Feasibility of commercial maritime nuclear propulsion

JF Marais

20513607

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Supervisor: Dr B Botha

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"Possession of surplus energy is a requisite for any kind of civilization, for if man possesses merely the energy of his own muscles, he must expend all his strength - mental and physical - to obtain the bare necessities of life." Hyman G Rickover, 1957

ABSTRACT

The goal of this study is to make the unique benefits of modern nuclear technology available to a wider sphere. The basic hypothesis is that the time is ripe to re-investigate nuclear propulsion for commercial shipping. As the pressure on fossil fuel is mounting, both in terms of supply as well as pollution prevention and carbon emission control, it is imperative that nuclear power be made available for large-scale propulsion. Making a nuclear engine small enough to power a car is, however, still an engineering challenge. Large ships do, however, pose a platform that is sufficiently large to contain a nuclear reactor.

The ultimate goal will be to design a nuclear power plant that is as safe as a conventional diesel engine, will fit into the same space and mass constraints, and is cost-competitive over the life of the ship.

During this study, the shipping industry was reviewed to select the Very Large Crude Carrier class of oil tankers as a favourable candidate for nuclear propulsion. A set of design-driving requirements was formulated for the propulsion of this type of ship. A stepwise design process was followed to realize a reactor and propulsion system, using proven technology and components as far as possible.

The selected configuration is a prismatic fuel high-temperature gas-cooled reactor driving a standard steam turbine in a Rankine cycle configuration. This configuration is compared with a diesel engine alternative to verify its life-cycle cost competitiveness.

It was found that even under conservative assumptions, using current fuel oil price and cost of capital, the nuclear propulsion system is a feasible alternative to a diesel engine over the life of the ship.

UITREKSEL

Die doelwit van hierdie studie is om die unieke voordele van moderne kernkragtegnologie aan 'n groter mark beskikbaar te stel. Die basiese aanname is dat die tyd nou ryp is om kernkrag te her-oorweeg vir die aandrywing van kommersiële skepe. Daar is tans baie groot en toenemende druk op die koolstofgebaseerde energiebronne. Druk kom as gevolg van beide die kleinerwordende bronne van olie en steelkool, asook die toenemende klem op voorkoming van besoedeling en druk om koolstofdioksiedvrystelling te beperk. Hierdie faktore noop ons om kernkrag vir grootskaalse vervoerdoeleindes te oorweeg. Tans lyk dit nie moontlik om 'n kernreaktor klein genoeg te maak om in 'n motorkar te kan pas nie. Groot skepe bied wel genoeg ruimte om 'n reaktor te kan bevat.

Die eind-doel is om uiteindelik 'n kernkragaandrywingstelsel te ontwerp wat net so veilig is as 'n ekwivalente dieselenjin, wat in dieselfde spasie kan pas, nie meer weeg nie, en wat nie meer sal kos oor die lewensiklus nie.

Hierdie studie het die skeepsindustrie bestudeer en die VLCC olietenkerklas as 'n gunstige kandidaat vir kernkragaandrywing gekies. 'n Stel vereistes is afgelei vir die aandrywingstelsel vir hierdie tipe skepe. 'n Sistematiese ontwerpsproses is toe gevolg om die reaktor en hitte-omsettingstelsel te ontwerp. Bestaande komponente en bewese tegnologie is sover moontlik gebruik om sodoende die risiko's so laag as moontlik te hou.

Die gekose argitektuur bestaan uit 'n hoë-temperatuur-gasreaktor met blokvormige brandstof. Die hitte-omsettingstelsel bestaan uit 'n Rankine-siklus met 'n standaard stoomturbine. Die lewensikluskoste van hierdie stelsel is vergelyk met die van 'n soortgelyke dieselenjin om die ekonomiese lewensvatbaarheid te bepaal.

Selfs met inagneming van baie konserwatiewe aannames, en teen huidige olieprys en rentekoerse, blyk dit dat die kernkragaandrywingstelsel goedkoper is as die diesel-alternatief oor die lewensduur van die skip.

Table of Contents

1. BACKGROUND 15

1.1 INTRODUCTION 15
1.2 BACKGROUND 15
1.2.1 Advances in Nuclear Technology 15
1.2.2 Increasing Scarcity of Fossil Fuel 16
1.2.3 Focus on Air Pollution and Carbon Emission 17
1.2.4 Public Perceptions of Nuclear Energy 17
1.2.5 Energy Density 18
1.3 PROBLEM STATEMENT 18
1.4 OBJECTIVE 19
1.5 METHODOLOGY 19
1.6 DOCUMENT OVERVIEW 19

2. LITERATURE OVERVIEW 20

2.1 HISTORICAL COMMERCIAL NUCLEAR SHIPS 20
2.2 ARCTIC ICE-BREAKERS 21
2.3 MILITARY NUCLEAR SHIPS 21
2.4 CURRENT SMALL NUCLEAR REACTOR DEVELOPMENT PROJECTS 21

3. REQUIREMENTS ANALYSIS 23

3.1 TARGET MARKET INVESTIGATION 23
3.1.1 Oil Tankers 25
3.2 PROPULSION REQUIREMENTS 28
3.2.1 Propulsion Force 28
3.2.2 Propulsion Power 29
3.2.3 Propeller Speed 30
3.3 AUXILIARY POWER 30
3.4 DIRECT VERSUS INDIRECT DRIVE 31
3.5 ENGINE SIZE AND WEIGHT CONSTRAINTS 32
3.6 COST CONSIDERATIONS 32
3.7 TYPICAL MISSION PROFILE 33
3.8 VALUE SYSTEM FOR A SHIP PROPULSION UNIT 35
3.9 SAFETY ASPECTS 35
3.9.1 Nuclear Safety in the Commercial Maritime Domain 35
3.9.1.1 Risk of dangerous nuclear exposure 35
3.9.1.2 Simplicity in operation and maintenance 35
3.9.1.3 Passive safety 35
3.9.1.4 Radiation shielding 36
3.9.1.5 Security 36
3.9.2 Regulatory Requirements 36
3.10 TABLE OF INITIATING REQUIREMENTS 37

4. CONCEPTUAL DESIGN 39

4.1 FUEL OPTIONS 40
4.1.1 Comparison of Prismatic versus Pebble Fuel in the Context of a Ship 41
4.1.2 Fuel Management 43
4.1.3 Pebble Refuelling Options 43
4.1.3.1 MEDUL (Mehrfachdurchlauf - German for 'multi-pass) 43
4.1.3.2 Once Through Then Out (OTTO) Cycle 43
4.1.3.3 Peu à Peu (little by little in French) (PAP) 44
4.1.3.4 Ordered pebble bed 44
4.1.4 Prismatic Refuelling Options 44
4.1.4.1 Traditional approach 44
4.1.4.2 One-piece removal 45
4.1.5 Conclusion on Fuel Options 45
4.2 PROPULSION MECHANISMS 46

4.2.1 Brayton Cycle Turbine 46
4.2.2 Direct Cycle Steam Turbine 47
4.2.3 Direct Drive Concept 48
4.2.3.1 Derivation of direct-drive propulsion force 48
4.2.3.2 Conclusion on direct-drive concept 51
4.2.4 Rankine Cycle Steam Turbine 52
4.2.4.1 HTR MODUL 53
4.2.4.2 MHTGR 54
4.2.4.3 Rankine ship engine design 55
4.2.4.4 Sizing and costing 58
4.3 NUCLEAR REACTOR 64
4.3.1 Design of Reactor and Support Systems 67
4.3.1.1 Core size 67
4.3.1.2 Proposed core layout 67
4.3.1.3 Reflector dimensions 68
4.3.1.4 RPV 69
4.3.1.5 Decay heat removal system 70
4.3.1.6 Radiation shielding 72
4.3.2 Cost Model 77
4.3.2.1 Material cost values 77
4.3.2.2 Reactor component cost assumptions 79
4.3.2.3 Reactor support system cost assumptions 82
4.3.2.4 Summary of reactor design 83
4.3.3 Fuel Temperature Analysis 84
4.3.3.1 Conduction 85
4.3.3.2 Convection 86
4.3.4 Pressure drop determination 88
4.3.5 Burn-up and Fuel Cycle 88
4.3.6 Reactor Orientation 89
4.3.6.1 Benefits of a horizontal reactor 89
4.3.6.2 Disadvantages of a horizontal reactor 90
4.3.6.3 Layout 90
4.3.6.4 Issues that need special attention in the horizontal reactor concept 91
4.3.6.5 Conclusion on orientation 93
4.4 SUMMARY OF PROPULSION SYSTEM DESIGN 94
4.5 LOGISTICS 97
4.5.1 Refuelling 97
4.5.2 Conclusion on Logistics 98

5. ECONOMIC INVESTIGATION 99

5.1 BENCHMARKS 99
5.1.1 MRX [87] 99
5.1.2 NEREUS Project [24] 99
5.2 COST MODEL FOR A DIESEL ENGINE ALTERNATIVE 101
5.3 COST MODEL FOR THIS DESIGN 101
5.3.1 Capital Cost 101
5.3.2 Running Cost 101
5.3.2.1 Maintenance and operation 101
5.3.2.2 Fuel cost 101
5.3.2.3 Opportunity cost of fuel space 102
5.3.2.4 Opportunity cost of carbon tax 102
5.3.2.5 Spent-fuel handling cost 102
5.3.3 Decommissioning Cost 102
5.3.4 Licensing Cost 103
5.4 COST COMPARISSON BETWEEN NUCLEAR AND DIESEL PROPULSION 103
5.5 CONCLUSION ON ECONOMIC INVESTIGATION 105

6. PROJECT RISK MATRIX 106

7. CONCLUSIONS 107

8. FUTURE WORK 108

9. BIBLIOGRAPHY 109

10. APPENDIX A: DIAGRAMS OF RELEVANT REACTOR PRECEDENTS 113

11. APPENDIX B: FUEL CALCULATIONS 118

12. APPENDIX C: RANKINE CYCLE EES MODEL 120

13. APPENDIX D: SHIP REACTOR CALCULATIONS 127

14. APPENDIX E: CURRENCY EXCHANGE RATES USED IN THIS STUDY 129

FIGURES

Figure 1: "Atoms for peace" postcard 15
Figure 2: Annual Oil Production 16
Figure 3: Public Acceptance of Nuclear Energy 18
Figure 4: Jacobs Power Conversion Cycle 22
Figure 5: Jacobs Nuclear Ship Engine 22
Figure 6: Size Comparison of Five of the Largest Ships [79] 25
Figure 7: Distribution of Tanker Classes (Number of Ships) 26
Figure 8: Distribution of Tanker Classes (Deadweight Tonnage) 27
Figure 9: Dates of Tanker Deliveries 27
Figure 10: VLCC Supply/Demand vs Earnings [1] 28
Figure 11: Propulsion Power Demand for VLCCs and ULCCs 29
Figure 12: Iran Delvar VLCC 30
Figure 13: BP VLCC 30
Figure 14: Typical Indirect coupling 31
Figure 15: Xenon Dead Time [47] 31
Figure 16: A typical 31 MW Wärtsilä Engine 32
Figure 17: Typical Mission Profile 34
Figure 18: TRISO-coated Particles 40
Figure 19: HTTR Fuel Configuration [44] 45
Figure 20: VLCC Propeller 46
Figure 21: Direct Brayton Cycle 47
Figure 22: Direct Steam Cycle 47
Figure 23: Prototype Pluto Engine 48
Figure 24: Direct-drive Concept 48
Figure 25: Direct Propulsion Force as Function of Heat Rate 51
Figure 26: Direct Propulsion Force as Function of Exit Steam Quality 51
Figure 27: A Standardized Marine Steam Turbine Unit (Dresser-Rand) 52
Figure 28: Rankine Cycle T-S Diagram 53
Figure 29: MHTGR Steam Generator Temperature Profile 55
Figure 30: Rankine Ship Engine Diagram 57
Figure 31: Heatric Compact Heat Exchanger 60
Figure 32: Compact Heat Exchanger Top View 60
Figure 33: Compact Heat Exchanger Layout 61
Figure 34: Siemens SST-600 Steam Turbine 62
Figure 35: Proposed Core Layout 68
Figure 36: Reflector saving as function of reflector thickness 68
Figure 37: Reactor Layout 70
Figure 38: Neutron Flux-to-dose Rate Conversion 73
Figure 39: NB-36H Shielded cockpit 74
Figure 40 : Savannah Neutron Shield 75

Figure 41: Decay Heat Removal and Shielding (Side View) 76
Figure 42: HTTR Fuel Compact [75][71] 84
Figure 43: Heat Transfer Model 85
Figure 44: Fuel Temperature Profile 87
Figure 45: Frontal View of Centre of Gravity Change 89
Figure 46: Reactor and Power Conversion Unit Next to a Typical Oil Tanker 90
Figure 47: Reactor Upright Inside Ship 91
Figure 48: Horizontal Reactor Inside Ship 91
Figure 49: Horizontal Control Rods 92
Figure 50: Vertical Control rods 92
Figure 51 CAD Model: Section 95
Figure 52: CAD Model: Layout 96
Figure 53: MRX Reactor Removal Concept [87] 98
Figure 54: MRX Cost Comparison 99
Figure 55: LS180 price [$/ton] 100
Figure 56: Annual Cost Comparison between Nuclear and Diesel-propulsion Options 104
Figure 57: Comparison of Annual Costs 105
Figure 58: HTGR Reactor Layout [29] 113
Figure 59: HTGR Core Layout – Top View [59] 114
Figure 60: HTTR Reactor Pressure Vessel [36] 115
Figure 61: HTTR Core Internals [36] 116
Figure 62: HTTR Horizontal Cross-section [36] 117
Figure 63: Conceptual Reactor Design by Lobet et al. [48] 117

Tables

Table 1: Characteristics of Commercial Nuclear Ship 20
Table 2: Small Nuclear Power Reactors 21
Table 3: Ship Categories 24
Table 4: Oil Tanker Types 25
Table 5: Resistance Forces 28
Table 6: Size and Weight Precedents 32
Table 7: Fossil Fuel Equivalent Engine Cost 33
Table 8: VLCC Routes [14] 34
Table 9: Initiating Requirements 37
Table 10: Fuel-related Requirements 40
Table 11: Comparison between Ceramic-coated Fuel and Metallic Pin Fuel 40
Table 12: Comparison between Pebble and Prismatic Fuel 41
Table 13: HTR MODUL Parameters 53
Table 14: MHTGR Steam Generator Parameters 54
Table 15: Rankine Ship Engine Parameters 55
Table 16 : Rankine Engine Equipment Power Rating 58
Table 17: Blower Sizing 58
Table 18: Shell in Tube Steam Generator 59
Table 19: Rankine Engine Equipment Size 62
Table 20: Rankine Engine Cost Model 63
Table 21: Reactor Requirements 64
Table 22: Reactor Precedents 64
Table 23: Comparison of Graphite and Beryllium Characteristics [26] 69
Table 24: Amount of Water Required for Passive Decay Heat Removal for Different Time Durations. 71
Table 25: Fuel Cost Calculations 78
Table 26: Fuel Characteristics 79
Table 27: Reactor Sizing and Costing 81
Table 28: Reactor Cost Summary 83
Table 29 : TRISO Burn-up 88
Table 30: Advantages and Disadvantages of Horizontal and Vertical Control Rods 92
Table 31: Total Mass budget 94
Table 32: NEREUS Capital Cost Structure 100

Table 33: NEREUS Life-cycle Cost Comparison to Diesel Alternative 100
Table 34: Summary of Capital Cost 101
Table 35: Annual Comparative Cost Components 104
Table 36: Risk Matrix 106
Table 37: Currency conversion factors 129

ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
BWR	Boiling-water Reactors
CAD	Computer-aided Design
DIA	Diameter
DPP	Demonstration Power Plant
DWT or dwt	Dead Weight Ton
EES	Engineering Equation Solver (software provided by F-Chart Software Corporation)
EI&C	Electrical Instrumentation and Control
EUA	European Union Allowances
GA	General Atomics
HFO	Heavy Fuel Oil
HPC	High-pressure compressor
HTGR	High-temperature Gas-cooled Reactor
HTTR	High-temperature Test Reactor
HVAC	Heating, Ventilation and Air-conditioning
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IMO	International Marine Organization
JAERI	Japan Atomic Energy Research Institute
KAERI	Korea Atomic Energy Research Institute
LMR	Liquid Metal Reactor
LPC	Low-pressure Compressor
MHTGR	Modular High-Temperature Gas-Cooled Reactor
MSR	Molten Salt Reactor
NEREUS	Naturally safe, Efficient, Reactor, Easy to operate, Ultimately simple and Small
OECD	Organization for Economic Cooperation and Development
OTTO	Once Through Then Out
PBMR	Pebble Bed Modular Reactor
PBMR (Pty) Ltd	Pebble Bed Modular Reactor (Pty) Ltd
PWR	Pressurized Water Reactor
RFR	Required Freight Rate

RIT	Reactor Inlet Temperature
ROT	Reactor Outlet Temperature
RPV	Reactor Pressure Vessel
SWU	Separative Work Unit (to enrich uranium)
TEU	Twenty Feet Equivalent Unit
TRISO	TRIstructural ISOtropic
ULCC	Ultra-large Crude Carrier
USA	United States of America
VLCC	Very Large Crude Carrier

LIST OF SYMBOLS

Symbol	Definition
°C	Degrees Celcius
A	Area [m^2] or Atomic Mass number
b	Reflector thickness
C_F	Resistance coefficient
D	Diameter [m]
D_k	Hydraulic diameter
D^R	Diffusion constant
E_D	Energy produced by decay
F	Force [N]
f	Friction factor
H	Height [m]
h	Convection coefficient
h_{fg}	Enthalpy of vaporisation [J/kg]
i	real interest rate
k	Conduction coefficient
Kn	Knots
L	Length
L^R	Diffusion length [cm]
M	Mass [kg]
$\dot{m}$	Mass flow rate [kg/s]
MWt	Mega Watt (thermal power)
MWe	Mega Watt (Electrical power)
Nu	Nusselt number
n	number of periods
n^{th}	Refers to a serial production unit (excluding the first few prototype units)
P	Pressure [P]
dP or ΔP	Pressure differential [P]
P_D	Power generated by decay
Pr	Prandtl number
Q	Power

$\dot{Q}$	Heat rate [W]
q'''	Volumetric power density
R or r	Radius [m]
R_A	Air resistance [N]
R_{drag}	Drag resistance [N]
R_E	Eddy resistance [N]
R_F	Frictional resistance [N]
R_T	Total resistance (or Towing Resistance) [N]
R_W	Wave-making resistance [N]
Re	Reynolds number
Rpm	Revolutions per minute
S	Entropy [J/kg . K]
S	Surface area
Sv	Sievert
T	Temperature [°C]
V	Speed or velocity [m/s]
$\dot{V}ol$	Volume rate [m^3/s]
Vol	Volume
W	Work [Watt]
x	Quality factor of steam
Chemical symbols	
NOx	nitrogen oxide
PyC	Pyrolitic carbon
SiC	Silicon carbide
SOx	sulphur oxide
UF_6	Uranium hexafluoride
UO_2	Uranium di-oxide
Greek letters	
α	Proportion
δ	Reflector savings
η	Efficiency [%]
ρ	Density [kg/m^3]
Σ_s	Scattering macroscopic cross-section [cm^{-1}]

Σ_t	Total macroscopic cross-section [cm^{-1}]
Σ_{tr}	Transport cross section
Ø	Diameter
ϕ_T	Thermal neutron flux [neutrons per cm^2 per s]
Monetary units	
$	US Dollar
$m	Million US dollar
€m	Million Euro
Rm	Million Rand
£m	Million British pound

1. BACKGROUND

1.1 INTRODUCTION

Reading the literature from the 1960s about the then "*new, emerging nuclear industry*", it is clear that there was an expectation that nuclear energy would revolutionize the world to a much larger extent than what we have experienced to date. Many projects and campaigns were launched such as "Atoms for Peace" (see Figure 1) and "The Friendly Atom" to show the public the wonderful advances that would be possible by harnessing nuclear energy. Many of these nuclear dreams have been forgotten, at least partly, because of the public perception that nuclear energy is too dangerous. Nuclear energy-generation has had limited application, despite its vast potential. Application in propulsion has been even less, with only four commercial maritime vessels produced to date, although hundreds of military nuclear ships and boats have been built and operated successfully. Of the four commercial ships, only one has had reasonable success and is currently operational – the Russian Ice-breaker, the Sevmorput. Operation of the other three was cut short decades ago, mostly for cost reasons.

This study looks into the potential to apply modern-day nuclear technology to propulsion of large ships, and reviews the different options to make nuclear propulsion feasible in the commercial shipping industry.

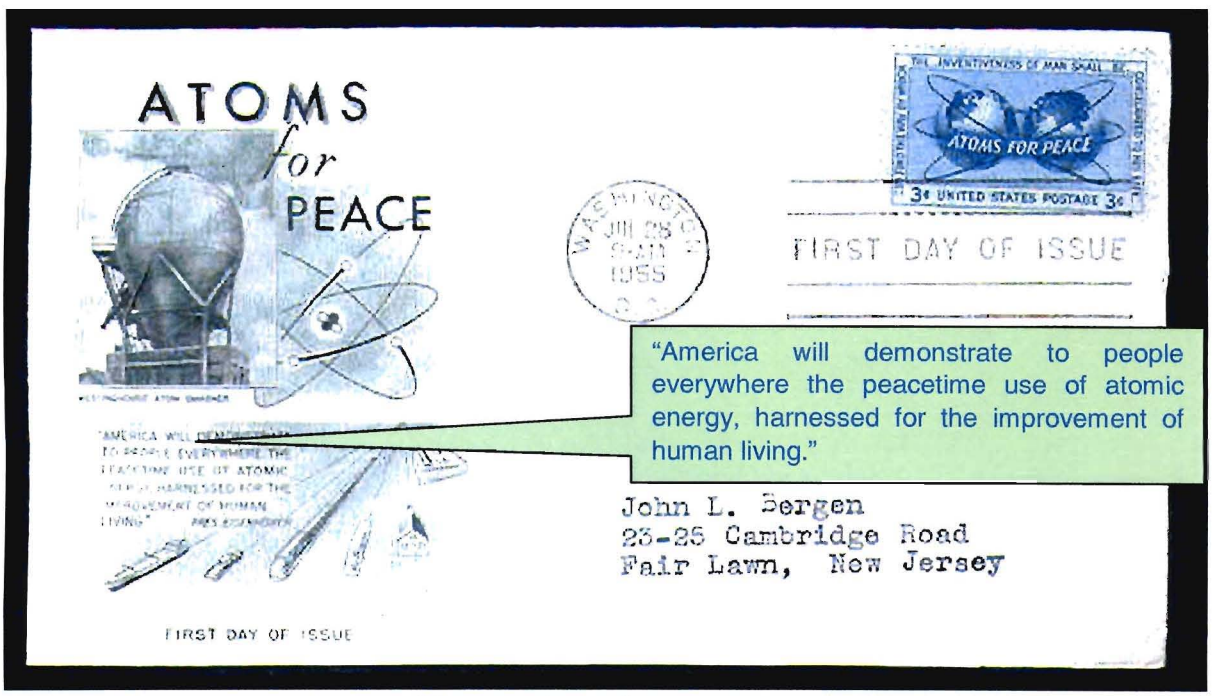


Figure 1: "Atoms for peace" postcard

1.2 BACKGROUND

The environment around commercial nuclear ship propulsion has changed significantly over the last few years. Some of the factors responsible for the change are discussed below:

1.2.1 Advances in Nuclear Technology

The Generation IV initiative is a worldwide attempt to establish a paradigm of nuclear technology. If the goals of this initiative can be accomplished, it is anticipated that many new nuclear applications will become feasible. The goals that are being pursued as part of the Generation IV initiative [30] are:

1. sustainable energy generation, by meeting clean-air objectives and effective fuel utilization;
2. sustainable energy generation, by minimizing waste and reducing the long-term waste storage responsibilities of future generations;
3. to have a significant life-cycle cost advantage over alternative energy sources;
4. to ensure that the financial risk associated with the project is not larger than that of alternative energy sources;
5. superior safety and reliability characteristics;
6. to achieve a very low probability of reactor core damage;
7. to eliminate the need for off-site emergency response.

Most of these goals are achievable with technology that is available today. High-temperature Gas-cooled Reactor (HTGR) designs, such as the PBMR that is currently under development, incorporate these technologies. Many of these goals are achieved by utilizing simplified systems that rely on physical properties, such as gravity rather than engineered safety systems that rely on external power to ensure safety.

1.2.2 Increasing Scarcity of Fossil Fuel

There is no doubt that the price of oil is steadily increasing, as the available oil reserves are declining. Figure 2 below, supplied by the United States Department of Energy [82], shows their prediction regarding the annual oil production until 2125. It shows that the available supply sources will come under severe pressure as soon as 2026. This clearly implies a need to develop alternatives to fossil-fuel based transport systems within the next two or three decades.

Oil is currently priced at approximately $70,00 per barrel, which is about 25 times the price that was applicable when it was decided in 1970 that the Savannah nuclear ship was not cost-effective [10].

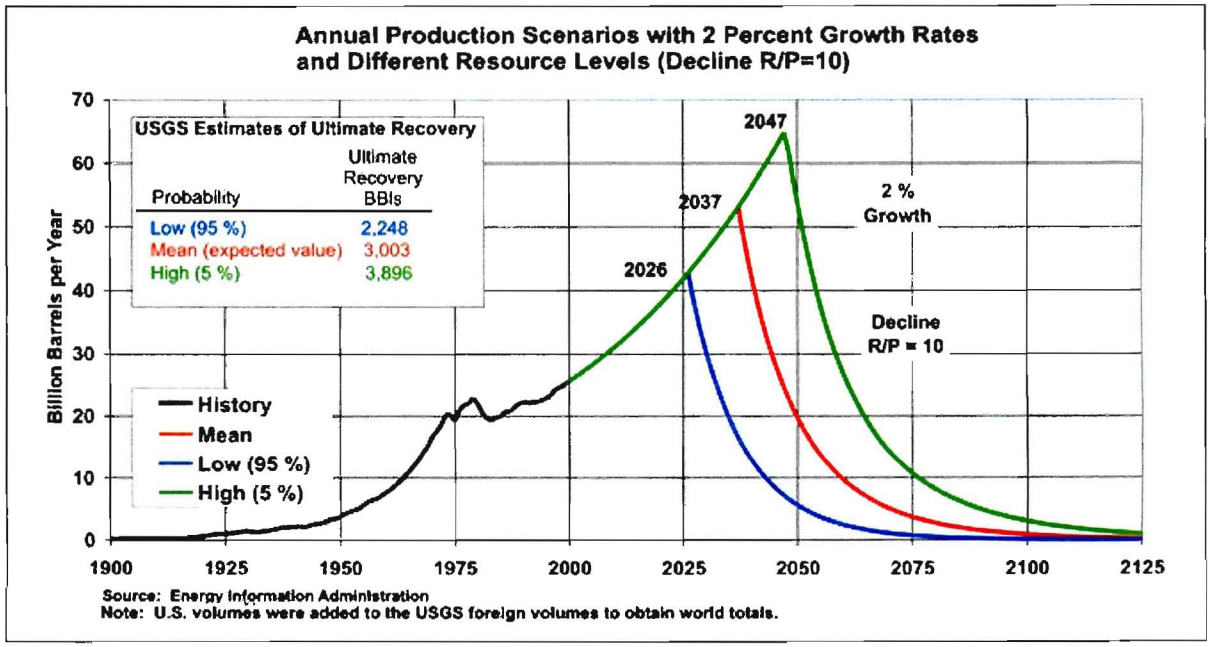


Figure 2: Annual Oil Production

1.2.3 Focus on Air Pollution and Carbon Emission

Many nations and international organisations like the United Nations are currently focusing on reducing carbon emissions. Carbon emission from ship propulsion is a surprisingly significant contributor, since these engines typically use the worst kind of fuel, resulting in disproportionately large emissions. According to a recent studies [20] and [76], it is estimated that just fifteen of the world's biggest ships emit as much pollution as all the cars in the world (estimated at around 760 million). The same studies estimates that up to 30% of all the nitrogen oxide (NOx) pollution and up to 9% of all sulphur oxide (SOx) pollution can be ascribed to large ships.

An inherent part of this problem is that the very large ships operate mostly outside national boundaries, where it is especially hard to regulate. Pollution by ocean-going vessels is subject mainly to regulations of the International Marine Organization (IMO), which is an organ of the United Nations. Currently the IMO has no mandatory regulations on the emissions of greenhouse gases from ships [41]. The expectation is, however, that this will soon change. It was recently reported [24] that sulphur oxide emission-control areas such as the Baltic Sea, and soon the North Sea and English Channel, will bar heavily polluting vessels, with recommendations to extend controlled areas to the Mediterranean, parts of the North Atlantic and the Pacific Rim.

"Carbon Tax" is also expected to start playing a significant role in the shipping industry in the near future, as more and more nations start to introduce carbon-emission taxes.

The obvious implication is that shipbuilders will face increased opposition (or regulatory constraints) to the expansion of polluting engines.

1.2.4 Public Perceptions of Nuclear Energy

Many nations around the world are experiencing a growing realization that nuclear energy is an acceptable alternative energy source. According to a survey published by the Accenture Corporation in April 2009 [74], over 66% of people around the world believe that their countries should start using or increasing the use of nuclear power. Figure 3 below, presented at a 2008 IAEA workshop, indicates the trend of public acceptance in the USA [70]. This changing public perception is based on improved knowledge and understanding, rather than emotional responses to events such as the Hiroshima bombings or the Chernobyl accident.

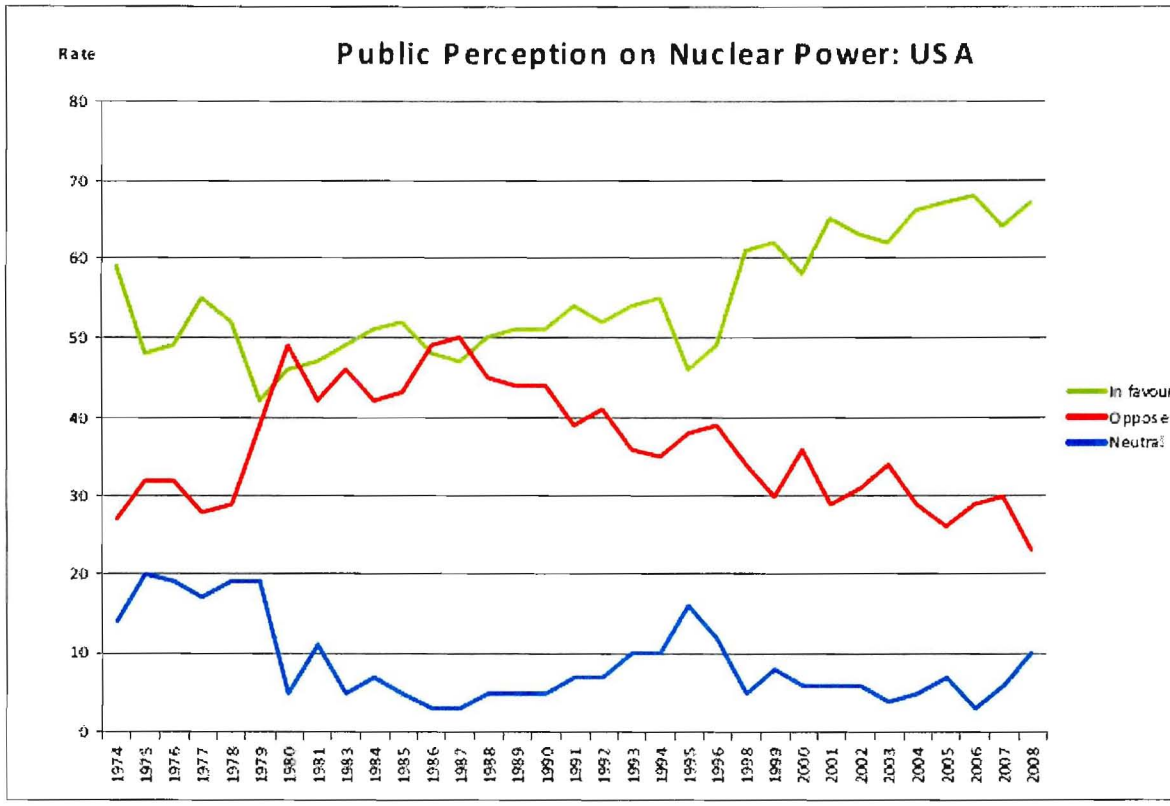


Figure 3: Public Acceptance of Nuclear Energy

1.2.5 Energy Density

Mass transport systems typically require very high energy density (power per unit of mass or size). This implies that the typical "Green Energy" options such as solar cells or wind turbines do not seem applicable. Nuclear energy potentially offers the highest energy density currently available. In fact, 1 kilogram (kg) of natural uranium can provide 20 000 times as much power as 1 kg of coal [86]. This has a huge impact on the logistics of fuel transport. These factors would suggest that nuclear engines should be well suited to propulsion systems, especially where extended periods of operation are required, with limited fuel storage.

1.3 PROBLEM STATEMENT

These changing factors in the technological, economic, environmental and political domains seem to indicate a need (or opportunity) to revisit nuclear propulsion for the wider commercial maritime domain. Nuclear reactors in general have many potential advantages over oil-based engines, such as:

a. relatively low fuel cost;
b. very small (or no) mass penalty for fuel that has to be carried for a voyage;
c. high fuel-energy density; and
d. long periods between refuelling.

Generation IV reactors add the following to these advantages:

e. Passive safety features that could make nuclear reactors as safe as conventional engines.
f. Resistance against proliferation of dangerous nuclear material.
g. Simplified operation.

Generation IV reactors could present a unique solution to the problems of pollution and oil scarcity for the shipping industry, without increasing safety risks.

1.4 OBJECTIVE

Based on the above, the objective of this study is to investigate the feasibility of designing a High-temperature Gas-cooled Reactor with sufficient safety, small enough and yet economically viable for propulsion of large commercial ships.

1.5 METHODOLOGY

The investigation consists of three phases:

a. Determine the requirements for a nuclear propulsion unit, by analysing the maritime propulsion market. Consider factors such as:
   i. propulsion power, force and speed;
   ii. typical size and weight of a propulsion system; and
   iii. mission profile.
b. Create a conceptual design of a nuclear propulsion unit, by selecting an optimal reactor and power-conversion system, using as far as possible standard or proven components.
c. Investigate the commercial viability of such a nuclear engine by comparing the life-cycle cost to conventional propulsion options.

1.6 DOCUMENT OVERVIEW

In accordance with the methodology presented above, this document is structured as follows:

Chapter 2 presents an overview of related literature;

Chapter 3 presents a set of requirements for a nuclear propulsion system;

Chapter 4 sets out a conceptual design for a propulsion system;

Chapter 5 contains an economic analysis of the proposed design concept;

Chapter 6 presents risk associated with this development;

Chapter 7 draws conclusions from the study, and

Chapter 8 proposes future work.

2. LITERATURE OVERVIEW

This section presents a brief overview of literature related to nuclear ship propulsion.

2.1 HISTORICAL COMMERCIAL NUCLEAR SHIPS

Four commercial nuclear ships have been built to date: the American Savannah; the German Otto Hahn; the Russian Sevmorput and the Japanese Mutsu. Table 1 below contains relevant details of these ships [83], [62] and [78].

Table 1: Characteristics of Commercial Nuclear Ship

	Savannah	Otto Hahn	Sevmorput	Japanese Mutsu
Comment	The ship was decommissioned after only eight years. It was a technical success, but not economically viable. This was to some extent due to low oil cost, a large specialized crew and expensive one-of-a-kind maintenance requirements.	Covered 1 200 000 km on 126 voyages in 10 years without any technical problems. However, it proved too expensive to operate and was converted to diesel.	Ice-breaker and cargo carrier. World's first nuclear-powered oil-drilling vessel. Still in operation Reactor pressure vessel is 4,6 m high and 1,8 m in diameter, enclosing a core 1 m high and 1,2 m in diameter.	Several technical and political problems resulted in an embarrassing failure. Never carried any commercial cargo.
Date Launched	1959	1964	1988	1972
Tonnage	22000	26200	61900	8240
Length (waterline) [m]	182	164	260	130
Beam [m]	23,8	23,4	32,2	19
Draught [m]	8,8	5,3	11,8	6,9
Fuel	Low Enriched Uranium	Low Enriched Uranium (3,5 to 6,6%)	High Enriched Uranium (90%)	Low Enriched Uranium
Reactor Power [MWt]	74	38	135	36
Reactor Volume		35 m^3		
Active Core Height		8,3 m	1 m	
Cost	$46,9m ($28,3m for reactor and fuel)			
Propulsion Power [MW]	16,4	8	29,4	7,5
Speed [knots]	21	17	20	17
Propulsion Plant Weight [ton]	2500			
Specific Weight [kg/kW]	151			

The conclusion derived from the history of these four ships seems to be that economics is the biggest hurdle to overcome.

2.2 ARCTIC ICE-BREAKERS

Nuclear propulsion has been shown to be economically and technically viable in the Russian Arctic area. In this region, nuclear ships have increased navigation from 2 to 10 months per year due to increased power levels and refuelling advantages. The first of these ships was the Soviet-built Lenin, which remained in service for an amazing 30 years despite two nuclear accidents [81]. These ships would seem to fit into a category between military and commercial ships.

2.3 MILITARY NUCLEAR SHIPS

According to the World Nuclear Association [84], roughly 150 ships worldwide are powered by more than 220 small nuclear reactors, and more than 12 000 reactor years of marine operation have been accumulated. This seems to be a well-understood and low-risk domain, which should be applicable to the wider commercial domain.

2.4 CURRENT SMALL NUCLEAR REACTOR DEVELOPMENT PROJECTS

Current small nuclear reactor development projects that may be relevant for ship propulsion are shown in Table 2 below, compiled by the World Nuclear Association [85]:

Table 2: Small Nuclear Power Reactors

VK-300	300 MWe PWR	Atomenergoproekt, Russia
CAREM	27 MWe PWR	CNEA & INVAP, Argentina
KLT-40	35 MWe PWR	OKBM, Russia
MRX	30-100 MWe PWR	JAERI, Japan
IRIS-100	100 MWe PWR	Westinghouse-led, international
SMART	100 MWe PWR	KAERI, S. Korea
NP-300	100-300 MWe PWR	Technicatome (Areva), France
PBMR	165 MWe HTGR	Eskom, South Africa, et al
GT-MHR	285 MWe HTGR	General Atomics (USA), Minatom (Russia) et al
BREST	300 MWe LMR	RDIPE (Russia)
FUJI	100 MWe MSR	ITHMSO, Japan-Russia-USA

More details on five of the most interesting developments are provided below:

i. Japanese MRX

The MRX is a 100 MWt integral-type PWR that is specifically aimed at ship propulsion. The reactor weight is 1600 tons and size is 1210 m^3, which is claimed to be half that of the Mutsu referred to in Table 1 above. The MRX intends to utilize passive engineered safety systems that are claimed to result in core damage frequency of two orders smaller than existing Pressurized Water Reactors [42].

ii. Adams atomic engines

This concept utilizes a pebble bed reactor in the range of 1 to 100 MWe, with a Brayton Cycle gas turbine, that uses nitrogen as coolant [10]. The concept design plans to have radial coolant flow from the outside inwards over the fuel spheres. The intention is for a core outlet temperature of 950 °C to enable superior Brayton cycle efficiency.

iii. Hyperion nuclear battery

This reactor uses uranium hydride as fuel, which is claimed to give it completely passive power control. The concept is to provide steam in accordance with demand, without any moving parts, much like a battery. The reactor is designed to produce 70 MWt for a period of up to 10 years [34].

iv. The NEREUS Project

This is a Dutch project that aims to develop a Naturally safe, Efficient, Reactor, Easy to operate, Ultimately simple and Small [23]. These are all aims that match the application of this study. The aim of the NEREUS project is to develop a 24 MWt plant with 40% thermal efficiency, with dimensions 10x10x10 m, and weighing less than 2000 ton [6].

v. Study by JGCC Jacobs of Delft University of Technology [43].

