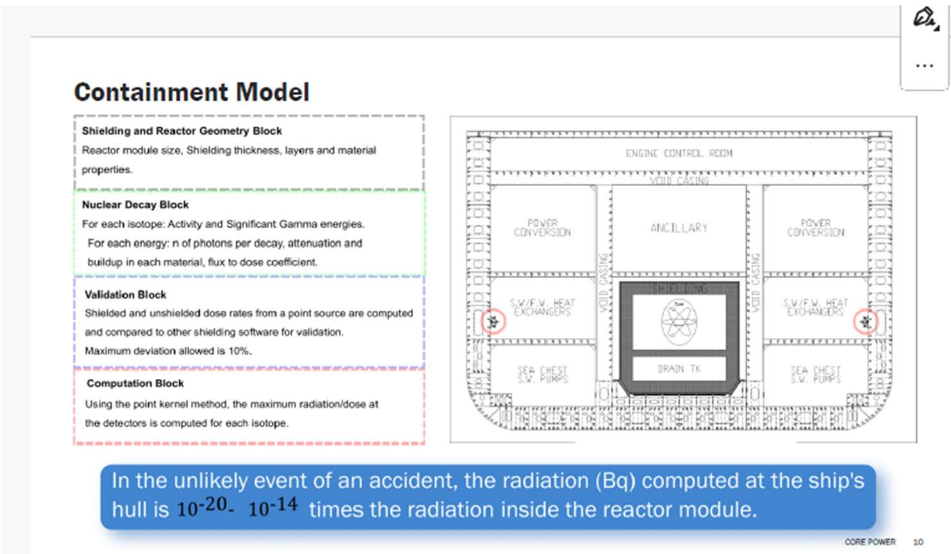

Figure 47: Vessel's containment model (Buitendijk, 2022)

5.2 Fuel irradiation and waste management

Nuclear fuel could reasonably be entitled as the most superior fuel as far as the Carbon emissions are concerned. The cost of being a zero emitter, however, is balanced with the hazard that stems from the radiation doses. The ultimate dilemma that must be put under the scope of pros and cons is definitely whether or not nuclear is a “greener” solution than fossil fuels without raising concerns about public health and keeping the total costs at affordable levels. A simple illustration of the relationship of fossil fuels and nuclear fuels according to Mikal Boe can be seen below.

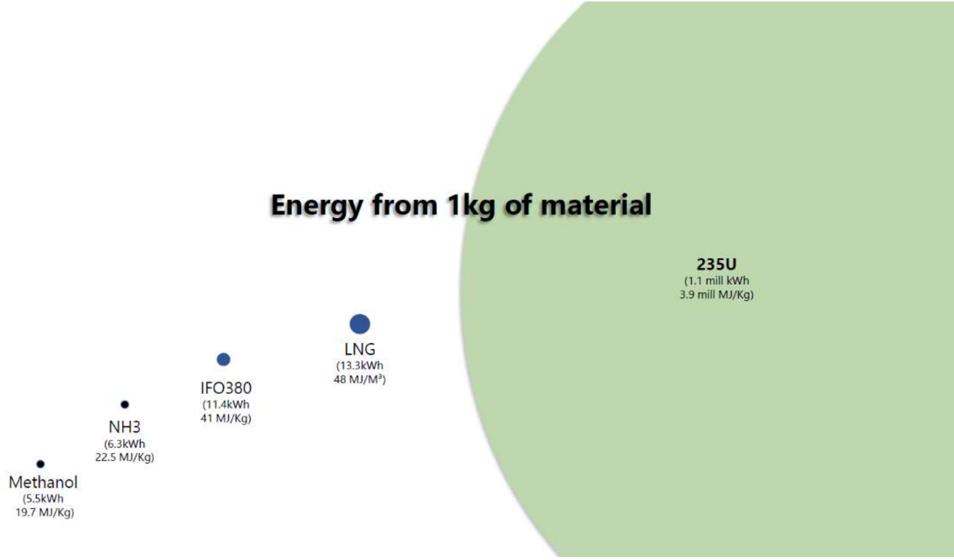

Figure 48: Energy Output from 1kg of various fuels (Core Power, 2022)

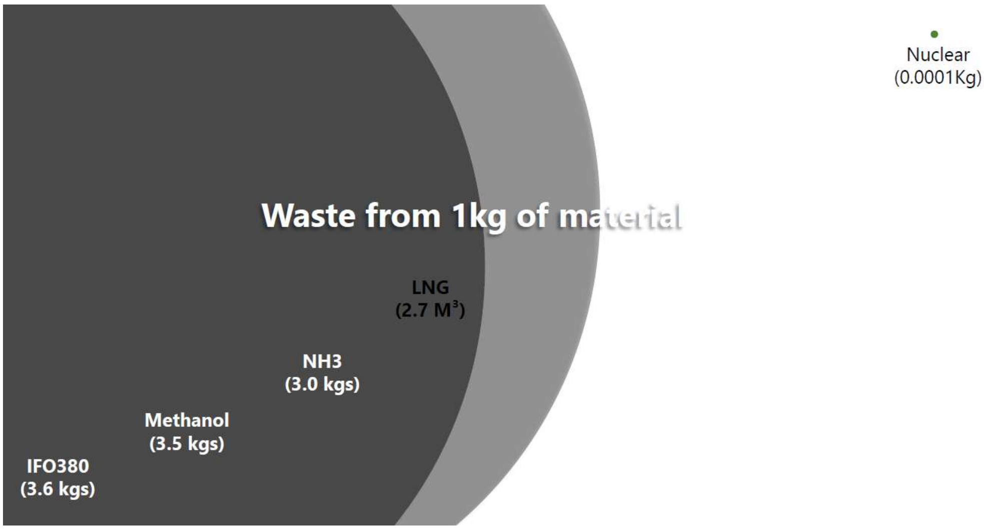

Figure 49: Waste generated from 1 kg of various fuels (Core Power, 2022)

Before comparing the estimated figures from the nuclear waste, it is essential to present the classification within the nuclear fuel. Nuclear waste is categorized into three main groups based on the levels of radioactivity in the materials left over. These three groups are:

1. High-Level Waste – HLW: fuel rods, core materials etc.

2. Intermediate-Level Waste – ILW: resins, chemical sludge, metal coverings etc.
3. Low-Level Waste – LLW: paper, clothing, tools etc.

The first two categories require much more attention in their storage since they need shielding and cooling until they are upgraded to the third one. Even though the high-level waste products account for less than 5% of the total waste volume, they are so pollutant so as to cover 95% of the totally produced radioactivity. The rule that all the materials obey is the one of natural decay where radioactive elements are turned into non-radioactive. What should be highlighted is that the more radioactive an element is, the faster it will decay. It is common knowledge that waste is produced in every step of a fuel cycle from the beginning of the process during mining until the very end when the actual "burning" takes place.

Another important parameter for the waste is time. In the occasion of nuclear energy, what is used to describe the profile of an element is the so-called "half-life" which is the length of time required for half of the radioactive atoms to decay. Typically, there are three categories based on the duration of the half-life once again:

1. Very Short Lived, <100 days
2. Short Lived, <31 years
3. Long lived, >31 years

Most of the indexes indicate that after 7 half-lives, less than 1% of the initial radiation has remained. (International Atomic Energy Agency, 2003).

Assuming that all the possible reprocesses have been done and that the fuel has the best utilization it could have, there is still uncertainty of what is the best way to handle the waste. Waste that is ready for disposal, is being stored in steel barrels which are filled with inert gas and sugar and then sealed. One of the problems that remain unsolved today is the selection of a remote and inaccessible location suitable for long-term storage. The problem is that this location might be inappropriate for any human intervention for possibly up to 100 years and this lengthy period could act as a boomerang for any of the future plans. Such locations are the bottom of the oceans (prohibited nowadays), deep underground facilities similar to mines, the space (quite risky and expensive for now), and the most accepted solution which are above-ground facilities near the nuclear power plant.

5.3 Regulations and policies

Apart from the last two topics that raise many concerns, a third field is believed to be very crucial for the continuation of the nuclear venture. Actually, according to Hagen and Megan, Licensing and Regulatory Framework for the certified operation of the reactors will be the most incomparable challenges. Historical data can act as the proof of what some of the potential obstacles might be however one of the major problems is going to be port access. There is no point of investing in such an expensive and high-risk technology if the outcome is already condemned.

Luckily, nuclear powered vessels have a long history of more than 60 years and thus some preliminary work on legislating level already exists. There are many instruments involved in the establishment of all the rules, resolutions, protocols and policies including among others: the United Nations, the IMO and the IAEA. A brief presentation of some of the most pivotal conventions is shown in the picture below.

Category	Conventions
Nuclear safety conventions	1979 Convention on the Physical Protection of Nuclear Material 1986 Convention on Early Notification of Nuclear Accident 1986 Convection on Assistance in the Case of a Nuclear Accident or Radiological Emergency 1994 Convention on Nuclear Safety 1997 Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radiation Waste Management
Nuclear liability conventions	1960 Convention on Third Party Liability in the Field of Nuclear Energy (Paris Convention) 1963 Vienna Convention on Civil Liability for Nuclear Damage (Vienna Convention) 1963 Convention Supplementary to the Paris Convention (Brussels Supplementary Convention) 1997 Convention on Supplementary Compensation for Nuclear Damage (CSC)
Maritime transport safety and liability conventions	1962 Convention on the Liability of Operators of Nuclear Ships (Brussels Nuclear Ship Convention) 1965 International Maritime Dangerous Goods Code (IMDG Code) 1971 Convention Relating to Civil Liability in the Field of Maritime Carriage of Nuclear Material 1974 International Convention for the Safety of Life at Sea (SOLAS) 1978 International Convention on Standards of Training, Certification and Watchkeeping for Seafarers (STCW) 1981 Code of Safety for Nuclear Merchant Ships 2015 International Code of Safety for Ships Using Gases or Other Low-Flashpoint Fuels (IGF Code)
Radioactive marine pollution control related conventions	1972 Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter (London Convention) 1973 International Convention for the Prevention of Pollution from Ships (MARPOL) 1982 United Nations Convention on the Law of the Sea (UNCLOS) 2000 Protocol on Preparedness, Response and Co-operation to Pollution Incidents by Hazardous and Noxious Substances (OPRC-HNS Protocol)

Figure 50: Legislation and Policies around Nuclear Energy

Nonetheless, the existing regulations are riddled with insufficiencies that neither they cover the field with specific instructions on many topics but also they hold back the efficient and broad development of nuclear powered vessels. These insufficiencies and shortcomings include issues with the application of nuclear convention principles, the ratification deadlock of specialized conventions for nuclear ships, a problematic reliance on flag states' regulation and regulation incoordination, and inadequate liability and compensation mechanisms for nuclear-powered merchant ships' environmental damage indemnity (Wang, Zhang, & Zhang, 2023).

6. CASE STUDY

At this chapter of the thesis, an exhaustive analysis for the most promising design is presented. This analysis includes some technical specifications of the reactor, the basics around the operating profile of the plant such as the working conditions and some estimations concerning the CAPEX and the OPEX of this project. Finally, the reactor is compared with a vessel equipped with a conventional diesel engine in a effort to outstand the environmental impact that each solution brings with.

6.1 Reactor's design selection

The selection of the most promising design it was made upon assumptions and under a degree of uncertainty. Several factors such as the applicability, the maturity, the cost and the safety of every design were acknowledged. Based on the data from the tables a hierarchy among the six designs was decided and is illustrated below.

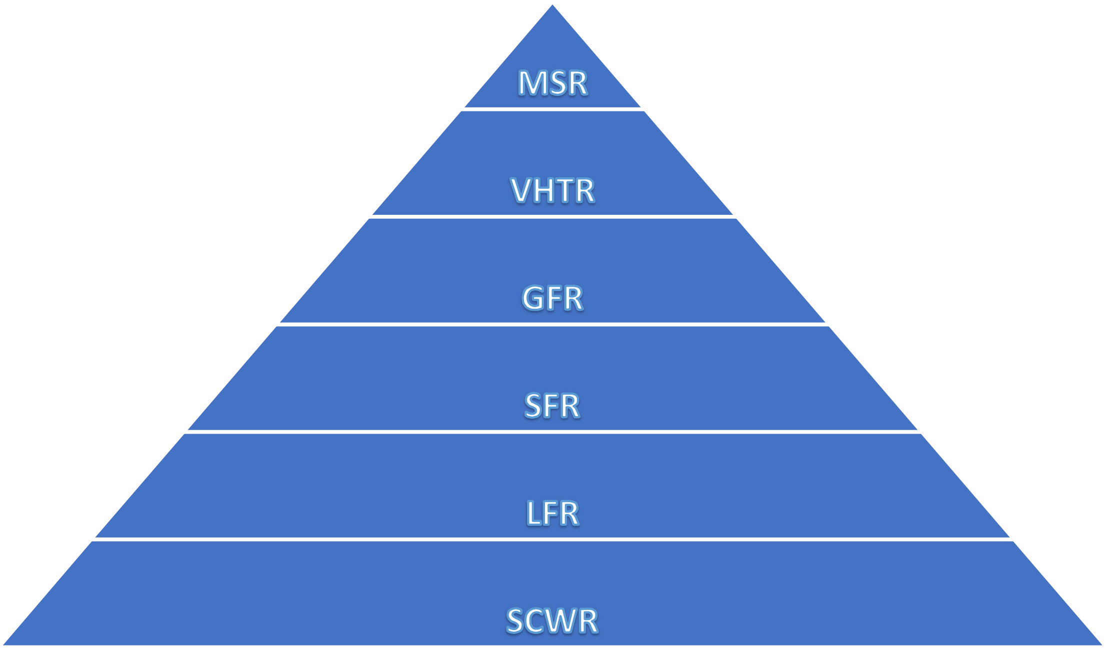

Figure 51: Hierarchy of the most promising Gen IV designs

From a technical perspective it seems that VHTRs and MSRs are probably the most promising designs. Despite the primary contention of Molten Salt Reactors which lies in the readiness level of this exceptional design as Canadian research outlines, the superiority of MSRs is visible to the unaided eye. Some of the performance characteristics that made this design so prominent were definitely its high burnup measure that indicates a decent fuel efficiency, the increased passive safety that the molten salt provides and the auspicious signs for thorium implementation within the cycle. Moreover, the timeline for the implementation of nuclear power to marine applications is projected from 10 to 15 years ahead and hence there is still enough time for research and try-outs.

6.2 Vessel's route selection

The primary question posed by the public concerns the vessel through which this technology will be implemented. As it was implied in the fourth chapter of this paper, nuclear power has significant capabilities and thus there is no point of implementing this technology in a vessel with medium or low energy demands. As a result, the selected vessel for this case study was a Bulk Carrier of 400k DWT. This type of vessels (Valemax) is owned by the Brazilian mining company Vale S.A. that is responsible for the safe carriage of iron ores from Brazil to many destinations around the world.

Figure 52: Seamar

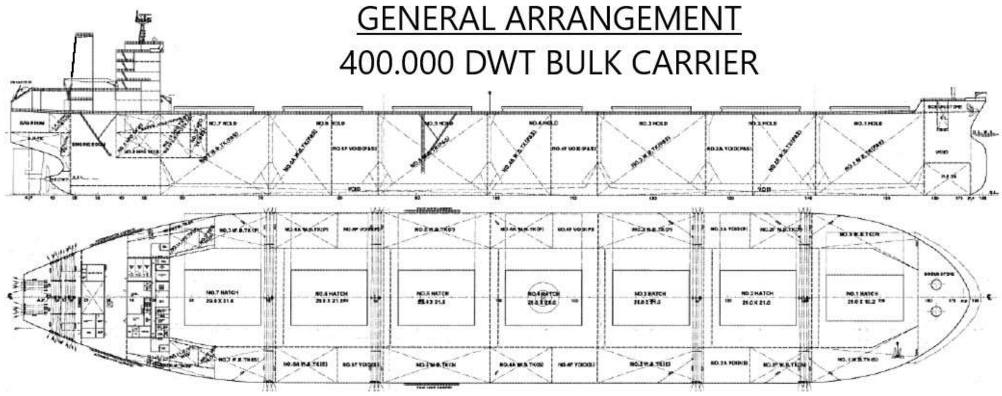

Figure 53: Seamar General Arrangement

6.3 Route selection

For the voyage plan of this vessel a realistic hypothesis will be made. Our vessel will be sailing with a service speed of 15.1 kns from Tubarao (BRA) to Qingdao (CHI) and back. This route is equal to 11,065 nms as it was calculated from the ShipAtlas app.

Figure 54: Vessel's Route[2]

The duration of a single route is expected to last approximately 30.5 days when the vessel is on laden condition (TUB→QIN), 27.9 days when is on ballast condition (QIN→TUB) and every visit in each port will last up to 3 days. Hence the total duration of a round voyage is estimated up to 65 days. A fundamental term for this hypothesis, is the efficient functioning in the company's logistics department and in the ore's supply chain which ensures timely execution, thereby avoiding delays in vessel's itinerary. On these grounds, the vessel is estimated to complete 5.5 voyages per year and with a projected lifetime of 25 years, the vessel could exceed 140 voyages in total. Within these calculations 5 days every 5 years are excluded for maintenance reasons.

Duration	30 days 12 hours
Distance	11.065 nm
Speed	15,1

Figure 55: Information for vessel's voyage

[2] https://www.maritimeoptima.com/shipatlas

Table 23: Vessel's itinerary per Voyage

VESSEL'S SCHEDULE/ VOYAGE				
DAYS OF LADDEN ROUTE	30.5	DAYS	732	hrs
DAYS OF PORT STAYING @ Qingdao (CHI)	3	DAYS	72	hrs
DAYS OF BALLAST ROUTE	27.9	DAYS	670	hrs
DAYS OF PORT STAYING @ Tubarao (BRA)	3	DAYS	72	hrs
DURATION OF TOTAL VOYAGE	**64.4**	DAYS	**1546**	hrs

Table 24: Vessel's itinerary per year

VESSEL'S SCHEDULE/ YEAR				
DAYS OF LADDEN ROUTE	172.9	DAYS	4149	hrs
DAYS OF PORT STAYING @ Qingdao (CHI)	17.0	DAYS	408	hrs
DAYS OF BALLAST ROUTE	158.1	DAYS	3795	hrs
DAYS OF PORT STAYING @ Tubarao (BRA)	17.0	DAYS	408	hrs
DURATION OF TOTAL VOYAGE	**365.0**	DAYS	**8760**	hrs

6.4 Engine and reactor Selection

The power needs of this vessel are extracted from the resistance tests in two conditions: the ballast and the laden. In the table below, the power that must be delivered for every speed is presented including the factor of sea margin as well. The highlighted cells indicate the required power for the service speed in the design condition, while for the ballast condition the achieved speed for the same power is calculated. The values of power below are related only to the propulsion needs of the vessel.

Table 25: Vessel's Resistance Results

Ballast			Ladden		
Vs	Delivered Power (kW)	Sea Margin 15%	Vs	Delivered Power (kW)	Sea Margin 15%
10	3789.1	4457.7	10	6191.7	7284.3
10.5	4487.9	5279.8	10.5	7171.3	8436.8
11	5273.9	6204.6	11	8249.3	9705.1
11.5	6153.2	7239.1	11.5	9430.6	11094.8
12	7132.2	8390.9	12	10719.7	12611.4
12.5	8217.3	9667.4	12.5	12121.4	14260.5
13	9415.1	11076.5	13	13640.6	16047.7
13.5	10732.1	12626.0	13.5	15281.9	17978.7
14	12175.2	14323.8	14	17050.0	20058.9
14.5	13751.4	16178.2	14.5	18949.8	22293.9
15	15467.7	18197.3	15	20986.0	24689.4
15.5	17331.2	20389.7	**15.1**	**21410.0**	**25188.2**
16	19349.2	22763.8	15.5	23163.3	27250.9

16.47	**21410.0**	**25188.2**	16	25486.5	29984.1
16.5	21529.1	25328.3	16.5	27960.3	32894.5
17	23878.3	28092.1	17	30589.5	35987.7

Hence if the desirable speed on the design condition is 15.1 kns and with respect to the sea margin the engine should be capable of delivering 25188.2 kW. To accomplish this output and with a loss factor of 0.92% in the transmission system the nominal continuous rating point of the engine should be approximately 27400 kW. In addition, the vessel's electric balance under the assumption that three auxiliary engines are operating at the same time indicates that the maximum power need is estimated at 1460 kW.

To make this hypothesis more realistic, the conventional engine was selected from MAN manufacturer. Specifically, the 6G80ME-C10.6-HPSCR model was chosen with an SMCR power rated at 25200 kW at 68 r/min. The main characteristics of this engine can be seen below:

Table 26: Main Characteristics of the Conventional Diesel Engine (MAN, 2024)

Engine type	6G80ME-C10.6-HPSCR	
SMCR power	kW	25200
SMCR speed	r/min	68
Turbocharger type		High eff.
NOx emission compliance		Tier III
Propeller type		Fixed pitch propeller
Cooling system		Central
Hydraulic power supply		Mechanical
Hydraulic control system		Unified
Constants		
LCV for fuel oil	[kJ/kg]	42700
Steam pressure	[bara]	7
ISO	Air	25
	Water	25

Since nuclear reactors of the fourth generation have not reached the market yet, the manufacturers are very limited and consequently most of their models remain on demonstrating level. However, an initial estimation for some of the key characteristics of the discussed Molten Salt reactor are presented. Based on the figures calculated before, this Molten Salt Reactor should be capable of delivering 29 MWe (27400 kW +1460 kW) if it is to replace MAN's engine successfully. Taking for granted an efficiency around 30%, the equivalent thermal measure will be up to 90 MWth.