This is a very interesting study that focused on the design of a helium-cooled prismatic reactor for a feeder container ship. The reactor has a capacity of 31 MWt and makes use of block fuel. The power-conversion cycle is indicated in Figure 4. A CAD model of the engine is shown in Figure 5. This study calculated that the nuclear engine would cost €154,5m.

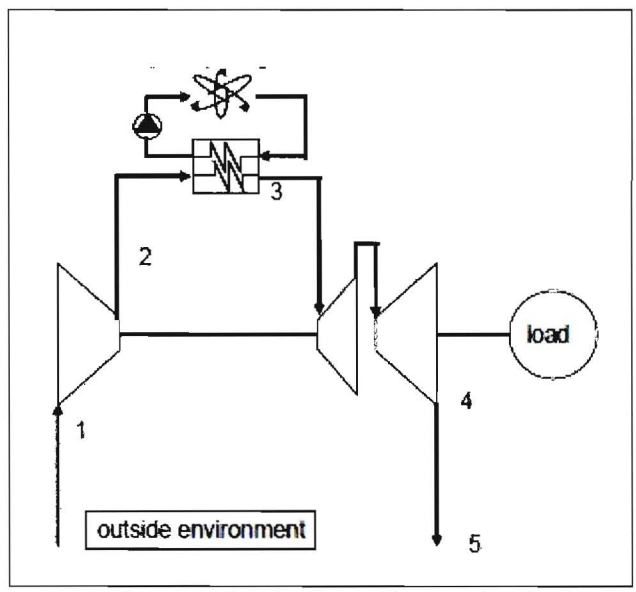


Figure 4: Jacobs Power Conversion Cycle

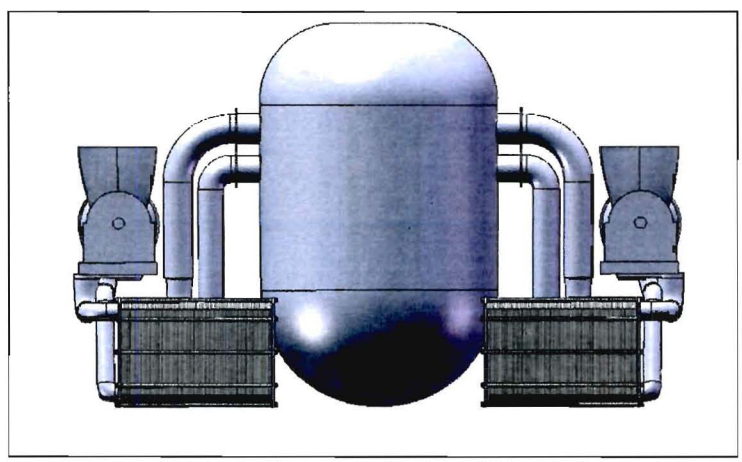


Figure 5: Jacobs Nuclear Ship Engine

3. REQUIREMENTS ANALYSIS

The intention is that this dissertation should follow a conventional systems engineering approach [40] to accomplish the design of a nuclear ship propulsion system. The process is, however, scaled down to fit within the limited scope of a dissertation. In principle, a set of solution-independent requirements are established as a first step. The second step is to synthesize a design that will comply with the set of requirements.

This section is dedicated to establishing the technical requirements for a nuclear ship engine and the main functions that have to be accomplished. The synthesis is presented in Section 4, which consists of a set of trade-off studies to select an optimal configuration. Section 5 describes the selected design.

3.1 TARGET MARKET INVESTIGATION

In order to establish a set of technical requirements that will drive the design of a propulsion unit, it is essential to first select a specific target market. The optimal target market should represent a good match between the type of ship and nuclear engines in general. The target market should also be sufficiently large, growing and profitable, to enable the establishment of a viable business case.

In selecting a target market, a few assumptions are made about nuclear engines in general:

a. Nuclear engines are typically very large and heavy (the power plant of the Savannah weighed 2500 ton). This is mostly due to radiation shielding (that does not scale linearly with reactor power) and large metallic pressure vessels. This implies that a larger ship requiring a more powerful reactor will be more appropriate for nuclear propulsion.
b. Nuclear systems do not require frequent refuelling, but do require a specialized home port for handling radioactive waste, spent fuel and fresh fuel.
c. Nuclear ships do not require fuel space for long voyages, liberating space for cargo.
d. Nuclear engines are capital-intensive.
    i. Such engines would therefore make sense only for a ship that can earn sufficiently high revenues.
    ii. In addition, a large, installed fleet would enable the significant learning curve advantages. In the aircraft industry, these learning curve advantages are known to have decreased unit production man-hours by as much as 20% with each doubling of the produced numbers [19].
    iii. Due to the costly capital nature of a nuclear engine and relatively low fuel cost, it makes the most sense to use these machines at the highest possible capacity. This is why most nuclear power plants are used as "base-load" stations where the plant runs at maximum capacity for about 90% of the time. In the shipping environment this implies that the most suitable ship for a nuclear engine would be a ship that is cruising at maximum power for most of the time. This translates into ships servicing long routes and having a very short port time.
e. Nuclear engines would require relatively sophisticated maintenance and operational support.
f. All nuclear systems are tightly regulated by authorities, and the latter would thus impose a regulatory burden on the ships that are powered by such reactors. The type of shipping industry that is selected should be compatible with such a regulatory framework.

In an article by Adams [11], criteria were set out for deciding on appropriate shipping applications for nuclear propulsion. These include:

a. Long trade route
b. Quick port turnaround
c. Large deadweight capacity
d. Emission limitations
e. Speed.

The article proposed that the following types of ships would be well suited to nuclear propulsion:

a. Large container ships
b. Automobile carriers
c. Refrigerated cargo carriers
d. Long-distance passenger ships
e. Bulk cargo carriers
f. Any ship that spends most of its time in operation.

Large modern-day ships can be divided into the following categories:

Table 3: Ship Categories

Dry Cargo	
	Bulk carrier
	Container ship
	Barge
	Roll-on-roll-off ship (motor cars)
Tankers	
	Oil tanker
	Liquid natural gas
	Chemicals
Passenger Ships	

Refer to Figure 6 for a diagrammatic comparison of five examples of the largest ship types.

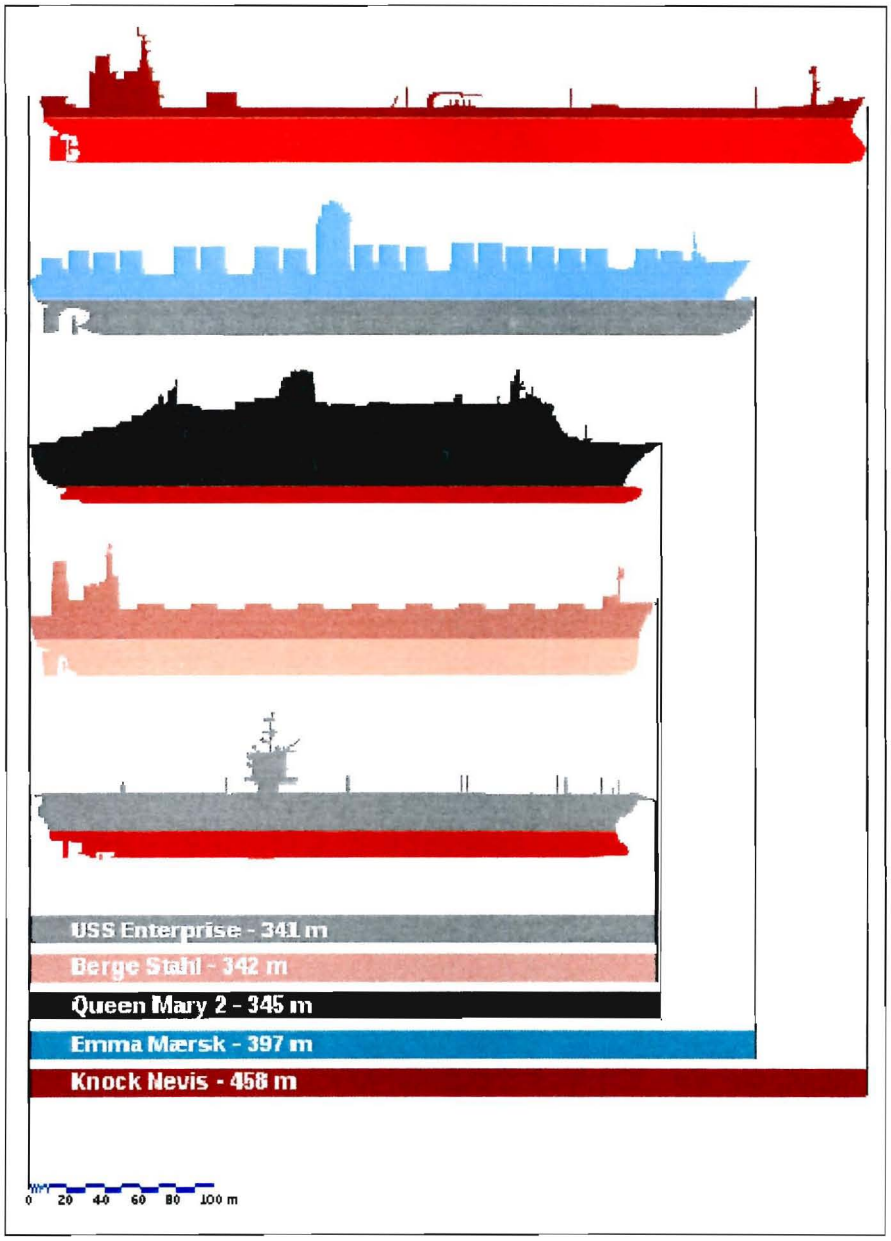


Figure 6: Size Comparison of Five of the Largest Ships [79]

Based on the factors listed above, the broad group of ships that seems best suited for nuclear propulsion is large oil tankers.

3.1.1 Oil Tankers

Oil tankers are typically classified in six size groups [80]:

Table 4: Oil Tanker Types

Class	Size in DWT	Typical Length [m]	Typical New Price
Product tanker	10 000 – 60 000		$43m
Panamax	60 000 – 80 000	220	
Aframax	80 000 – 120 000	230 to 250	$58m

Class	Size in DWT	Typical Length [m]	Typical New Price
Suezmax	120 000 – 200 000	250 to 270	
VLCC (Very Large Crude Carrier)	200 000 - 320 000	300	$120m
ULCC (Ultra-large Crude Carrier)	320 000 - 550 000	> 300	

***Note**: DWT or Dead Weight Tonnage is the mass that the ship can carry (cargo, fuel, ballast, etc.)*

Because of their large size, VLCCs and ULCCs can often not enter or leave a port fully loaded, and are therefore equipped to take on their cargo at off-shore platforms. At the other end of the journey they often pump their cargo off to smaller tankers at designated points off-coast. For this reason, these tanker routes are generally long, requiring these ships to stay at sea for extended periods of up to and beyond seventy days at a time [80]. This characteristic, coupled to their large size and cost, would indicate that they should be well suited for nuclear propulsion.

Figure 7, Figure 8 and Figure 9 were compiled by the MAN Diesel A/S Company [52], based on their market research. These figures show that the VLCC segment is dominant in terms of DWT capacity and has almost 10% market share in the number of tankers world-wide. The same research from MAN also indicates that the demand for ULCCs seems very low, with almost all of the current fleet of ULCCs having been built in the 1970s. Figure 9 indicates that even though the VLCC class did not show the fastest growth in terms of numbers of ships during the period 1992 to 2006, on average 35 VLCCs were built per year. According to a March 2009 report in the Petroleum Economist [21], there are currently 501 VLCCs in use worldwide and another 227 on order. Figure 10 shows that despite volatility, the daily earnings of a VLCC have increased substantially over the last two decades.

Based on all these factors, it is concluded that the VLCC market segment is the preferred choice for this study.

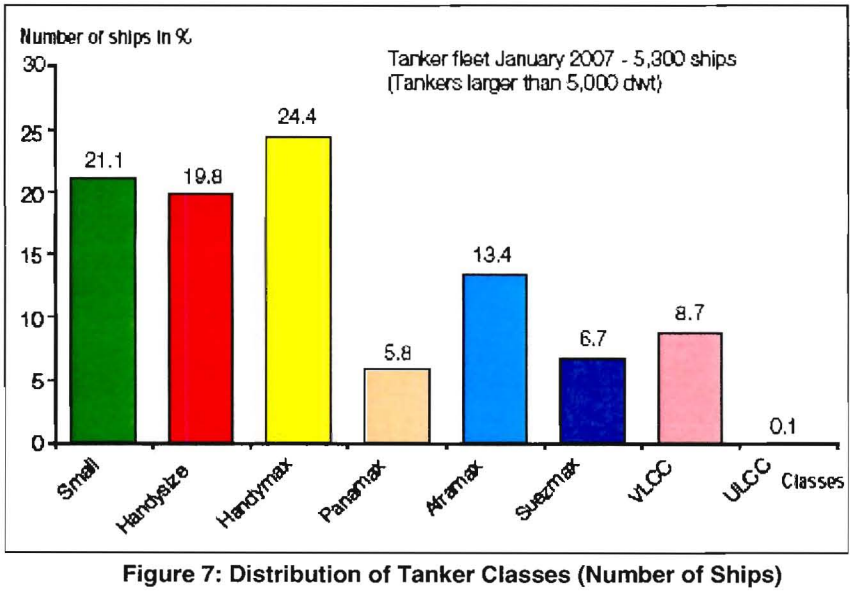


Figure 7: Distribution of Tanker Classes (Number of Ships)

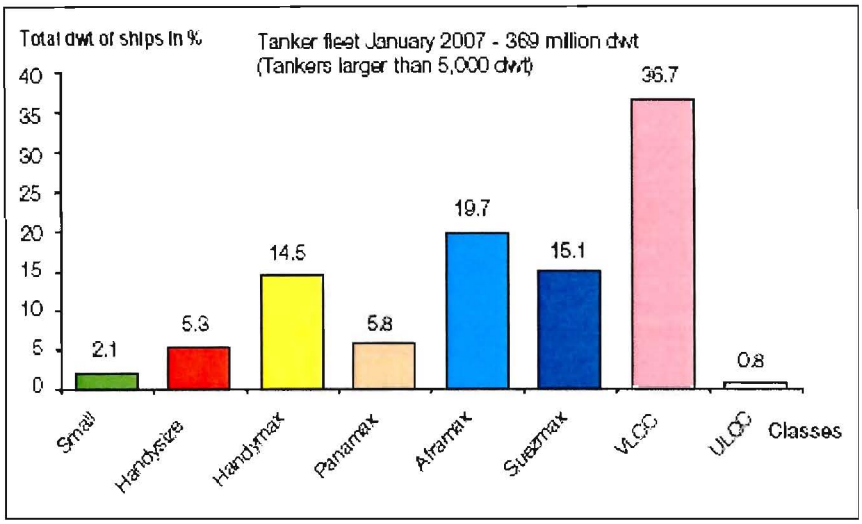


Figure 8: Distribution of Tanker Classes (Deadweight Tonnage)

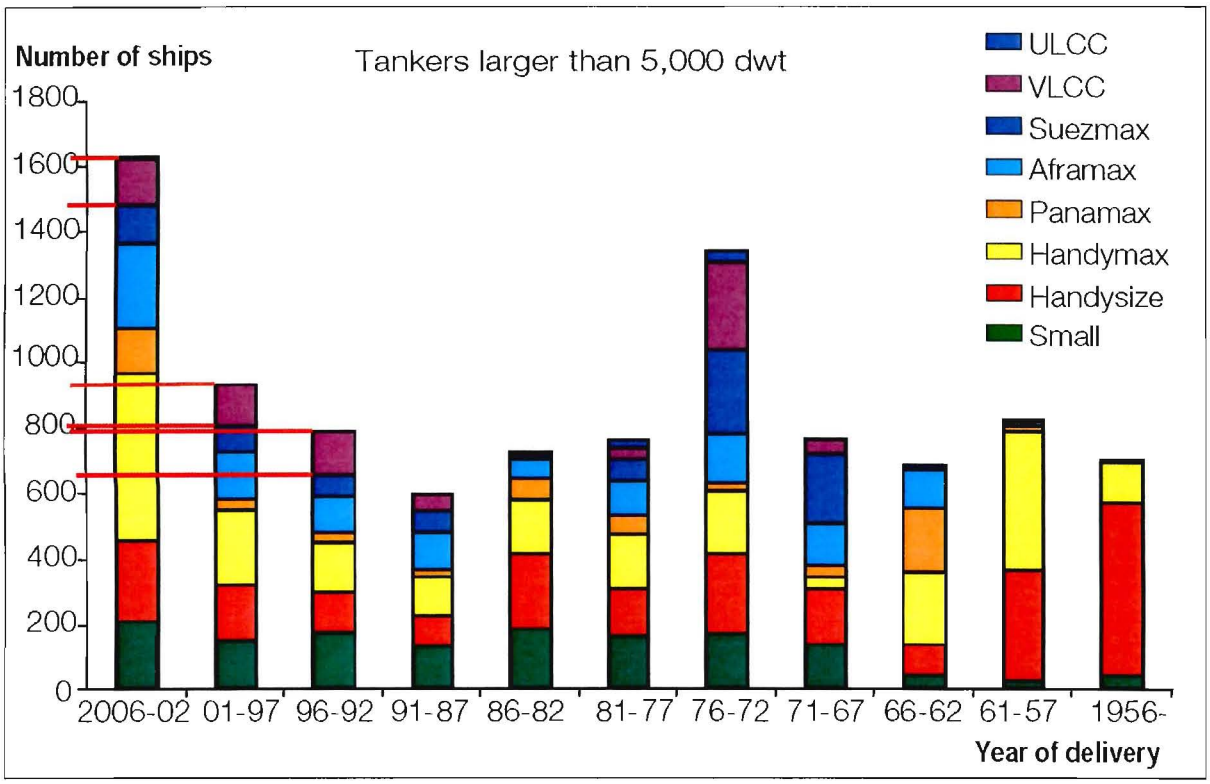


Figure 9: Dates of Tanker Deliveries

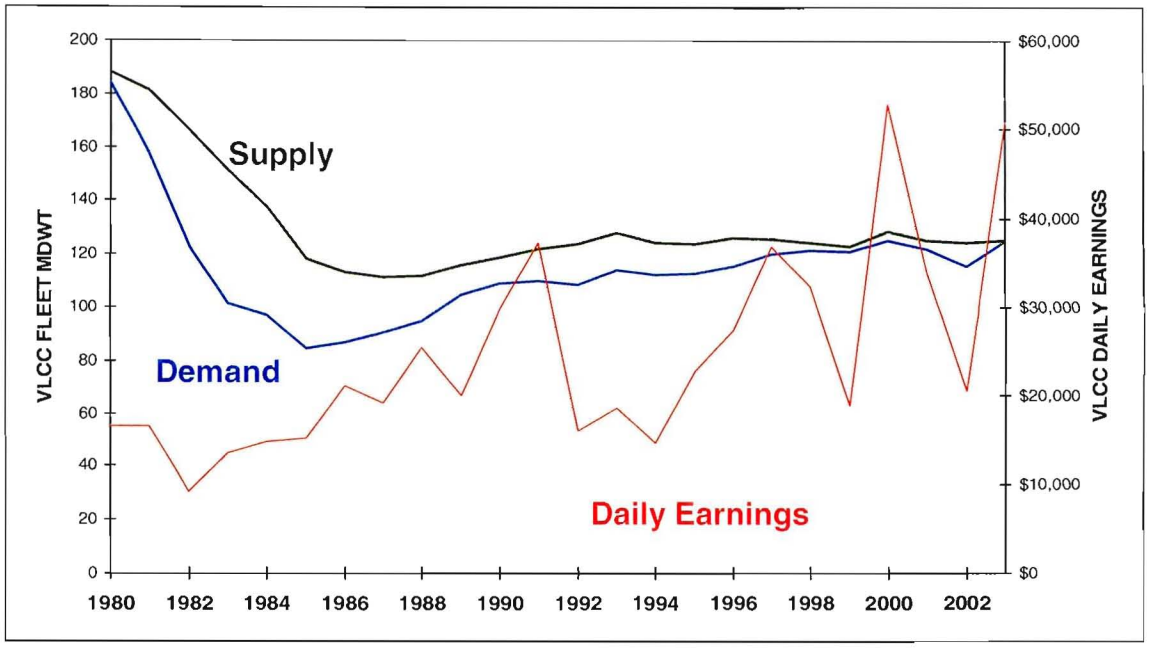


Figure 10: VLCC Supply/Demand vs Earnings [1]

3.2 PROPULSION REQUIREMENTS

A set of requirements relating to the propulsion for a VLCC are developed in this section. Propulsion is characterized here by the force required, the power of the engine and the propeller speed.

3.2.1 Propulsion Force

The propulsion force required for a ship is dominated by the total resistance experienced by the specific ship at the cruising speed. The total resistance (R_T) is often referred to as the 'towing resistance' of the ship and consists of the frictional resistance (R_F), the air resistance (R_A), the wave-making resistance (R_W) and the eddy resistance (R_E).

The typical relative contributions of these types of resistance for tanker ships are shown below [50]:

Table 5: Resistance Forces

Resistance Type	Percentage of Total Resistance (for Low-speed Ships like Tankers)
R_F	90
R_W	5
R_E	3
R_A	2

It is evident that for tankers, the frictional resistance is dominant.

The frictional resistance depends on the square of the ship speed (V), the wetted surface area (S_{wetted}), the density of the water (ρ_{sea_water}) and the resistance coefficient (C_F), which is a unique characteristic of the ship.

$$R_F = \frac{1}{2} \times \rho_{sea_water} \times S_{wetted} \times C_F \times V^2 \quad \text{.....(Eq 3.1)}$$

A typical value for C_F for an oil tanker is 0,002. This number is based on information from three different reports [66], [9] and [52]. More accurate calculations can be based on methods developed by Holtrop and Mennen [33], but the methods were not used in this study.

The R_F for an average VLCC of 280 000 DWT (length 320 m, breadth 58 m, draught 19 m) cruising at 15,5 knots with C_F of 0,002 can be calculated as:

$$R_{F_VLCC} = \frac{1}{2} \times 1000 \times [320 \times (58 + (2 \times 19))] \times 0.002 \times (15.5 \times 0.514)^2 = 245\text{kN} \quad \text{.....(Eq 3.2)}$$

In practice, an additional margin should be added onto this amount, to account for weather and ageing. For this study, an additional 15% will be used. On top of that margin, an additional 15% is estimated to be required for accelerating the ship to cruising speed.

$$R_{F_VLCC} = 245kN \text{ *1.15*1.15 = 324kN} \quad \text{.....(Eq 3.3)}$$

3.2.2 Propulsion Power

The propulsion power required for a ship is the product of the ship speed and the propulsion force, divided by the efficiency of the propulsion system. Efficiencies for propeller-driven ships vary over a wide range of 50% to 90%. In this study, a value of 90% will be used, in correspondence with the typical values presented by MAN [50]. In addition, the engine capacity should be scaled up to make allowance for deterioration of the ship's hull smoothness, sea state, and wind. An additional 15% power is typically added for a 'sea margin' and another 10% for an engine-ageing margin [52].

Figure 11 below is based on calculations by MAN Diesel [52] for VLCC and ULCC tankers with a nominal speed of 15,5 knots and an allowance of 15% for sea margin as well as an engine margin of 10%. This indicates that an engine for a VLCC needs a maximum continuous output of between 19 MW and 33 MW, with 25 MW being the nominal value, corresponding to a 280 000 DWT ship with a maximum operating speed of 15,5 knots.

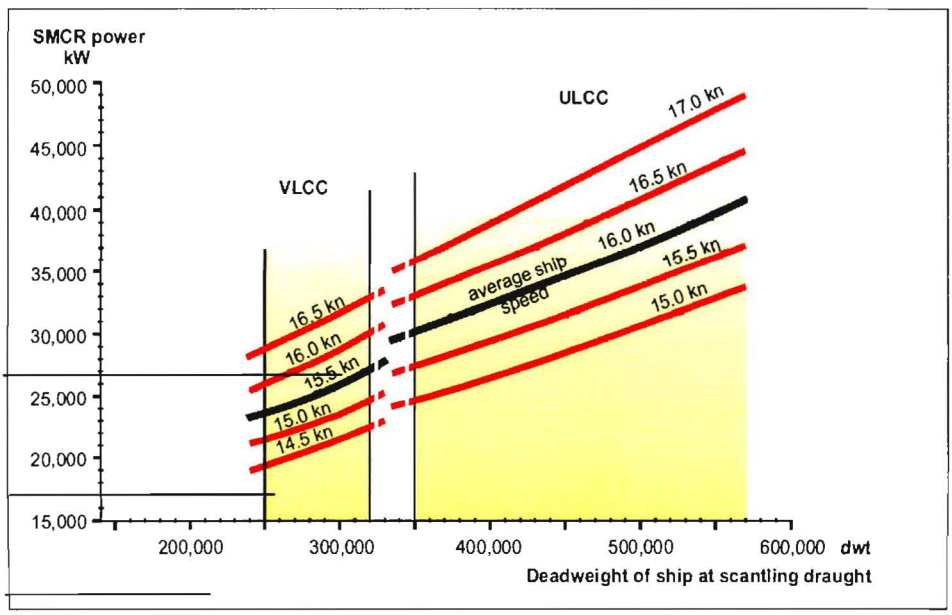


Figure 11: Propulsion Power Demand for VLCCs and ULCCs

3.2.3 Propeller Speed

When a propeller is used for propulsion, a larger propeller rotating at a slower speed is advantageous for overall efficiency. A typical propeller speed for a VLCC ship is between 65 and 85 revolutions per minute [51]. This very slow speed of rotation implies that fast-rotating machines such as turbines would require a gearbox with a large speed-reduction ratio.

3.3 AUXILIARY POWER

A modern VLCC typically uses a single large engine, coupled directly to a single propeller, but also incorporates smaller secondary engines for house load while in port or in emergency situations. A typical example is the Iran Delvar VLCC that has a 27 MW main engine and three auxiliary engines that each has a capacity of 1,5 MW (refer to Figure 12). Other VLCCs use a diesel-electric propulsion system where the propeller is coupled to an electrical drive, which is driven by generators connected to diesel engines. Examples of this type of propulsion are the BP-VLCC fleet, as shown in Figure 13. These vessels make use of two 6,6 kV motors provided by the Alstom Corporation.



Figure 12: Iran Delvar VLCC

When a nuclear engine is used as the main propulsion system, it is foreseen that the secondary engine(s) would have to be used for low-speed propulsion as well as providing house-load power while in port or in emergency conditions.



Figure 13: BP VLCC

Nuclear reactors are typically less adept at providing power at low part-load than diesel engines. This is due mostly to the effect of fission product poisons such as Xenon, which builds up in a reactor when power is stepped down. The effect of these poisons must be countered by adding positive reactivity into the reactor to avoid sub-criticality. This positive reactivity addition is usually by means of withdrawing control rods, or cooling down the fuel in the reactor. For reactor safety considerations it is advantageous to have this amount of reactivity that can be added, with control rods as small as possible. The reason behind this constraint is to limit the amount of reactivity that could be added by accidental withdrawal of the control rods. Typically, HTGRs are designed to be able to operate at a minimum of 50% of full power. For a VLCC this would equate to a speed reduction from an operational speed of 15,5 knots to an estimated 11 knots. This implies that auxiliary engines will be needed to power the ship at low speed and to provide electrical power while the ship is stationary.

There is an economic trade-off that has to be made between the requirement to operate at low power with the nuclear reactor and the capacity of the auxiliary diesel engine. This trade-off will depend on the details of the reactor control rods and the mission profile of the ship. These details are beyond the scope of this study. For this study, the requirement will be to operate the nuclear reactor at a minimum of 50% power and for at least two auxiliary diesel engines that can provide 1,5 MW each.

3.4 DIRECT VERSUS INDIRECT DRIVE

As indicated in Section 3.3, VLCCs are built with both direct and indirect coupling via electrical power (refer to Figure 14). In military nuclear ships, the Russian, USA and British navies use direct drives, while the French and Chinese navies utilize turbo-electric propulsion. In the application of nuclear propulsion there are two very compelling reasons to use indirect coupling:

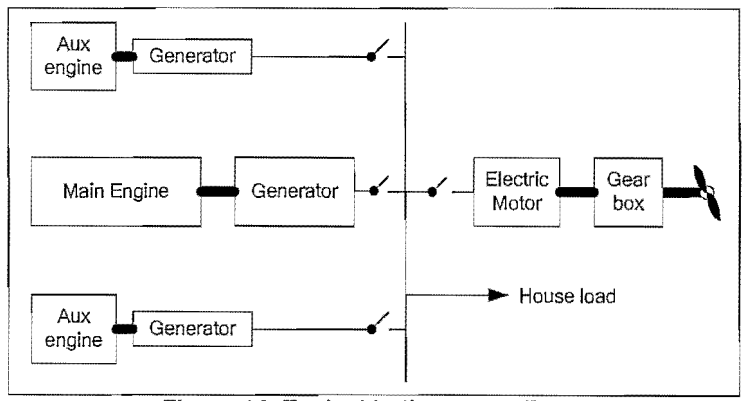


Figure 14: Typical Indirect coupling

a. Indirect coupling allows the flexibility to separate the ship propulsion dynamics from the reactor dynamics. It allows the possibility of stopping propulsion power while maintaining the reactor critical at minimum power (implying a mechanism to dissipate the reactor power into the environment). This is a significant advantage, since Xenon poisoning as a result of a reactor shutdown causes a dead period of several hours during which it is impossible to restart the reactor. Refer to Figure 15 for an illustration of the dead time for a few different reactor flux levels.

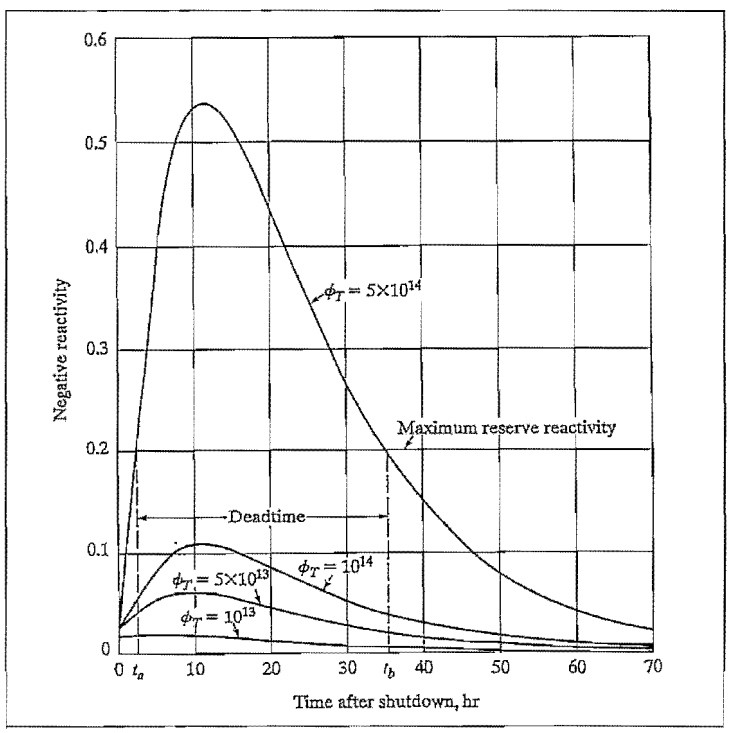


Figure 15: Xenon Dead Time [47]

b. The indirect coupling enables auxiliary engines to power the same propeller while the nuclear reactor is not in operation.

These reasons are seen to outweigh the cost and (small) efficiency disadvantages of an indirect drive system.

3.5 ENGINE SIZE AND WEIGHT CONSTRAINTS

Table 6 illustrates a few precedents that can be used to establish size and weight constraints for the nuclear ship engine.

Table 6: Size and Weight Precedents

Basis	Weight [ton]	Size
81MW RTA96-C Wärtsillä Diesel	2300	27 m long by 13 m high
NEREUS 24 MWt nuclear ship engine design in [24]	2000	10 m cube
Jacobs nuclear ship engine (31 MWt) [43]	1370 (Reactor)	Reactor only: 11,5 m high; 9 m wide
MRX 100 MWt nuclear ship engine	1600	1210 m^3
NSS Savannah power plant (74 MWt)	2500	Approximately: Reactor: 6 m long by 5 m wide by 4 m high Machinery: similar to reactor
MAN B&W 7S80MC, 25 MW [49]	996	12,6m long; 13 m high; ~7 m wide.
MAN B&W 6S80ME-C8, 25 MW [49]	820	11,43 m long; 14,3 m high; ~7 m wide.

From these precedents a weight goal of 2000 ton and size goal of 1000 m^3 is deduced.

In addition to the size and weight constraints discussed above, it is important to consider the impact of the engine on the ship dynamics. Tall, heavy structures will tend to destabilize a ship, while heavy structures near the keel of a ship will enhance dynamics.

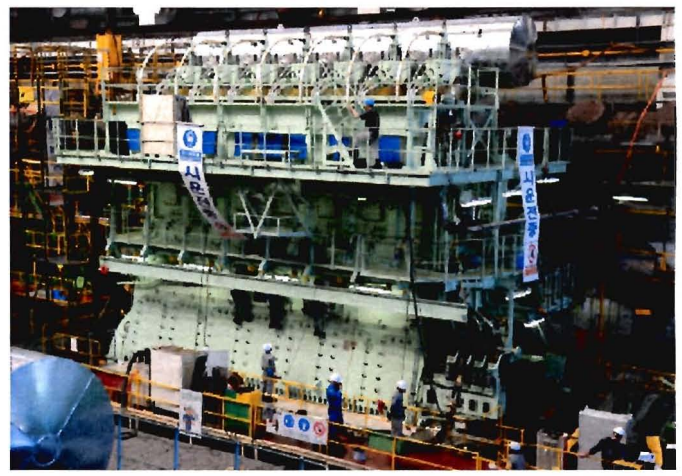


Figure 16: A typical 31 MW Wärtsilä Engine

3.6 COST CONSIDERATIONS

It is not possible or desirable to directly compare the capital cost of a nuclear engine with that of a diesel engine, because the life-cycle cost breakdown is significantly different. This section *does, however,* attempt to

establish the capital cost of an equivalent diesel engine, to enable life-cycle cost comparisons.

a. Based on an enquiry directed to MAN Diesel Corporation, the estimated cost of a type 6S80ME-C8 engine, which has a 25 MW power rating at 78 revolutions per minute (rpm), would be €6,8m. The 7S80MC-C8 with a power rating of 29,26 MW at 78 rpm is estimated to cost €7,5m.
b. Combined-cycle gas turbines in the 25 MW capacity range are nominally priced at $820/kW, according to the Gas Turbine World Handbook for 2009 [3]. This equates to $20,5m for a new 25 MW machine.
c. Information from two studies that compared diesel versus nuclear propulsion was also used: the NEREUS project compared its reactor to an 8 MW diesel engine, while the Jacobs study made a comparison with an 8,4 MW diesel engine. These cost estimates were linearly scaled up to 25 MW, even though that would probably lead to an overestimation of the engine cost.
d. Cost information about rebuilt diesel generators was also obtained, although not for models as large as 25 MW. These prices were scaled linearly to 25 MW, and it was assumed that a refurbished engine would cost about 80% of the cost of a new engine.

Table 7: Fossil Fuel Equivalent Engine Cost

Engine Type	Capacity [MW]	Price [$m]	Scaled Price of a 25 MW Engine	Escalated for a New Engine
MAN 6S80ME-C8	25	8,16 (€6,8m)	8,16	8,16
MAN 7S80MC-C8	29,26	9 (€7,5m)	9	9
Combined cycle gas turbines [3]	25	20,5	20,5	20,5
NEREUS diesel engine comparison [24]	8	8 (€6,4m)	25	25
Jacobs diesel engine comparison [43]	8,4	3,36 (€2,8m)	10	10
Wartsila Sulzer 16 ZAV40S overhauled [8]	10,5	3,29	38.00	30.00
Caterpillar-MAK engines (4 x 7,1 MW), 2001 model [7]	28	9,5	9.55	11.25
Average				**19.18**

The average of these six estimates was calculated as $14m, but the most direct comparison is the MAN 6S80ME-C8 engine, which will be used as a benchmark against which to compare the nuclear plant.

3.7 TYPICAL MISSION PROFILE

A typical mission profile can be used to determine the required availability and power profile of the propulsion system over a typical year.

a. Typical oil tanker routes [14] are shown below in Table 8.

Table 8: VLCC Routes [14]

VLCC Tanker Routes	Distance (miles)	Time (days)
Saudi Arabia to Japan	6 611	46
Saudi Arabia to US Gulf Coast	12 225	82
Saudi Arabia to Singapore	3 708	28
Saudi Arabia to South Korea	6 234	44
Saudi Arabia to Red Sea	3 061	23
North Sea to Canada	2 966	23

Note: All voyages represent a round trip, estimating four days for loading and discharging

The Saudi Arabia to Japan route is taken as the norm for this study. A speed profile for this route is shown in Figure 17.

b. A large ship has a typical life expectancy of between 15 and 20 years [63]. For this study it will be assumed that the engine would have to have a useful life of 20 years.

c. A typical VLCC has a utilization level of around 90% [1], making it very suitable for nuclear propulsion.

d. For this study, it will be assumed that the ship will require major maintenance every 5 years, each maintenance period having duration of about 2 months. Annual maintenance of 2 weeks per year is assumed, while routine maintenance is assumed to be done during operation and while cargo is being loaded.

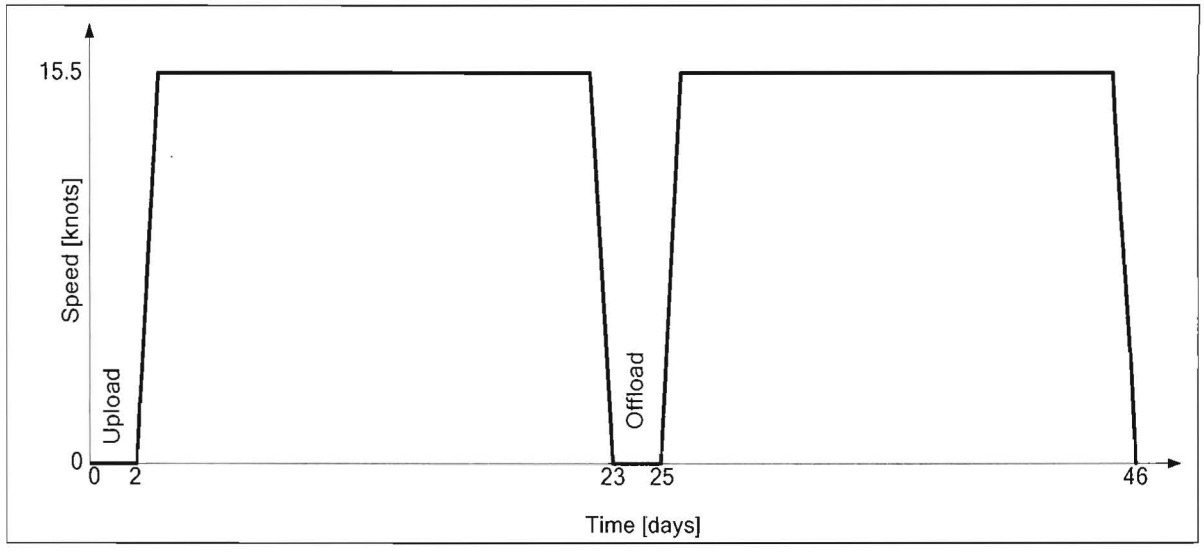


Figure 17: Typical Mission Profile

e. Based on these assumptions, the following can be deduced:

i. While in operation, the engine would run at full power for 87% of the time, run at low power for 4% of the time, and be idle for the remaining 9% of the time.

ii. The total, anticipated maintenance duration would be 6 months plus 40 weeks out of a total life of 240 months, representing 6,3% of the time.

iii. This implies that over the life of the engine it would run at full power for approximately 80% of the time, at low power for approximately 4% of the time and be idle (or maintained) for the remaining 16%.