6.5 Assumptions of nuclear-powered bulk carrier

6.5.1 Nuclear consumption

As the consumption of fossil fuels is measured for an engine of internal combustion, an identical procedure is established for the nuclear fuel. The basic difference is that during the operation of the reactor, the nuclear fuel remains inside the system. Nonetheless, the fuel at this stage is in a depleted form that is incapable of providing further energy. The rate at which the nuclear fuel is depleted is the requested consumption. This value is affected by many other factors such as the type, the burn-up, the efficiency, the load and the operating profile of the reactor.

Table 27: MSR's key info

Type of reactor	Molten Salt Reactor
BurnUp	30 MWd/kg fuel
Efficiency	40 %

The figure in the field of BurnUp was calculated with the method of linear interpolation of the already constructed MSRs. This graph includes the equation that helped to reach to this result. It has to be noted all of the MSRs refer to power outputs bigger than the one discussed in this case however they were the more logical to be used.

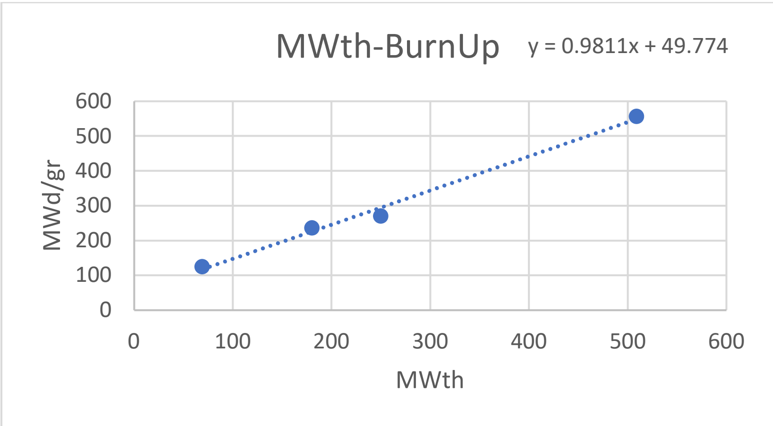

Figure 56: Correlation of Thermal Output with BurnUp

Table 28: Correlation of Thermal Output with BurnUp

Thermal Output [MWth]	BurnUp [MWd/ Kg fuel]
125	69
236	180
270	250
125	69
557	509
y = 0.9811x + 49.774	
72	30

Now, assuming that the approach of 38 MWd/ kg fuel does not deviate from the reality, this figure is equal to

$$30\frac{MWd}{kg\ of\ fuel} = 30000kW * 24\ hrs\ /1000\ gr = 720\frac{kWh}{gr\ of\ fuel}$$

In order to get the specific consumption for the MSR, the efficiency of the conversion in the turbine had also to be taken into account. With an efficiency rate close to 33% (explained later), the SFOC of the nuclear reactor was calculated to be 0.0038 gr/ kWh ≈ **0.004 gr/kWh**. It has to be noted that the SFOC of the reactor can get up to 0.010 gr/ kWh when the load (%) goes beneath 50%. This increase in the SFOC at these loads is justified by the very inefficient use of the turbine. For visual purposes, the next table shows typical values of the consumption for several loads.

Table 29: SFOC of reactor according to Load (%)

Load %	10%	25%	50%	100 %
SFOC gr/kWh	0.01	0.008	0.0040	0.0038

Taking into consideration that this vessel will be chartered under long-term contracts with a fixed route, the operating profile could be described under these two potential scenarios, where the values in the table represent the portion of the total annual time of the vessel. Obviously, the operating time of a vessel/ year is every hour of the year. Under this hypothesis, the total time of every vessel annually is equal to 24 hrs * 365 days = 8760 hrs.

Table 30: Operating profile scenarios

Scenarios	Time at Full Load (%)	Time at Part Load (%)	Time with no Load (%)
A	65	10	25
B	90	10	0

Before moving to further calculations, the terms of Load (%) have to be specified.

- Full Load: 87.5% of MWe → 25,3 MWe
- Part Load: 40% of MWe → 11,6 MWe
- No Load: 20% of MWe → 5,8 MWe

Under these boundaries, the power required from the vessel in kWh was calculated as it follows:

Scenario	kWh at Full Load	kWh at Part Load	kWh at no Load
A	143680699.5	10105016.23	12631270
B	198942506.9	10105016.23	0

If the upper values are multiplied with the SFOC recorded in the table then the mass of the necessary fuel will have approximately been estimated:

Table 31: Total Consumption in grams per load level (%)

Scenario	gr/year at Full Load	gr/year at Part Load	gr/year at no Load
A	548780.4493	46314.6577	91664.43

B	759849.8529	46314.6577	0

For these two scenarios and for a Bulk Carrier that carries 400k DWT the year consumption of nuclear fuel is given in kgs in the table below:

Table 32: Total annual consumption in kgs per year

Scenarios	Consumption (kgs/year)
A	687
B	806

6.5.2 Conversion method

As examined in the previous chapter, the methods for converting the reactor's thermal energy into useful mechanical power are not numerous. In more detail, two are the dominant methods nowadays and these are either the Rankine Cycle (steam turbine) or the Brayton Cycle (gas turbine). Many variations that improve the power output of the system have been developed for both. Apart from enhancing the efficiency of the cycle, these variations are also used to work under conditions that the simple cycles are not capable of operating smoothly. The improvement in the efficiency however is balanced from the increase in the volume, the size and the complexity of the construction.

According to Houtkoop's research, a safe projection for the implementation of a simple superheated steam turbine along with the nuclear reactor would result in cycle efficiency of around 29%. The operating conditions would not exceed 4 MPa and 480°C in pressure and temperature accordingly. With the use of reheating techniques however and despite the increased complexity of the system, the efficiency would possibly reach 38%. The conditions would also become harsher with the temperature and pressure reaching 525°C and 9MPa accordingly! For the sizing estimation of the steam cycle these values have been assumed:

Table 33: Sizing ratios for the steam turbine

Weight		Volume		Dimensions (L:W:H)		
1.5	ton/MWe	3.5	m3/ MWe	2:1:1		
43.2	tons	100.9	m3	8.5	4.2	4.2

In the case of the Closed Brayton Cycle, there are two potential working mediums, helium and supercritical CO_2. The estimated net efficiencies will be around 32% while more optimistic supporters claim for efficiencies that approach 40%. In addition, with the use of a recuperator, an extra increase of 5% is expected.

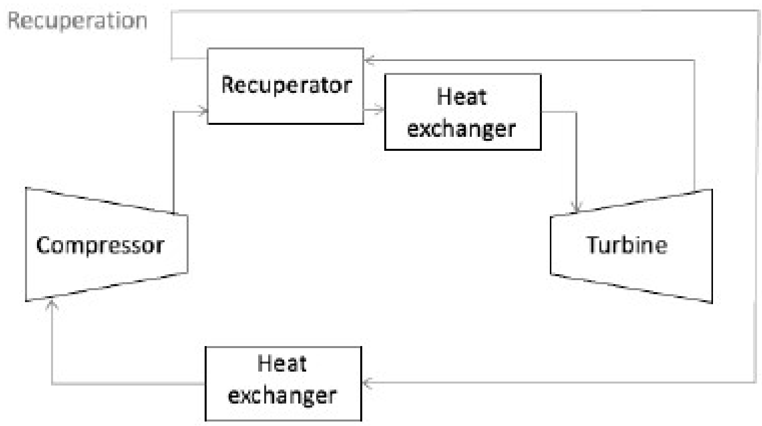

Figure 57: Recuperated CBC

The maximum temperatures within the system will be really close the outlet temperature of the molten salt solution. The working pressure on the other hand in the case of sCO_2 will be increased compared to helium's option (about 8 MPa-20 MPa). Supercritical CO_2 has the supplemental advantage of the plant's downsize compared to Helium or the Rankine Cycle. The size reduction with the implementation of a gas turbine instead of a steam is depicted on the table below (* indicates the times decreased compared to the steam turbine values):

Table 34: Sizing ratios for Gas Turbine

Weight		Volume		Dimensions (L:W:H)		
1.45	ton/MW	2.2	m3/ MW	2:1:1		
41.8	tons	63.4	m3	6.5	3.2	3.2

The upper figures refer only to the turbine size however the whole island where conversion happens includes and some other components. These components are the heat exchangers where heating and cooling take place, however on the grounds that there is no change in the phase of their mediums, heat exchangers are addressed as really compact components with indiscernible contribution.

The most favourable choice among the presented ones seems to be the implementation of the recuperated Closed Brayton cycle with the sCO_2. Not only is the improvement of the efficiency but also there is the advantage of releasing important space in the engine room. Intercooling would be another suggestion to take into consideration however it comes at the price of the increased system complexity and for this reason will not be examined any more. To conclude with and from a more conservative view of the topic, an efficiency of 40% along with the recuperator will be assumed.

Table 35: Efficiency of the cycle

Efficiency of the sCO2 recuperated Closed Brayton Cycle
40%

6.5.3 Load following

The load response of a gas turbine that employs sCO_2 as medium is limited from the heat exchanger's capabilities. It seems that the load response will be similar to the reactor's load following capabilities and in order to quantify this a reasonable power change rate would be 4-7 %/min for the window of 50% to 100% of the initial reactor's power. In the cases where less than 50% of the name power is needed there are thoughts for supplementary mechanisms where the working medium that carries additional non useful energy for this period will be sent to a separate tank and will be stored there until an increase in the power is needed again. This solution however has not been implemented ever before and is still on experimental level.

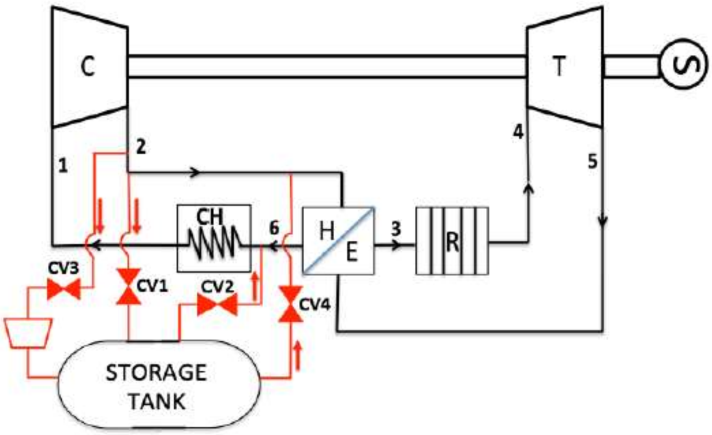

Figure 58: Storage Tank Modification (Houtkoop, 2022)

6.5.4 Heat exchangers

Generally, heat exchangers are fabricated into various types as mentioned before. As Oh, Kim and Patterson highlight in their research these are the values for three of the main categories.

Table 36: General characteristics of Heat Exchangers

Measurements	Shell and Tube	Helical Coil	PCHE
Heat Transfer Coefficient (W/m^2*K)	500	1000	2000
Surface density (m^2/m^3)	75	80	1100

The expansion of the PCHE heat exchanger should be thought as a turning point for the nuclear engineering if it can be implemented successfully. As it can be seen in the upper table PCHE is by far more superior contrary to the rest two options since the better heat transferring properties are not accompanied with the problem of oversizing. However, Helical Coil and Shell/Tube heat exchangers are way more developed and will always be a proven solution and for this reason will be recommended at this point.

In more detail, the conversion island will include an intermediate loop with Helium as medium for safety and waste management reasons. With the use of this loop the radioactive components will be separated, and every maintenance task will be simplified. In parallel, the danger of radiation expansion is circumscribed to the first loop. The volume of this helical coil heat exchanger is estimated at 0.16 m^3/MWth ending up to 11.52 m^3. The second heat exchanger that will connect the intermediate loop with the secondary loop where sCO_2 will be circulating is estimated at 0.12 m^3/MWth ending up to 10.8 m^3.

Table 37: Volume of Primary and Secondary Heat Exchangers

Primary - Intermediate	Intermediate - Secondary
0.16 m^3/MWth	0.12 m^3/MWth
11.52 m^3	10.8 m^3

Moreover, another heat exchanger is necessary for the important purpose of decay cooling. Its function is destined for cooling the reactor when the latter is has stopped or generally is out of order. The maximum decay heat from a non-working reactor can reach up to 7% of the maximum thermal power. However, due to the use of seawater as coolant to this heat exchanger, PCHE cannot be used (only clean fluids are allowed within the PCHE). Consequently, the heat exchanger for this purpose will be again Helical Coil and its volume is estimated up to 0.15 m^3.

Table 38: Volume of Decay Cooling Heat Exchanger

Decay Cooling
0.0016 m^3/MWth
0.15 m^3

6.5.5 Pumps and fans

There are two more categories that are so crucial for the smooth operation of the whole system that they cannot be disregarded. As it happens in the case of a conventional diesel engine, pumps and fans are necessary to ensure that every fluid flows in adequate quantities and under the proper conditions. Epigrammatically, a series of the required pumps and fans are suggested below along with their expected electrical consumption in kWs. It has to be noted that the estimated power of the pumps was calculated for 30m height which is the maximum vertical distance within the vessel and that all of the pumps are centrifugal.

Table 39: Pumps, fans and total power demands in kWs

Purpose	No. of pumps	No. of active pumps	Operating Media	ΔT (°C)	Total Power	
					kW/MWth	kW
Pumps						
Main coolant pump	3	2	Molten Salt	100	1.67	120
Seawater pump decay cooler	2	0	Seawater	40	0.27	19
Freshwater circulation pumps gearboxes	2	1	Freshwater	50	0.01	1

Freshwater circulation pumps electromotors	2	1	Freshwater	50	0.02	1
Seawater pump general	3	2	Seawater	25	0.05	4
Fans						
Secondary coolant fan helium	3	2	Helium	210	0.35	25
Additional Loads						
Control Electronics & Minor Systems					4	288
Total					6.37	459

6.5.6 Batteries implementation

An additional thought that comes along with the employment of nuclear technology is the electrification of the vessels with the use of batteries. At first sight, it seems very reasonable that with such generated power, some portion of this load could be stored for future/emergency use. The problems however that must be addressed here is that nowadays with the common battery technology either the stored energy will be inadequate, or the storage space will not be manageable. Despite the fact that batteries cannot cover the vessel needs in total for a long duration, they could be a supplemental solution for cases that the reactor cannot meet the load following demands itself. This supplemental role could be fulfilled either in cases where the vessel faces bad weather conditions, or in case of an emergency. It must be noted that batteries could work also as the final stage of the storage tank referred above when an immediate speed decrease is desired (in ports for example).

The battery pack that will be examined for this project are models of the Corvus company[3]. The selected product of the company will be containerized battery packs. This new technology which is an approved solution fits many battery packs into the standardized size of a 20 ft long TEU. Through this patent of Corvus, the battery equipment of the vessel is more compact, the installation time is significantly reduced, the classification approvals demand less paperwork and obviously the maintenance or even the replacement tasks related to the batteries are becoming considerably easier.

In the application of Seamar, two containerized battery packs were selected (Corvus Energy, 2023).

[3] Corvus Energy: https://corvusenergy.com/

Figure 59: Concept of the Containerized Batteries (Corvus Energy, 2023)

According to the next table, the total battery capacity of the vessel will be approximately 3MWh. The total space of the battery room is calculated to reach 88m^3 and its total weight will not exceed 55 tns.

Table 40: Technical characteristics of containerised battery (Corvus Energy, 2023)

Specifications of the containerized battery	
Max Energy Capacity (kWh)	1492
Height (m)	3.000
Length(m)	6.058
Width (m)	2.438
Volume (m^3)	44.43
Weight (kg)	26.65
No. of packs	7-12

6.5.7 Shielding

This paragraph examines the importance of the so-called shielding of the reactor. Of course, this part of the reactor is probably the most vital since it plays the most significant role in preserving the radiation inside the reactor's limits. With respect to the limits introduced by IAEA for the maximum exposure to radiation per year which are presented below:

Table 41: Limits of Radiation Exposure

Radiation to:	Worker	Public
Yearly Exposure (mSV)	50	1

As implied in the first chapter of this report, there are four harmful radiation types: A, B, G and X rays. Due to the fact that A and B types are not so penetrating such as the other two the design of every shielding around the world is focused on the absorption of the Gamma and the X rays. For these two, water and lead are combined to mitigate any harmful consequence. Concrete is also a solution for Gamma radiation however its establishment is not so promising in naval applications. The combined shielding will include a spherical core of water and on the outer layer of this core there will be a layer of lead.

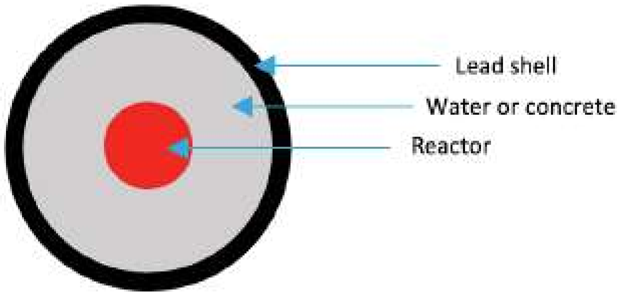

Figure 60: Shielding Concept (Houtkoop, 2022)

The specifications for the shielding (radius, dimensions, weight) were extracted from Houtkoop's paper again:

Table 42: Shielding dimensioning

	Lower Bound	Upper Bound	Exact estimations
Radius (cm)	500	600	550
Weight (tons)	1500	3300	3000
Size (m^3)	780	1700	1500

6.5.8 Layout

Before entering the more technoeconomic aspects of nuclear propulsion, a plot where most of the details analysed before are combined all together must be given. To be more specific, firstly it had to be decided what will be the connection between the reactor, the conversion island, the auxiliary systems, the emergency system and the propeller shaft. Generally, there are two methods for connecting all these components within the same system. Either all the above will be indirectly connected (electrically) or they will be connected directly.

Of course, if every component is connected in a different way, multiple layouts can be created. Nonetheless, two factors must be reviewed thoroughly before any layout is decided. The first factor that has to be evaluated is the optimization of the system and how each additional component contributes to the overall efficiency. A table where approximate values of the efficiency of the values is presented.

Table 43: Efficiency of every compartment

Gearbox	0.98
Generator	0.97
Switchboard	0.98
E-motor	0.97
Shaft	0.99

As Houtkoop implies, the most optimal layout will be the fully electric layout. This layout despite the fact that is not the most efficient (0.90%), it disposes the most compact gearbox and seems the most favourable for a potential connection with batteries for extra supplementation. The layout of course is doubled to eliminate the possibilities of leaving the vessel ungoverned due to a single failure.

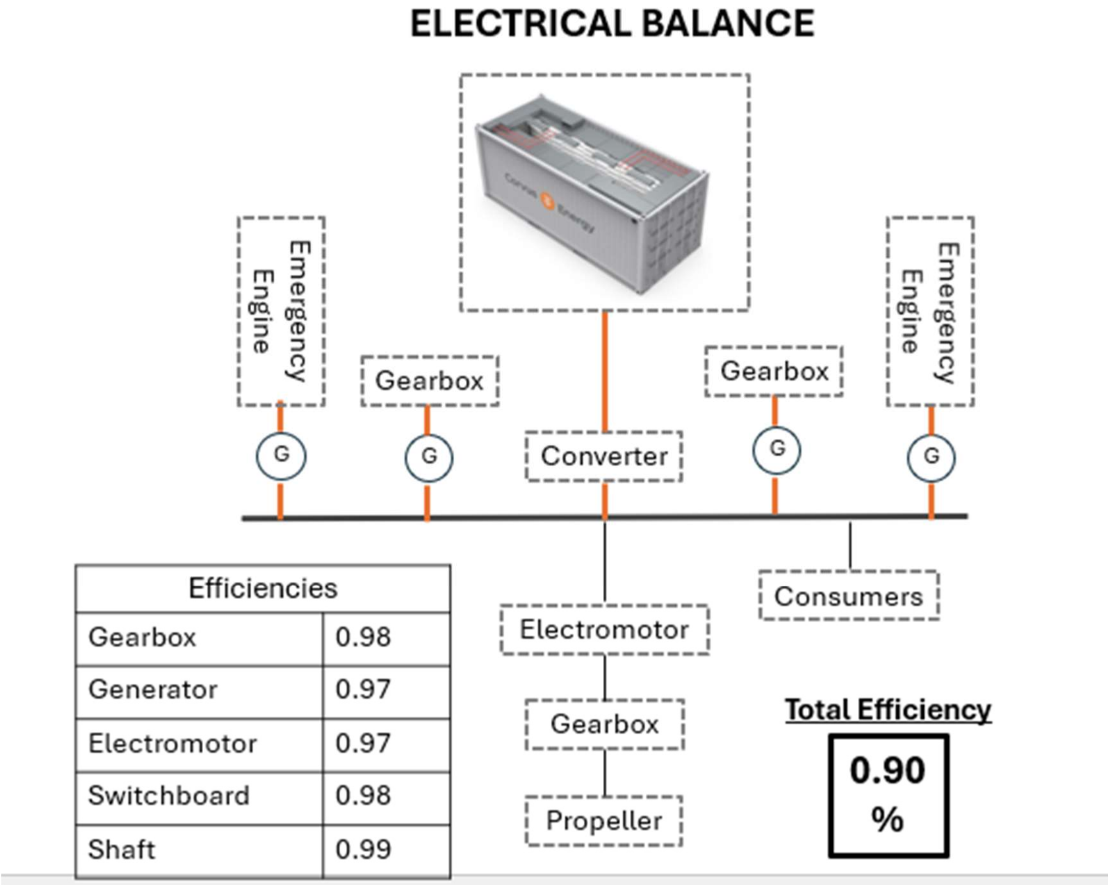

Figure 61: Nuclear-powered Propulsion layout

Finally, a scheme of the nuclear vessel's propulsion and power generation system is presented.