3.8 VALUE SYSTEM FOR A SHIP PROPULSION UNIT

Safety in a commercial ship context is equivalent a hygiene factor, as described by Frederick Herzberg [22], which would not motivate a potential client to select nuclear propulsion above other modes, but if not treated properly will lead to a negative bias against nuclear propulsion.

Factors that are considered to be important in determining the value of a ship propulsion unit are:

a. Economical cost (over the life cycle)
b. Reliability
c. Lightweight
d. Compact (engine as well as fuel)
e. Simple
f. Easy to operate
g. Easy to maintain

3.9 SAFETY ASPECTS

This section aims to establish the requirements related to safety for a nuclear propulsion unit of a VLCC ship.

3.9.1 Nuclear Safety in the Commercial Maritime Domain

3.9.1.1 Risk of dangerous nuclear exposure

Even though public perceptions of nuclear energy are changing, as indicated in Section 1.2.4, there is still widespread belief that any nuclear reactor could potentially explode like a nuclear bomb. This was enhanced by the explosion of the Chernobyl reactor in 1986, due to a prompt supercritical excursion, that led to a core meltdown. Obviously, carrying this type of reactor on an oil tanker is inconceivable! Based on this, one can derive a requirement that a nuclear reactor for VLCC propulsion should be designed in such a way that it is impossible to achieve prompt criticality, impossible for the core to melt or explode, independent of the actions of the operating crew.

3.9.1.2 Simplicity in operation and maintenance

A commercial maritime reactor has to be simple and robust to operate and maintain. This would contrast with the current trend to have very highly skilled operators and even regulatory agents at each nuclear plant. Experience with simplified nuclear plants would be required to convince nuclear regulators that requirements for highly skilled personnel could be changed.

3.9.1.3 Passive safety

The concept of passive or inherent safety is relatively new in the nuclear industry. The concept aims to ensure nuclear safety without the need for operator actions or external power sources. Usually this concept relies on natural forces such as gravity or buoyancy to accomplish safety functions. Passive safety would be highly advantageous in a commercial shipping environment. This characteristic would reduce the need for highly skilled operators

and would obviate the need for redundant and often diverse back-up safety systems, resulting in cost reduction, maintenance reduction, space saving and reduced complexity.

3.9.1.4 Radiation shielding

An obvious requirement is that personnel have to be shielded against radiation exposure – this should be in line with international practice. The International Commission on Radiological Protection (ICRP) publishes recommendations on the amount of radiation that is deemed acceptable. According to the ICRP publication 103 [38], any single worker that is occupationally exposed to radiation should not receive more than 20 mSv/year, averaged over 5 years. Radiation exposure of a member of the public should not exceed 1 mSv/year.

In addition, it is important to ensure that the oil cargo (or any other cargo) does not get activated by a neutron field, since this would lead to widely distributed radioactive contamination once the oil is offloaded and distributed through the oil supply chain. A thermal neutron flux limit of $2x10^5$ neutrons cm^{-2} s^{-1} is assumed as an acceptable level, based on experience.

3.9.1.5 Security

One of the risk factors that prevented significant commercialization of nuclear reactors in the past is the fear that highly radioactive material on the one hand or highly enriched nuclear fuel on the other hand could get lost or fall into the wrong hands. Both of these substances could be very dangerous in the wrong hands, whether via ill-intent, or unintended. Proliferation resistance is one of the stated goals of the Generation Four Initiative, as indicated in Section 1.2.1.

A relatively new shipping-related risk is related to piracy. During 2008 there were 293 reported piracy events, amongst those the capture of a VLCC [21]. This risk of piracy placed even more emphasis on the need to prevent proliferation of nuclear fuel.

An obvious requirement for a new ship reactor would therefore be to minimize the potential for loss of nuclear fuel and in addition, utilize fuel that would not be attractive for making nuclear weapons.

3.9.2 Regulatory Requirements

The nuclear industry is probably the most strictly regulated of all industries. Each country that utilizes nuclear power has its own nuclear regulatory body that is legally authorized to protect the public and the environment against the dangerous effects of ionizing radiation. For this purpose, most regulatory bodies have developed a set of regulations to govern all nuclear-related installations and activities via the granting of licenses.

The International Atomic Energy Agency (IAEA), which is part of the United Nations, fulfils the role of an international quasi-regulator, even though it is not a regulator which has the power to grant a license for a nuclear system. The IAEA attempts to foster convergence amongst the different national regulatory bodies, by developing standards that can be applied by member countries. Unfortunately, the process of convergence is slow and has not made much impact yet, with the result that each of the major nuclear countries has its own set of regulations that are in general not well aligned with each other.

This fragmented international regulatory framework has a direct impact on the development of a commercial nuclear ship engine, since the nuclear ship might have to be separately licensed for each country that would be visited during operation. This would result in a very

costly and time-consuming process, since the process to obtain a license for a new type of nuclear reactor could easily extend over more than a decade!

For this study, the most significant safety requirements from the IAEA Safety Requirements publication, "Safety of Nuclear Power Plants: Design", NS-R-1 [37] are used:

a. Utilize the Defence in Depth concept to ensure multiple independent layers of protection.
b. Ensure that the three Fundamental Safety Functions are always maintained:
    i. Control of the reactivity.
    ii. Removal of heat from the core.
    iii. Confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases.
c. Two diverse reactor shutdown systems shall be provided.

3.10 TABLE OF INITIATING REQUIREMENTS

The requirements that were developed in Chapter 3 are listed in Table 9 below. These requirements are used in Chapter 4 to develop a conceptual design for a nuclear propulsion system.

Table 9: Initiating Requirements

No.	Requirement	Criteria
P1	Propulsion force	280 kN
P2	Propulsion power	25 MW
P3	Propeller speed (if applicable)	75 rpm
P4	Minimum power level from reactor	50% of full power
P5	Auxiliary diesel power	2 times 1,5 MW
P6	Indirect drive via electrical connection to propeller	-
P7	Ship speed (maximum continuous rating)	15,5 knots
C1	Capital cost constraint	\$8,16m (n^{th} capital cost)
C2	Size constraint	1000 m^3
C3	Weight constraint	2000 ton
O1	Avoid tall, heavy structures: optimize the centre of gravity to be as low as possible	-
M1	Typical power profile during a typical mission	Refer to 3.6
V1	Value system	Life-cycle cost
S	Safety-related requirements:	-
S1	Impossibility of a prompt critical reactor.	-
S2	Impossibility of a core meltdown	-
S3	Impossibility of a nuclear explosion independent of the actions of the operating crew.	Frequency less than 10^{-6} per year
S4	Simple and robust to operate and to maintain	-
S5	Passive safety, without requiring interventions	-

No.	Requirement	Criteria
	by personnel or back-up power sources	
S6	Limit personnel exposure	Effective dose < 20 mSv per year during normal operation for radiation workers Effective dose < 1 mSv/ year for non-radiation workers.
S7	Limit neutron activation of cargo	Thermal neutron flux below $2x10^5$ neutrons cm^{-2} s^{-1}
S8	Minimize the potential for loss of nuclear fuel	-
S9	Utilize fuel that would not be attractive for making nuclear weapons	-
S10	Utilize the Defence in Depth concept in the design and operation	Reference: NS-R-1
S11	Control of the reactivity	Reference: NS-R-1
S12	Removal of heat from the core	Reference: NS-R-1
S13	Confinement of radioactive materials	Reference: NS-R-1
S14	Two diverse reactor shutdown systems	Reference: NS-R-1

4. CONCEPTUAL DESIGN

This section presents a systematic synthesis of a nuclear propulsion system that aims to comply with the requirements that were developed in Section 3. The synthesis is done by means of trade-off studies, among several alternatives, to select the most appropriate design solution. The overriding goal is to use proven technology or standard components as far as possible to reduce developmental and licensing risks.

For the purpose of this study, the nuclear propulsion system is simplistically divided into three major systems: fuel, reactor and power-conversion system. Different subsections deal with each of these systems.

The type of fuel is dealt with first, since that largely determines the reactor design. The power-conversion system is treated secondly, since its design drives many of the reactor requirements. Finally, the reactor design is treated in Section 4.3.

4.1 FUEL OPTIONS

The type of fuel used in the reactor plays a central role in determining the reactor type and configuration, the safety concept, the operations concept, as well as the radioactive waste management concept.

The requirements that were identified in Table 9, pertaining to the selection of the fuel type, are listed in Table 10 below:

Table 10: Fuel-related Requirements

No.	Requirement
S2	Impossibility of a core meltdown
S5	Passive safety, without requiring interventions by personnel or back-up power sources
S9	Utilize fuel that would not be attractive for making nuclear weapons

Although there are many different types of nuclear fuels, the most common type is metal-encased uranium-dioxide as used in Light Water Reactors. A second type of fuel that has been applied in several reactors is ceramic-coated uranium-dioxide particles. An example of this type of fuel is the TRISO particles shown in Figure 18 [71]. The three requirements listed in Table 10, lead to a preference for a ceramic-type coated particle fuel, rather than metallic fuel pins. Table 11 below lists some of the characteristics, indicating the main differences between these two types of fuel.

Although other types of nuclear fuel can also be used, these are not investigated here, because such investigation is considered to be outside the scope of this project.

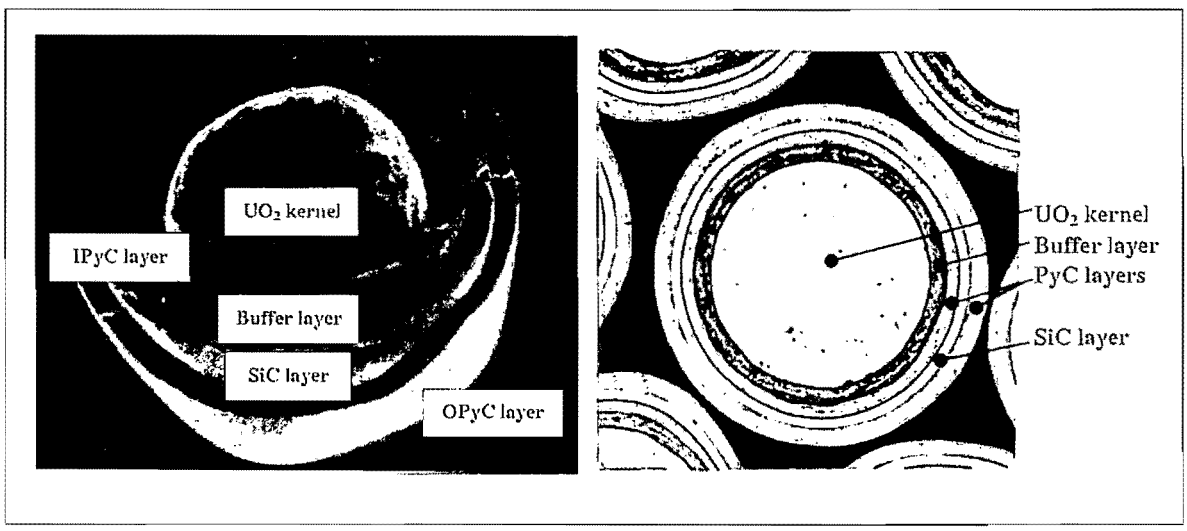


Figure 18: TRISO-coated Particles

Table 11: Comparison between Ceramic-coated Fuel and Metallic Pin Fuel

No.	Requirement	Ceramic-coated Fuel	Metallic Pin Fuel
S2	Impossibility of a core meltdown	Stable barrier below 1600 °C. Impossible to melt: carbon sublimates at 2800 °C.	Barriers break down above 1200 °C [72]. Zircaloy fuel tubes melt between 1760 and 2040 °C (dependent on the

No.	Requirement	Ceramic-coated Fuel	Metallic Pin Fuel
			oxygen concentration) [13].
S5	Passive safety, without requiring interventions by personnel or back-up power sources	Passive safety has been demonstrated. Safety concept relies on millions of small pressure vessels (in the form of a silicon carbide layer), containing fission products at the source.	Safety concept typically relies on active cooling systems and a concrete containment structure.
S9	Utilize fuel that would not be attractive for making nuclear weapons	The layers of the coated particles are difficult to remove, making it extremely difficult to remove plutonium from spent fuel.	Well-established processes can be used to remove plutonium from spent fuel.

In addition to these factors, coated particle fuel has several other advantages:

a. The stable ceramic fuel provides long-term stability in a waste repository.
b. The volume of spent fuel can easily be reduced by removal of the graphite matrix around the fuel particles.
c. Higher fuel burn-up can be achieved, resulting in less waste per volume of heavy metal used.

In order to comply with requirement S9 above, the fuel used in this application should make use of low enrichment (below 20%).

Based on these factors, ceramic-coated fuel (TRISO particles) is selected for this application.

4.1.1 Comparison of Prismatic versus Pebble Fuel in the Context of a Ship

There are basically two types of ceramic-coated fuel, categorized in accordance with the shape of the fuel elements. These are pebbles (or spheres) on the one hand and prismatic blocks or tubes on the other. Pebbles are typically used in a random bed structure such as the Pebble Bed Reactor, while blocks or tubes are stacked in a fixed structure. The advantages and disadvantages of the two fuel shapes in a ship application are shown in the table below:

Table 12: Comparison between Pebble and Prismatic Fuel

Aspect	Pebble Bed Reactor	Prismatic Fuel Reactor
Structural stability	Inferior: dynamic core can be compacted by mechanical impact (such as rough sea, or bumping into another ship).	Superior: fixed core The HTGR ship engine design, presented in [24], preferred prismatic block fuel, based on concerns about pebble movement in the maritime environment.
	Shutdown elements cannot be located inside the fuel zone.	Fixed-structure cores allow for shutdown elements inside the core
Safety	If on-line refuelling is used, very low excess reactivity is	Large excess reactivity just after

Aspect	Pebble Bed Reactor	Prismatic Fuel Reactor
	required Fuel elements experience isotropic thermal gradient	refuelling Increased personnel radiation exposure during open refuelling Fuel elements experience a significant thermal gradient that could lead to kernel migration (amoeba effect) at elevated burn-up, and increased fission product attack of the silicon carbide boundary.
On-line refuelling	Possible, but removed, spent fuel may be problematic on a ship	Impossible: batch refuelling
Batch refuelling	Possible	Possible
Burn-up management	More complex to predict: influenced by the random nature of fuel movement through the reactor. Could be measured individually in a multi-pass reactor system, but this would probably not be practical in a ship application.	Can be predicted with high accuracy
Fuel temperature	Lower peak fuel temperatures: improved safety	Higher peak fuel temperatures (typically 200 ºC higher for the same reactor outlet temperature and coolant flow rate [65]), resulting in higher fission product release from fuel at the same reactor outlet temperature.
Flux shape	Axial shape can be improved by cycling the same fuel several times through the reactor. Radial shape can be improved by utilizing several radial fuelling zones.	Flux shape can be managed during refuelling by radial and axial enrichment zoning.
Reactor length	Longer for same core height due to loading and unloading cone	-
Pressure drop (efficiency)	Inferior	Superior
Orientation of reactor	The reactor has to be in the shape of an upright (vertical) cylinder	Orientation independent

4.1.2 Fuel Management

Spent-fuel management on a ship is significantly different from a typical land-based nuclear power plant. A nuclear-powered ship would have the opportunity to unload its spent fuel only when it enters a special port that is equipped and licensed to handle high-level nuclear waste. Loading of fresh fuel onto a ship would require a licensed facility with security controls to ensure that safeguards against loss (accidental or malicious) are in place. The safety concerns are, however, less significant, because the radiation risk from fresh fuel is much lower.

It is evident that a logistics supply system would have to be developed to support the fuel operations of a nuclear ship, along with the other support requirements of such a ship.

From a safety and security perspective, it would be a significant advantage if there was no need to carry spent fuel on the ship.

4.1.3 Pebble Refuelling Options

Several different pebble fuelling regimes can be used to maintain the desired amount of excess reactivity in the core. The choice has a significant impact on the spent-fuel management, the capital cost of the system as well as the neutron flux shape in the reactor. A few options are discussed below:

4.1.3.1 MEDUL (Mehrfachdurchlauf - German for 'multi-pass)

In the MEDUL regime, each fuel element is passed through the core several times before it reaches its final burn-up and is discarded as spent fuel. A significant advantage of passing each fuel element several times through the reactor is that the axial flux distribution is flattened to some extent. This is a result of the reduced reactivity gradient over the axis of the reactor. With only a single pass (OTTO), all the fuel at the top of the reactor would be fresh, contributing maximum reactivity, while all the fuel at the bottom would contribute very little reactivity.

A significant cost and complexity disadvantage of the MEDUL regime is that the burn-up of each fuel element has to be measured in order to decide whether it should be discarded or returned to the reactor. A second disadvantage of the MEDUL regime, especially pronounced when many passes are utilized, is additional mechanical wear on the fuel elements, which will produce dust in the primary circuit.

In a ship application, the MEDUL regime seems impractical because of the resultant continuous stream of spent fuel that has to be carried on board the ship, requiring cooling, shielding, sub-criticality control and security.

4.1.3.2 Once Through Then Out (OTTO) Cycle

The OTTO cycle is a special case of the MEDUL regime, where fuel is passed through the core only once and then discarded. This regime is very attractive because of its inherent simplicity: no burn-up measurement is required and the fuel management is significantly simpler than that of the MEDUL. A significant disadvantage of this fuelling regime is the unfavourable axial flux shape that results in a much higher flux at the top of the reactor, compared to the bottom. According to Mulder and Teuchert, this problem is especially significant in reactors producing more than 120 MWt [57].

The OTTO regime has the same disadvantages as the MEDUL in a shipping application and seems impractical.

4.1.3.3 Peu à Peu (little by little in French) (PAP)

This is the simplest way of fuelling a pebble bed reactor: fuel is simply added on a regular basis to compensate for burn-up. At the end, all fuel is removed in one step. The significant disadvantage of this mechanism is that the reactor vessel has to be much taller to accommodate all the fuel until (infrequent) fuel-removal maintenance is carried out. As indicated in Table 9, a tall reactor is undesirable for ship propulsion.

4.1.3.4 Ordered pebble bed

Mr Tian from the Tsinghua University proposed an ordered bed modular reactor for use in maritime applications [68], [67], [69]. According to Tian, the randomly-packed bed is unsuited for maritime applications because vibration or mechanical shocks (that could be caused by ship movement, waves or docking) may cause local or general compaction of the pebble bed, causing sudden changes to the reactivity and/or coolant distribution.

In an ordered bed, the fuel spheres are stacked in a regular pattern (similar to a crystal structure) and would remain fixed until unloading.

Advantages of such an ordered bed include:

a. Enhanced structural stability in comparison to a randomly-packed bed.
b. Structured beds offer the possibility of significantly increasing the pebble packing ratio from the typical 61% up to 76%. This offers the possibility of increasing the specific power of the core.
c. Lower pressure drop over the core (as compared to a randomly-packed bed), resulting in improved cycle efficiency.
d. Batch refuelling is better suited to shipping operations.

Disadvantages of an ordered bed:

a. High excess reactivity after a new loading of fresh fuel, which has safety implications.
b. Due to the fixed positions and orientations of the fuel spheres, this type of core will suffer the same disadvantages as a prismatic core, such as temperature and flux gradients, that could result in creep and kernel migration.
c. Burn-up of fuel elements will be uneven, in accordance with the flux shape, with much higher burn-up close to the centre as compared to the outer layers of fuel. This could be improved by the use of graphite balls, or perhaps utilizing different fuel zones with different enrichments.
d. Complex and time-consuming processes will be required to remove spent fuel and refill the reactor with fresh fuel.

In essence, the ordered bed is very similar to the prismatic fuel form, with the difference that the ordered bed would be much slower to load and unload.

4.1.4 Prismatic Refuelling Options

4.1.4.1 Traditional approach

The tradi*tional approach* that has been followed successfully in prismatic-type reactors is similar to that followed in LWRs, where a portion of the fuel, typically one third, is replaced every 12 months. The different grades of enriched fuel (fresh, one year and two years old) are then arranged in an optimal way to achieve as flat as possible a flux shape as well as to reduce neutron leakage from the core.

This method requires sophisticated remote manipulators and a large, well-shielded space next to the reactor to stack out all the fuel elements for rearrangement.

4.1.4.2 One-piece removal

This refuelling method has been proposed for the MRX Japanese marine reactor design [87], where the entire reactor vessel with internals is replaced in one piece with another reactor vessel, which is pre-loaded with fresh fuel. This procedure would significantly simplify the dockyard operations from a radiological perspective and would also require less time than re-shuffling fuel elements in an open reactor. This method would, however, require very heavy lifting equipment and a ship designed for easy access and removal of the whole reactor.

4.1.5 Conclusion on Fuel Options

From this discussion it seems that prismatic TRISO fuel would be more suited for a ship reactor, especially because:

a. it provides superior safety properties;
b. it enhances stability during ship motion;
c. the capital cost of an on-line fuel handling system on each ship is saved in favour of a more complex fuel replacement operation in a specialized dock;
d. much of the Pebble Bed advantage is lost when on-line refuelling is not utilized;
e. prismatic fuel allows the possibility of a horizontal reactor that is investigated in Section 4.3.

Together with the selection of the fuel type, the moderator is selected to be graphite and the coolant to be helium, following the HTGR industry trend. Figure 19 below shows an example of a typical prismatic fuel element.

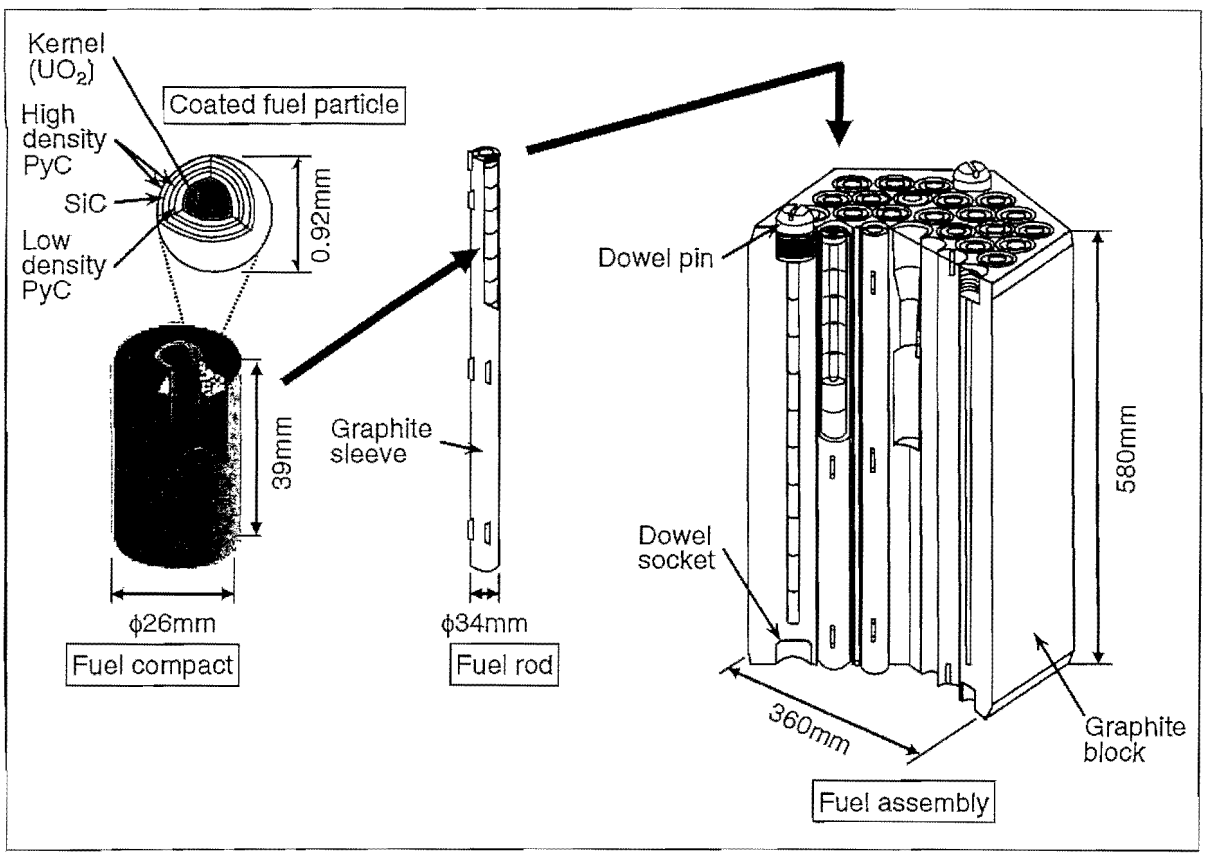


Figure 19: HTTR Fuel Configuration [44]

4.2 PROPULSION MECHANISMS

The vast majority of ships use propellers for propulsion. Large ships such as the VLCC class under consideration for this study utilize large-diameter propellers (typically 10 m diameter) that run as slow as 70 rpm. Refer to Figure 20 below for a picture of a VLCC propeller – note the person below the propeller. These propellers are mostly driven by two-stroke diesel engines.

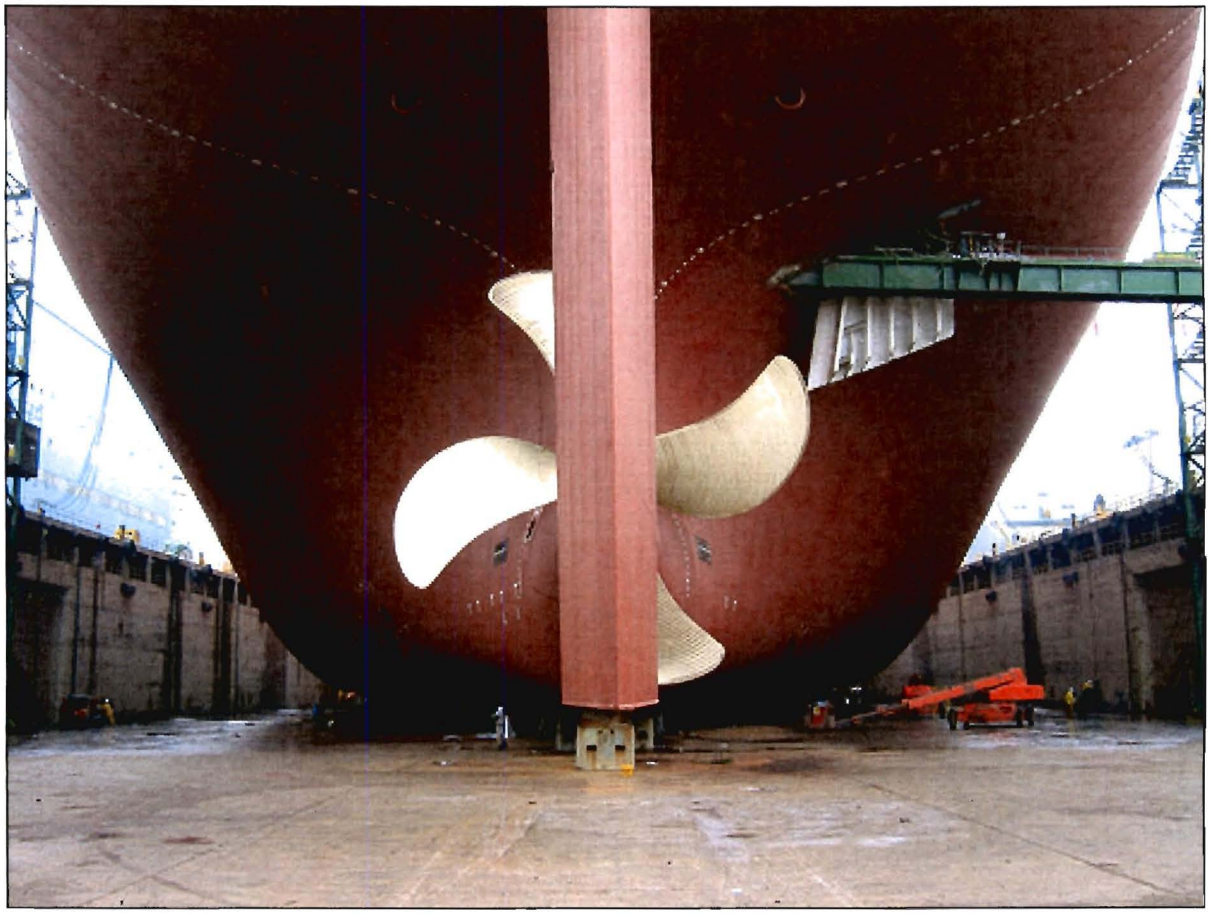


Figure 20: VLCC Propeller

This section considers several different mechanisms for converting fission heat into propulsion. Several power-conversion systems are discussed and one specific method is developed in sufficient detail to enable a cost, size and mass model.

4.2.1 Brayton Cycle Turbine

A Brayton cycle (see Figure 21) offers the possibility of smaller components and higher efficiency. The NEREUS design [6][24], Adams atomic engines [10] as well as the Jacobs study [43] investigated this approach in much detail. The PBMR DPP-400 is an example of such a design that is expected to achieve a thermal efficiency of more than 40%. The high cycle efficiency is, however, dependent on a high reactor outlet temperature, which brings with it material-related risks for the nuclear fuel, hot gas ducts as well as turbine blades. One of the significant disadvantages of a direct Brayton cycle coupled to a nuclear reactor is that the entire power-conversion circuit is exposed to potential radioactive contamination. An indirect Brayton cycle (as proposed by Jacobs [43]), which uses an intermediate heat exchanger, could remove this problem, but at the cost of an expensive and large additional

heat exchanger. Due to the complexities of the maintenance on the relatively large and radioactive primary loop, this design is not preferred for a ship engine.

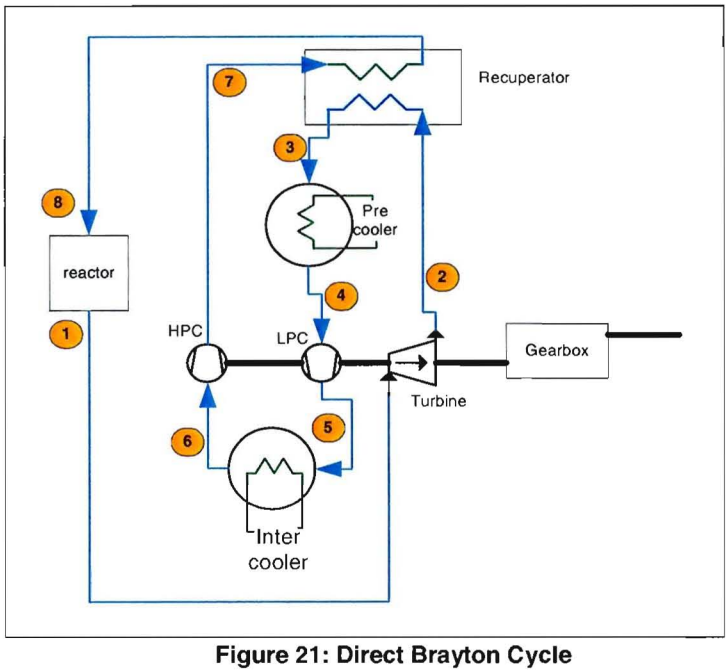


Figure 21: Direct Brayton Cycle

4.2.2 Direct Cycle Steam Turbine

A novel variant of the Rankine cycle that should be investigated in future studies is to pump water directly through an HTGR core. Refer to

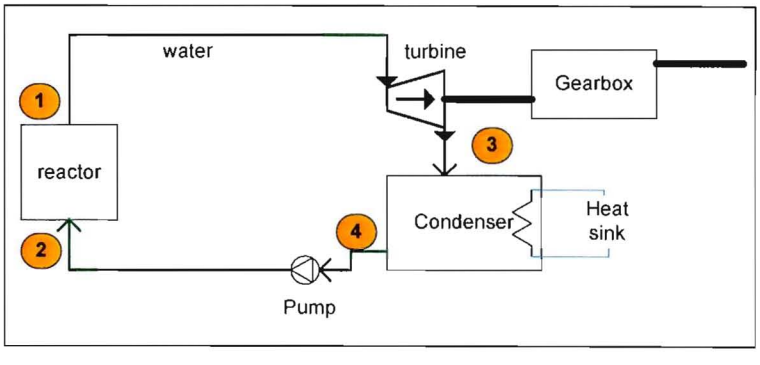


Figure 22: Direct Steam Cycle

Figure 22 for a cycle diagram. If the steam outlet temperature is sufficiently low (< 500 ˚C), steam can be generated inside the core of the reactor in zircalloy tubes. Zircalloy has a very high melting point (~2000 ˚C) and would thus be able to withstand the high temperatures during passive decay heat-removal transients. The neutronic design should be such that the absence of liquid water (moderator) would result in a sub-critical reactor. The steam from the reactor is then directly fed to a steam turbine, and after condensing pumped back into the reactor. This holds the promise of a very simple engine, but increases the risk of accidental water ingress into the fuel zone of the reactor.

4.2.3 Direct Drive Concept

An unconventional propulsion concept that provides direct propulsion from the nuclear heat source is proposed in Figure 24. This concept does not require a turbine to convert heat into rotation and then into a propulsion force. Instead, this concept uses a similar principle as a ramjet to provide propulsion from heat. A nuclear-powered ramjet has been developed in the American Project Pluto for a nuclear-powered missile, where incoming air was forced under high pressure directly through the core of a reactor – refer to Figure 23.



Figure 23: Prototype Pluto Engine

4.2.3.1 Derivation of direct-drive propulsion force

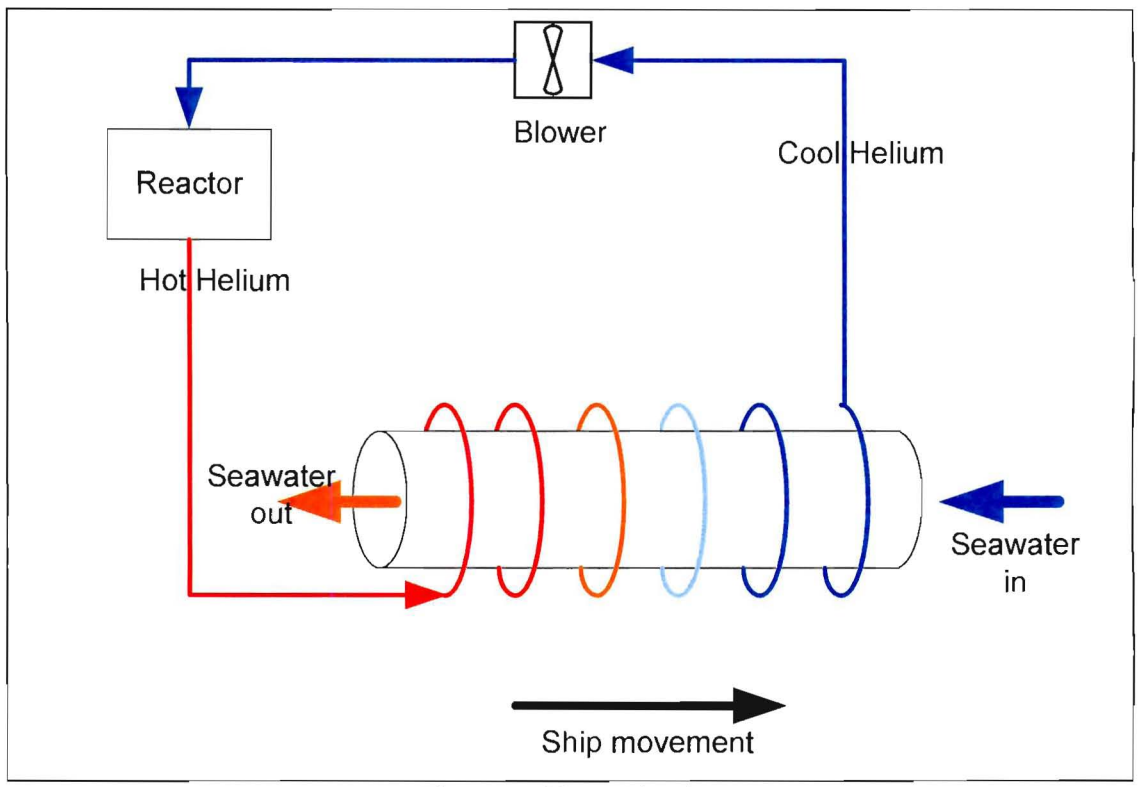


Figure 24: Direct-drive Concept

The propulsion force from such a system ($F_{propulsion}$) can be described by:

$$F_{propulsion} = \dot{m}_{water}\left(V_{water_out} - V_{water_in}\right) \quad \text{.....(Eq 4.1)}$$

Where $\dot{m}_{water}$ represents the water mass flow rate and V_{water_out} and V_{water_in} respectively the water velocity at the outlet and inlet of the pipe. Obviously the ship will need some minimum forward velocity before this propulsion force will become effective. This would have to be supplied by a secondary engine.

But $$V_{water_out} = \frac{\dot{m}_{water_out}}{\rho_{out} \times A_{out}} \quad \text{.....(Eq 4.2)}$$

Where A represents the area of the heat transfer pipe and ρ the density of the water.

Thus $F_{propulsion} = \dot{m}_{water_out}\left(\frac{\dot{m}_{water_out}}{\rho_{out} \times A_{out}} - \frac{\dot{m}_{water_in}}{\rho_{in} \times A_{in}}\right)$ .....(Eq 4.3)

But $\dot{m}_{water_out} = \dot{m}_{water_in}$

Thus $F_{propulsion} = (\dot{m}_{water})^2\left(\frac{1}{\rho_{out} \times A_{out}} - \frac{1}{\rho_{in} \times A_{in}}\right)$ .....(Eq 4.4)

For simplicity, assume that the inlet and outlet area of the pipe is equal: $A_{in} = A_{out}$ (in actual operation, a pump should be used at the inlet and a diffuser at the outlet of the pipe).

The pipe diameter is related to the water mass flow:

$\dot{m}_{water} = A_{in} \times V_{water_in} \times \rho_{seawater}$ .....(Eq 4.5)

Thus $F_{propulsion} = (A \times V_{water_in} \times \rho_{seawater})^2\left(\frac{1}{\rho_{out} \times A} - \frac{1}{\rho_{in} \times A}\right)$

$F_{propulsion} = A \times (V_{water_in})^2 \times (\rho_{seawater})^2 \times \left[\frac{1}{\rho_{out}} - \frac{1}{\rho_{seawater}}\right]$ .....(Eq 4.6)

The density of the seawater at the exit of the pipe is dependent on the amount of energy added to the incoming water. For simplicity, ignore the energy required to heat the water up to boiling point. The exit density $\rho_{out} = x \times \rho_{steam} + (1-x) \times \rho_{seawater}$, where x is the quality factor of the exiting fluid.

The quality factor would be dependent on the rate of heat supplied $\dot{Q}$ to the stream of water. When $\dot{Q} = \dot{m}_{water} \times h_{fg}$ (with h_{fg} enthalpy of vaporisation for water), then the quality factor would be 1, while the quality factor would be zero if no heat is added. Thus $x = \frac{\dot{Q}}{\dot{m} \times h_{fg}}$ over the range of x between 0 and 1.

Substituting ρ_{out} into (Eq 4.6):

$F_{propulsion} = A \times (V_{water_in})^2 \rho_{seawater} \times \left[\frac{\rho_{seawater}}{\rho_{seawater}[1-x] + x \times \rho_{steam}} - 1\right]$ .....(Eq 4.7)

Substituting x into (Eq 4.7) gives:

$$F = A \times V^2 \rho_{sea} \times \left[\frac{\rho_{sea}}{\rho_{sea}\left[1 - \frac{\dot{Q}}{A \times V \times \rho_{sear} \times h_{fg}}\right] + \rho_{steam}\frac{\dot{Q}}{A \times V \times \rho_{sea} \times h_{fg}}} - 1\right] \quad \text{.....(Eq 4.8)}$$

From Section 3.2.1, the required propulsion force is calculated from

Where A represents the area of the heat transfer pipe and ρ the density of the water.