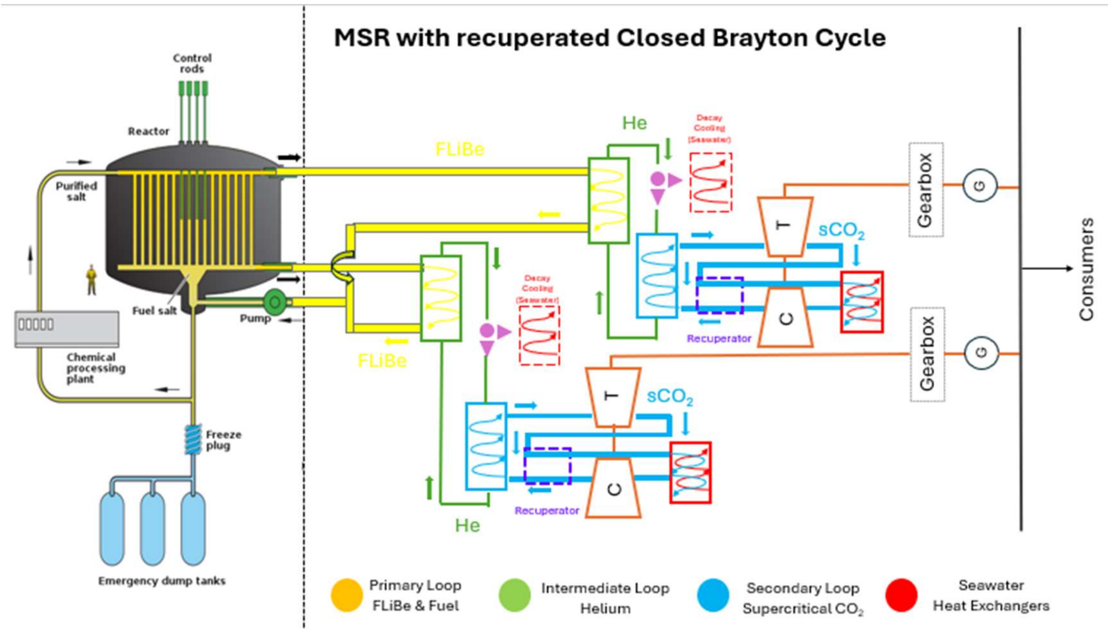

Figure 62: Nuclear Powered Vessel total layout

6.5.9 Weight of components

The table below summarizes the weight of the major components that a nuclear reactor intended for marine use must have. Some of the weights have already been presented while others are extracted from Houtkoop's thesis.

Table 44: Weight of each reactor component

Component	Dimensioning	Amount	Weight (tns)
Reactor	-	1	150
Shielding	-	1	3000
Heat Exchangers	3 sizes of Hel.	8	175
Turbines	1.45 ton/MW	2	82
Generators	0.0027 t/kW	4	317
Electromotor	0.0027 t/kW	2	142
Gearboxes	0.00017 t/kW	4	204
Switchboards	0.5 t/cabinet		24
Emergency power supply	0.0086 t/ kW	2	69
Batteries	26.65/ container	2	54
Total			41

6.5.10 Volume of components

The table below summarizes the weight of the major components that a nuclear reactor intended for marine use must have. Some of the weights have already been presented while others are extracted from Houtkoop's thesis.

Table 45: Volume of each reactor component

Component	Dimensioning	Amount	Volume (m^3)
Reactor	-	1	Included in shielding
Shielding	-	1	1500
Heat Exchangers	3 sizes of Hel.	8	65
Turbines	2.2 m3/MW	2	126
Generators	0.0017 t/kW	4	204
Electromotor	0.0017 t/kW	2	92
Gearboxes	0.00017 t/kW	4	97
Switchboards	0.94 m^3/cabinet	46	44
Emergency power supply	0.011 m^3/ kW	2	88
Batteries	44 m^3/ container	2	8
Total			2257

6.5.11 Waste Management

Nuclear wastes are believed to be the sticking point of this technology and probably one of the most challenging problems that have to be addressed before this technology expands. As stated before, waste is composed of many different levels that are directly related to the vessel's operating profile. Generally, the different categories of waste that are examined in this research are:

- High-Level waste
- Waste produced during fabrication - Tails
- Intermediate Level waste
- Low-Level Waste
- Decommissioning Waste

It has to be noted that only for the high-level waste the selection of a closed over open cycle for the fuel diminishes the estimated waste from 1.5 to 3 times. According to calculations, the waste stream for a 25 year period for a 400k DWT is summarized on this table:

Table 46: Waste Estimation

Waste Categorization	Cumulative Waste for 25 years (in m^3) - Scenario A		Cumulative Waste for 25 years (in m^3) - Scenario B	
	Lower	Higher	Lower	Higher
High Level	3.5	11.5	4.5	15
Tails	2.5	12.5	3.5	20
Intermediate Level & Low Level	40	50	50	60
Decommissioning	550	1200	550	1200
Total	596	1274	608	1295

6.5.12 Finance

This part of the case study examines probably the most determining part of the whole research. Preparing a financial plan for such a venture can be a quite tough task, nevertheless every investor would argue that is more than necessary. As every budget, the costs will be split to CAPEX and OPEX. CAPEX is focused in the initial investment and basically in the cost of building and installing a nuclear reactor of this power, while OPEX is oriented to the maintenance and fuel costs.

6.5.12.1 CAPEX

Before analysing the estimations for the CAPEX costs, it has to be highlighted what will be included within the costs. The prices referred below refer only to components connected with vessel's propulsion and power generation systems. The exact cost breakdown is presented on the table below where any percentage is subject to change. Also, the cost for further development can be decreased or even abandoned since the total number is already overwhelming.

Table 47: Nuclear Reactor's CAPEX breakdown

Design, Establishment, Licensing	12 %
Project Management and Supervision	8 %
Installation of:	
Nuclear Reactor & Shielding	30 % & 5 %
Conversion Island	20 %

Balance of Plant	15 %
Transportation	3 %
Further development	7 %

The construction cost of a nuclear reactor is given in proportion with the installed kWs of the reactor. SMRs cases are once again limited and hence most of the calculations are based on these cases or they derive from estimations. The most optimistic supporters (Energy Options Network's paper) of nuclear technology support that the initial construction costs of an MSR reactor can begin from 3982 $/kWe, while the US Department of Energy do not believe that the price can drop below 6800 $/kWe.

Apart from the installation and construction costs, a significant part that has to be factor in the total price is the decommissioning cost. Even though this cost does not seem so important, actually the deactivation, the removal and the disposal of the reactor is 2000$ /Mwe.

Thus, the total CAPEX costs are recapitulated to the table below:

Table 48: Nuclear Reactor's CAPEX estimations

CAPEX for 400k DWT Bulk Carrier	
Lower Bound ($)	172,512,006
Upper Bound ($)	253,778,946

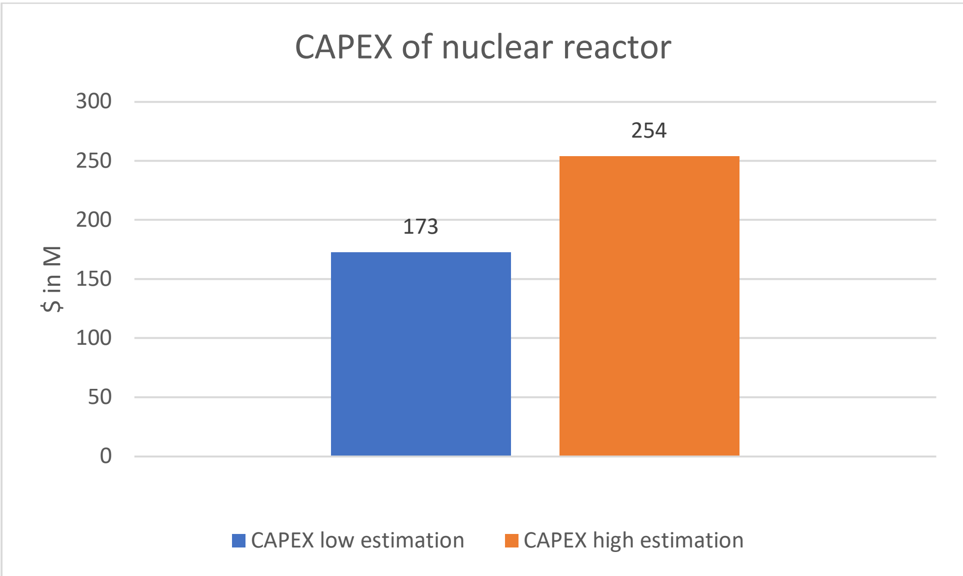

Figure 63: CAPEX estimations for nuclear reactor

6.5.12.2 OPEX

In this section and contrary to the initial investment presented above, all the maintenance and fuel costs are presented. Before defining the fuel cost, the total price/kg of mined Uranium has to be presented. In addition, some extra costs for the procedures that the mined Uranium undergoes have to be counted in. All this information can be found in the table below:

Table 49: Fuel Price per kg and annual estimations for nuclear reactor

Process for producing 1kg of UO_2 from U_3O_8			
Uranium	8.9 kg of U_3O_8 * 91$	810	
Conversion	7.5 kg of U x 15$/kg	112	
Enrichment	7.3 SWU x 52$	380	
Fuel Fabrication	300 /kg	300	
Total ($ per kg)		1,602 $/kg	
Total ($ per year)		Scenario A	Scenario B
		1,100,188 $	1,291,475 $
Molten Salt- FLiBe	306 *150 $/kg	45,961	
	Total ($ per year)	9,162	
Total ($ per year)		Scenario A	Scenario B
		1,109,381 $	1,300,667 $