Thus $F_{propulsion} = \dot{m}_{water_out}\left(\frac{\dot{m}_{water_out}}{\rho_{out} \times A_{out}} - \frac{\dot{m}_{water_in}}{\rho_{in} \times A_{in}}\right)$ .....(Eq 4.3)

But $\dot{m}_{water_out} = \dot{m}_{water_in}$

Thus $F_{propulsion} = (\dot{m}_{water})^2\left(\frac{1}{\rho_{out} \times A_{out}} - \frac{1}{\rho_{in} \times A_{in}}\right)$ .....(Eq 4.4)

For simplicity, assume that the inlet and outlet area of the pipe is equal: $A_{in} = A_{out}$ (in actual operation, a pump should be used at the inlet and a diffuser at the outlet of the pipe).

The pipe diameter is related to the water mass flow:

$\dot{m}_{water} = A_{in} \times V_{water_in} \times \rho_{seawater}$ .....(Eq 4.5)

Thus $F_{propulsion} = (A \times V_{water_in} \times \rho_{seawater})^2\left(\frac{1}{\rho_{out} \times A} - \frac{1}{\rho_{in} \times A}\right)$

$F_{propulsion} = A \times (V_{water_in})^2 \times (\rho_{seawater})^2 \times \left[\frac{1}{\rho_{out}} - \frac{1}{\rho_{seawater}}\right]$ .....(Eq 4.6)

The density of the seawater at the exit of the pipe is dependent on the amount of energy added to the incoming water. For simplicity, ignore the energy required to heat the water up to boiling point. The exit density $\rho_{out} = x \times \rho_{steam} + (1-x) \times \rho_{seawater}$, where x is the quality factor of the exiting fluid.

The quality factor would be dependent on the rate of heat supplied $\dot{Q}$ to the stream of water. When $\dot{Q} = \dot{m}_{water} \times h_{fg}$ (with h_{fg} enthalpy of vaporisation for water), then the quality factor would be 1, while the quality factor would be zero if no heat is added. Thus $x = \frac{\dot{Q}}{\dot{m} \times h_{fg}}$ over the range of x between 0 and 1.

Substituting ρ_{out} into (Eq 4.6):

$F_{propulsion} = A \times (V_{water_in})^2 \rho_{seawater} \times \left[\frac{\rho_{seawater}}{\rho_{seawater}[1-x] + x \times \rho_{steam}} - 1\right]$ .....(Eq 4.7)

Substituting x into (Eq 4.7) gives:

$$F = A \times V^2 \rho_{sea} \times \left[\frac{\rho_{sea}}{\rho_{sea}\left[1 - \frac{\dot{Q}}{A \times V \times \rho_{sear} \times h_{fg}}\right] + \rho_{steam}\frac{\dot{Q}}{A \times V \times \rho_{sea} \times h_{fg}}} - 1\right]$$ .....(Eq 4.8)

From Section 3.2.1, the required propulsion force is calculated from

$$R_{drag} = \frac{1}{2} \times \rho_{sea_water} \times S_{wetted} \times C_F \times V^2 \quad \text{.....(Eq 4.9)}$$

The propulsion force requirement for this study is 324 kN.

In order for the ship to sustain a steady speed in the water, $F_{propulsion}$ must equal F_{drag}.

By equating (Eq 4.6) and (Eq 4.9):

$$A = \frac{0.5 \times C_F \times S_{wetted}}{\left(\frac{\rho_{sea}}{\rho_{out}}\right) - 1} \quad \text{.....(Eq 4.10)}$$

From Section 3.2.1, the parameters C_F = 0,002 and the wetted area is 30720 m^2. The heat transfer pipe area A, from (Eq 4.6) has to be 0,031 m^2 (pipe radius 100 mm) if it is assumed that the water is completely in gas form at the exit of the pipe. This pipe dimension is independent of the ship forward velocity (as long as the drag force equation (Eq 4.9) remains valid).

For the desired normal ship speed of 15,5 knots, as determined in Section 3.2, it implies a mass flow rate (from (Eq 4.5)) of:

$\dot{m}_{water} = 0.031 \times 7.97 \times 1000 = 247$ kg/s

The reactor power required to boil all this water amounts to:

$\dot{Q} = \dot{m}_{water} \times h_{fg}$. For water $h_{fg} = 2.257E6$ J/kg

Thus $\dot{Q} = 247 \times 2.257E6 = 557$ MW, which is 22 times more than the conventional 25 MW diesel engine that is typically employed by a VLCC tanker!

The exit gas velocity would be 6300 m/s, which is much faster than the speed of sound in water (1500 m/s) and will probably result in the pipe choking the flow of the water!

Figure 25 and Figure 26 below provide an indication of the sensitivity of the propulsion force (for a given pipe diameter) to the amount of heat supplied to the water. It is clear that for maximum efficiency, the quality of the exit steam should be 1 and that almost no propulsion force is created if the exit steam quality is less than 0,99.

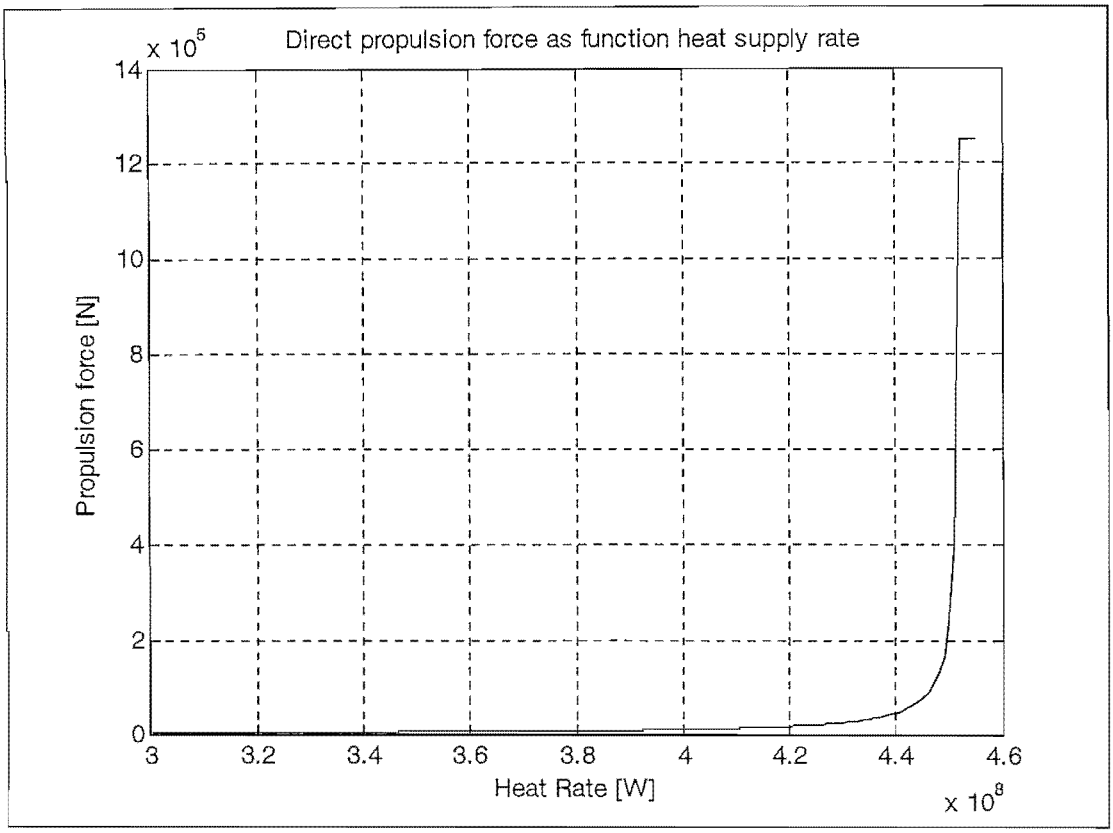


Figure 25: Direct Propulsion Force as Function of Heat Rate

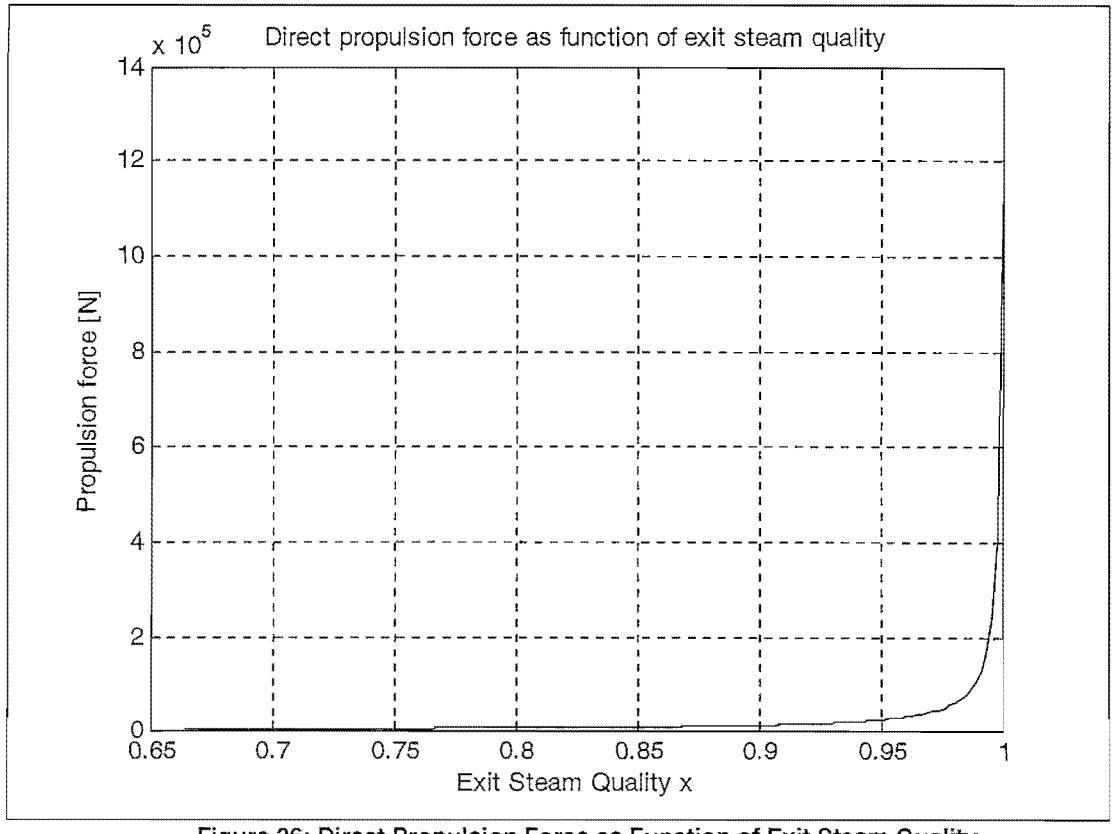


Figure 26: Direct Propulsion Force as Function of Exit Steam Quality

4.2.3.2 Conclusion on direct-drive concept

Although this direct-drive concept offers huge improvements in simplicity, it does not seem feasible, due to its inefficient energy use and impractical exit steam velocity.

4.2.4 Rankine Cycle Steam Turbine

This approach uses the power-conversion unit that is traditionally used in LWR power systems. A primary loop (that for an HTGR typically utilizes He as working fluid) is coupled to a second steam loop via a steam generator. The steam loop consists of a power turbine, a condenser cooled by a heatsink, a feed-water re-heater and a feed-water pump that supplies water into the steam generator. In a typical LWR application, such a cycle has an efficiency of around 33%, using sub-critical saturated steam. In an HTGR application, the steam can be superheated to enable efficiencies of more than 40%. Steam temperatures of around 600 °C can be utilized by standard steam turbines without special material concerns, although then at slightly lower efficiencies.

Two very significant advantages of using a Rankine cycle power-conversion system are:

a. The steam loop is very similar to conventional steam turbine cycles, which have been used for decades in the shipping industry. See a typical unit in Figure 27. This will enhance acceptance in the shipping industry and will also simplify the development process, due to the availability of standardized steam power-conversion units. This supports the cost constraint C1 as well as the simplification Requirement S4.

b. The steam generator serves as a very effective boundary for any radioactive fission products or activation products, implying that the secondary loop is virtually free of radioactivity. This enhances maintenance and operation in support of Requirement S4 and S6.



Figure 27: A Standardized Marine Steam Turbine Unit (Dresser-Rand)

Due to these advantages as well as the availability of engineering information, the Rankine cycle is the preferred choice for this study.

Two very useful HTGR designs are available that utilize a Rankine cycle power-conversion unit: the HTR MODUL and the MHTGR. These designs can be used to establish a basis for the ship engine and provide information for scaling.

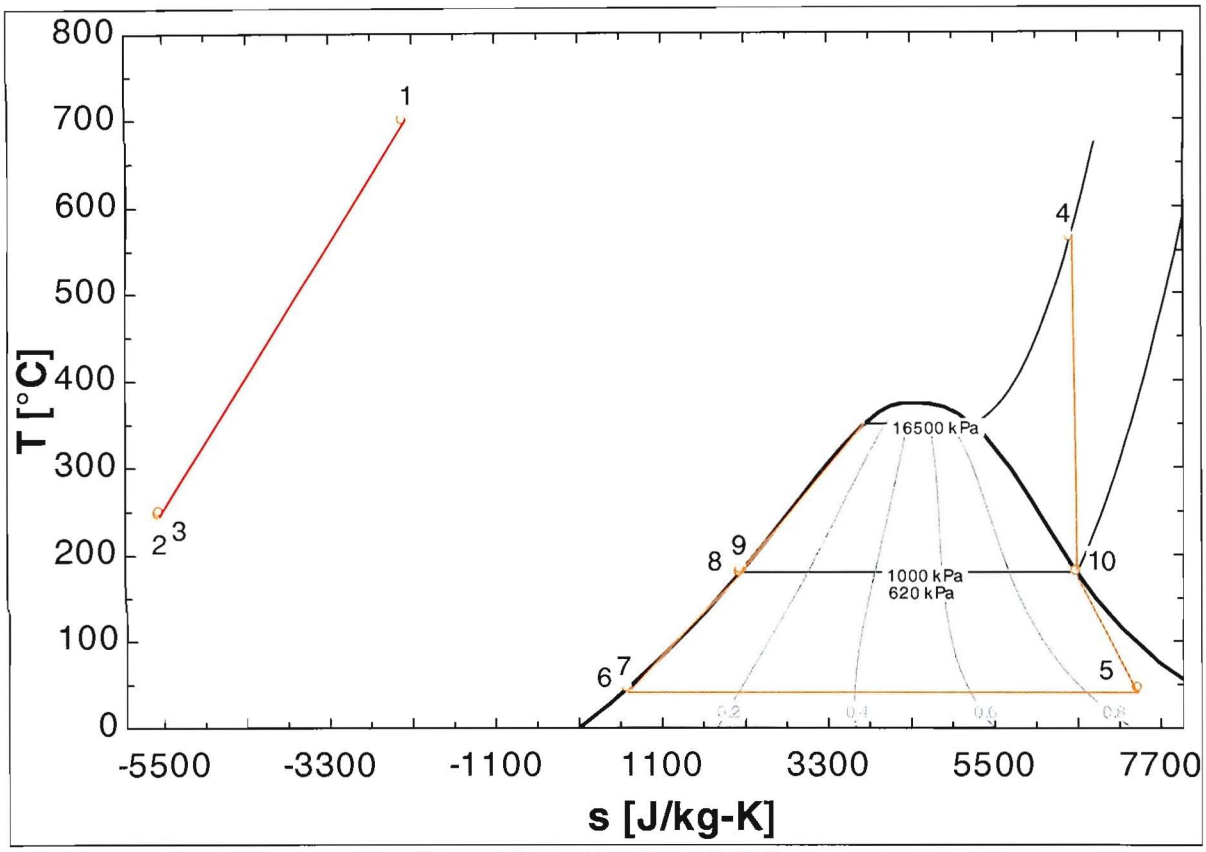


Figure 28: Rankine Cycle T-S Diagram

4.2.4.1 HTR MODUL

The HTR MODUL was designed in Germany during the late 1980s by Siemens Interatom, but was never constructed, due to political pressure after the Chernobyl accident in 1985. Much detail about this design is provided by Kugeler and Schulten in their handbook on Hochtemperaturreaktortechnik [46]. Pertinent parameters for this design are listed in Table 13.

4.2.4.1.1 MODUL steam generator

In the MODUL design, the steam outlet pressure was selected to be 190 bar at a temperature of 530 ˚C, while the feed-water temperature to the steam generator was selected as 170 ˚C. The MODUL primary loop was selected to have a reactor outlet temperature of 700 ˚C and a reactor inlet temperature of 250 ˚C. A once-through helical tube heat exchanger was selected with the water inside the tubes. This steam generator was designed to have a heat exchanger surface area of 2100 m^2. The steam generator vessel (incorporating the primary loop-blower) was designed to be 21,7 m high.

Table 13: HTR MODUL Parameters

Parameter	Unit	Primary Side Value	Secondary Side Value
Thermal power	MW	200	
Mass flow rate	kg/s	85	77
Inlet temperature	˚C	700	170
Outlet temperature	˚C	245	530

Parameter	Unit	Primary Side Value	Secondary Side Value
Outlet pressure	bar	60	180
Pressure drop	bar	0,4	20
Heat exchanger surface area	m^2	2100	
Steam generator vessel height	m	Approximately 22 (including blower)	

Another variant of this design is the HTR-100, which combined the reactor and steam generator into a single pressure vessel that is approximately 30 m long and 6 m in diameter. A 30 m high pressure vessel would not be practical on a ship, but if this vessel can be horizontally oriented, it would suit a ship application very well. This option is investigated in Section 4.3.6.

4.2.4.1.2 MODUL blower

The MODUL blower is integrated into the top shell of the steam generator. The concept design was for a single-stage radial impeller. This blower had the following functional characteristics:

Mass flow: 85 kg/s

Motor rating (electrical): 2,95 MW

Pressure head: 1,5 bar

The dimensions for this blower are: 3,8 m length with a diameter of 1,9 m. The anticipated blower mass is approximately 11,5 tons.

4.2.4.2 MHTGR

The MHTGR design [12] was developed by General Electric in the USA during the 1980s. Table 14 and Figure 29 contain details of this design. It is interesting to see in Figure 29, the different regions where the water is heated up, where steam is formed and where the steam is superheated. This unit has a 350 MWt capacity. In this design, the steam generator has a total weight of 232 ton without any water and 245 ton with water in. The dimensions of the steam generator are: 4,34 m diameter and 17,7 m height.

Table 14: MHTGR Steam Generator Parameters

Parameter	Units	Value
Inlet helium flow rate	kg/s	157,2
Inlet helium temperature	°C	685
Inlet helium pressure	MPa	6,31
Outlet helium temperature	°C	255
Steam generator helium pressure drop	kPa	25,5
Inlet feed-water flow rate	kg/s	137,2
Inlet feed-water temperature	°C	193

Parameter	Units	Value
Inlet feed-water pressure	MPa	20,7
Outlet steam temperature	°C	540
Outlet steam pressure	MPa	17,3

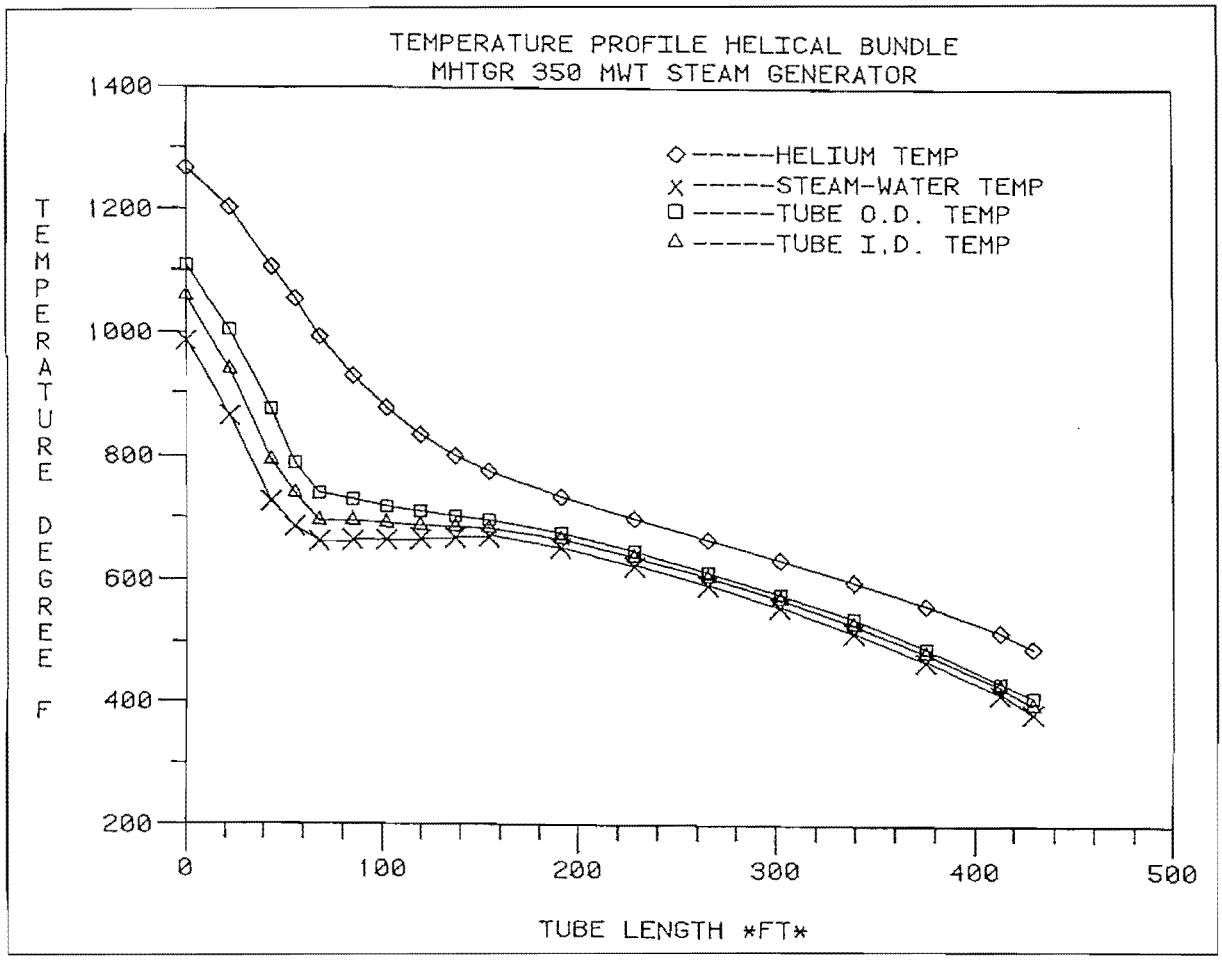


Figure 29: MHTGR Steam Generator Temperature Profile

4.2.4.3 Rankine ship engine design

Based on these precedents, the most important parameters for a conceptual Rankine engine for the ship are presented below in Table 15.

Table 15: Rankine Ship Engine Parameters

Parameter	Value	Basis
Reactor pressure drop [kPa]	28	Refer to Section 4.3.4
Reactor thermal power	80 MW	Assume overall efficiency of 31%
Reactor diameter	2 m	Refer to Section 4.3
Reactor height	6 m	Refer to Section 4.3
ROT [°C]	700	MODUL and MHTGR design
RIT [°C]	250	MODUL and MHTGR design
Primary loop pressure [MPa]	6	MODUL and MHTGR design

Parameter	Value	Basis
Outlet steam pressure [MPa]	16,5	Achievable with the standard Siemens SST-700 steam turbine
Condenser pressure [kPa]	10	Typical value maintained by a vacuum pump
Steam exit quality at turbine	0,9	Assumption
Blower isentropic efficiency	0,9	Assumption
Blower electrical efficiency	0,95	Assumption
Steam generator pressure drop on primary side [kPa]	40	MODUL design
Steam generator pressure drop on secondary side [kPa]	2000	MODUL design
Steam temperature [°C]	565	Achievable with the standard Siemens SST-700 steam turbine
Feed-water reheat	Optimized for cycle efficiency	

These parameters were used to construct an EES model as shown in Appendix C for a system as set out in Figure 30. The T-s diagram of this system is shown in Figure 28. Note that the cycle efficiency can be improved significantly by increasing the steam temperature. In this conceptual design, the conservative approach is taken to keep the steam temperature within the bounds of commercially available steam turbines.

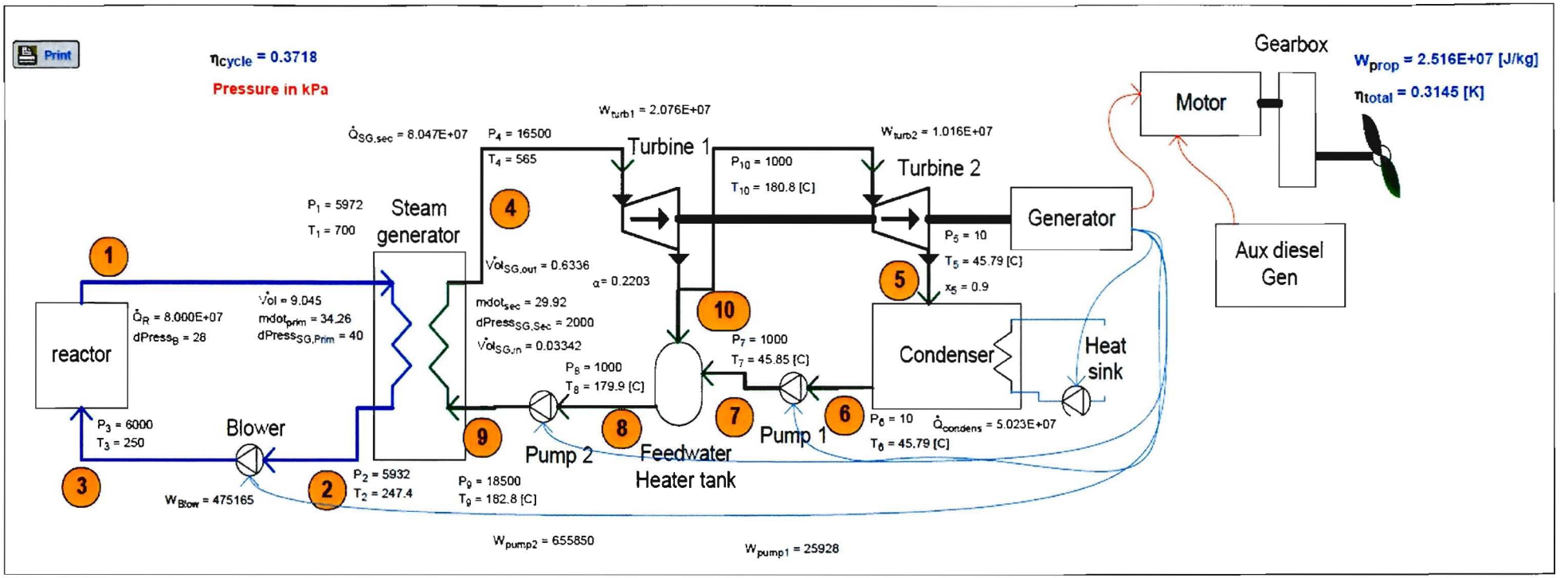


Figure 30: Rankine Ship Engine Diagram

4.2.4.4 Sizing and costing

Based on this conceptual design, the power rating of the main equipment is as follows:

Table 16 : Rankine Engine Equipment Power Rating

Equipment	Rating
Blower	500 kWe (assuming 95% electrical efficiency)
Steam generator	80,5 MW
Turbine	31 MW
Condenser	50 MW
Feed-water pump no. 1	27 kWe (assuming 95% electrical efficiency)
Feed-water pump no. 2	690 kWe (assuming 95% electrical efficiency)
Gearbox	25 MW

Determination of the approximate physical sizes of the major equipment is dependent on many design decisions, many of which go beyond the scope of this study. In this study, size approximations will be based on scaling of similar equipment in similar applications. Table 19 lists the dimensions of the major equipment.

Capital cost estimates for the most significant equipment are based on an average cost per unit mass for the different components. These cost measures are obviously crude and cannot make provision for unexpected design complexities or the very significant nuclear regulatory process. Unit cost values were collected from individuals with significant experience in designing and building nuclear power and fossil-fired power plants. Table 20 lists the cost estimates for the Rankine engine.

4.2.4.4.1 Primary loop blower

The cost and size estimates are based on personal discussions with Mr Lieb Liebenberg of PBMR (Pty) Ltd. A 500 kW centrifugal blower is estimated to be 1 m long, with an outer diameter of approximately 0,7 m, operating at about 8000 RPM. Such a blower would cost approximately R200k. Electromagnetic bearings would cost another R200k. The blower motor cost is estimated at R1000k, with electrical drive equipment estimated at R375k. The total blower cost is thus estimated to be R1,8m.

A second estimate of the blower dimensions is derived by linearly scaling the MODUL design in accordance with the power of the blower. The ship engine blower power is 16% that of the MODUL blower. It is accepted that equipment parameters do not scale down linearly over a large range and the estimates of Mr Liebenberg are accepted as more accurate.

The two estimates are compared in Table 17 below.

Table 17: Blower Sizing

Parameter	Lieb Liebenberg's Estimate	Scaled MODUL Design
Weight	-	$=\frac{0.5}{3.2}\times 11.5 = 1.8$ ton

Parameter	Lieb Liebenberg's Estimate	Scaled MODUL Design
Length	1 m	$=\frac{0.5}{3.2}\times 4=0.6$ m
Diameter	0,7 m	$=\frac{0.5}{3.2}\times 2=0.3$ m

4.2.4.4.2 Steam generator

Two options are considered for steam generators: a helical shell-in-tube design that is similar to that used in the design of the HTR MODUL, and secondly, a compact heat exchanger similar to the design of the PBMR DPP 400 recuperator. This compact heat exchanger uses small, etched grooves in plates that are diffusion-bonded into a stack, also called a core. Refer to a schematic diagram in Figure 31.

4.2.4.4.2.1 Shell-in-tube

The cost estimate for the shell-in-tube steam generator is based on a PBMR-proprietary model developed by Wilma van Eck [4]. Based on this model, the steam generator has the parameters as indicated in Table 18. For comparison purposes, the HTR MODUL steam generator is scaled down in accordance with its power.

Table 18: Shell in Tube Steam Generator

Parameter	Calculated Value	Scaled HTR MODUL Steam Generator
Number of tubes	219	-
Heat exchanger area	787 m^2	832 m^2
Material selection	2.25 Cr 1 Mo for sub-cooled and evaporation area Alloy 800 H for super-heater	Incoloy 800
Pressure vessel height	11,7 m (allowing space for the blower)	8,6 m
Pressure vessel diameter	3,2 m	-
Mass (dry)	133 ton	-
Cost	€15m (50% of which is for the pressure vessel)	-

4.2.4.4.2.2 Compact heat exchanger

Cost and size estimates for this heat exchanger are based on personal discussions with Mr Lieb Liebenberg, an experienced designer at PBMR (Pty) Ltd, as well as information supplied by Heatric, a division of Meggitt (UK) Ltd. Based on the parameters as shown in Figure 30, it was determined that the heat exchanger should consist of 24 cores, with each core having dimensions of 600 mm by 300 mm by 465 mm. Based on information from Heatric, this is estimated to cost £1 300 000. The weight of the 24 cores is estimated to be 10,6 ton (dry).

These numbers exclude the pressure vessel that has to contain the heat exchanger. A pressure vessel that contains the heat exchanger cores as indicated in Figure 32 is 4,2 m

long, with an outer diameter of 2,4 m as indicated in Figure 33. This vessel would weigh 13 tons and cost about €300k. This heat exchanger is also amenable to be integrated into the Reactor Pressure Vessel as discussed in Section 4.3.1.4.

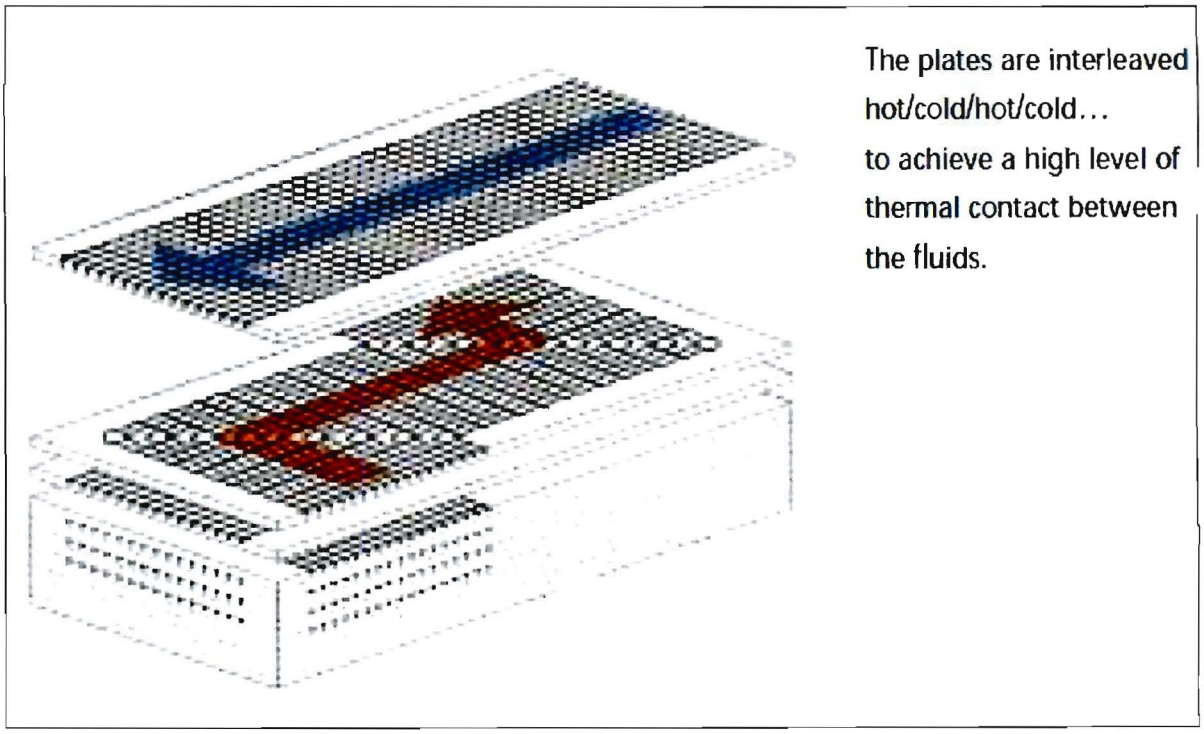


Figure 31: Heatric-type Compact Heat Exchanger

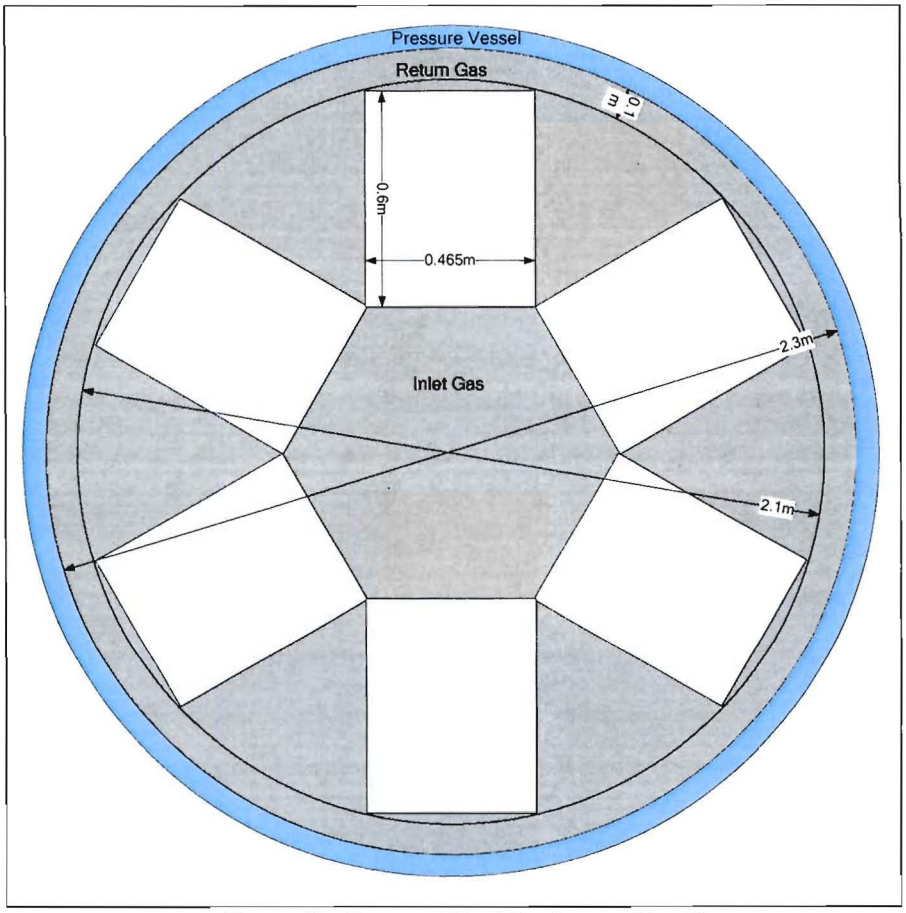


Figure 32: Compact Heat Exchanger Top View

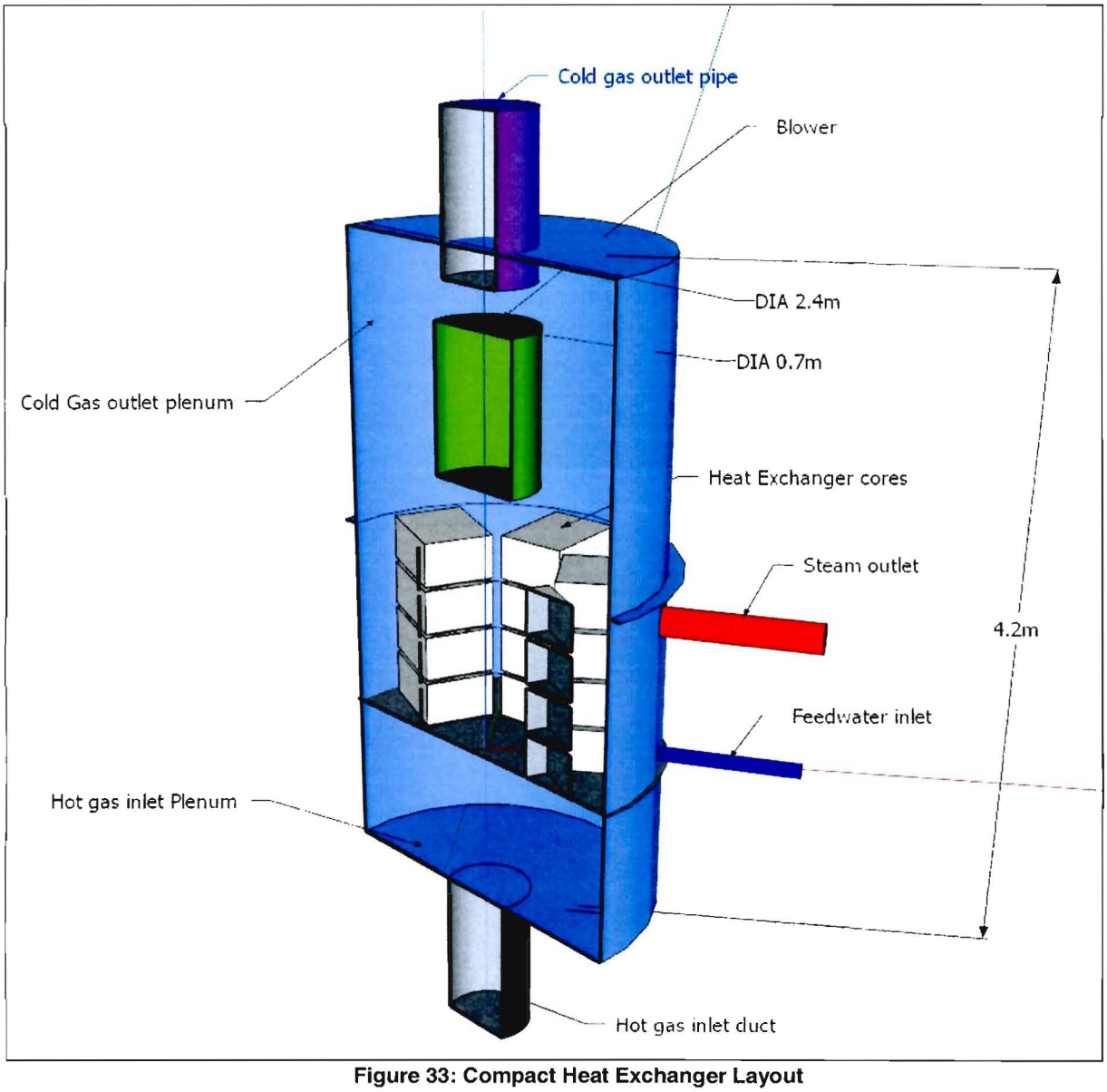


Figure 33: Compact Heat Exchanger Layout

4.2.4.4.3 Steam power-conversion unit (including generator)

The cost estimate for the steam plant is based on work done by Rod Speedy of PBMR Pty (Ltd), with inputs from the "Gas Turbine World 2009 GTW Handbook" [3]. The cost of the steam plant, excluding the steam generator, is estimated to be $741 / kW output power for a 31 MW turbine. For the design under consideration, this translates to $23 million. This estimate is based on 2009 USA dollar values. This value includes the capital cost for the turbine, generator, starting, lubrication and control systems as well as the pipes, condenser and feed-water pump and vacuum pump.

The size of the steam power-conversion unit is based on the 35 MW Siemens SST-600 steam plant, as indicated in Figure 34 [64].

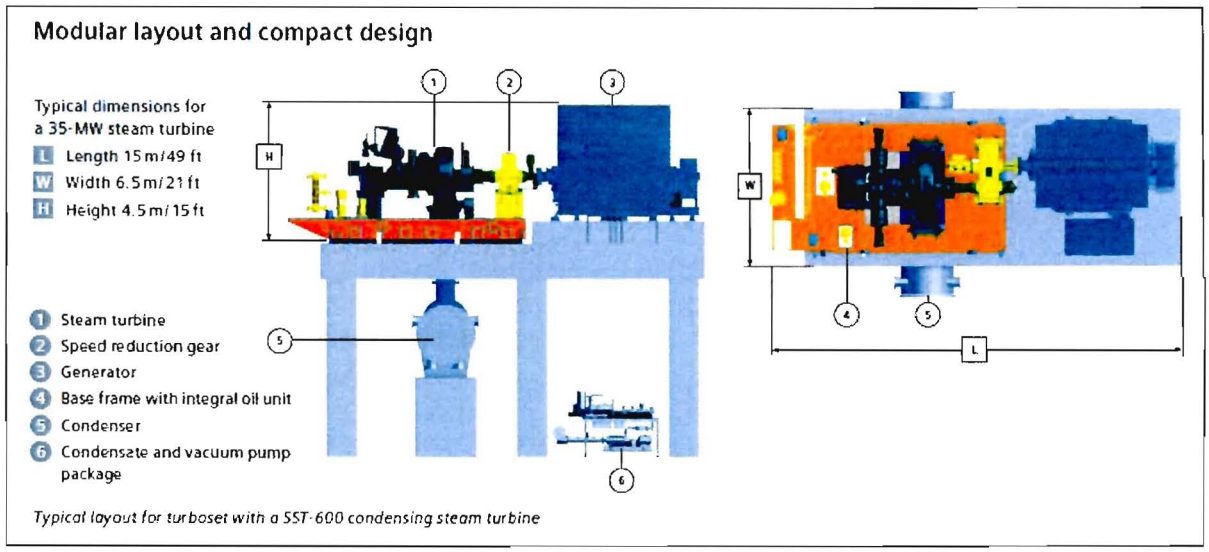


Figure 34: Siemens SST-600 Steam Turbine

4.2.4.4.4 Electrical drive motor

The cost of the 25 MW electrical drive motor with its drive electronics and speed reduction gearbox is estimated to be $4m, in line with estimates made for the Jacobs reactor design [43]. Based on e-mail communication with Mr Philippe Graber from the Converteam Corporation, the drive motor and power electronics would have a mass of 280 ton and the size would be 9 m long, 7 m wide and 5.6 m high.

4.2.4.4.5 Helium

Wilma van Eck's model [4] was used to estimate the helium inventory of this plant. The helium mass is calculated to be 300 kg, costing about €30k, which is negligible.

4.2.4.4.6 Primary-loop ducts, pipes and valves

Wilma van Eck's model [4] was also used to calculate an approximate cost for the hot gas duct that leads the helium from the reactor to the steam generator and the pipe that leads the cold helium back into the reactor. It was assumed that the hot gas duct would be 5 m long. One valve was also included in the primary loop on the cold side of the steam generator to enable blocking of natural circulation through the primary loop during shutdown conditions. The cost of the pipes and valve was estimated to be €5m.

4.2.4.4.7 Control and instrumentation

Control system cost is discussed in Section 4.3.2.3.3.

Table 19: Rankine Engine Equipment Size

Equipment	Size	Mass [ton]
Reactor	Refer to Section 4.3	
Blower	1 m long 0,7 m diameter	1,8 ton

Equipment	Size	Mass [ton]
Compact steam generator	2,1 m diameter 1,2 m high	10,6
Compact steam generator pressure vessel *Note1*	4,2 m long (including the blower) 2,3 m diameter	13
Steam generator shell-in-tube *Note2*	Outer diameter: 3,2 m Height: 10,5 m	130
Steam turbine with condenser and feed-water pump	Length: 15 m Width: 6,5 m Height: 4,5 m (without condenser); estimated 10 m with condenser, vacuum pump and feed-water pump	152 (based on mass of the Siemens SGT-600)
Drive motor and gearbox	Length: 9 m Width: 7 m Height 5.6 m	280

***Note1**: This pressure vessel may not be required if the steam generator is incorporated into the reactor pressure vessel.*

***Note2**: This type of steam generator will not be used in this study*

Table 20: Rankine Engine Cost Model

Item	Cost	Dollar Cost [$m][note1]
Compact steam generator	£1,3m	1,95
Steam generator pressure vessel	€300 k	0,36[note2]
Electrical drive motor and gearbox		4
Blower	R1,8m	0,23
Steam power-conversion unit	$23m	23
Duct, pipes and valves	€5m	6
Total		**35,54**

***Note1**: Refer to Appendix E for currency exchange rates used in this study*

***Note2**: This pressure vessel may not be required if the steam generator is incorporated into the reactor pressure vessel.*

4.3 NUCLEAR REACTOR

The requirements that drive the reactor design come from the requirements analysis in Section 3, as well as the design choices that were made in the fuel selection (Section 4.1) and propulsion design (Section 4.2). The pertinent requirements are consolidated in Table 21 below:

Table 21: Reactor Requirements

No.	Requirement	Value
S	Safety-related requirements as listed in Table 9	-
DP1	Thermal power	80 MW
DF1	Fuel	Prismatic TRISO
DF2	Coolant	Helium
DP2	Coolant outlet temperature	700 °C
DP3	Coolant inlet temperature	250 °C
DP4	Coolant pressure	6 MPa
DP5	Minimum power operation	50%
	Optimization and constraint parameters:	-
C1	Cost	< $14m
C2	Size	< 1000 m^3
C3	Mass	< 2000 ton
O1	Low centre of gravity	-

Instead of starting the reactor design with the proverbial "clean sheet", several precedents are used in this section from which a conceptual reactor design is derived. Table 22 lists the most significant precedents that were considered. Drawings of these reactor designs are contained in Appendix A.

Table 22: Reactor Precedents

	Thermal Power [MW]	Pressure [MPa]	Fuel Elements	Status
PBMR DPP-400	400	9	Pebble bed 6 cm Ø	Basic design
HTR MODUL	200	6	Pebble bed 6 cm Ø	Basic design
HTGR design by General Atomics	350 and 2240	9	Hexagonal blocks: 36 cm flat to flat; 79,3 cm long	Basic design
The high-temperature reactor for ship propulsion by Lobet *et al.* [48]	180	4.5	Hexagonal blocks: 43,7 cm flat to flat	Conceptual
JAERI HTTR	30	4	Hexagonal blocks: 36 cm flat to flat; 58 cm long	Operational since 1998

The reactor design that is developed here is a conventional prismatic High-temperature Gas-cooled Reactor (HTGR). The major elements of the reactor are discussed below:

a. Reactor fuel consists of tiny uranium-dioxide kernels inside TRISO particles (typically less than 1 mm diameter). The coating layers surrounding the uranium provide the primary fission product retention capability. The TRISO particles are embedded in a graphite matrix. This graphite serves as a convenient structural material, heat removal path, neutron moderator and the second fission product release barrier.

b. The reactor core is constructed from graphite blocks, which provide the structural material for the core, but also moderate fission neutrons to sufficiently slow energy levels to be absorbed into fuel to sustain a chain reaction. The fuel elements are inserted into sleeves in the graphite blocks. The graphite blocks also have holes through which the coolant can flow to remove the fission heat.

c. A reflector surrounds the core on all sides to reduce neutron leakage from the core. Graphite is typically used as a reflector, but other materials with a high neutron scattering cross-section and a low neutron absorption cross-section can also be used. In this study, beryllium is also considered as a reflector.

d. The reflector is surrounded by a core barrel, which has the function of maintaining the core elements in the correct location and orientation for all operational as well as accident conditions.

e. The core barrel is surrounded by a Reactor Pressure Vessel (RPV). This is the main pressure-containing vessel and also provides structural support to the core barrel. The RPV also forms the third barrier to fission product release.

f. Fission heat is removed from the core by helium coolant, which has a very small neutron cross-section, does not get activated and does not change phase above -269 °C. The coolant is typically injected into the bottom of the reactor and flows up through the side reflector to provide cooling to the outer region of the reactor. At the top of the reactor, the gas is directed downwards by a plenum constructed of graphite blocks. At the bottom of the reactor where the gas emerges from the core graphite blocks, the gas is collected in a plenum and directed to the reactor outlet pipe.

g. Reactivity control

   i. The reactivity of the core is controlled mainly by ensuring a large negative temperature reactivity coefficient for the reactor. Typical values are -5×10^{-5} $\Delta k/k/°C$. When more heat is produced than removed by the coolant, the fuel heats up and the Doppler resonance broadening of the uranium-238 immediately results in increased neutron absorption and reduced neutronic power. Operators can control the power of the reactor by controlling the amount of heat removed from the reactor. This can conveniently be controlled by the speed of the primary loop blower. This temperature coefficient can be used as a rapid shutdown means (by stopping the primary blower) if all the components in the core can withstand the resulting elevated temperatures. This means can, however, not provide permanent cold shutdown capability, because the core will become critical (at low power) as soon as it cools down sufficiently.

   ii. In order to limit the excess reactivity of the core over the fuel cycle, burnable poisons are added to the core. This can be in the form of gadolinium or boron that is coated on the fuel elements, or separate poison rods inserted into separate sleeves in the graphite blocks. The amount of poison is calculated to be steadily consumed by neutron absorption as the core ages so that at the end of the fuel cycle these poisons have virtually no effect on reactivity. This burnable poison control has significant safety benefits because it reduces the amount of reactivity that can be added in an accident as well as reduces the requirements on control rods.

iii. A second reactivity control system consists of control rods that move inside sleeves in the graphite blocks. The control rods can move inside the side reflectors or inside the fuel zone, depending on the desired shape of the neutron flux. These control rods contain Boron-10, which has a very high neutron absorption cross-section. Operators can control the fission rate by moving the control rods in or out of the core. The control rods can also be used for start-up and shutdown.

h. Instrumentation and control: the primary sensors for monitoring the reactor operation are neutron flux detectors and coolant temperature detectors.

i. The neutron flux measurement is a direct and immediate indication of the fission energy. The neutron flux is also used to derive the reactor period, which indicates the rate of change of the fission power. The flux level is monitored by detectors located outside the RPV, to enable maintainability and to increase the life of the detectors. These detectors are designed to monitor the neutron flux level over an extremely wide range, from a few neutrons per second to double normal power levels, typically 10^{10} neutrons/s/cm^2 at the detector location.

ii. The coolant outlet temperature is an indirect indication of the fuel temperature while the coolant is flowing. Coolant temperature is measured at the inlet and outlet of the reactor, using thermocouples that can withstand the radiation environment. The temperature difference over the core is used in conjunction with the coolant flow rate to calculate the heat removal rate.

iii. The reactor is tripped automatically when the neutron flux becomes too high, when the reactor period becomes too short or when the outlet temperature becomes too high. This automatic trip functionality is provided by a high-integrity Reactor Protection System which typically drops the control rods into the core under gravity and immediately renders the core sub-critical. This Reactor Protection System along with its sensors and actuators are designed to the highest possible standards to ensure nuclear safety under all conditions, including all postulated accident scenarios. These systems typically use three or four redundant channels that are completely independent and utilize an inter-channel voting system to determine when the reactor should be tripped.

iv. An operational control system is used to control the normal operation of the reactor as well as the power-conversion system. This system consists of a distributed control system, which interfaces to all the equipment as well as the operators. This system also provides dedicated investment protection functions, which prevent damage to expensive equipment.

4.3.1 Design of Reactor and Support Systems

A stepwise design is followed in this section, starting from the reactor core and in turn addressing the elements of the reactor in a radial sequence.

4.3.1.1 Core size

The size of the active core is determined by thermodynamic concerns, since the drivers for neutronic criticality are not dominant for the size of reactors under consideration. Core power density is selected to facilitate passive decay heat removal in accordance with the safety requirements. The power density should be between 3 MW/m^3 (or 3 W/cm^2 surface flux density) as in the HTR MODUL design and 4,8 MW/m^3 as in the PBMR design [46]. Different power density options are compared in Table 27.

To facilitate passive safe decay heat removal, the ratio of height/diameter ($\frac{H}{D}$), should be kept around three, as determined by Kugeler [46]. This is in contrast to a traditional low leakage core design, which has an optimal ratio of $\frac{H}{D} = 0.93$ [45]. Following the Kugeler approach with the ratio of $\frac{H}{D} = 3$, the core volume can be written as:

$$V_{core} = 3 \times D \times \pi \times \frac{D^2}{4} = \frac{3}{4} \times \pi \times D^3 \quad \text{.....(Eq 4.11)}$$

Different core volumes and height-to-diameter options are shown in Table 27. The yellow highlighted column in Table 27 indicates the selected core power density of 4 MW/m^3, with a core power of 80 MW.

4.3.1.2 Proposed core layout

A proposed core layout is shown in Figure 35, using 37 hexagonal blocks of the same cross-sectional dimensions as used by the Fort Saint Vrain reactor, the General Atomics HTGR and Japanese HTTR reactor designs. For an 80 MWt reactor with power density of 4 MW/m^3, this layout is similar to the HTTR core, except that almost double the height of the HTTR core is required, implying 10 blocks of 612 mm high stacked on top of each other. Each of the fuel blocks would have 33 sleeves, each containing 15 fuel elements. At the same core height, 30 of the 37 blocks would contain fuel, while the other 7 would contain control elements.

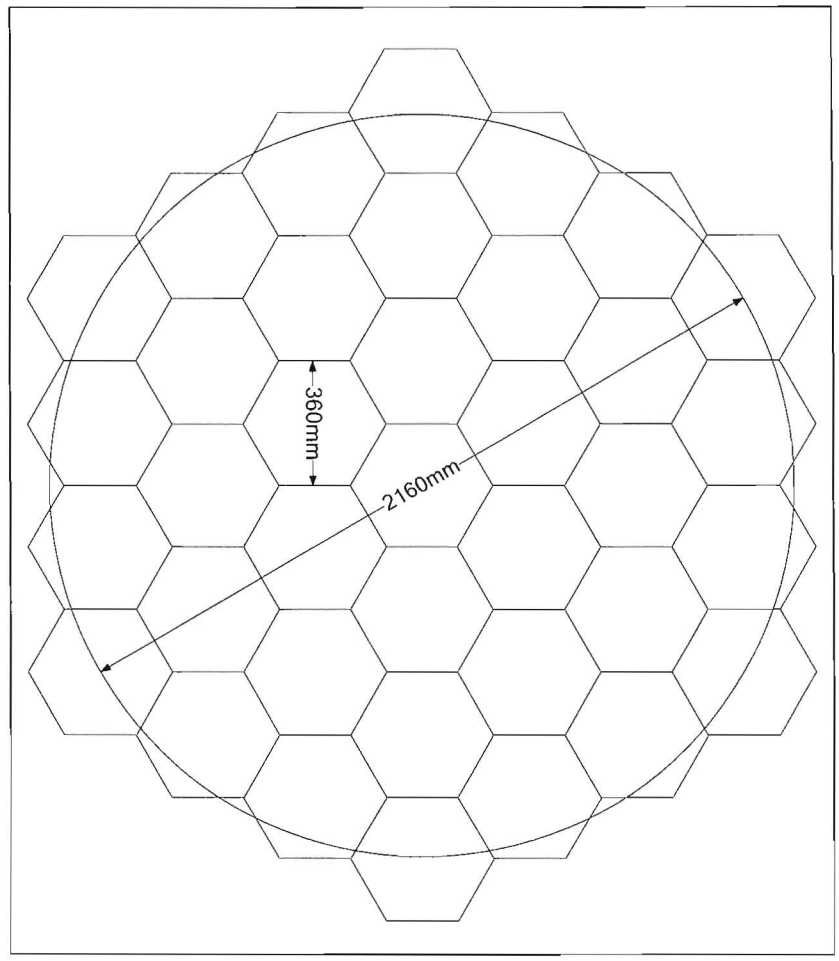


Figure 35: Proposed Core Layout

4.3.1.3 Reflector dimensions

From a neutron economy perspective, the reflector thickness (b) should be several diffusion lengths (L^R): $b \rangle\rangle L^R$ according to Duderstadt and Hamilton [26]. The guidance provided by Kloosterman, of Delft Technical University [45], is that the optimal reflector thickness should be twice the diffusion length. Figure 36 shows a comparison of reflector savings (δ) as a function of reflector thickness, expressed as $\frac{b}{L^R}$. From this figure it is clear how the incremental reflector savings diminish for reflector thickness beyond twice the diffusion length.

The diffusion length can be calculated as $L^R = \sqrt{\frac{D^R}{\Sigma_a{}^R}}$ with the diffusion coefficient defined by [26]:

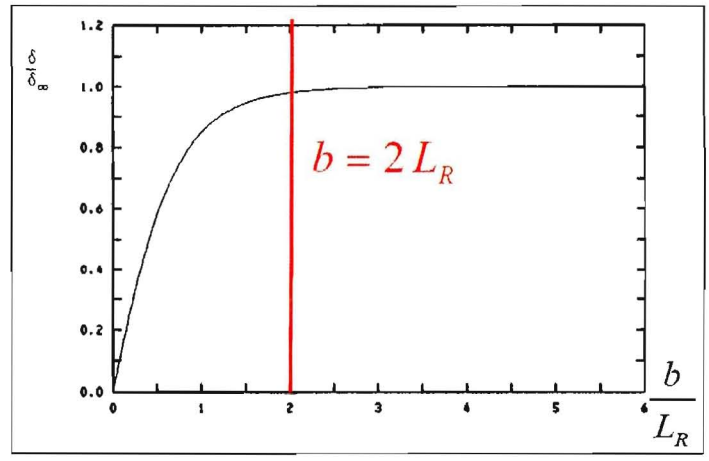


Figure 36: Reflector saving as function of reflector thickness

$$D^R = \frac{1}{3 \times \Sigma_{tr}} \quad \text{.....(Eq 4.12)}$$

where Σ_{tr} ,the transport cross-section, is defined as: $\Sigma_{tr} = \Sigma_t - \Sigma_s\left(\frac{2}{3 \times A}\right)$. In this definition, Σ_t is the total cross section, Σ_s the scattering cross section and A the atomic mass *number.*

For a graphite reflector $\Sigma_{tr} = 0.359cm^{-1}$, $D^R = 0.93cm$ and thus $L^R = 58.7cm$.

An optimal graphite reflector should therefore be 1,2 m thick. This thickness was used in the calculations in Table 27. In comparison, a reflector made from beryllium, has $L^R = 21\,\text{cm}$, implying an optimal reflector thickness of 0,42 m. Such a reflector could significantly reduce not only the size of the reflected core, but in turn also the core barrel and RPV.

Table 23 lists characteristics for graphite and beryllium as possible reflectors. These values are used in Table 27 to compare the relative weight and cost of using beryllium and graphite as reflectors.

Table 23: Comparison of Graphite and Beryllium Characteristics [26]

Parameter	Graphite	Beryllium
Melting point [°C]	3652	1278
Density [g/cm^3]	1,6	1,85
Moderation ratio [$\frac{\xi\Sigma_s}{\Sigma_a}$]	192	143
Migration length [cm]	62	23
Diffusion Length [L^R in cm]	59	21

It is clear from Table 27 that although allowing an almost 50% weight reduction for the complete reactor, the cost of beryllium would be prohibitive. An additional risk associated with beryllium is that it is poisonous to machine, requiring additional safety measures.

For this study, the graphite reflector with 1,2 m thickness will be the selected choice.

4.3.1.4 RPV

The diameter of the RPV is determined by provisions for a core barrel on the outside of the reflector. For the calculations in Table 27, the RPV wall thickness is assumed to be 180 mm, 60 mm is allowed for core carrel thickness and 160 mm for the space between the RPV and core barrel. These values are conservatively based on the PBMR DPP-400 design [46], which has a system pressure of 9 MPa and operates at higher temperatures than this design.

To determine the RPV height, it is assumed that the gas plenum at the top of the core is about 1 m high and the bottom plenum containing the core outlet is approximately 3 m high.

An additional 2 metres are added to the RPV height to allow space for incorporating the compact steam generator and primary loop blower inside the RPV (refer to Section 4.2.4.4.2.2). This packaging option will save space, reduce heat loss and significantly reduce the risk of primary loop pipe breaks. This configuration introduces a risk of water entering the reactor core, which could lead to graphite oxidation and even fuel damage. This risk has to be reduced to acceptable levels by means of the mechanical design, but also by ensuring reactor shutdown in case of accidental water ingress.

For simplicity, the domes on the ends of the RPV are assumed to be hemispherical. These could be optimised in the future to save space and material.

The RPV of the HTTR project can be used as a benchmark for comparison of the size of this design. The core of the ship reactor is of the same diameter as the HTTR and approximately double the height of the HTTR. The HTTR RPV has a diameter of 5,74 m and height of 13,22 m (approximately 9 m, excluding top and bottom domes), as shown in Figure 60. It is estimated that the additional core height would add an additional 3,2 m to the height of the RPV, resulting in a vessel height of 16,42 m (12,2 m, excluding top and bottom domes). This agrees well with the equivalent value of 12.52 m shown in Table 27 (if the additional 2 m height for the steam generator is not taken into consideration).

The option selected for this study (in yellow highlight in Table 27) has an RPV with outer diameter of 5,28 m and edge-to-edge length of 19,8 m. See Figure 37 for a conceptual RPV layout.

The radiation exposure of the RPV is one of the main determinants of the life of the reactor. A typical guideline is that the RPV should not receive exposure of more than $4x10^{18}$ neutrons/cm^2 over its lifetime. Detailed neutronic models must be developed to determine the service life of the RPV. Those calculations are, however, outside the scope of this study.

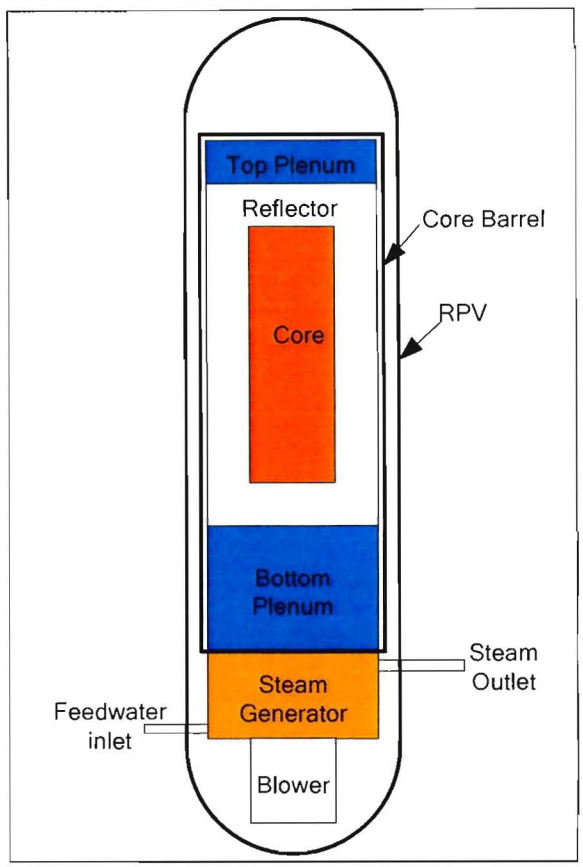


Figure 37: Reactor Layout

4.3.1.5 Decay heat removal system

As shown in the initiating requirements in Table 9, "Passive safety, without requiring interventions by personnel or back-up power sources" is one of the drivers of this reactor design. The concept of passive decay heat removal was developed by Kugeler and Schulten as part of the design of the HTR MODUL [46]. The concept relies on the following provisions:

i. Low core power density (~ 3 MW/m^3);.
ii. A core shape where the height-to-diameter ratio is around 3.
iii. A large heat capacity in the core, to enable slow temperature rise under decay heat conditions.
iv. A ceramic core where all components, including the fuel, can withstand the elevated temperatures of up to 1600 °C, without degradation.
v. A large difference between the normal operating fuel temperature and the safety limits of the fuel.
vi. Provision of a water-cooling system on the outside of the reactor vessel that can passively remove core heat by boiling off water into the atmosphere.

In the ship application, a water jacket can be provided around the RPV to remove decay heat during emergencies by evaporation of water into the atmosphere. During normal shutdown

conditions, the water jacket will be cooled by means of pumps and heat exchangers that dump heat into the sea. Important parameters of such a system are determined below:

a. The ideal solution would be to have a sufficiently large quantity of stored water so that no operator action or external power is required for some period of time. The required amount of time is dependent on the emergency scenario and also on regulatory requirements. The required amount of water is estimated below:

The decay heat power from a reactor (P_D) can be approximated by:

$$P_D(t) = P_{th} \times 0.062 \times t^{-0.2} \quad \text{.....(Eq 4.13)}$$

if the reactor has been operational for a long time at steady-state power of P_{th}.

The total amount of decay energy (E_D) produced over time can be calculated by integrating P_D over time:

$$E_D(T) = \int_0^T P_D(t)dt \quad \text{.....(Eq 4.14)}$$

$$E_D(T) = \frac{0.062}{0.8} \times P_{th} \times t^{0.8}\Big|_0^T \quad \text{.....(Eq 4.15)}$$

To determine the mass of water required (M_{water}) to dissipate this amount of heat via evaporation, the evaporation heat constant for water is used: $h_{fg} = 2.257E6$ J/kg. In this calculation, heating up the water to boiling point is conservatively ignored.

$$M_{water} = E_D / h_{fg} \quad \text{.....(Eq 4.16)}$$

$$M_{water} = 0.0775 \times \frac{P_{th}}{h_{fg}} \times T^{0.8} \quad \text{.....(Eq 4.17)}$$

Table 24 below shows the amount of water required for different time durations:

Table 24: Amount of Water Required for Passive Decay Heat Removal for Different Time Durations

Tons of water		Reactor power [MW]		
		75	80	85
days	0.5	13	14	15
	1	23	24	26
	10	145	154	164
	30	348	371	395
	365	2,570	2,741	2,912

b. From this calculation, it seems reasonable to make provision for water tanks for 24 hours, equating to 24 tons of water for the 80 MW reactor design. This implies that in an emergency situation the crew would have up to 24 hours time to restore cooling flow, or to refill the boil-off tanks. The water can be stored in two tanks of 2,5 m diameter and 2,5 m height. Two tanks are required to ensure that a single failure cannot completely disable the function. These tanks would have to be above the RPV to enable natural circulation between the water jacket and the tanks. The water quality should be such that corrosion of the tanks and water pipes is prevented over the life of the system. In

addition, the water should not contain elements that can become radioactive due to neutron capture and in turn result in gamma radiation.

c. The water jacket can be constructed from stainless steel tubes arranged side by side to effectively form a water wall around the RPV. A typical pipe for this water jacket would be a schedule 40, 65 mm pipe with an outer diameter of 73 mm [27]. It is assumed that the water jacket is attached on the outside of the RPV, with a gap of 100 mm between the RPV and the pipe network. This concept is indicated in Figure 41 for a reactor with a horizontal orientation as discussed in Section 4.3.6.

d. To calculate the amount of piping that is required for the water jacket, it is assumed that the RPV is 17,8 m long, as calculated in Section 4.3.1.4 and adding an additional 2 m length for incorporating the steam generator inside the RPV as discussed in Section 4.3.6.1.

e. The RPV length of 19,8 m implies 264 pipes next to each other, with a 2 mm gap between the pipes. Each of the pipes would have a length of $2 \times \pi \times R_{pipe} = 17{,}5$ m to be able to reach around the reactor, as indicated in Figure 41. This implies 4600 m of piping to cover the sides of the RPV.

f. The weight of the piping would be 8,6 kg/m and the water inside the pipe 3,1 kg/m [27]. This equates to a total weight of 53,8 ton.

g. Detailed thermodynamic calculations would be required in future studies to confirm the natural circulation in this configuration and to confirm that the maximum temperatures on all structures are within the design limits.

4.3.1.6 Radiation shielding

This section discusses high-level considerations relating to radiation shielding:

a. The shielding system must provide sufficient protection to personnel, systems of the ship and cargo against neutrons and gamma radiation. The allowable dose limits are defined in Table 9 as:
   i. 20 mSv/year for maintenance workers in a limited access area just outside the shield;
   ii. 1 mSv/year for personnel in unrestricted areas;
   iii. thermal neutron flux below 2×10^5 neutrons $cm^{-2}s^{-1}$ for the cargo of the ship.

b. During reactor operation it can be assumed that the dominant dose would be from neutrons, and gamma radiation can be ignored. The converse is true during reactor shutdown periods, when gamma dose would be dominant.

c. Assumptions have to be made regarding the amount of time that a maintenance worker has to spend in close proximity to the reactor, to be able to determine the allowable dose rate on the outside of the shield wall. For this study it is assumed that a radiation worker would have to spend 1 hour in the vicinity of the reactor every week while the reactor is operational. It is also assumed that the non-radiation workers will spend 80% of their time on board the ship at a distance of at least 10 m from the reactor shield (refer to Figure 48).

d. Using the assumption that the neutron dose is dominant during reactor operation, the approximate neutron flux to dose rate conversion shown in Figure 38 can be used to determine the allowable neutron flux on the outside of the shield wall [32]. On the further assumption that the shielding wall has to be sufficiently thick so that only thermal neutrons leak out in any significant strength, a dose rate of 1 millirem/hr would be equivalent to $2{,}6 \times 10^6$ neutrons.$m^{-2}s^{-1}$.

e. The 1 mSv/year limitation for non-radiation workers equates to a permanent dose rate of $1{,}14 \times 10^{-8}$ millirem/hr. Using the conversion from Figure 38, this translates to a

permanent neutron flux of $3x10^{-2}$ neutrons.$m^{-2}s^{-1}$ for non-radiation workers. If personnel spend 80% of a typical year on board the ship, the allowable flux increases to $3,75x10^{2}$ neutrons.$m^{-2}s^{-1}$. If the non-radiation workers can be assumed to be at least 10 m away from the shield, the neutron flux on the outside of the shield should not exceed $4,7x10^{5}$ neutrons.$cm^{-2}s^{-1}$. This is based on the assumption that the flux is emitted equally in all directions and will thus reduce as a function of $\frac{1}{4 \times \pi \times radius^2}$.

f. The neutron flux limit associated with radiation workers can be determined in a similar way as $1x10^{6}$ neutrons.$cm^{-2}s^{-1}$. This shows that the non-radiation worker limit is the more restrictive in this application.

g. The cargo neutron flux limitation of $2x10^{5}$ neutrons.$cm^{-2}s^{-1}$ is assumed to be at a distance of at least 2 m from the shield. This implies a thermal neutron flux limit of $1x10^{7}$ neutrons.$cm^{-2}s^{-1}$ just on the outside of the shield, which is enveloped by the non-radiation worker limitation calculated above..

h. More work is required to establish limits and shielding against gamma radiation.

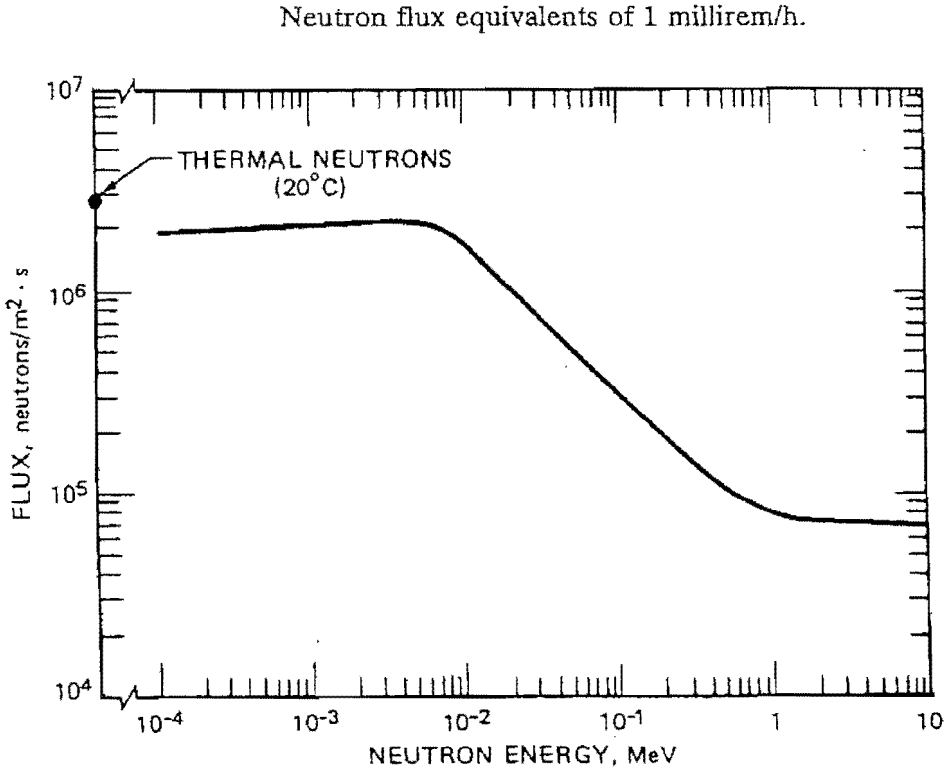


Figure 38: Neutron Flux-to-dose Rate Conversion

i. In land-based reactors, neutron shielding is typically provided by concrete, which contains hydrogen in the form of water as well as heavy material additives such as barites, iron or boron. The hydrogen atoms slow down the fast neutrons that escape from the reactor until these neutrons can be absorbed by heavy atoms. Typical examples of these concrete shields are the General Atomics HTGR with a thickness of 1,5 m, the HTR MODUL with a thickness of 2,25 m and the PBMR with a shield thickness of 2,5 m.

j. Gamma shielding is provided by heavy materials such as iron (as carbon steel or stainless steel) or lead, which typically forms a liner on the concrete walls, or is added into the concrete mix.

k. A conceptual layout of a traditional 2-metre thick concrete shield with a stainless steel liner on the outside of 2 cm thickness is shown in Figure 41. The volume of concrete can be calculated from the concept presented in Figure 41:

$$Volume = \frac{length \times \pi}{4}\left[(D_{inner} + 2 \times thickness)^2 - D_{inner}{}^2\right] + 2 \times thickness \times \pi \times \left(\frac{D_{inner}}{2}\right)^2 \quad \text{.....(Eq 4.18)}$$

The inner diameter is assumed as being 6 m, in accordance with Figure 41. The length is assumed to be 20 m, based on an RPV that is long enough to contain the steam generator. For a 2 m thick shield, the Volume is 1118 m^3. At a typical density for reactor shielding concrete (Barytes concrete) of 3,5 ton/m^3 [32], this translates to a shield weight of almost 4000 ton, which is deemed impractical. Even reducing the thickness to 1 m results in a shield mass of 1740 ton, which results in the total propulsion system exceeding the 2000 ton weight limitation contained in Table 9.

l. Several alternative techniques can be used to reduce the size and mass of the radiation shield to more acceptable values:

i. Multiple layers of materials that alternately slow down and absorb neutrons and gamma rays can be used. This approach was followed in the design of the American nuclear-powered bomber aircraft, the NB-36H [15]. In that application, the innermost layer consisted of a beryllium oxide reflector, the next layer of liquid sodium coolant, the third layer of cadmium (for absorbing slow neutrons), the fourth layer of paraffin wax for slowing down fast neutrons and the fifth layer a steel shell for absorbing slow neutrons and gamma rays. The shield for the NB-36H weighed about 16 ton.

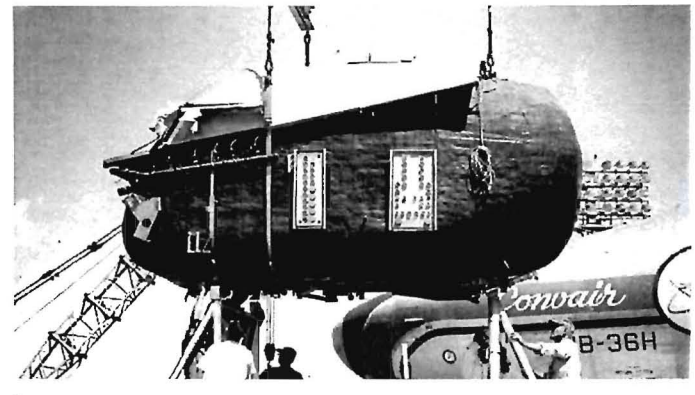


Figure 39: NB-36H Shielded cockpit

ii. Another improvement that can be implemented is to use borated graphite on the outer layer of the core reflector. This technique was proposed in the HTR MODUL design as well as the GA HTGR design. This methodology will significantly reduce the thickness and weight of the shield.

iii. The Thermo Electron Corporation markets lightweight radiation shielding that consists of high-purity polyethylene [18]. According to Thermo, polyethylene contains more hydrogen atoms per volume than any other material, which makes it particularly suitable for slowing down fast neutrons. Boron is embedded into the polyethylene to absorb the neutrons after slowing down. This boron-impregnated polyethylene is easily shaped and has good structural properties. Thermo also offers a gamma shielding material called "Enviro-Shield" as an alternative to lead-based shields. This material has a typical density of 3 g/cc.

iv. The radiation shield for the Savannah nuclear ship consisted of concrete at the bottom and a combination of water, lead and polyethylene sheets at the upper parts. Figure 40 shows a diagram of the shielding employed for the Savannah.