Considering the consumption of the nuclear fuel for every year the total fuel cost can be calculated. This consumption was calculated before, with the estimations being between 687kg- 806 kg per year according to the scenario. Also, a more precise approach would include the maintenance costs (fixed and variable):

Table 50: Nuclear Reactor's OPEX annual breakdown

Fuel Costs	1602 $/kg + 60 kg of Molten Salt	Scenario A	Scenario B
		1,109,381 $	1,300,667 $
Fixed Maintenance Costs	95 $/ kW/ Year	2,739,659 $/year	
Variable Maintenance Costs	3 $/ MWh	762,928 $/year	
Insurance Costs		1000000 $/ year	
Total ($/Year)		Scenario A	Scenario B
		5,611,969 $	5,803,256 $

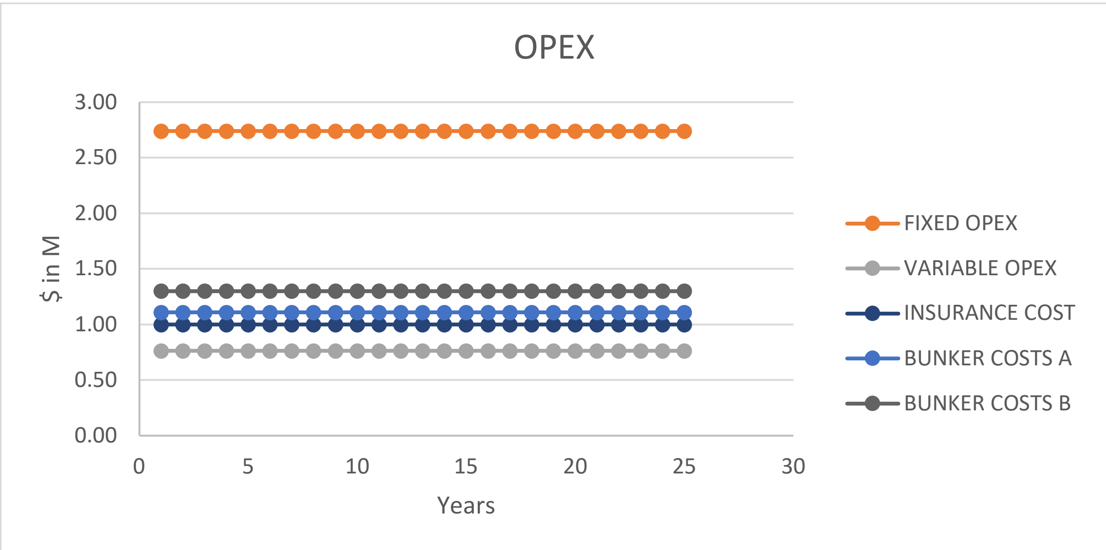

Figure 64: OPEX figures for nuclear-powered vessel

Finally, the OPEX costs for the 25-year period are summarized in the table below:

Table 51: Nuclear Reactor's OPEX estimation for vessel's lifetime

OPEX for 25 years		
25-year period	Scenario A	Scenario B
	140,299,221	145,081,391

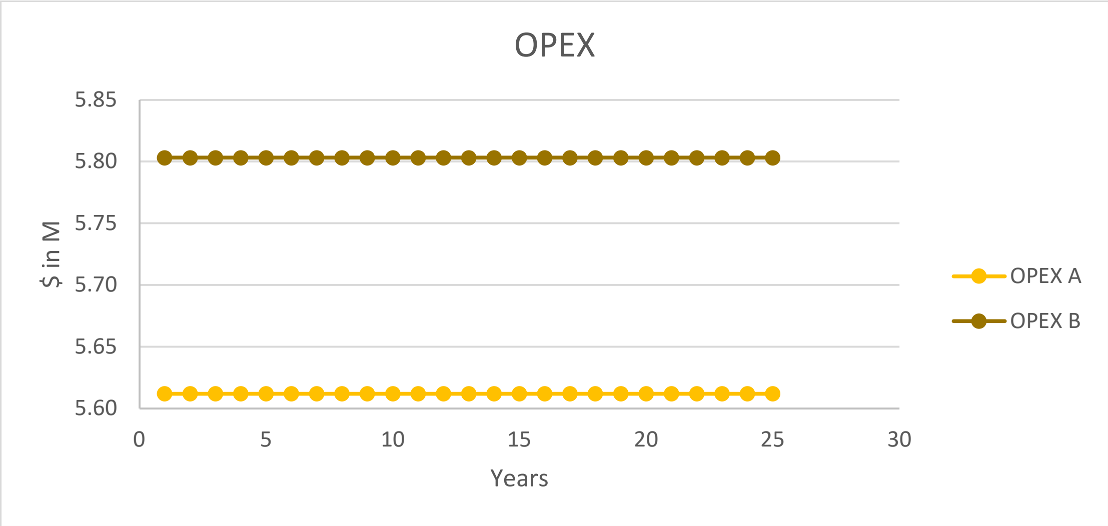

Figure 65: OPEX Cashflows for the two scenarios

6.5.12.3 CAPEX & OPEX & CASHFLOWS

As a result, the sum both of CAPEX and OPEX for a nuclear-powered vessel with two potential operating profiles is calculated and presented below:

Table 52: Total Costs' estimations for nuclear-powered vessel's lifetime

CAPEX & OPEX for 25 Years		
	Scenario A	Scenario B
Lower Bound	312,811,227	317,593,397
Upper Bound	394,078,167	398,860336

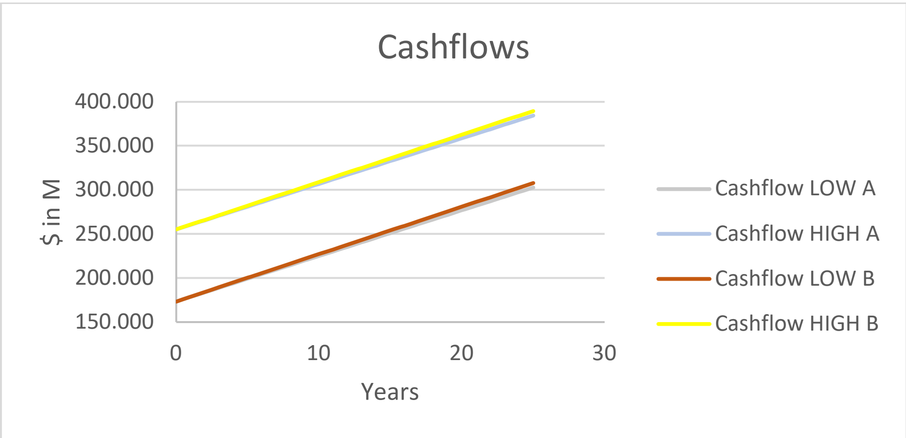

Figure 66: Total Cashflows for two scenarios and two CAPEX estimations

6.6 Conventional engine

If the same vessel is equipped with the conventional diesel engine which was presented in the first section of the case study, then the following figures should be expected for a 25-year period.

6.6.1 CAPEX

To begin with, an estimation for the vessel's CAPEX had to be established. For this reason, in a more than simplified approach a specific figure for the total CAPEX/ kW installed was selected. With the assistance of different researches, the value described above was set to 400$ / Kw (Boulios, 2024). Taking this value as an average for a 25,200 kW installed engine the CAPEX will be approximately:

Table 53: CAPEX of vessel with conventional diesel engine

CAPEX for a 25,200 kW engine (in $M)
11,34 $

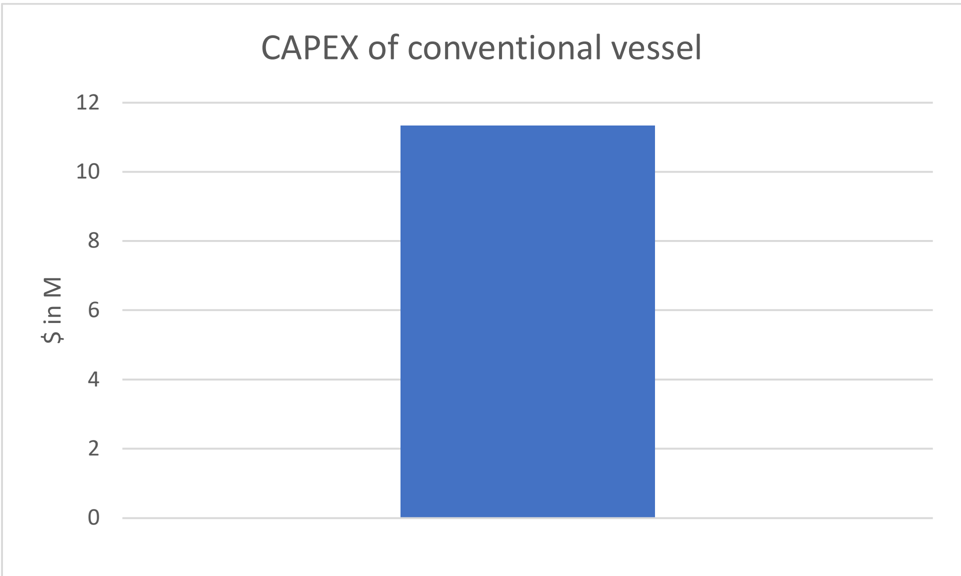

Figure 67: CAPEX estimation for conventional vessel

6.6.2 OPEX

OPEX as it will be proved later is the most expensive cost of a vessel with a conventional diesel engine. As it can be seen, the bunkering costs constitute more than 90% of the total OPEX. It has to be reminded that in this study the costs that are examined are related only to the propulsion system and its maintenance. Insurances, port charges and other expenses are not included.

To calculate the OPEX of this vessel during the 25-year period it was more than necessary to frame the vessel's schedule. In this table the vessel's schedule is summarize both for the duration of a round trip but also for period of a whole year. As it was stated in the beginning of this case study, the vessel with such itinerary, is capable of completing more than 5 round tris/ year (5.56 actually).

Taking as granted the information from the table above, the next step was to estimate the mass of fossil fuel that the vessel needs for the period of 1 year. The mass will be given if the energy demands and the SFOC of the engine are both known. Vessel's energy demands are calculated by multiplying the power that every consumer absorbs times the hours that this consumers operates. Then we will multiply again this figure with the SFOC to get the desired outcome. It has to be highlighted that since the Auxiliary Engines are powered on a daily basis, a different SFOC had to be used in the calculations. This also happened during the days that the vessel is within port limits. The two SFOC are extracted from MAN's catalogue.