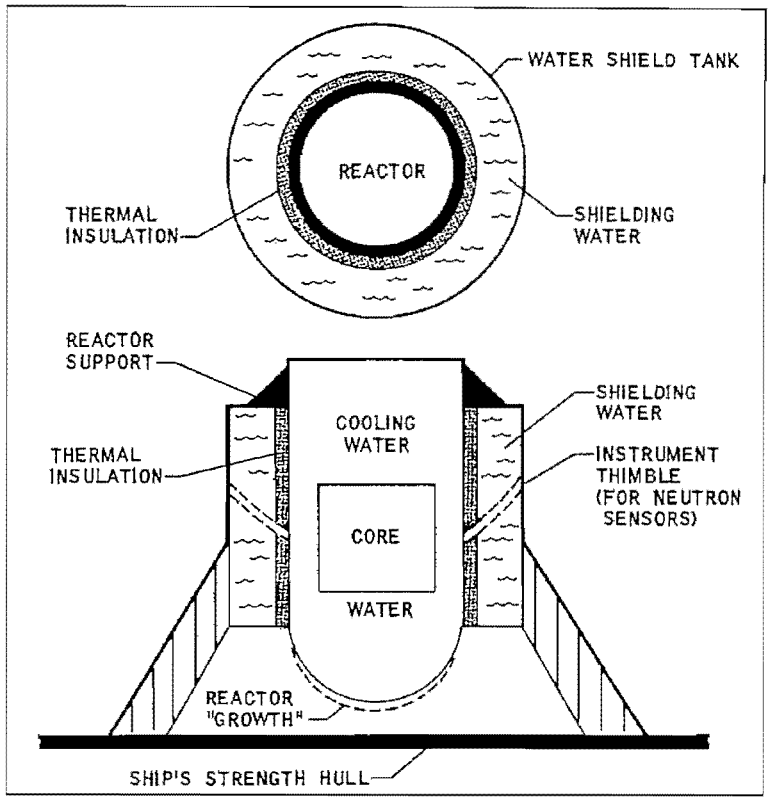


Figure 40 : Savannah Neutron Shield

m. Detailed radiation analysis would, however, have to be done in the future to design an optimal shield.

n. A secondary function that could potentially be allocated to the shielding structure is that of confinement of radioactivity during depressurization accidents. This function is, however, subject to a more detailed safety analysis.

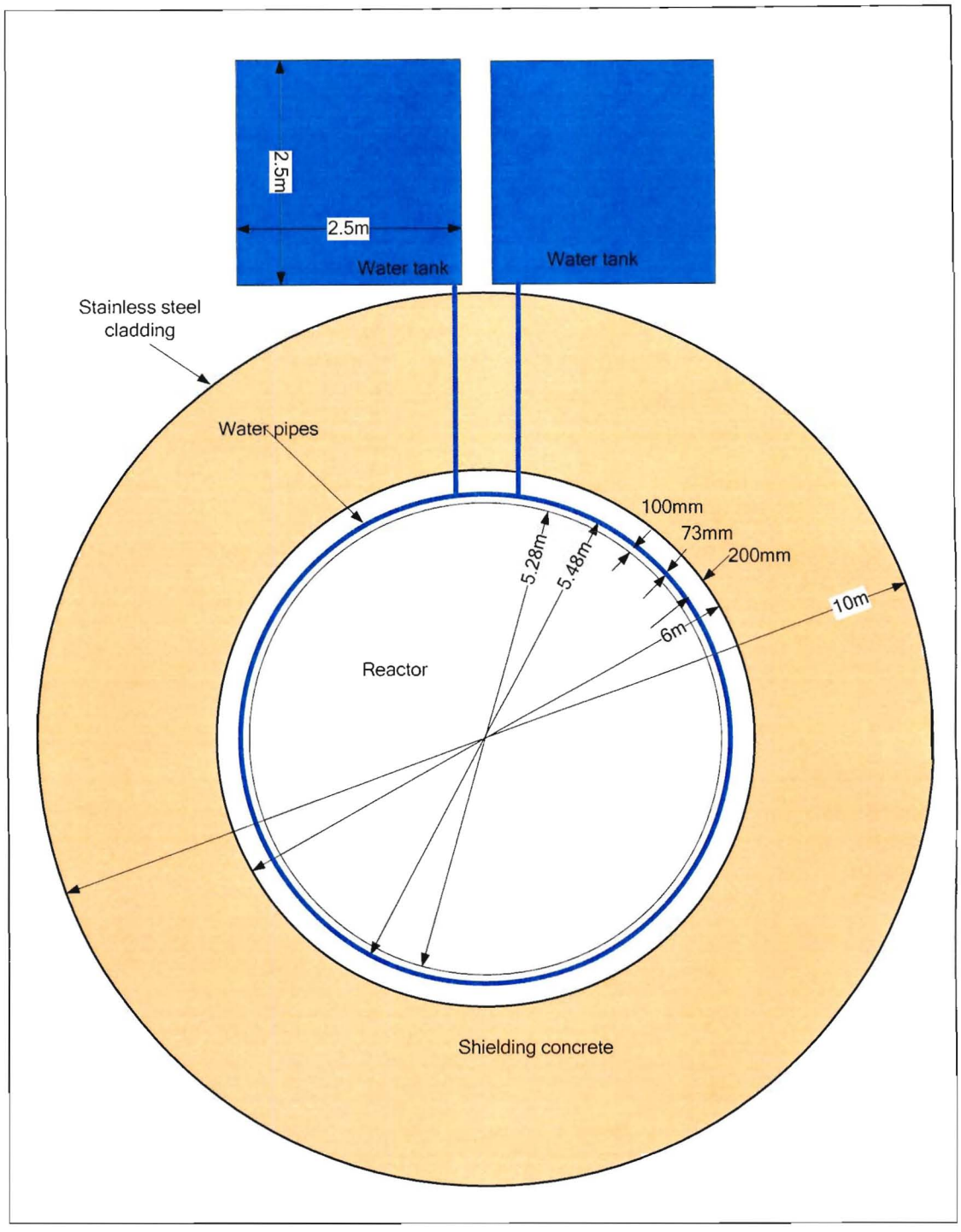


Figure 41: Decay Heat Removal and Shielding (Side View)

4.3.2 Cost Model

This section establishes a cost estimate for the reactor system as defined in Section 4.3.1. The basic assumption is that the reactor cost can be based on the mass of the different materials used. A second basic assumption is that the capital cost for the n^{th} unit is estimated, disregarding the development and industrialization costs.

Section 4.3.2.1 below establishes a cost per kg for each material type, based on literature and experience. This *'material cost'* includes the cost of the raw material as well as manufacturing and assembly.

Section 4.3.2.2 sets out the specific assumptions required for each of the reactor systems to enable cost estimate, while Section 4.3.2.3 sets out the assumptions relating to the reactor support systems. A cost model for the reactor is presented in Table 27, where different options for reactor power, power density and reflector material are compared.

4.3.2.1 Material cost values

4.3.2.1.1 Fuel

The fuel unit cost is based on an annular fuel compact with the same dimensions and uranium content as the HTTR fuel. It is assumed that 6% or 8% enriched fuel is used, with 13000 TRISO particles per fuel compact. This must be confirmed via neutronic analysis in future studies.

The fuel cost consists of the following main components:

a. Uranium feedstock.
b. Enrichment cost measured in Separative Work Units (SWU).
c. TRISO particle manufacturing cost.
d. Fuel compact manufacturing cost.

The Ux Consulting Company provides an on-line calculator [73] that can be used to calculate the cost of enriched uranium products, based on the current uranium spot price and the current SWU price to enrich the uranium. Using this calculator, it was determined that 1 kg of 6% enriched uranium would require 12,5 kg of UF_6 and 10,1 SWU. At the current spot price of uranium, UF_6 costs 117,5 $/kg and an SWU costs $160. These values add up to $3085 for 1 kg of 6% enriched uranium. Similarly, 1 kg of 8% enriched uranium would cost $4278, requiring 16,8 kg of UF_6 and 14,4 SWUs. These calculations are shown in Table 25.

It is complex to estimate the cost of producing the TRISO particles and the fuel compacts without specific assumptions regarding the fuel production plant. A useful cost estimate can be based on research done by the "Electricity Policy Research Group" of the Cambridge-MIT Institute [58]. According to their research, the enriched uranium cost is typically 32% of the actual fuel cost. The cost of spent fuel management is assumed to be 11% of the fuel cost, with fuel manufacturing making up 57%. It is acknowledged that the Cambridge-MIT research is probably biased towards LWR technologies and that the cost structure will be slightly different for HTGR fuel. This cost structure does, however, provide a good initial cost estimate.

Using the assumption that the enriched uranium makes up 32% of the final fuel cost, together with the cost of enriched uranium as calculated above, results in fuel cost of $9640 for fuel element material containing 1 kg of 6% enriched uranium. Similarly, 1 kg of fuel

containing 8% enriched uranium would cost $13370. These calculations were implemented in an EES programme as shown in Appendix B.

Table 25: Fuel Cost Calculations

	Unit Cost [$]	6% Enrichment		8% Enrichment	
		Units	Cost [$]	Units	Cost [$]
UF_6	117,5	12,5	1469	16,8	1974
SWU	160	10,1	1616	14,4	2304
Enriched uranium cost			3085		4278
Fuel fabrication cost			6564		9092
Fuel cost per 1 kg of uranium			9640		13370

4.3.2.1.2 Nuclear-grade graphite

Based on industry norms, nuclear-grade graphite typically costs three times the price of high-purity electrode graphite. The Wall Street Journal published the price of electrode graphite from the SGL Corporation to be €5,22/kg [5] in September 2009. This translates to approximately $18,8/kg for nuclear-grade graphite.

Another source of the nuclear–grade graphite price is contained in a report by the "China Carbon Graphite Group, Inc". In this document, the market price of nuclear-grade graphite that is imported into China is given as $17,5/kg [28]. This price is used in the cost calculations because the reference provides a more direct price indication.

It is assumed that the material cost makes up 60% of the final manufactured product cost, based on industry experience.

4.3.2.1.3 Beryllium

The United States Geological Survey [25] shows that the cost of beryllium as a metal was $400/pound ($880/kg) in 1998. Wikipedia indicates the beryllium price as $745/kg in 2001 [77]. No newer price information could be obtained for beryllium.

This 1998 price is conservatively used in the cost calculations in Table 27 because it is the higher of the two indications.

The assumption is made that the material cost is 60% of the final product.

4.3.2.1.4 RPV Steel

The RPV will use traditionally accepted materials. For costing purposes, the following is assumed:

a. SA508 carbon steel is used.
b. Carbon steel cost is taken as €8,7/kg (~$12/kg) from the MEPS website [56] for international, average hot-rolled steel.
c. Material cost makes up 25% of the final product cost, based on industry experience with nuclear pressure vessels.

4.3.2.1.5 Core internals stainless steel (SS-316)

Due to the high operational temperature and the high neutron flux that the core barrel will have to withstand, it is normally constructed from stainless steel. For costing purposes, the following is assumed:

a. Stainless steel SS-316 is used.
b. Stainless steel cost is assumed to be $33/kg from the MEPS website [56].
c. Material cost makes up 40% of the final product cost, based on industry experience.

4.3.2.1.6 Shielding concrete

In order to estimate the cost of the radiation shield, the following is assumed:

a. The price of high-density shielding concrete is R1500/m^3 ($200/$m^3$), excluding rebar, based on information from Lafarge South Africa.
b. Rebar cost is estimated as R15/kg ($2/kg), utilizing 240kg/$m^3$ in the shield construction. The rebar would thus cost $480/$m^3$
c. The shield can be cast in sections that can be assembled inside the ship.
d. The final cost of the shield will be double that of the materials.
e. The concrete shielding cost would thus be $1360/$m^3$. At a typical density of 3,5 ton/m^3 [32], the mass-based cost is $0,39/kg.

4.3.2.2 Reactor component cost assumptions

4.3.2.2.1 Fuel

The fuel cost is based on the following:

a. Fuel element: 39 mm long,
b. Fuel element outer radius: 13 mm
c. Fuel element inner (hole) radius: 5 mm
d. Uranium particle diameter 0,6 mm
e. Number of TRISO particles per fuel element: 13000

From these values the parameters listed in Table 26 were calculated in the EES programme contained in Appendix B:

Table 26: Fuel Characteristics

Parameter	6% enrichment	8% enrichment
Mass of fuel element	41,3 g	
Mass of uranium in fuel element	13,6 g	
Density of fuel element	2342 kg/m^3	
Cost per fuel unit containing 1 kg of enriched uranium (from Section 4.3.2.1.1)	$9640	$13370
Cost per fuel element	$131	$182
Cost per mass of fuel element:	$3175/kg	$4403/kg

4.3.2.2.2 Core graphite structure

The holes in the graphite blocks for directing flow, control rods and fuel are conservatively ignored in the volume and cost calculations.

Additional graphite material is added for the top and bottom plenum. In the volume and cost calculations, these blocks are conservatively assumed to be solid.

4.3.2.2.3 Reflector

A layer of reflector material is assumed to cover the core around the sides as well as the top and bottom of the core.

The holes in the reflector blocks for coolant flow, control rods, instrumentation etc. are conservatively ignored in the volume and cost calculations.

4.3.2.2.4 Core barrel

It is assumed that the core barrel is constructed of SS-316 stainless steel.

Wall thickness is assumed to be 60 mm all round.

It is assumed that the barrel is a right cylinder with flat ends on both sides.

4.3.2.2.5 RPV

It is assumed that the RPV is constructed of SA508 carbon steel.

The vessel is assumed to have hemispherical domes at the top and bottom.

The vessel is assumed to have a wall thickness of 180 mm all over.

Table 27: Reactor Sizing and Costing

	Graphite Reflector						Beryllium Reflector
Reactor Power [MW]	**75**	**75**	**75**	**80**	**80**	**80**	**80**
Power Density [MW/m^3]	**3**	**4**	**4.5**	**3**	**4**	**4.5**	**4**
Core Volume [m^3]	25.00	18.75	16.67	26.67	20.00	17.78	20.00
Core Diameter [m]	2.20	2.00	1.92	2.25	2.04	1.96	2.04
Core Height [m]	6.59	5.99	5.76	6.74	6.12	5.88	6.12
Core Graphite Mass [ton]	40.00	30.00	26.67	42.67	32.00	28.44	32.00
Core Graphite Cost [$ mil]	1.17	0.88	0.78	1.24	0.93	0.83	0.93
Fuel Rod Volume [m^3]	4.41	3.31	2.94	4.71	3.53	3.14	3.53
Fuel Rod Mass [ton]	10.33	7.75	6.89	11.02	8.26	7.35	8.26
Fuel Rod Cost (6% enrich) [$ mil]	32.50	24.60	21.87	34.99	26.24	23.32	26.24
Fuel Rod Cost (8% enrich) [$ mil]	45.49	34.12	30.32	48.52	36.39	32.35	36.39
Reflector Thickness [m]	1.20	1.20	1.20	1.20	1.20	1.20	0.42
Reflected Reactor diam [m]	4.60	4.40	4.32	4.65	4.44	4.36	2.88
Reflected Reactor height [m]	8.99	8.39	8.16	9.14	8.52	8.28	6.96
Reflector Volume [m^3]	190.67	169.33	161.52	195.94	173.83	165.74	51.39
Reflector Mass [ton]	305	271	258	314	278	265	95
Reflector Cost [$ mil]	8.90	7.90	7.54	9.14	8.11	7.73	139.44
Core Barrel Volume [m^3]	2.4	2.2	2.2	2.5	2.3	2.2	1.1
Core Barrel Mass [ton]	19.5	18.0	17.4	19.9	18.3	17.7	8.5
Core Barrel Cost [$ mil]	1.61	1.48	1.44	1.64	1.51	1.46	0.70
RPV Height (excluding domes) [m]	14.99	14.39	14.16	15.14	14.52	14.28	12.96
RPV Inner Diameter [m]	5.08	4.88	4.80	5.13	4.92	4.84	3.36
RPV Volume [m^3]	29.45	27.18	26.34	30.01	27.66	26.79	16.01
RPV Weight [ton]	230.93	213.10	206.47	235.27	216.89	210.06	125.50
RPV Cost [$ mil]	11.08	10.23	9.91	11.29	10.41	10.08	6.02
Total Mass [ton]	**606**	**540**	**516**	**622**	**554**	**529**	**269**
Total Cost [$ mil] (8% enriched fuel)	**68.2**	**54.6**	**50.0**	**71.8**	**57.4**	**52.5**	**183.5**

Material Properties	Graphite	Beryllium	Fuel (6%)	Fuel (8%)	SS-316	SA508	**Units**
Density	1.6	1.85	2.342	2.342	8.03	7.84	**ton/m^3**
Cost (machined articles)	29.2	1467	3175	4403	82.5	48	**$/kg**

4.3.2.3 Reactor support system cost assumptions

Cost estimates for the decay heat removal system and the shielding system are based on the design presented in Section 4.3.1. Other systems that are required to support the reactor are not modelled in detail in this study. The basic assumption that is made, to be able to cost these systems, is that the cost of the support systems can be scaled from the cost of the reactor. Zhang and Sun from the Tsinghua University developed a cost model whereby the reactor support systems' cost can be related to the cost of the reactor for an HTGR [88].

The support systems required in the ship application are discussed briefly below:

4.3.2.3.1 Decay heat-removal system cost

The decay heat-removal system is described in Section 4.3.1.5.

Based on information from the Matthew Davis Company [55], the cost of SS 316 schedule 40 piping is assumed to be AU$310/running metre. At the current exchange rate, the cost would equate to $288/metre.

The cost of the piping as calculated in Section 4.3.1.5 is thus $1,3 m. It is assumed that the tanks and water purification equipment would cost an additional $2 m.

4.3.2.3.2 Shield cost

The shielding system is described in Section 4.3.1.6.

The cost for a concrete shield as indicated in Figure 41 is $675 000, based on a concrete price as shown in Section 4.3.2.1.6.

4.3.2.3.3 Electrical, instrumentation and control system cost

The Electrical, Instrumentation and Control (EI&C) system of the nuclear propulsion system is used for the control of normal operation as well as emergency reactor trip functionality.

A cost estimate for this system is based on the work by Zhang and Sun [88]. They estimated that the RPV and reactor internals (excluding fuel) for an HTGR make up about 54% of the reactor plant cost, while the EI&C system makes up about 10% of the reactor plant cost. Based on the assumption that this cost ratio is valid in this application and the default reactor option in Table 27 is used, the EI&C system would cost about $\frac{\$20mil}{54\%}10\% = \$3{,}7m$. The system is estimated to weigh about 0,5 ton.

4.3.2.3.4 Helium management and purification system cost

This system is required to maintain the desired amount of helium in the primary loop, while the helium is slowly leaking out of the system. A secondary function of this system is to continually purify the helium to remove contaminants from the primary loop.

According to Zhang and Sun [88], the "Reactor Auxiliary Systems", including the helium purification system, make up about 6% of the reactor plant cost, while the RPV and reactor internals make up 54% of the plant cost. This implies a cost of about $\frac{\$20mil}{54\%}6\% = \$2{,}2$ m.

This system is estimated to weigh 5 ton.

4.3.2.3.5 Reactivity control systems cost

The reactivity control system will be used by operators for start-up and shutdown of the reactor. It typically consists of control rods containing Boron-10 carbide. The rods are moved by actuators in sleeves inside the core to control the reactivity of the core.

No information could be obtained to establish a basis for the cost and weight of this system. Based on judgement, it is estimated to cost $2m and weigh about 1 ton.

4.3.2.3.6 Heating, ventilation and air-conditioning system cost

A Heating, Ventilation and Air Conditioning (HVAC) system will be required to control the environment around the reactor and power-conversion systems. The HVAC is expected to perform the following functions:

a. The HVAC will be required to keep moisture content as well as temperature within acceptable limits.
b. The HVAC must also ensure a sufficient air exchange in enclosed compartments to ensure a safe environment for maintenance workers.
c. A third function of the HVAC system is to filter the air that is released into the atmosphere to ensure that no radioactive gases or particles are released.

Since no information could be obtained to establish a basis for the cost and weight of this system, judgment was used to estimate the cost to be $2m and the weight to be 5 ton.

4.3.2.4 Summary of reactor design

Table 28 contains a cost and mass summary of the selected reactor design.

Table 28: Reactor Cost Summary

System	Cost Estimate [$m]	Mass [ton]
Reactor	215.00	556
Core Graphite	0,93	32
Graphite Reflector and Gas Plenums	8,11	278
Core Barrel	1,51	18,3
Reactor Pressure Vessel	10,41	216,9
Fuel	36,39	26,2
Reactor Support Systems	692.70	4096.5
Reactivity Control System	2	1
Decay Heat Removal System	3,3	77,8
Shielding System	0,675	4000[note1]
Electrical, Instrumentation and Control System	3.7	0.5
Helium Management System	2,2	5
HVAC	2	5
Total	**767.70**	**4124.5**

***Note1**: The shielding system mass can be reduced in line with the proposals in Section 4.3.1.6*

4.3.3 Fuel Temperature Analysis

This section presents a preliminary analysis of the operational temperature in the fuel. This analysis is required to confirm that the normal operational fuel temperature is sufficiently far removed from the fuel-failure temperature limit of 1600 °C [46]. Several assumptions are made to enable hand calculations:

a. All the heat generated inside the fuel is transported out by the coolant and an insignificant amount of heat is transported from the fuel into the graphite blocks.
b. An insignificant amount of heat is transported in the axial direction of the fuel.
c. The neutron flux in the fuel region is symmetrical

These assumptions reduce the heat transfer to a one-dimensional problem as indicated in Figure 43, which can be solved by means of 'hand calculations'.

A fuel element (equivalent to the HTTR design [71], [75]) inside a graphite block is shown in Figure 42. This model is used to calculate the fuel temperature. Each of the graphite blocks has 33 sleeves that can be filled with fuel elements. The total volume of the fuel in the ship reactor would thus be:

$$Vol_{fuel} = 6.12 m_{high} \times 33_{sleeves} \times 30_{blocks} \times \pi\left(R_f{}^2 - R_h{}^2\right) = 2.74 \text{ m}^3 \quad \text{.....(Eq 4.19)}$$

This implies an average fuel power density of $q''' = \frac{80MW}{2.74m^3}$ =29.19 MW/m^3.....(Eq 4.20)

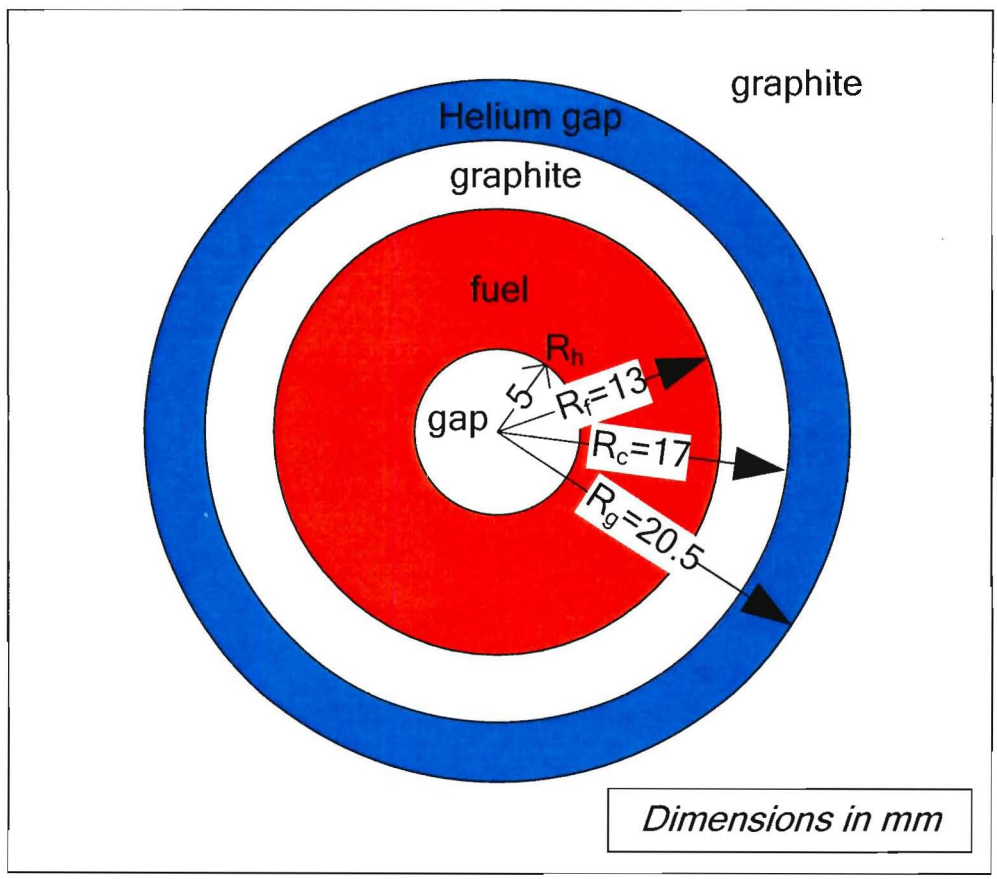


Figure 42: HTTR Fuel Compact [75][71]

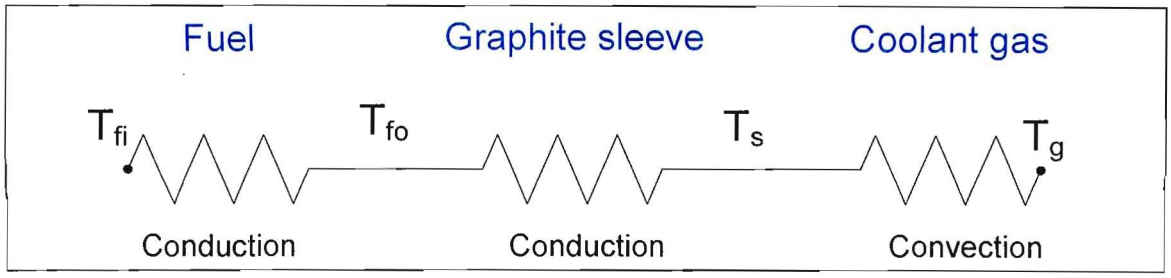


Figure 43: Heat Transfer Model

4.3.3.1 Conduction

The radial conduction is described by:

$$\frac{d}{dr}\left(r \times \frac{dT}{dr}\right) = -\frac{q'''}{k} \quad \text{.....(Eq 4.21)}$$

where k is the conduction coefficient, T the temperature (as defined in Figure 43), r the radial distance from the centreline and q''' the volumetric power density. For UO_2, the thermal conductivity is 2,5 W/m.K [26], while that of graphite is 30 W/m.K [35].

(Eq 4.21) can be integrated twice and solved, using boundary equations:

a. No heat will be transferred over the inside fuel boundary since the fuel design prevents any heat removal over this boundary:

$$-k_f \left.\frac{dT_f}{dr}\right|_{r=R_h} = 0 \quad \text{.....(Eq 4.22)}$$

b. The temperature on the outside of the fuel zone equals $T(r = R_f) = T_{fo}$ .....(Eq 4.23)

The temperature inside the fuel zone can be calculated by:

$$T_f(r) = \frac{q'''}{2k_f}\left\{\frac{1}{2}\left({R_f}^2 - r^2\right) + {R_h}^2\left(\ln(r) - \ln(R_f)\right)\right\} + T_{fo} \quad \text{.....(Eq 4.24)}$$

The temperature rise from the outside surface to the inside of the fuel is therefore:

$$\Delta T_f = \frac{q'''}{2k_f}\left\{\frac{1}{2}\left({R_f}^2 - {R_h}^2\right) + {R_h}^2\left(\ln(R_h) - \ln(R_f)\right)\right\} \quad \text{.....(Eq 4.25)}$$

In the graphite sleeve, (Eq 4.21)simplifies, because there is no heat generation:

$$\frac{d}{dr}\left(r \times \frac{dT}{dr}\right) = 0 \quad \text{.....(Eq 4.26)}$$

Again this equation can be integrated twice and solved, using boundary conditions:

a. Continuity of temperature at fuel-graphite boundary:

$$T_c(r = R_f) = T_{fo} \quad \text{.....(Eq 4.27)}$$

b. All the heat generated in the fuel will flow through the fuel-graphite boundary:

Fourier's law: $q''' \times Vol_{fuel} = -k_c \times Area_{fuel_outer} \times \frac{dT_c}{dr}$ (at $r = R_f$) .....(Eq 4.28)

The temperature distribution inside the graphite can therefore be described by:

$$T_c(r) = T_{fo} - \frac{q'''}{2k_c}(R_f^2 - R_h^2)\left[\ln(r) - \ln(R_f)\right] \quad \text{.....(Eq 4.29)}$$

So that the temperature rise across the cladding can be expressed as:

$$\Delta T_c = \frac{q'''}{2k_c}(R_f^2 - R_h^2)\left[\ln(R_c) - \ln(R_f)\right] \quad \text{.....(Eq 4.30)}$$

4.3.3.2 Convection

The convection is described by

$$Q = h_{conv} \times A_{clad} \times \left(T_{surf} - T_{gas}\right) = q''' \times Volume_{fuel} \quad \text{.....(Eq 4.31)}$$

with Q representing the power that must be transferred, A_{clad} the area of the cladding and h_{conv} representing the convection coefficient.

The temperature rise between the gas and the cladding surface can be calculated as:

$$\Delta T_{conv} = \frac{q'''}{2hR_c}\left(R_f^2 - R_h^2\right) \quad \text{.....(Eq 4.32)}$$

The Dittus-Boelter relationship [39] can be used to derive h_{conv}:

$$Nu = 0.023 \times \mathrm{Re}^{0.8} \times \mathrm{Pr}^{0.4} \text{ (for heating conditions)} \quad \text{.....(Eq 4.33)}$$

$$h_{conv} = \frac{Nu \times k_{coolant}}{D_k} \quad \text{.....(Eq 4.34)}$$

Where:

Nu is the Nusselt number

Re is the Reynolds number

Pr is the Prandtl number

k represents the conduction coefficient of the coolant

D_k represents the equivalent hydraulic diameter, which in an annular flow configuration equals the difference between the outer diameter and the inner diameter.

This Dittus-Boelter relationship is valid if:

0,6 < Pr < 160	(0,66 in this case)
Re > 10000	(~15000 in this case)
Length/Diameter > 10	(874 in this case)

An EES model was developed to calculate the temperature drops and a MATLAB script was used to plot the temperature distribution, as shown in Figure 44. The EES and MATLAB models are contained in Appendix D. The Nusselt and Reynolds numbers that were calculated agree well with values reported for the HTTR in [2].

At the average power density of the core, the temperature rises 366 °C above the coolant temperature. If the power density at the bottom of the reactor is assumed to be the same as the average power density, the fuel temperature rises to 1066 °C. Assuming a very conservative power peaking ratio of 1,2 at the bottom (outlet) of the reactor, the maximum fuel temperature is calculated as 1140 °C. This calculation model was used to calculate the equivalent maximum fuel temperatures for the HTTR reactor configuration to confirm that the model was correct. A maximum fuel temperature of 1340 °C was calculated for a 1,2 peaking ratio at the bottom of the core, which is roughly in agreement with published peak fuel temperatures.

The calculated maximum fuel temperature leaves 460 °C available for decay heat-up before the fuel design limit of 1600 °C is reached. Future analysis would have to confirm whether this can be achieved by the passive means described in Section 4.3.1.5.

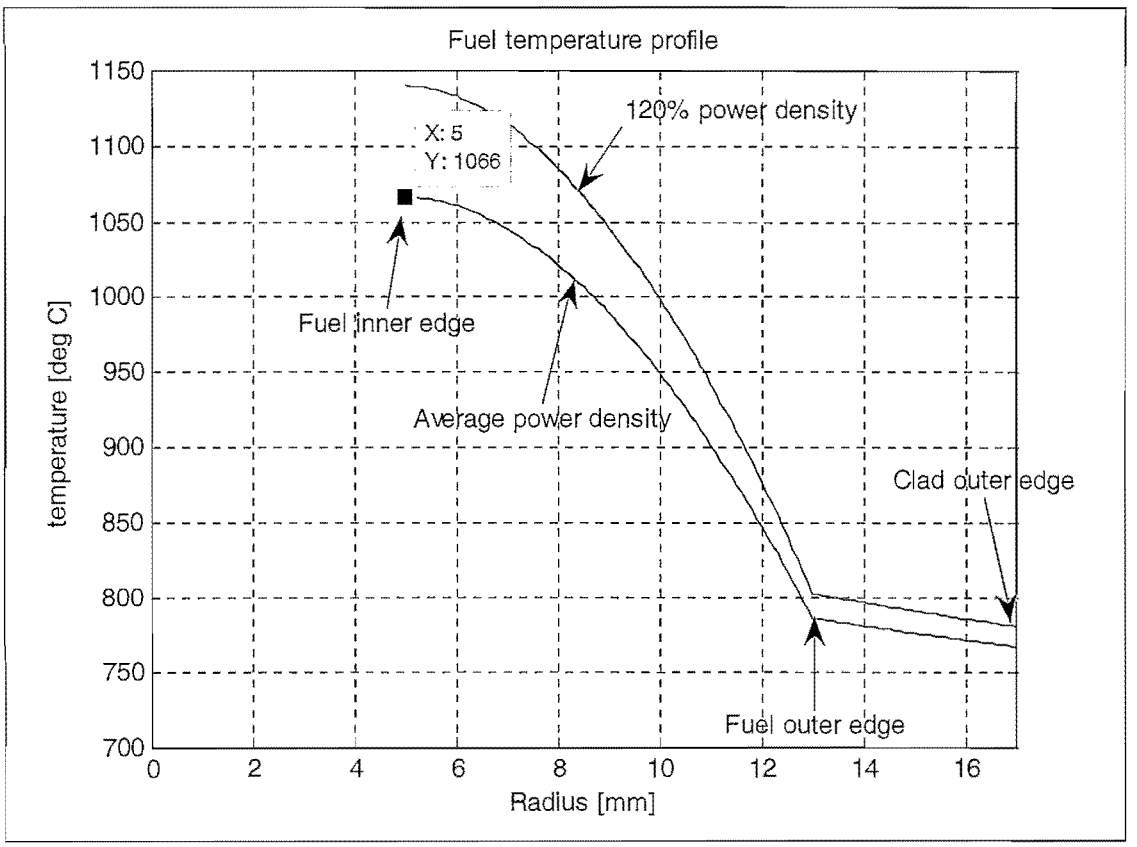


Figure 44: Fuel Temperature Profile

4.3.4 Pressure drop determination

The pressure drop over the reactor is required as an important input to the design of the power-conversion unit, since it influences the power required by the He circulator. The pressure drop can be determined by using the Moody chart and assuming that the graphite sleeves can be considered as smooth surfaces. The pressure drop can be calculated by the Darcy-Weisbach equation [39]:

$$\Delta P = f \frac{L}{D_k} \frac{\rho \times V^2}{2} \quad \text{.....(Eq 4.35)}$$

Where

f is the friction factor determined from the Moody chart (dependent on the Reynolds number and the surface smoothness)

L is the length of the reactor

D_k is the hydraulic diameter of the flow area

ρ is the density of the helium coolant

V is the flow velocity

For the chosen geometry and flow conditions, the helium velocity in the channels is 16,7 m/s and the Re number is ~12000. From the Moody chart [39] f is determined to be 0,03 for a smooth surface. This results in a pressure drop of 14 kPa. This value should be doubled to account for surface roughness and to make allowance for the inlet and outlet plenums. These equations were implemented in an EES programme that is contained in Appendix D.

4.3.5 Burn-up and Fuel Cycle

The burn-up attainable from TRISO fuel has been studied in many analyses, experiments and actual reactors. Table 29 shows reference designs from several countries involved in development of HTGRs [35].

Table 29 : TRISO Burn-up

	Germany	USA (fissile)	JAPAN (HTTR-30)	USSR	China	RSA (DPP-400)
Enrichment %	10,6	19,9	6	8	17	9,6%
Design temperature limit [°C]	1250	700 - 1250	1495	1400	1000	1130
Burn-up limit [% FIMA]	8 to 11 (THTR: 11%)	<26	3,6	8,8	11	9,5
Neutron fluence [10^{25} m^{-2}]	2 to 3	4,5	1,3	3,0	1,0	2,7

a. From this table it is clear that the HTTR burn-up limit of 3,6% FIMA is very conservative. In the ship engine application where the reactor outlet temperature is significantly lower than the HTTR design of 950 °C, it can reasonably be expected that the fission product release from the coated particles would be very small.

b. EES calculations (contained in Appendix B) show that the target burn-up would be reached after 864 days of full-power operation. At a maximum utilization of 90%, this equates to a refuelling cycle of 1018 days or 2,8 years.

c. An ideal refuelling cycle would fit in with the scheduled maintenance cycle of a VLCC ship, which is approximately five years. A five-year cycle implies that the discharge burn-up has to be 6,5%. A detailed neutronic analysis would have to be performed to determine the optimal enrichment level, burnable poison and heavy metal loading to enable this burn-up goal. As a first-order estimate it can be assumed that the enrichment would have to be 8%. The cost impact of such a change is shown in Table 27.

d. It would not be optimal to use only one enrichment level throughout the core. Different enrichment zones should be used to flatten the flux profile as far as economically viable, to reduce the difference between the maximum and average fuel temperatures. In addition, different enrichment zones are required to ensure an even burn-up of all the fuel. This study does, however, not consider this aspect in the cost estimates, but simplistically assume that all the fuel will have 8% enrichment.

4.3.6 Reactor Orientation

The reactor design that is proposed in Section 4.3.1 is independent of the vertical or horizontal orientation of the reactor.

As stated in Section 3.5, it is an important consideration to have the centre of gravity of the propulsion system as low as possible to enhance the stability of the ship. From this point of view, it is obvious that a horizontal reactor lying close to the keel of the ship is much more desirable than an upright reactor with a height of almost 20 metres.

Most nuclear reactors that have been built to date are cylindrical and upright. A spherical shape would have been optimal to prevent neutron losses. However, the preference for the cylindrical shape is because it facilitates uniform coolant flow through the reactor core. An upright reactor has many obvious benefits such as buoyancy forces that can enhance natural circulation in shutdown conditions, gravity-assisted control rod insertion and gravity- assisted fuel-element motion in a pebble bed reactor. This horizontal reactor concept was also considered by Jiafu Tian [68] for ship propulsion. In this application, the benefits of a horizontal reactor could outweigh the loss of the above-mentioned benefits.

4.3.6.1 Benefits of a horizontal reactor

a. Centre of gravity

The height of the centre of gravity of the reactor will change from 10 m for a vertical orientation to 2,6 m for a horizontal orientation. For the total ship, this change in centre of gravity is not expected to make a significant impact. A more localized effect could, however, be more significant. These effects could be analysed by means of a finite element analysis in a future study.

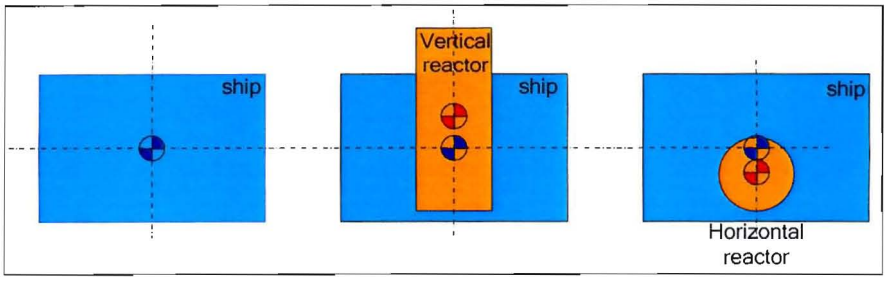


Figure 45: Frontal View of Centre of Gravity Change

b. Incorporation of the steam generator into the RPV

A horizontal reactor opens the possibility of incorporating the steam generator into the RPV without additional centre of gravity penalty.

c. Mechanical load distribution

Since the reactor has a weight of around 550 ton, it will be a challenge to mount an upright reactor sufficiently sturdy to withstand all loads. A horizontal reactor will provide significantly better load transfer from reactor to ship hull. This factor is especially important when dynamic loads from rough sea conditions are considered.

4.3.6.2 Disadvantages of a horizontal reactor

a. Installation of a 20 m-long horizontal reactor in a ship may be difficult, depending on the internal structure of the ship.
b. In addition, replacement of a horizontal reactor may require special and potentially complex changes to the ship layout.
c. A horizontal reactor with a steam generator inside the RPV would increase the probability of water ingress into the core of the reactor.

4.3.6.3 Layout

Figure 46, Figure 47, and Figure 48 show scaled diagrams of the reactor (with a 1 m thick shield) and power-conversion units next to and inside an oil tanker that is somewhat smaller than a typical VLCC. It seems from these diagrams that a vertically oriented reactor would be significantly higher than the main deck of the ship, while the horizontal reactor would seem to be able to fit. Obviously more detailed ship structural drawings would be required to be able to assess the layout.

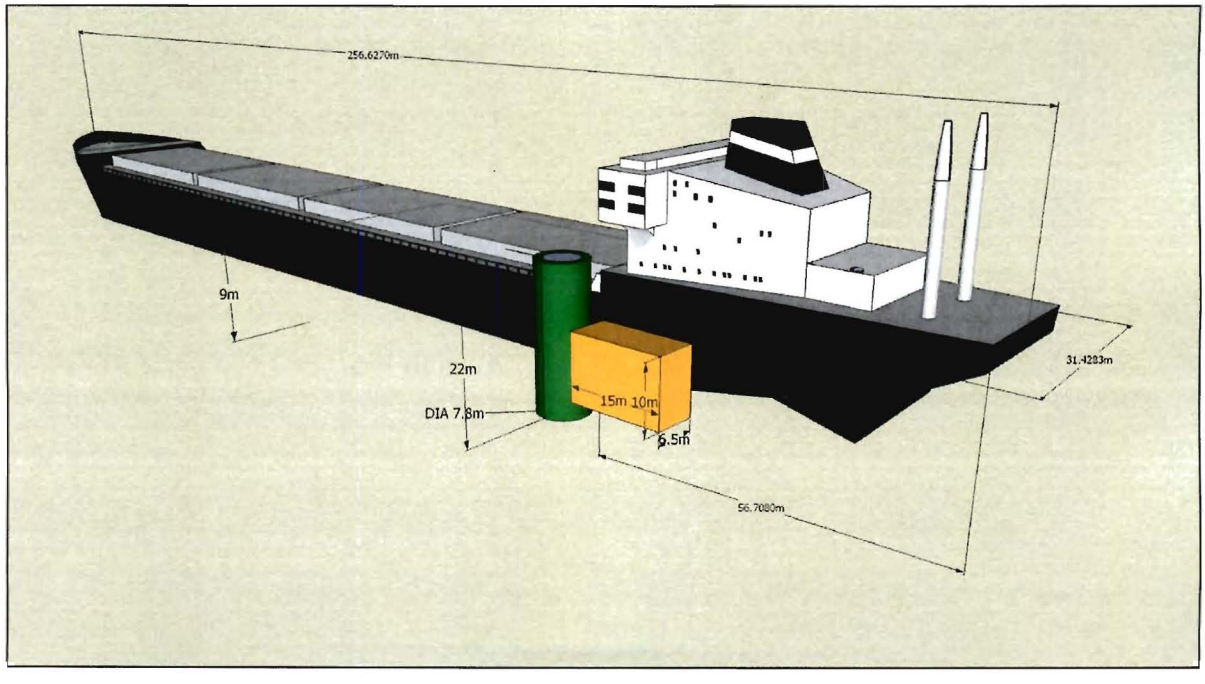


Figure 46: Reactor and Power Conversion Unit Next to a Typical Oil Tanker

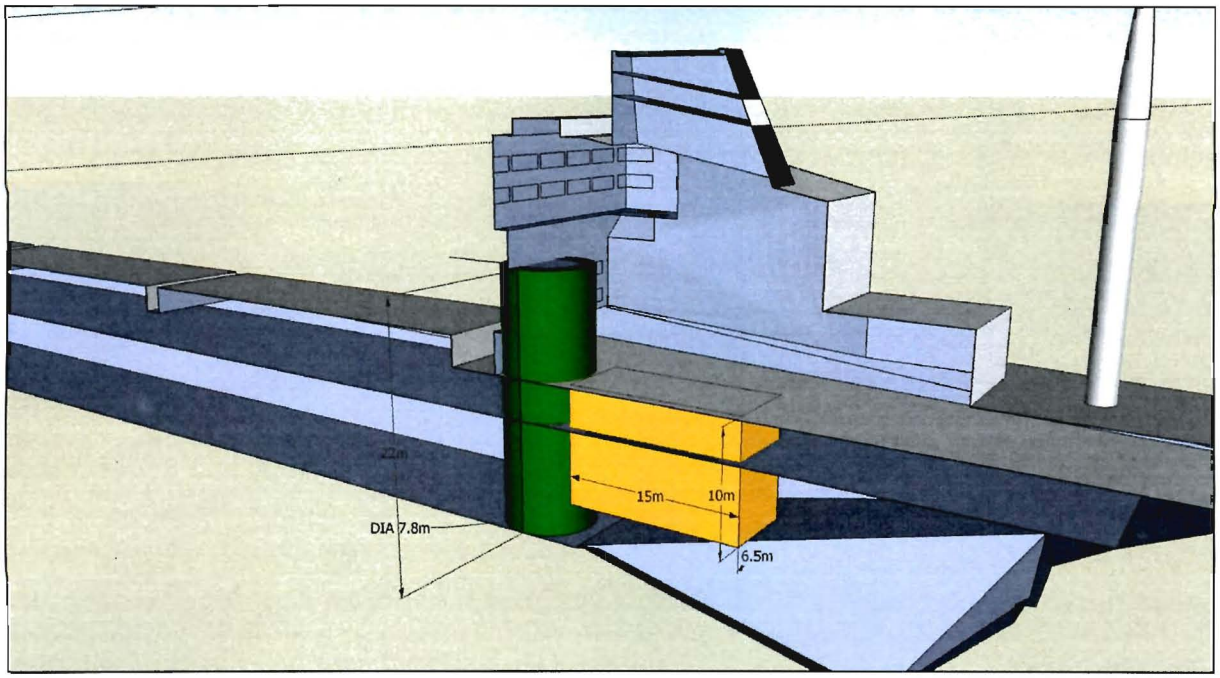


Figure 47: Reactor Upright Inside Ship

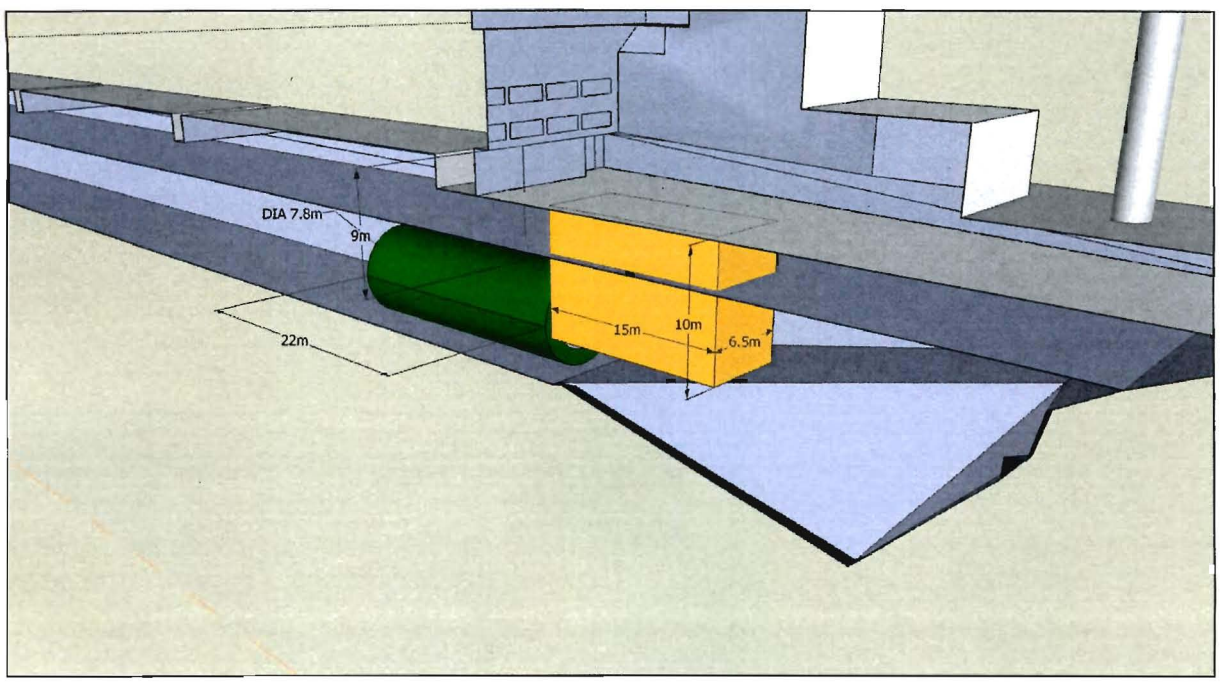


Figure 48: Horizontal Reactor Inside Ship

4.3.6.4 Issues that need special attention in the horizontal reactor concept

a. Control rod motion

In most power reactors, control rods can drop into the core under gravity to trip the reactor. This has the significant safety advantage that external power is not required to shut down the reactor in an emergency. Boiling Water Reactors (BWRs) are, however, a notable exception to this concept. BWRs have steam driers at the top of the reactor, requiring the control rods to enter the core from the bottom. These control rods are forced into the core by means of hydraulic power.

In a horizontal reactor, the control rods could be oriented horizontally and slide in graphite sleeves, or the control rods could enter the RPV from the top as indicated in

Figure 49 and Figure 50below. The relative advantages and disadvantages are compared in Table 30.

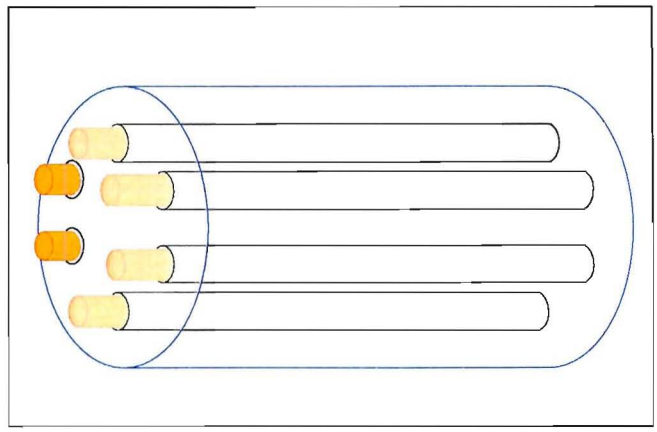


Figure 49: Horizontal Control Rods

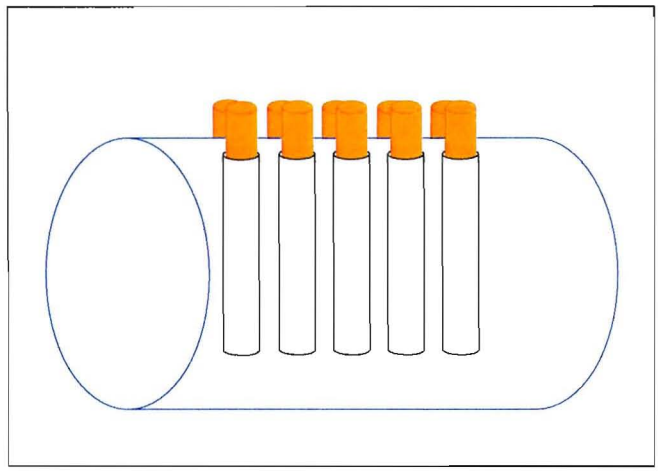


Figure 50: Vertical Control Rods

Table 30: Advantages and Disadvantages of Horizontal and Vertical Control Rods

Aspect	Horizontal Rods	Vertical Rods
Friction and dust	Friction between rods and bottom of sleeves will erode rods and sleeves, and will generate dust that will carry radioactive contaminants through the primary loop.	Much lower friction and dust.
Reactor trip	A reactor trip requires external power to push rods into the core.	Gravity will insert rods into the core, requiring no external power.
Refuelling	Easier to maintain sub-criticality while replacing fuel blocks. Leave most rods inside core while removing fuel from one sector at a time.	More complex to maintain sub-criticality while removing fuel blocks.
Bypass flows	Engineered bypass flow is used to cool rods	May be difficult to limit unwanted bypass flows and achieve desired rod-cooling

Aspect	Horizontal Rods	Vertical Rods
		bypass flow.
Reactor shape factor	The space required for the reactor becomes significantly longer	Better shape factor, since height is increased, but the control rods will be shorter than horizontal control rods.
Neutronic coupling between fuel and control rods	Better: fewer longer rods can be used	Weaker: requiring more control rods, implying more RPV penetrations.

Although both options seem feasible, the vertical-control rod option is selected for this application, mainly because of the advantage of passive-control rod insertion. Special attention must be given to the fuel replacement operations and coolant bypass flows.

b. Refuelling operations

The refuelling concept as discussed in Section 4.5.1 is based on replacing the entire RPV, rather than replacing fuel blocks inside the ship. With this concept in mind, refuelling operations can take place with the reactor in a vertical orientation, irrespective of its operational orientation.

c. Temperature distribution

The buoyancy effect that results from temperature-induced density changes causes natural circulation in vertical reactors. During shutdown conditions, this is often advantageous, because it enhances heat removal from the fuel. In a horizontal reactor this natural circulation would be much less pronounced, although it can be expected that the temperature in the upper part of a shutdown reactor will become hotter than the bottom parts. This is, however, not expected to have a significant effect.

4.3.6.5 Conclusion on orientation

Although several issues would have to be resolved to enable the horizontal reactor concept, this is deemed the preferred choice in the ship propulsion application.

4.4 SUMMARY OF PROPULSION SYSTEM DESIGN

A CAD model was developed to visualize the reactor and power conversion units and to establish the overall size of the equipment. In this CAD model the radiation shield was assumed to be a 1 m thick structure, because it is deemed possible to achieve the desired shielding with such a structure if the appropriate materials are used, as discussed in Section 4.3.1.6.

A section through the CAD model is shown in Figure 51 below, while the layout of the whole system is shown in Figure 52. The overall dimensions as indicated in Figure 52 are 37.2 m long, 7.8m wide and 10m high, excluding the electrical motor drive unit. It is clear that the intended size objective of 10 m^3 as set out in Table 9, could not be met with this design.

The total mass budget is shown in Table 31. In this mass budget, the mass of the compact steam generator and blower was included with that of the reactor. The mass of the shield is shown separately since it has not yet been optimized. The table shows that the mass objective of 2000 ton set out in Table 9, is not met with this design. The mass of the system without the shield is, however, close to 1000 ton, which implies that the mass objective could be reached if the shield mass can be reduced to 1000 ton.

Table 31: Total Mass budget

Subsystem	Mass [ton]
Turbine power conversion system	152
Drive motor	280
Reactor	566.4
Reactor support systems (excluding the shield)	89.3
Radiation shield	4000
Total	5087.7

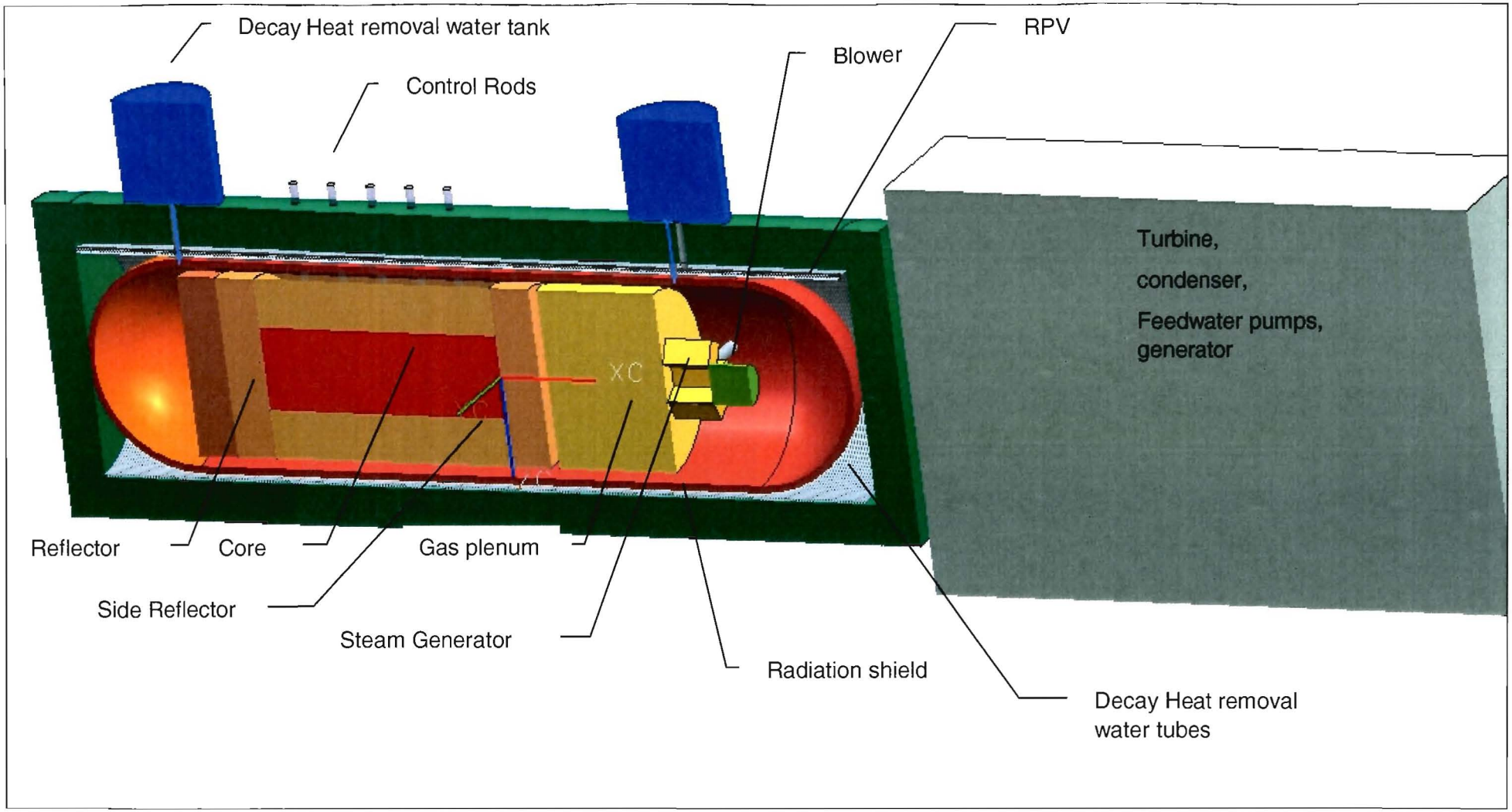


Figure 51 CAD Model: Section

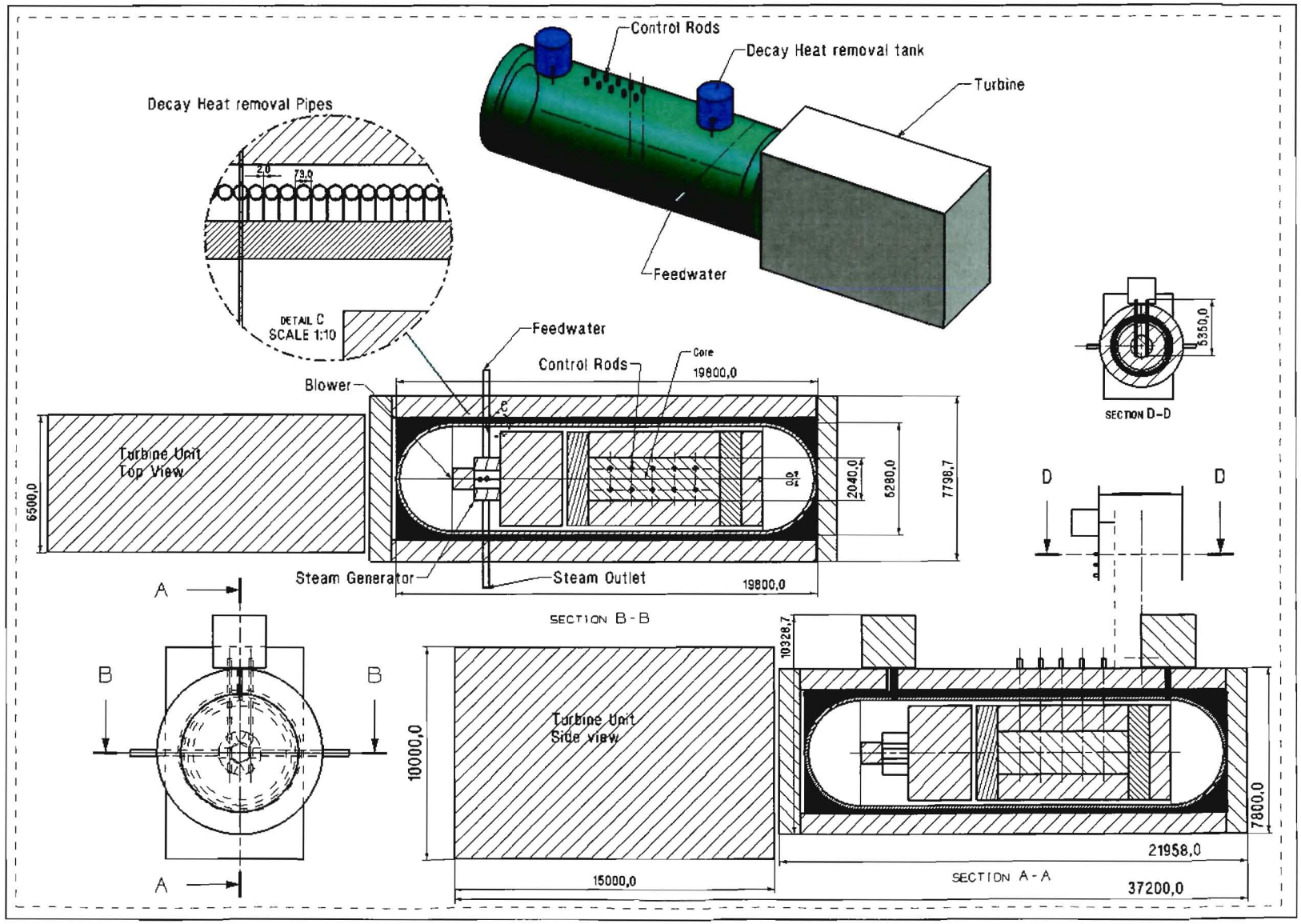


Figure 52: CAD Model: Layout

4.5 LOGISTICS

The logistics system that is required to support a fleet of nuclear ships would be extensive. Many complex issues would have to be resolved to enable smooth and reliable operation of the propulsion system:

a. Specialized home dock to perform reactor maintenance and refuelling.
b. Fresh-fuel handling from factory to reactor refuelling.
c. Spent-fuel handling to safely remove used fuel from the reactor, temporary storage and transport to final repository or reprocessing facility.
d. Safeguards against loss of nuclear material.
e. On-board maintenance, testing and inspections.
f. Annual scheduled maintenance, testing and inspection.
g. Refuelling of the reactor.
h. Decommissioning of the reactor and other radioactive components or materials at the end of the service life.

Some of these issues were briefly addressed in Section 4.1.2 under 'Fuel Management'. Only the refuelling concept will be developed further in this study. Other issues have to be resolved in future studies.

4.5.1 Refuelling

Removing the spent-fuel blocks on board the ship is not deemed a desirable action. Instead, it is proposed that the entire reactor with its gamma shielding is removed from the ship as a unit, as discussed in 4.1.4.2. Fuel replacement can then be done in a special maintenance facility where it will be easier to ensure radiation protection of workers. In addition, such a facility can ensure that contamination does not enter the reactor and that radioactive waste does not contaminate the ship. This concept will, however, require a specially adapted ship structure to allow removal of the reactor without the need to remove other equipment or structures. In addition, the concept would require heavy lifting and transport equipment, radiological shielding as well as a specialized clean facility close to the home port.

This reactor replacement concept has also been proposed for the Japanese MRX reactor, as shown in Figure 53 [87].

When a sufficiently large number of nuclear ships are in operation, the reactor can be replaced with a freshly-loaded reactor without requiring the ship to wait for the fuel replacement.

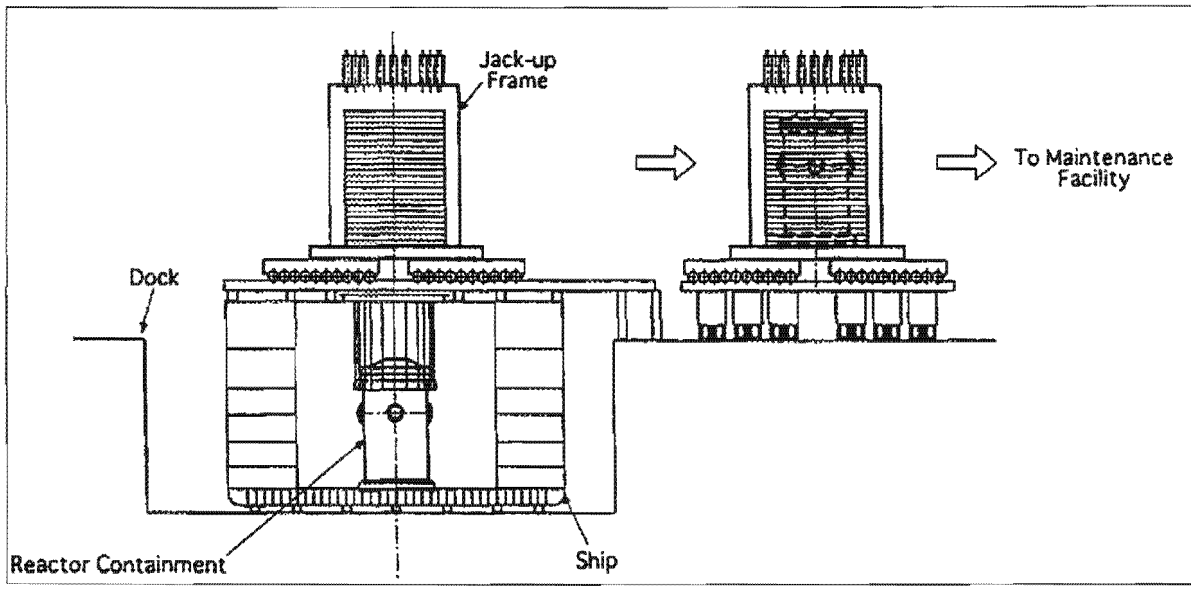


Figure 53: MRX Reactor Removal Concept [87]

4.5.2 Conclusion on Logistics

Based on this brief investigation, the logistic support system for a nuclear ship would be extensive and require significant investment. These aspects need to be investigated in future studies.

5. ECONOMIC INVESTIGATION

This chapter evaluates the economic viability of the concept design that was developed in previous chapters. Two benchmarks from other studies are evaluated in section 5.1, the cost model for diesel propulsion is provided in section 5.2, while the cost model for the concept that was developed in this study is presented in section 5.3. Section 5.4 provides a cost comparison between this nuclear propulsion system and the diesel alternative.

5.1 BENCHMARKS

Cost comparisons from two similar projects are provided below to enable comparison to the results obtained in this study:

5.1.1 MRX [87]

Figure 54 shows a cost comparison of the Japanese MRX to a diesel equivalent [87]. The diagram compares the Required Freight Rate (RFR) in $ per Twenty feet Equivalent Unit (TEU) for a container ship.

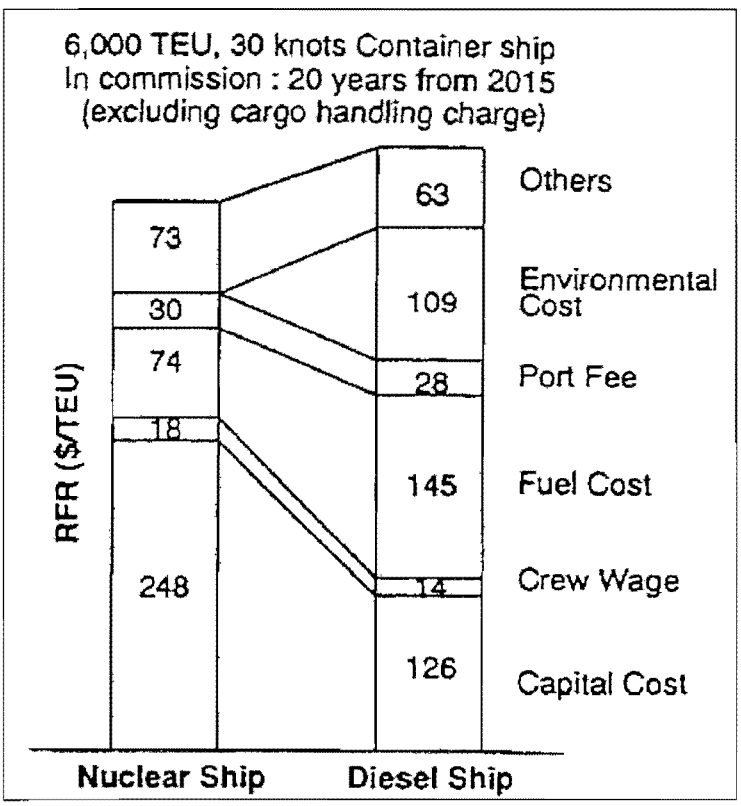


Figure 54: MRX Cost Comparison

From this comparison it is seen that the cost discriminators between the two types of propulsion are: capital, fuel and environmental cost. The other cost components are essentially similar for the two options.

5.1.2 NEREUS Project [24]

The NEREUS cost comparison was done for an 8 MWe propulsion unit. The benchmark diesel engine with an 8 MWe capacity was determined to cost $8m and the life-cycle cost to be $65 per MWh. The cost of the NEREUS HTGR propulsion system is set out in Table 32 and Table 33:

Table 32: NEREUS Capital Cost Structure

Item	Cost [$m]	
Nuclear plant		17
Reflector (75 ton @ $25/kg)	2	
Control rods, reactor internals and core barrel	1	
RPV (32 ton @ $45/kg)	1,5	
Shield (cast iron @ $1000/ton)	2	
Helium blower	1	
Passive decay heat removal system	0,5	
Power-conversion system		6
Total capital cost		14

Table 33: NEREUS Life-cycle Cost Comparison to Diesel Alternative

Item	NEREUS		Diesel Engine	
	Cost [$/MWh]	Cost [$m/year]	Cost [$/MWh]	Cost [$m/year]
Interest on capital and depreciation	44,4	2,24	25	1,2
Fuel	5	0,252	34	1,72
Lubrication	0	0	2	0,1
Maintenance, personnel and insurance	3	0,152	4	0,2
Total	56	430.00	65.00	79.00

***Note 1**: The assumption was that the ship engine will operate at 90% capacity for 80% of the time: 7000 hr per year, delivering 50500 MWh per year.*

***Note 2**: Capital interest and depreciation is based on 16% per year.*

Based on this calculation, the NEREUS programme is considered to be cost-competitive with the equivalent diesel-alternative.

When the capital cost for the NEREUS is compared with the capital cost calculated for this ship engine, it appears that the NEREUS cost (scaled according to power) is almost 50% lower than those calculated in this study. The NEREUS fuel cost is also significantly cheaper (30 times) than calculated here. This difference can be explained partly by the rising cost of materials since the NEREUS report was compiled in 2006 and partly by more conservatism applied in this study.

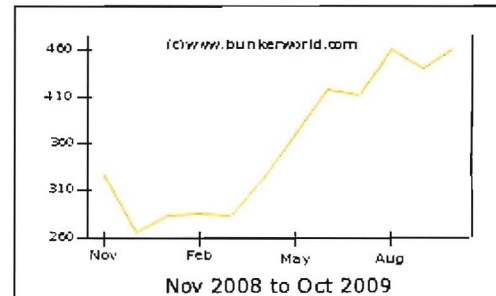


Figure 55: LS180 price [$/ton]

5.2 COST MODEL FOR A DIESEL ENGINE ALTERNATIVE

In order to enable a life-cycle cost comparison, a 25 MW diesel engine from MAN, type 6S80ME-C8, is used. As mentioned in Section 3.6, this engine is estimated to cost €6,8m, or $8,2m, based on an enquiry to the supplier. Based on supplier information, this engine uses fuel at a rate of 170 g/kWh, while the lubrication oil consumption is typically 0,1 g/kWh, and cylinder oil consumption is given as 0,6 g/kWh [53].

Based on the assumption in Section 3.7 that the engine would work at full power for 80% of the time, it can be calculated that the engine would produce 365.25*24*0.8*25 = 175,320 MWh per average year. This equates to 29 804 ton of fuel and 122,7 ton of oil per year.

Currently (October 2009), the LS180 Heavy Fuel Oil (HFO) price is $470/ton [17]. See Figure 55 for the recent trend in the price of LS180. This equates to an annual fuel cost of $14m. The current price of SAE 30 lubricating oil is around $800/ton [54], which equates to a lubrication oil cost of $100 000 per year. The LS180 price varied between $290/ton and $500/ton over the past year.

5.3 COST MODEL FOR THIS DESIGN

5.3.1 Capital Cost

The capital cost of the nuclear propulsion unit is based on the Rankine cycle propulsion system design in Section 4.2, with the compact heat exchanger inside the RPV, and the reactor system design in Section 4.3. These values are for the n^{th} production unit and excludes all development and industrialization cost. An additional 20% is added to the estimated price to make allowance for uncertainties and risk.

Table 34: Summary of Capital Cost

System	Estimated Capital Cost [$m]
Power-conversion system	35,54
Reactor (without fuel)	20,96
Reactor support systems	13,88
Risk and uncertainty (20%)	13,7
Total	**326**

5.3.2 Running Cost

5.3.2.1 Maintenance and operation

In line with the MRX comparison above and the Jacobs study [43], it is assumed that the maintenance and operational costs of this simplified nuclear propulsion system are equivalent to that of a diesel engine and are therefore not considered in the cost comparison.

5.3.2.2 Fuel cost

The fuel cost was calculated in Sections 4.3.2.1.1 and 4.3.2.2.1 and is shown in Table 27 to be $36,39m for a full reactor load. Based on the burn-up calculations in Section 4.3.5, it will

be assumed that the entire fuel load will be replaced every 5 years. The annual fuel cost is thus calculated to be the amortization of a five-year loan of $36,39m.

5.3.2.3 Opportunity cost of fuel space

An unique factor in the life-cycle cost comparison between nuclear propulsion and diesel propulsion is the large amount of space and weight associated with the diesel fuel that is saved when using nuclear propulsion.

On the cargo-carrying leg of the typical voyage, a VLCC will carry 280 000 ton of cargo and fuel space for 170 kg/MWh * 25 MW * 20 days = 2040 ton of fuel must be allowed. This implies that a nuclear-powered ship can carry 0,7% more cargo than a diesel-powered equivalent. This factor is deemed insignificant in terms of cost and is excluded from the cost comparisons.

The weight and space that is freed up can, however, be made available to the equipment of the nuclear engine. At a density of 930 kg/m^3, the fuel space is estimated be 2200 m^3.

5.3.2.4 Opportunity cost of carbon tax

It is generally accepted that most governments will start phasing in carbon taxes as a mechanism to reduce global warming. The current value of the "European Union Allowances" (EUA) carbon credit is €15/ton of CO_2, with forecasts of around €37/ton for the year 2016 [61]. This forecast is in line with the €30/ton used in a March 2009 analysis done by the International Council of Shipping [31].

The amount of CO_2 emitted by a ship engine is not common knowledge. One source has estimated that the 80 MW Emma Maersk container vessel emits about 300 000 tons of CO_2 per year. If it can be assumed that the amount of CO_2 emission can be scaled in accordance with the engine capacity, the equivalent 25 MW engine would emit about 94 000 ton of CO_2 per year. This opens up an opportunity to sell EUA carbon credits worth 15 x 94 000 = €1,4m or $1,7m per year at the current prices, or $4,2m at 2016 forecast prices. This value will be taken into consideration in the diesel versus nuclear cost comparison.

5.3.2.5 Spent-fuel handling cost

The cost of spent-fuel management and storage is already included in the cost of the fuel as calculated in Section 4.3.2.1.1, and no additional cost will be built into the cost structure.

5.3.3 Decommissioning Cost

It is difficult to estimate the cost of decommissioning a nuclear reactor.

Examples include the Fort ST Vrain HTGR that was decommissioned in 1989 for $195m, which equates to $513/kWe.

An OECD study in 2003 established that decommissioning of western PWRs typically costs $200 to $500 per kWe, while the decommissioning cost for gas-cooled reactors is estimated to be about $2600/kWe [60]. This value is judged to be excessive, and for this study, a decommissioning cost of half this quoted value will be used, which is still four times as much as the LWR benchmark.

If this value is assumed to be correct and applicable, the decommissioning cost for the nuclear ship engine would be $32,5m. This cost is, however, due only at the end of the 20-year life of the engine. The annual amount of money that must be invested to cover this

expense at the end of the reactor life is dependent on the net interest rate that can be expected on the investment.

5.3.4 Licensing Cost

The cost of licensing a new nuclear system is probably the single biggest risk factor. In this application, the risk is exacerbated by the fact that the ship propulsion unit would have to be acceptable to multiple nuclear regulatory bodies, with different requirements. As pointed out in Section 3.9.2, the safety design requirements for this system are based on the requirements of the IAEA, which should in theory enable international acceptance. The cost of licensing is, however, not quantified in this study.

5.4 COST COMPARISSON BETWEEN NUCLEAR AND DIESEL PROPULSION

When diesel propulsion cost is compared with nuclear propulsion cost, there are two variables that play a significant part: the oil price and the interest rate on capital expenditure. The interest rate plays a significant role in the cost of the nuclear option, while the oil price plays a large role in the diesel propulsion option. The sensitivity to these two variables is analyzed in this section.

The cost comparison that is applied here is to convert all relevant life-cycle cost components into annual cost components at current dollar values. The capital cost of the propulsion system is converted into an annual payment over the 20-year lifespan of the engine. The fuel cost is converted into an annual instalment over the five-year refuelling cycle. It is assumed here that the same interest rate would be applicable to finance the initial capital cost, the five-yearly fuel cost, and also to the decommissioning investment that will be built up over the life of the ship. The interest rate that is applicable here is the real (or net) interest rate, which is the nominal interest rate, minus the inflation rate.

An annuity payment on a loan can be calculated by [16]:

$$PMT_{loan} = \frac{principal \times i}{1-(1+i)^{-n}}, \quad \text{.....(Eq 5.1)}$$

while an annuity for a future investment value can be calculated by [16]:

$$PMT_{invest} = \frac{Future_Value \times i}{(1+i)^{n}-1} \quad \text{.....(Eq 5.2)}$$

where i represents the real interest rate and n the number of periods.

Maintenance and operational costs are ignored in this comparison, because they are assumed to be similar for the two applications.

Table 35 lists the various annual cost components:

Table 35: Annual Comparative Cost Components

Component	Diesel Propulsion [$m]	Nuclear Propulsion [$m]
Capital loan repayment	$\frac{8.2\times i}{1-(1+i)^{-20}}$	$\frac{84.08\times i}{1-(1+i)^{-20}}$
Fuel	30 000 ton x fuel price	$\frac{36.39\times i}{1-(1+i)^{-5}}$
Lubrication oil	0.1	0
Decommissioning investment	0	$\frac{32.5\times i}{(1+i)^{20}-1}$
Carbon tax	0	-4,2

Note: *i is the real annual interest rate.*

These cost items were compared as a function of interest rate and fuel cost. The result is shown in Figure 56. From this graph it can be seen how the annual cost of the two alternatives change as a function of interest rate and Heavy Fuel Oil cost.