Table 54: SFOC of the main and auxiliary engines

SFOC (gr/kWhr)	
VLSFO	MGO
181.4	195

With this method the total grams of fossil fuel are calculated in the table below:

Table 55: Annual fuel mass for the conventional engines in grams and tons

FUEL MASS DEMANDS (gr/ tons/ year)			
19012200406	gr	19012.2	tons
76391552.8	gr	76.4	tons
17391488241	gr	17391.5	tons
76391552.8	gr	76.4	tons
854100000	gr	854.1	tons
37410571753	gr	**37410.6**	tons

Having this information available and with the current prices of VLSFO and MGO in the port of Qingdao the final estimation for the OPEX is given below:

Table 56: Current Bunkering Prices in Qingdao Port (2024)

BUNKER PRICES ($/ton)	
VLSFO	MGO
674	802

Table 57: Fuel Costs of vessel with conventional engine

Total OPEX in $M	
1 year	**26.8**
25 years	**670**

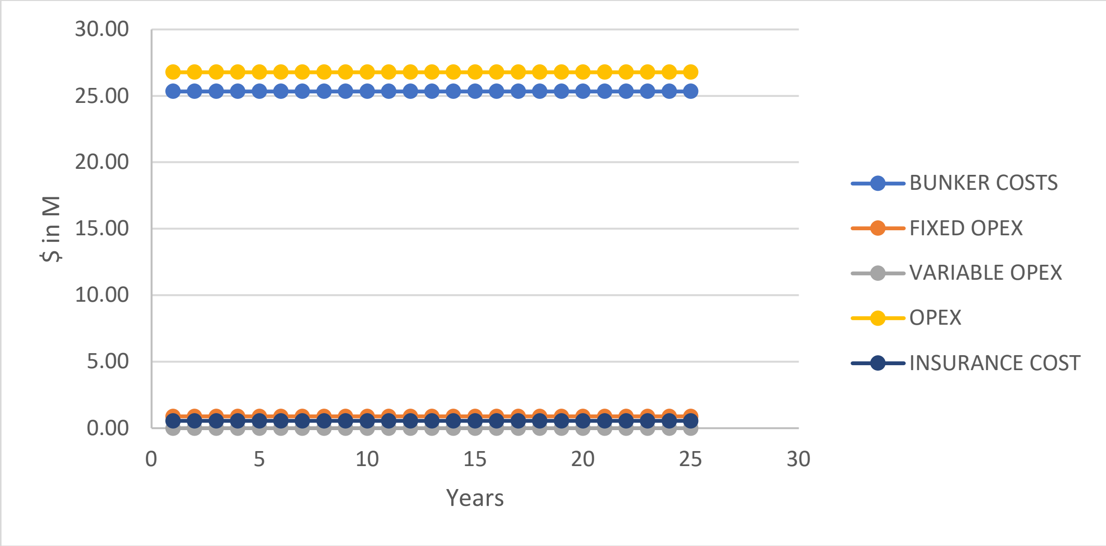

Figure 68: OPEX total costs for conventional vessel

The sum of CAPEX and OPEX is summarized below along with a timeline that depicts better how and when there is cashflow.

Table 58: Total Costs of vessel with conventional diesel engine

CAPEX & OPEX (in $M)
681.1

6.6.3 CAPEX & OPEX & CASHFLOWS

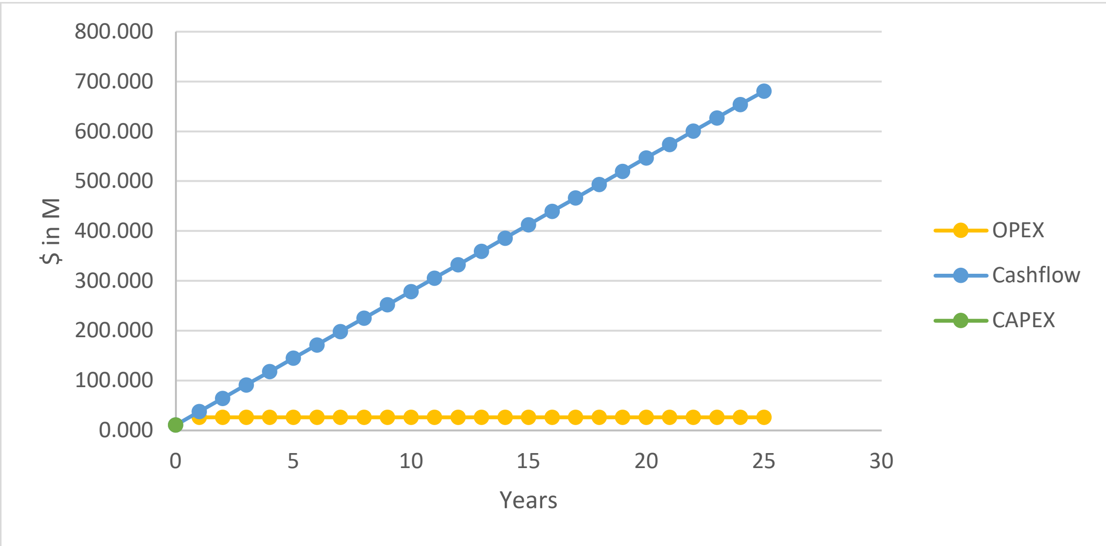

Figure 69: Total Cashflows for conventional engine

What can easily be concluded from the next graph are the payback periods for each case:

Table 59: Payback Periods

Low Capex A, B	7.8 years
High Capex A, B	11.7 years

6.7 Financial comparison

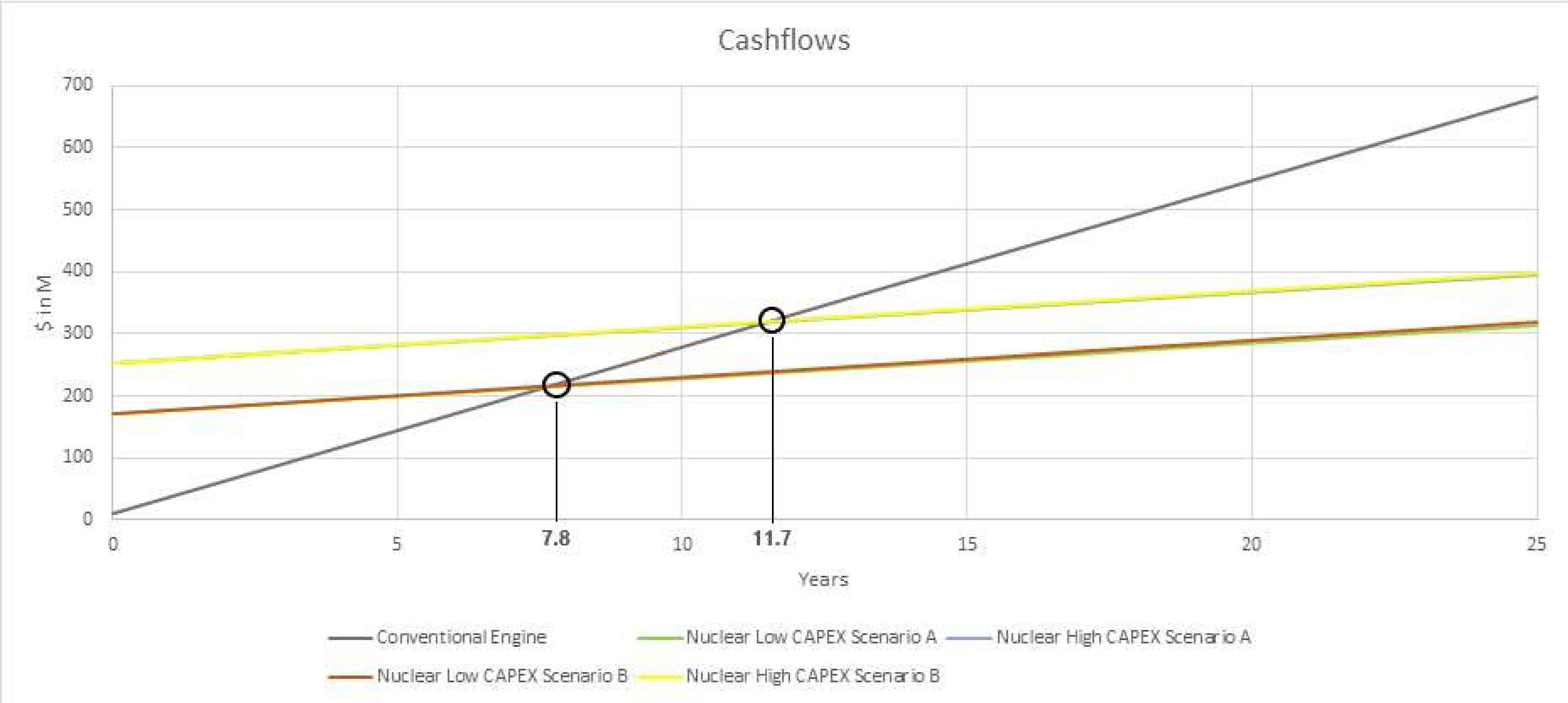

Figure 70: Financial Comparison between the two options

6.8 Environmental comparison

Apart from the financial perspective that as explained is very promising, the implementation of nuclear energy to the Maritimes sector aimed to the reduction of the environmental footprint. As it was ensured from the very beginning of this report, nuclear powered vessels were estimated to be more than competitive as far as CO_2 emissions were concerned. To prove this statement, a guaranteed methodology is to compare the emissions that correspond to the tons of fossil fuel burned. For example, with a typical burn-up of 45,000 MWd/t, one tonne of natural uranium made into fuel will produce as much electricity as 17,000 to 20,000 tonnes of black coal.

When the problem of CO_2 arises, three notions have to be explained first. As International Council of Clean Transportation describes, there are three classifications within emissions. These are: the well to tank emissions (well-to-tank), the tank to wake emissions (tank-to-wake) and the well to take emissions (well to wake). Of course, all these classes of emissions are deeply connected all together. Actually, each one of them is a link in a common chain. Well to tank emissions include the emissions generated during the production and the transportation of the fuel, tank to wake include the emissions created during the burning of the fuel and well to wake emission are the summarization of the other two. The last one contains the equivalent CO_2 generated during the whole lifetime of the fuel. The upper values both for VLSFO and MGO can be seen below:

Table 60: CO2 emissions per type of fossil fuel

CO_2 emissions (g CO_2/ g Fuel)		
Type	VLSFO	MGO
Well-to-tank	0.675	0.756
Tank-to-wake	3.114	3.206
Well-to-wake	3.789	3.962

Now the two cases have to be compared. For the case of the conventional engine the total mass of each fuel was calculated on an annual basis. The estimations showed that a 400k DWT Bulk Carrier produces during its lifetime 3550kt of CO_2 approximately.

Table 61: CO¬2 total emissions in grams per year and per vessel's lifetime

CO_2 Equivalent	
In kt/ year	141.9
In kt for 25 yrs	3548.1

At the same time and since nuclear fuel also produces some gr of CO_2 during the fuel production and during the reactor's construction there is another transformation that leads to equivalent kt of CO_2. Only that in this case the equalization is given in $grCO_{2e}$/kWh. According to the World's Nuclear Association, for every kWh produced from nuclear plant 0.012 grs of CO_2 are produced.

Hence, for our 29,000kW reactor under a 25-year-period then estimated amount of equivalent CO_2 can be seen below:

Table 62: Comparison with emissions from the nuclear-powered vessel

CO_2 Equivalent	
In kt/ year	2.52
In kt for 25 years	63
Decline	98.2%

The difference in the period of the 25 years is chaotic. In pure figures, the decline in the pollutants exceeded 97% which is a tremendous step for the whole industry. Moreover, a considerable aspect of the emitted CO_2 is the financial impact in the OPEX of the vessel in the case that carbon taxes are included in the international legislation. Assuming that every ton of CO_2 costs 75$ approximately, the annual OPEX of the vessel will be increased by 10,642,500$ or by 40%. The updated charts for the new version (inc. carbon taxes) can be seen below:

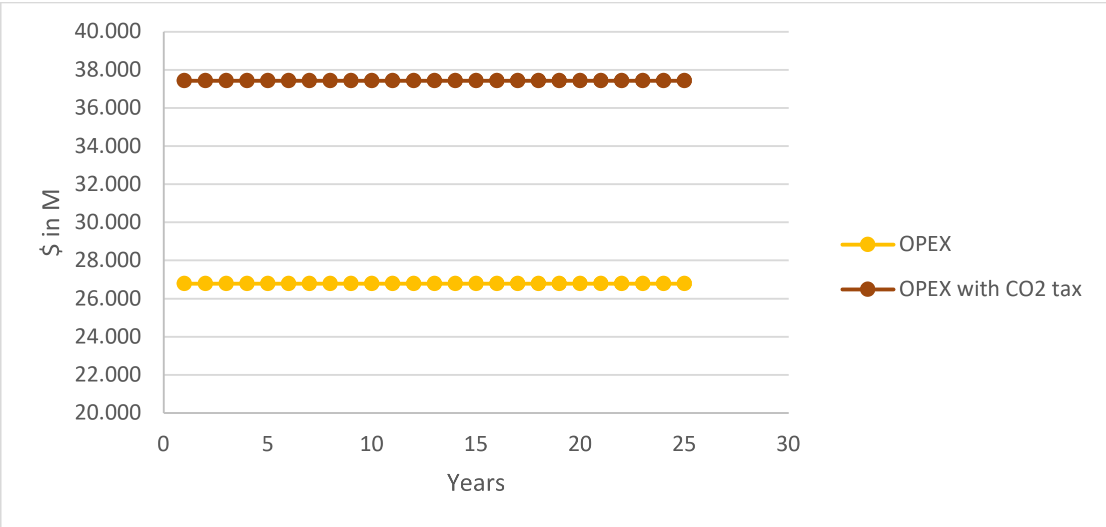

Figure 71: Difference in OPEX with and without carbon tax

What can easily be concluded from the next graph are the payback periods for each case:

Table 63: Payback Periods

	Without Tax	With Tax
Low Capex A, B	7.8 years	5.1
High Capex A, B	11.7 years	7.6

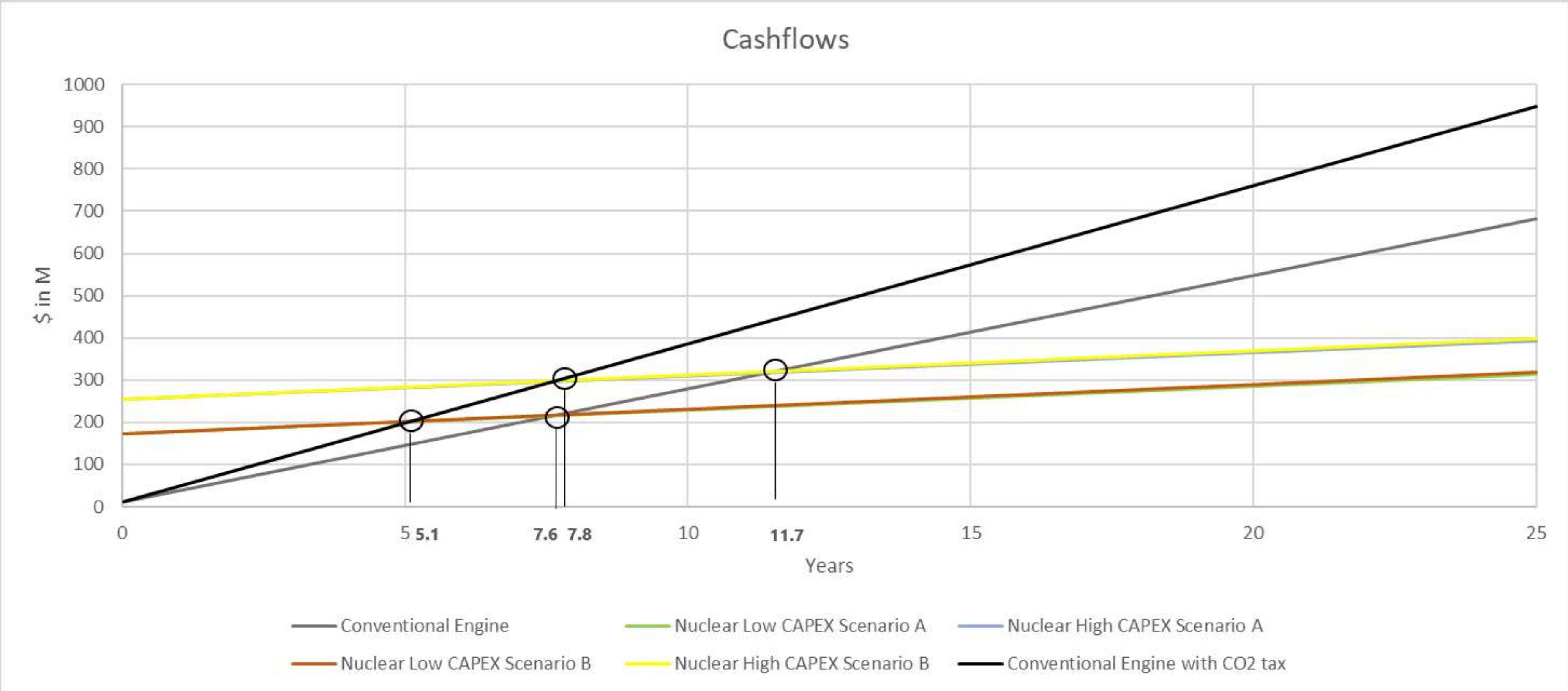

Figure 72: Comparison of nuclear reactor with conventional engine at every scenario

6.9 Dimensions' comparison

Typical values for the weight and the volume of a Valemax vessel of 400k DWT are briefly presented in the next table.

Table 64: Dimensions of machinery for the conventional vessel

Component	Weight (tons)	Volume (m^3)
Main Engine	1115	880
Aux Engine	13	17
Fuel	7000	6980
Electrical Components	8	17
Total	8136	7894

Given the values calculated for an identical vessel powered by a nuclear reactor, an initial prediction for the trend of the volume and the weight can be done.

Table 65: Dimensions' comparison between vessels

Case	Weight (tons)	Volume (m^3)
Vessel with Conventional Engine	8136	7984
Nuclear Powered Vessel	4122	2236
+/- Difference in %	-49%	-72%

From the upper results, it can be concluded that a nuclear-powered vessel will require less space and weight for its machinery. Thus, this is translated to a potential increase in the cargo holds that will automatically lead to an overall increase of the Deadweight.

7. CONCLUSIONS

To sum up the key points of this thesis, nuclear power is a challenge for the Shipping Industry. Not only there are the visible obstacles both from a technical and a legislating perspective, but in parallel many other problems will certainly arise. Even though the application of this powerful form of energy was never truly welcomed in this sector, the fundaments exist and should be used as a guide for any further attempts. Governments along with the scientific society have no other choice but to cooperate and encourage any effort to implement nuclear marine propulsion in a commercial level.

The idea of the expansion of nuclear power on a commercial level should not be abandoned. The applications from other industries are numerous and the power generation from the Gen-II and Gen-III reactors on an industrial scale provide a strong background in knowledge and experience. Nonetheless, the fresh air that Gen-IV models bring with them, suggest the path that must be developed. Increased safety, improved consumption, reduced size, tempting budgeting and of course minimized environmental impact are only some of the key features that characterize the new designs.

This thesis focused only on one design out of the six, the Molten Salt Reactor. The selection of this design was based on subjective criteria and for this reason all the designs must be examined in more detail. The results of this research seem to be the proof to all these expectations. The harmful emissions could be reduced by up to 97%, while at the same time the OPEX of the vessel are likely to decrease significantly. Based on the most optimistic projections of this thesis, the total reduction in the costs of the vessel during its whole lifetime (25 years) could reach 40%. On top of these quite impressive measures, there are certain indications that all of the advantages described above will be accompanied by reduction in the total weight and volume of the new system. In more detail, a summary table with the key values is presented.

Table 66: Comparison of the main values

Propulsion Method	CO_2 emissions (in ktns)	OPEX (in \$M)	CAPEX (in \$M)	Weight (in tons)	Volume (in m^3)
Conventional Engine	3548.1	670	11	8136	7984
MSR	63	145	253	4122	2236
Difference in %	-98.2	-78.3%	+95%	-49%	-72%

Apart from the techno economical terms which were discussed, the safety of the reactor is another field where the MSR excels. The reactor will not rely solely on its shielding in case radioactive material escapes, but the increased safety that the salt provides along with the freeze plugs and the natural circulation are significant upgrades. The proliferation resistance of the MSR reactor can also be the x-factor, especially if the thorium with a closed fuel cycle is employed (about 23% extra reduction only in the fuel costs).

To conclude with, as climate crisis escalates on such an extent, new serious actions must be undertaken on the near future. Unfortunately, there is no plenty of time to set a totally new plan, nevertheless this does not mean that irrational decisions must be made in hurry. Of course, Gen-IV reactors cannot be granted with any indemnity as far as their environmental effects are concerned. However, the findings of the case study point towards a more favourable direction and undoubtedly the establishment of nuclear power into commercial shipping is a project that worths the risk.

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