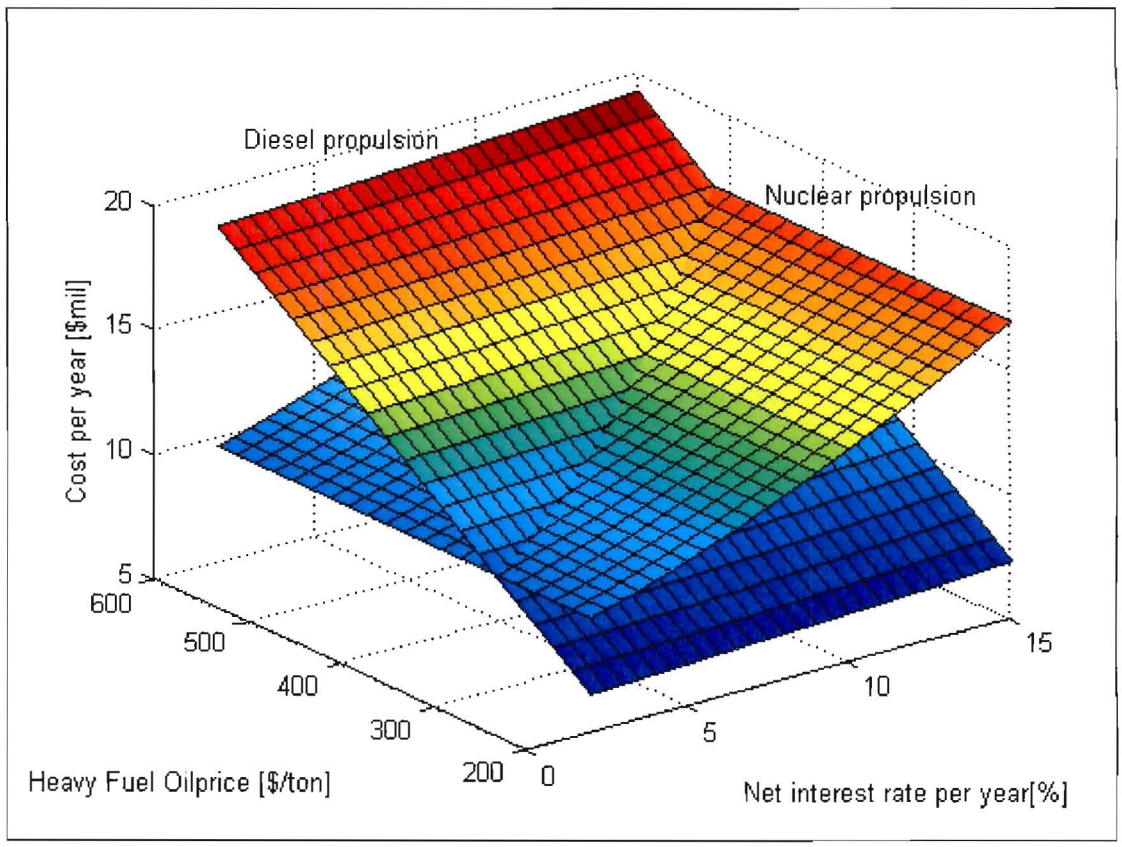


Figure 56: Annual Cost Comparison between Nuclear and Diesel-propulsion Options

Figure 57 below maps the relative cost competitiveness of diesel propulsion versus nuclear propulsion. The green area indicates the oil price and interest rate combinations for which nuclear would be cheaper than diesel propulsion. The horizontal dashed line indicates the current Heavy Fuel Oil (LS180) price. The dotted line shows where the equal-cost boundary between the two options would be in the absence of carbon credits.

This graph shows that even if capital is be borrowed at a net rate of less than 12,6%, the nuclear propulsion system would still cost less than a diesel alternative. This is significant, since the current real interest rate in South Africa is only about 4%, while the real interest rate in the USA is very close to zero at this stage.

The dotted curve shows that even if carbon credits are ignored, the nuclear-propulsion system would offer cost benefits at heavy-fuel oil prices above $500/ton at the current South African real interest rates.

This cost model was also used to calculate that the cost benefit of this nuclear propulsion relative to a diesel alternative would be around $4,26m per year at the current fuel oil price and 4% real interest rate.

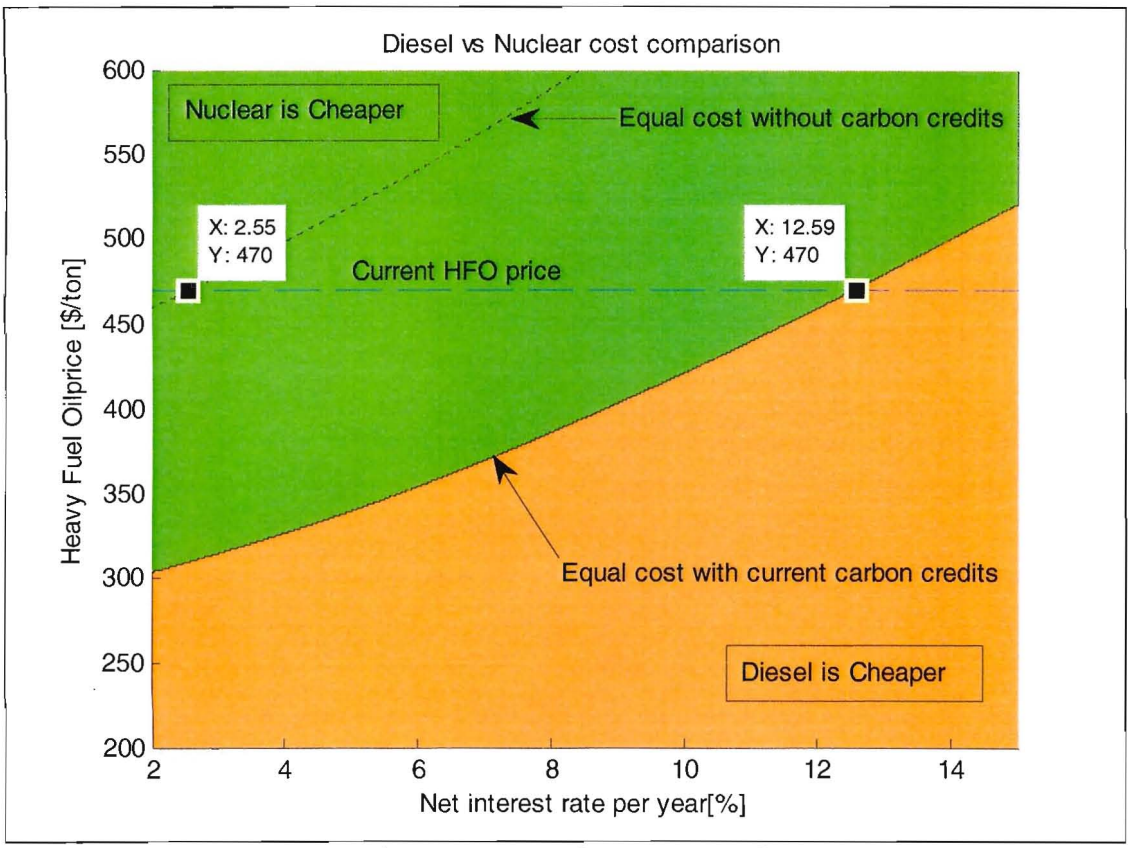


Figure 57: Comparison of Annual Costs

5.5 CONCLUSION ON ECONOMIC INVESTIGATION

The economic evaluation presented above shows that a mature nuclear propulsion system would be competitive with a diesel alternative if the life-cycle cost is considered. More work is, however, needed to evaluate the development and licensing cost of the first system.

6. PROJECT RISK MATRIX

A new system such as the one proposed in this study will always be subject to some elements of risk. Table 36 below shows a qualitative judgement of the various risks.

Table 36: Risk Matrix

Probability of occurrence	Low	Medium	High
High	Wiegner energy build-up in graphite blocks due to too low temperatures	The size of the nuclear engine (with shielding) is larger than a diesel engine equivalent	Multi-national nuclear licensing
Medium		The ship design may have to change to accommodate the reactor and specifically allow for replacing the reactor every few years for refuelling	Public and industry acceptance of nuclear power Ensure maintainability of Blower on the inside of the RPV
Low	The current radiation shielding system is too heavy and too large	Natural circulation in decay heat-removal system may not work In a horizontal reactor Power/heat dumping system during "islanding" state could be large.	Water from a steam generator leak could enter the reactor core and lead to fuel damage.

Impact of Occurrence

The risk items listed here would have to be investigated and resolved in future studies.

7. CONCLUSIONS

At the onset of this study, the objective was to determine whether a nuclear propulsion unit could be designed to be a feasible alternative to a fossil fuel-powered ship engine.

The Very Large Crude Carrier class of ships was selected as a suitable platform for nuclear propulsion. Based on this ship type, a set of requirements for the propulsion system were developed to guide the development effort. The required propulsion power was determined to be 25 MW.

A High-temperature Gas-cooled Reactor, using block fuel, was selected as the energy source. For the heat-to-propulsion conversion, a Rankine cycle was selected, which is based on a standard steam turbine. The overall efficiency of the power-conversion system was determined to be 31,5%, leading to a reactor thermal power of 80 MW.

The major subsystems required for the propulsion system were designed with sufficient detail to enable a first-order mass, size and cost estimate.

The size and mass of the designed system exceeded the objectives established for the project. However, if the space that is normally conserved for diesel fuel can be utilised, the size objective can almost be achieved. The mass objective is deemed to be achievable by utilising appropriate materials. The mass and size of the proposed solution is within an acceptable range for a VLCC. The concept design therefore seems technically viable.

Safety will always be an important aspect in a nuclear propulsion system. Although much attention has been given to nuclear safety in this study and inherent safety has been designed into the concept, the full scope of analysing the safety of the system was not encompassed in this study. It is thus not yet possible to determine if the nuclear engine is as safe as an equivalent diesel engine. This should be determined in future work.

The cost estimate was compared with the cost associated with an equivalent diesel engine. Despite the fact that many assumptions were required and many simplifications were made to enable a like-for-like comparison, it is clear that the nuclear propulsion system that was designed conceptually is economically viable for the selected ship. The life-cycle cost comparison shows that even at the current fuel oil price and interest rates, the nuclear engine is significantly cheaper than the diesel alternative. Considering trends in oil prices, this definitely justifies further investigation.

Based on the abovementioned, the study therefore suggests that nuclear propulsion should once again be considered as a viable alternative for commercial maritime propulsion. Although significant development cost and licensing effort will be required, trends with regards to fossil fuel prices and environmental pressures only amplifies the justification to support such a development.

8. FUTURE WORK

Many issues would have to be resolved in future studies:

a. The cost to develop and license a prototype nuclear propulsion system needs to be determined to establish the viability of developing such a system.

b. A complete safety concept for the propulsion unit would have to be developed, based on a specific regulatory framework. The chosen regulatory framework would depend on the nationality of the launch customer.

c. An operating concept for the engine needs to be developed.

d. A logistic support concept needs to be developed.

e. One of the very important issues is to optimize the shielding around the reactor to reduce the weight and size of the shield to more acceptable values. Several proposals are provided in Section 4.3.1.6.

f. A mechanism has to be developed to remove power from the reactor under "islanding" conditions, when the reactor is critical at minimum power (50%), but disconnected from the propulsion system.

g. Investigate the changes that would be required to this propulsion system to make it applicable to large container ships. The economic advantages might be as attractive as or even better than calculated in this study because of the significant effect of fuel oil cost on diesel engine propulsion.

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10. APPENDIX A: DIAGRAMS OF RELEVANT REACTOR PRECEDENTS

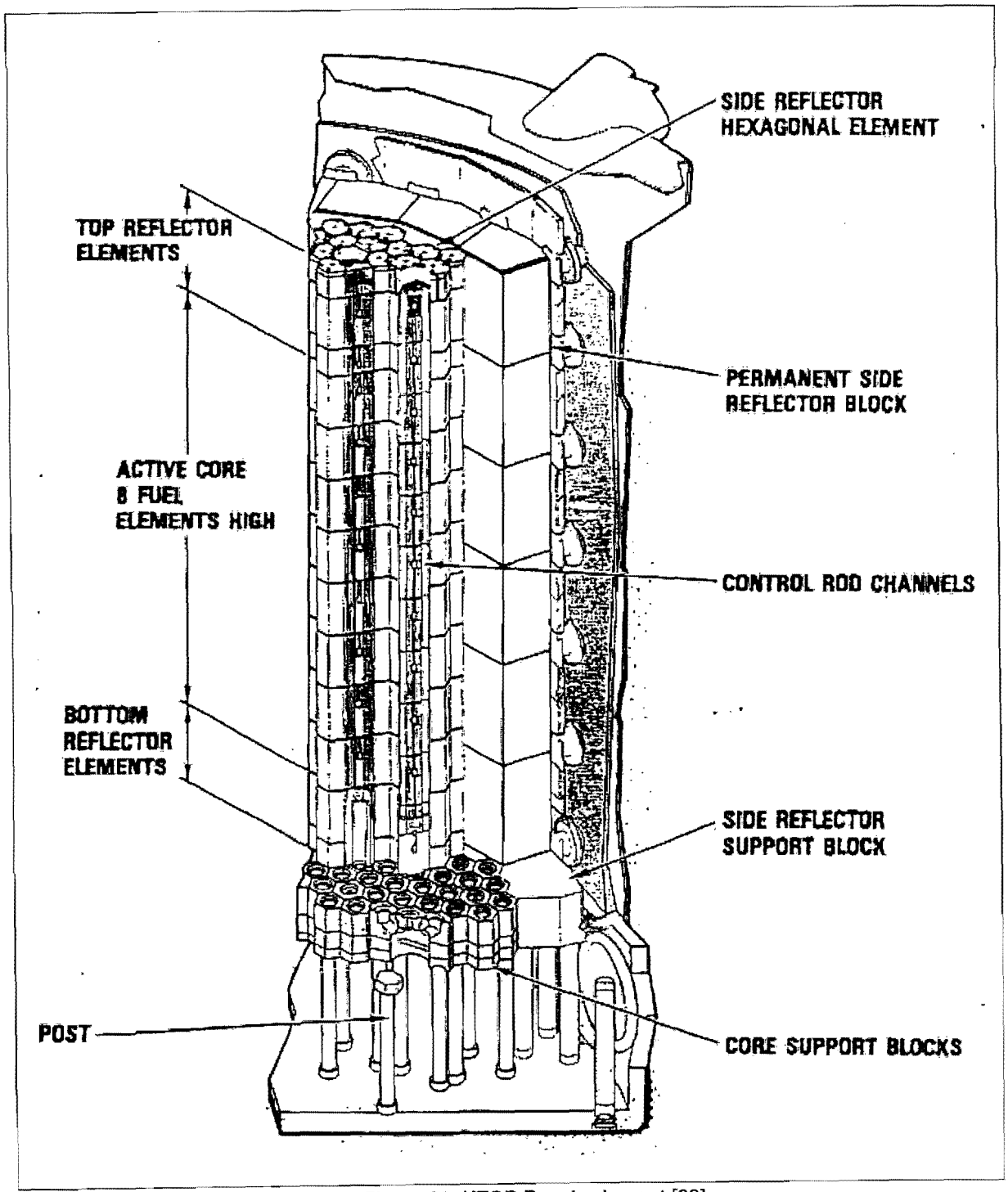


Figure 58: HTGR Reactor Layout [29]

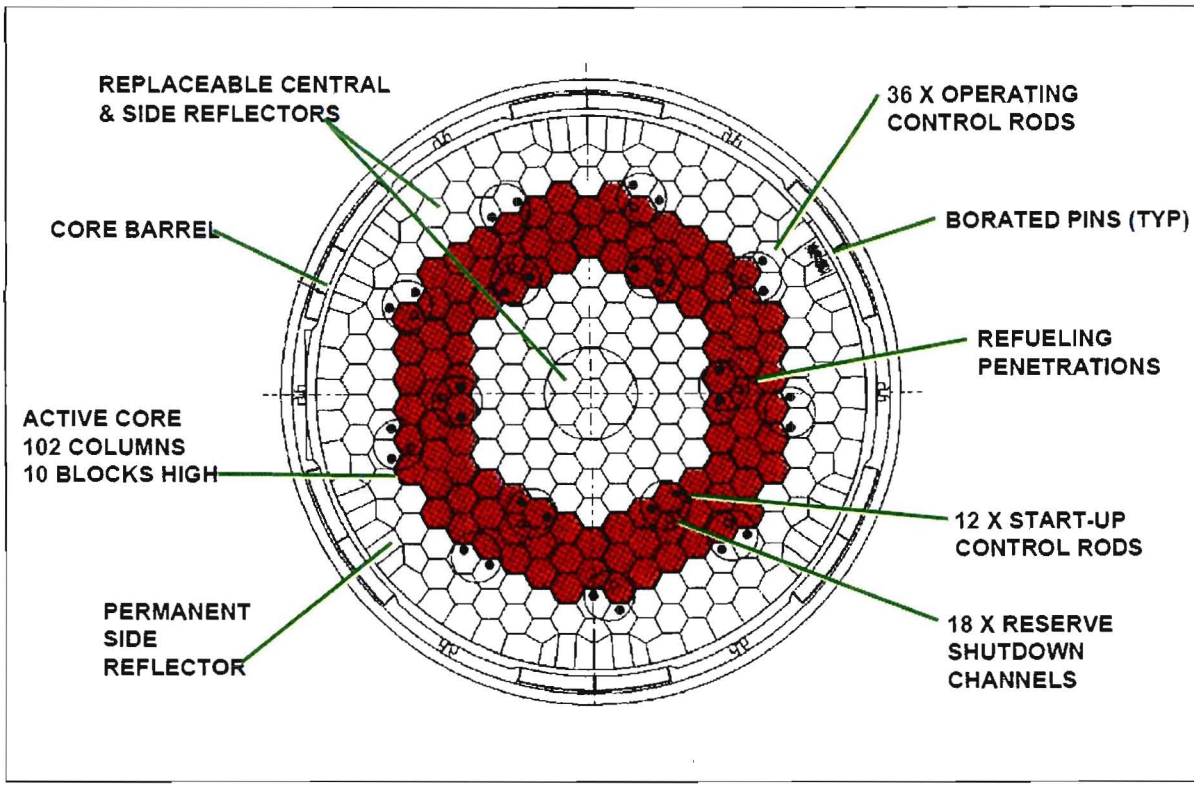


Figure 59: HTGR Core Layout – Top View [59]

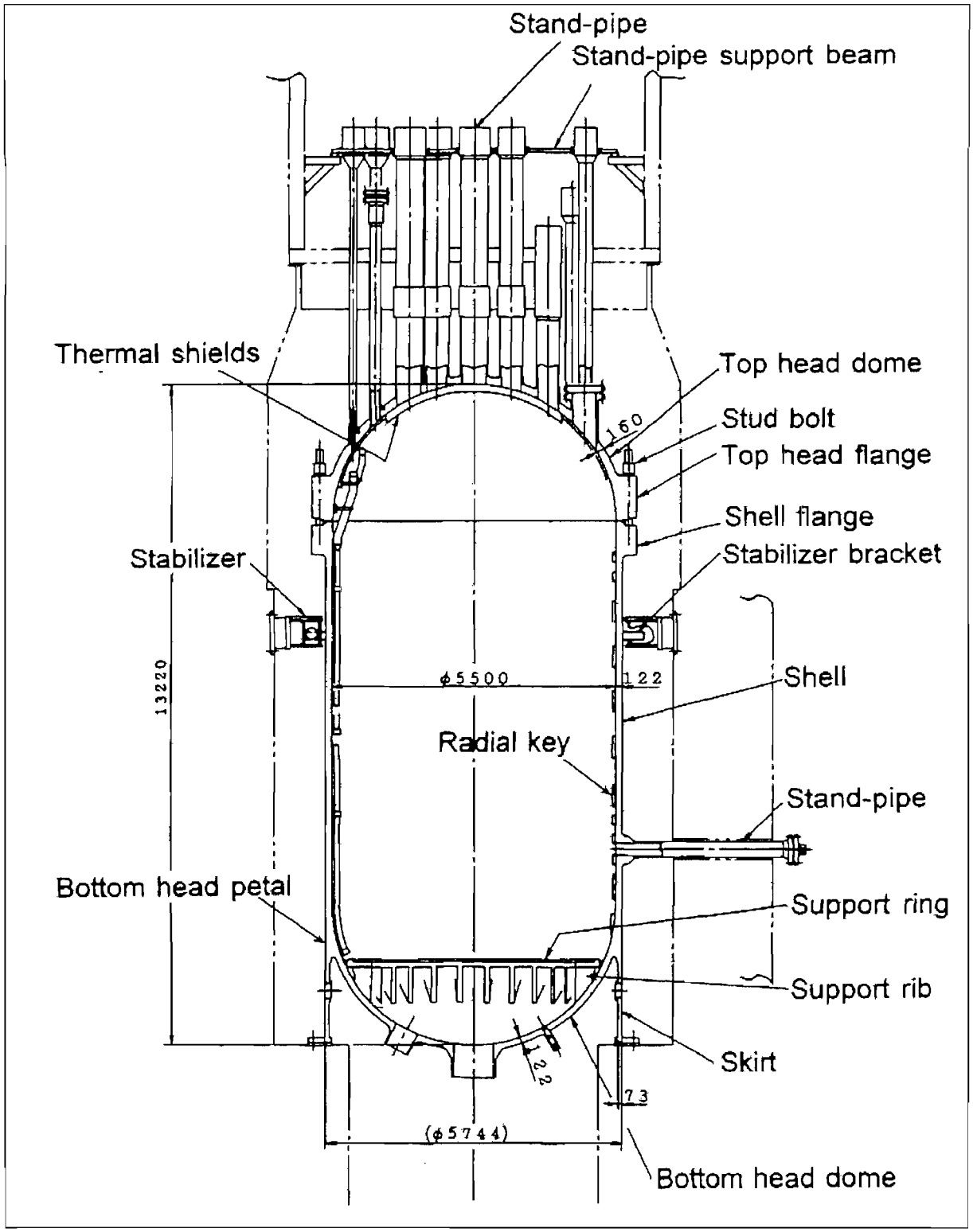


Figure 60: HTTR Reactor Pressure Vessel [36]

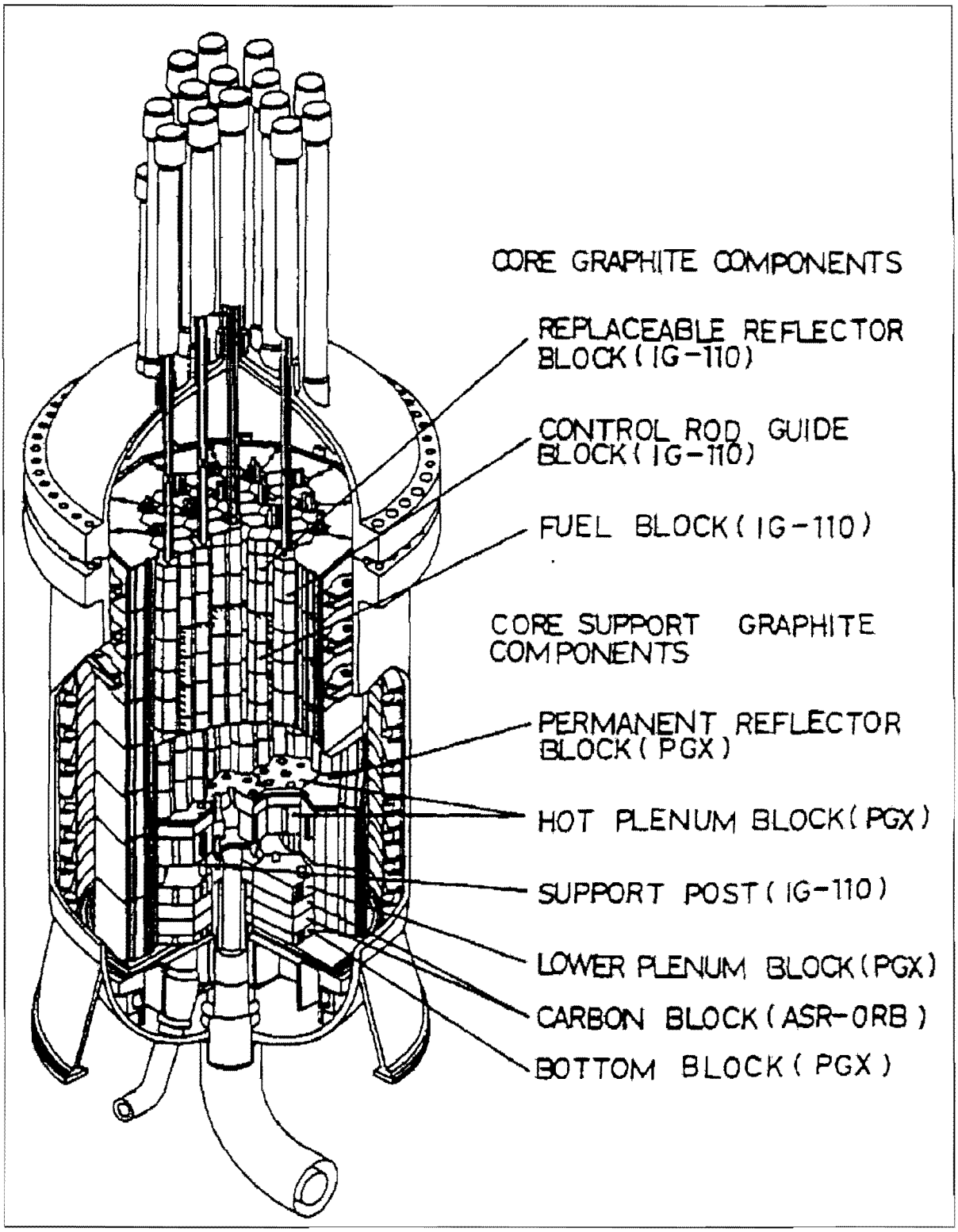


Figure 61: HTTR Core Internals [36]

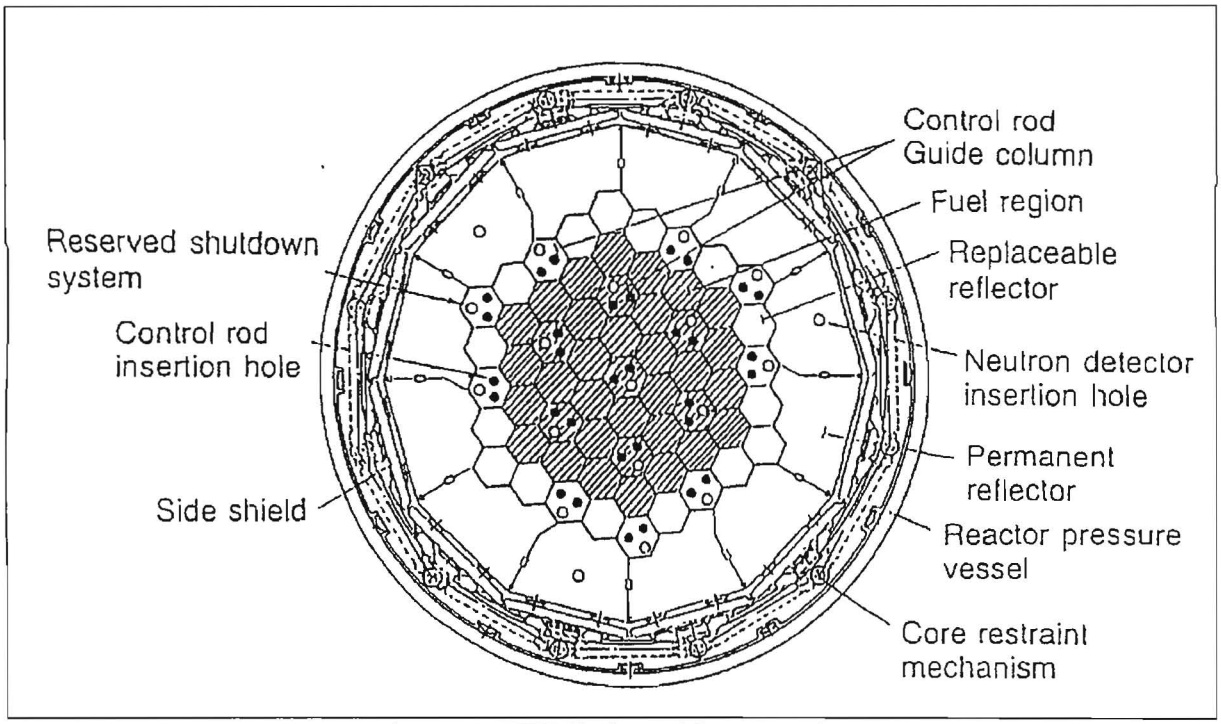


Figure 62: HTTR Horizontal Cross-section [36]

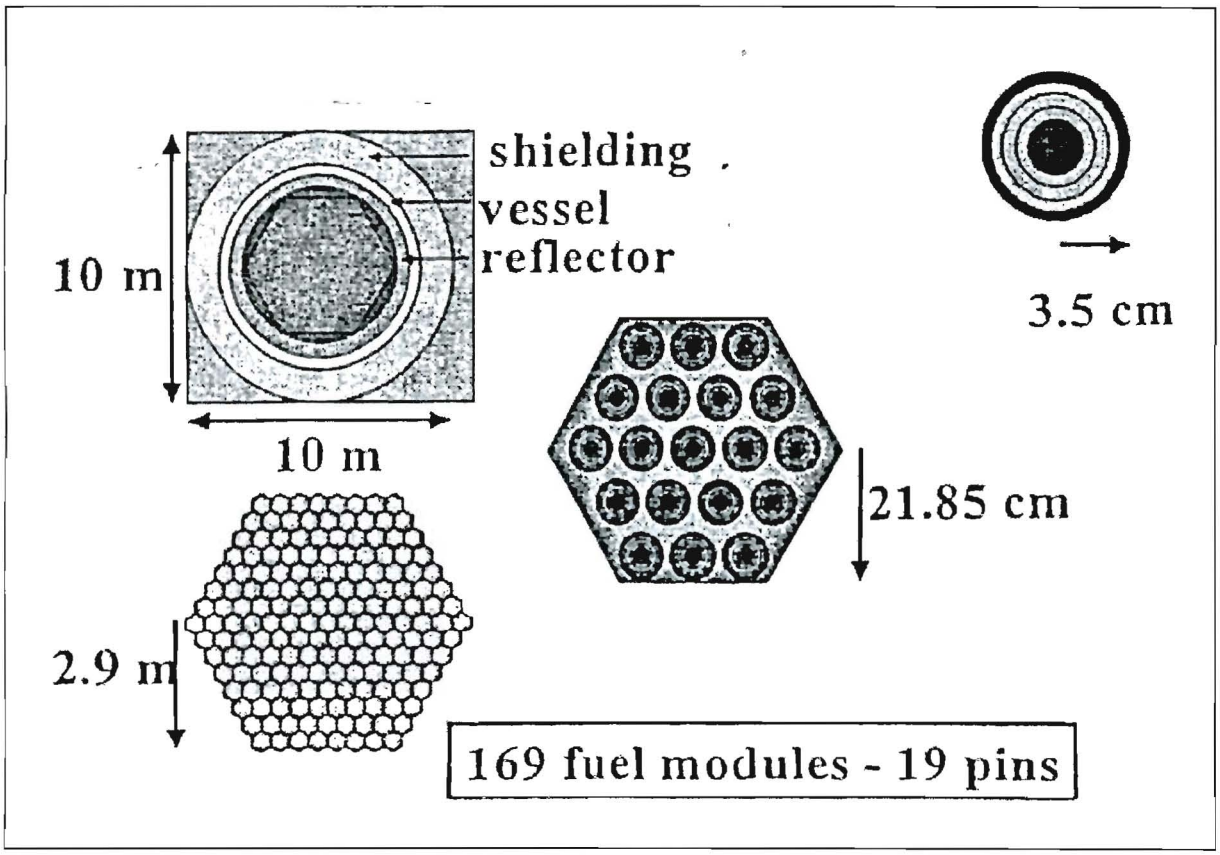


Figure 63: Conceptual Reactor Design by Lobet et al. [48]

11. APPENDIX B: FUEL CALCULATIONS

========== *Fuel Rod calculations* ==========

$L = 0.039$ *length*

$R_r = 0.013$

$R_h = 0.005$

$N_{particles} = 13000$ *number of TRISO particles per fuel rod*

$D_k = 0.0006$ *uranium kernal diameter*

$\rho_{UO2} = 10500$ *[kg/m^3]*

$\rho_G = 1600$ *Graphite density*

$$V_{kernal} = 4 / 3 \cdot \pi \cdot \left[\frac{D_k}{2}\right]^3$$

$$V_U = V_{kernal} \cdot N_{particles}$$

$$V_{rod} = L \cdot \pi \cdot [R_r^2 - R_h^2]$$

$$V_G = V_{rod} - V_U$$

$Mass_G = V_G \cdot \rho_G$ *Graphite mass per fuel element*

$Mass_{UO2} = V_U \cdot \rho_{UO2}$ *Mass of UO2 per fuel element*

$$Mass_U = Mass_{UO2} \cdot \left[\frac{238}{238 + 16 + 16}\right]$$ *Mass Uranium per fuel element*

$Mass = Mass_G + Mass_{UO2}$ *Mass of fuel element*

$$Density = \frac{Mass}{V_{rod}}$$

---------- *cost calcs* ----------

$Cost_{UF6} = 117.5$ *[$/kg] in October 2009*

$Cost_{SWU} = 160$

$Mass_{feed} = 16.8$ *mass of UF6 required per kg of 8% enriched product*

$N_{SWU} = 14.4$

$MaterialCostRatio = 0.32$ *Ratio of final fuel element cost to material cost*

$$Cost_{rate} = \frac{Cost_{UF6} \cdot Mass_{feed} + Cost_{SWU} \cdot N_{SWU}}{MaterialCostRatio}$$

$ for fuel element containing 1kg of enriched Uranium

$Cost_{element} = Cost_{rate} \cdot Mass_U$

$$Cost_{per_ass} = \frac{Cost_{element}}{Mass}$$

---------- *Burnup* ----------

$cal_{years} = 5$ *Goal*

$Power = 75$ *[MW]*

$N_{elements} = 10 \cdot 15 \cdot 33 \cdot 30$ *10 layers; 15 elements per sleeve; 33 sleeves per bloc; 30 fuel blocks at each level*

$Mass_{U,core} = N_{elements} \cdot Mass_U$

$$Burnup = Power \cdot \frac{N_{days}}{\frac{Mass_{U,core}}{1000}}$$

$$N_{years} = \frac{N_{days}}{365}$$

$Energy = Power \cdot 10^6 \cdot 60 \cdot 60 \cdot 24 \cdot N_{days}$ *[Joule]*

$N_{fissions} = 3.1 \times 10^{10} \cdot Energy$

$$N_{Uatoms} = 6.02 \times 10^{23} \cdot \frac{Mass_{U,core}}{0.238}$$

$$FIMA = \frac{N_{fissions}}{N_{Uatoms}} \cdot 100$$

$$cal_{days} = \frac{N_{days}}{0.9}$$ *90% utilisation*

$$cal_{years} = \frac{cal_{days}}{365}$$

SOLUTION

Unit Settings: [kJ]/[C]/[kPa]/[kg]/[degrees]

Burnup = 60960
cal_{years} = 5
$Cost_{per,mass}$ = 4403
$Cost_{SWU}$ = 160
Density = 2342
Energy = 1.064E+16
L = 0.039
$Mass_{feed}$ = 16.8
$Mass_U$ = 0.01361
$Mass_{U,core}$ = 2021
N_{days} = 1643
$N_{fissions}$ = 3.299E+26
N_{SWU} = 14.4
N_{years} = 4.5
ρ_G = 1600
R_f = 0.013
V_G = 0.00001617
V_{rod} = 0.00001764

cal_{days} = 1825
$Cost_{element}$ = 181.9
$Cost_{fuel}$ = 13369
$Cost_{UF6}$ = 117.5
D_k = 0.0006
FIMA = 6.455
Mass = 0.04131
$Mass_G$ = 0.02588
$Mass_{UO2}$ = 0.01544
MaterialCostRatio = 0.32
$N_{elements}$ = 148500
$N_{particles}$ = 13000
N_{Uatoms} = 5.111E+27
Power = 75
ρ_{UO2} = 10500
R_h = 0.005
V_{kernal} = 1.131E-10
V_U = 0.00000147

No unit problems were detected.

12. APPENDIX C: RANKINE CYCLE EES MODEL

++++++++++++++++Rankine Ship Engine++++++++++++++++++

++

================Design assumptions==================

$D = 2$ *Core diameter*

$H = 6$ *Core Height*

$\dot{Q}_R = 8 \times 10^7$ *Reactor Power [Watt]*

$ROT = 700$ *Reactor Outlet Temperature*

$RIT = 250$ *Reactor Inlet Temperature*

$T_{Steam} = 565$ *Maximum steam temperature*

$P_{prim} = 6000$ *kPa*

$P_{steam} = 16500$ *kPa*

$P_{tap} = 1000$ *guess*

$P_{condens} = 10$ *[kPa] controlled by vacuum pump*

$x_{turb} = 0.9$ *Quality of steam leaving the turbine*

$dPress_{SG,Prim} = 40$ *[kPa] Design decision*

$dPress_{SG,Sec} = 2000$ *[kPa] Design decision*

$\eta_{Blow,is} = 0.9$ *isentropic efficiency*

$\eta_{Blow,el} = 0.95$ *electrical efficiency*

$\eta_{T1} = 0.95$ *isentropic efficiency*

$\eta_{pump} = 0.9$ *pump efficiency*

$\eta_{Gen} = 0.98$ *Generator Efficiency*

$\eta_{Motor} = 0.96$ *Propulsion motor efficiency*

$\eta_{Gear} = 0.9$ *Propulsion gearbox efficiency*

alpha = 0.2;

Turbine bleed-off fraction

$speed = 2$

==========Reactor==========

$T_1 = ROT$ *deg C*

$T_3 = RIT$ *deg C*

$C_{P,H} = \mathbf{Cp}\left[\text{'Helium'}, T = Tav, P = P_3\right]$

$\dot{Q}_R = mdot_{prim} \cdot C_{P,H} \cdot [T_1 - T_3]$

---------Pebble Bed Pressure drop according to KTA equation--------

$dp = 0.06$ *Pebble diameter [m]*

$eps = 0.39$ *Pebble Bed Porosity*

$Tav = 0.5 \cdot [RIT + ROT]$

$TavK = Tav + 273$

$$\rho = 48.14 \cdot \left[\frac{\frac{P_3}{101 \cdot TavK}}{1 + 0.446 \cdot \frac{\frac{P_3}{101}}{TavK^{1.2}}} \right]$$ *KTA 3102.1, 1988 P in bar*

He dynamic viscosity

$\mu = 3.674 \times 10^{-7} \cdot TavK^{0.7}$ *KTA 3102.1, 1988*

flow velocity

$$FA = \pi \cdot \left[\frac{D}{2}\right]^2$$

$$\dot{Vol} = \frac{mdot_{prim}}{\rho}$$ m^3/s

$$vel = \frac{\dot{Vol}}{FA}$$ m/s

$$Re = \rho \cdot vel \cdot \frac{dp}{\mu}$$

$$validity = \frac{Re}{1 - eps}$$ *Needs to be between 1 and 10E5*

$$\psi_k = 320 \cdot \left[\frac{1 - eps}{Re}\right] + 6 \cdot \frac{[1 - eps]^{0.1}}{Re^{0.1}}$$ *KTA*

$$dPress_k = H \cdot \psi_k \cdot \left[\frac{1 - eps}{eps \cdot eps \cdot eps \cdot dp}\right] \cdot 0.5 \cdot \rho \cdot vel \cdot vel$$ *[Pa]*

--------Block Reactor Pressure Drop----------

$dPress_B = 28$ *[kPa] calculated from Darcy-Weisbach*

$P_1 = P_3 - dPress_B$

$h_1 = \mathbf{h}['Helium', T = T_1, P = P_1]$

$s_1 = \mathbf{s}['Helium', T = T_1, P = P_1]$

$Qdot_{chk} = mdot_{prim} \cdot [h_1 - h_2]$

==========Blower==========

$P_3 = P_{pnm}$

$C_P = \mathbf{Cp}\left['Helium', T = RIT, P = P_3\right]$

$C_V = \mathbf{Cv}\left['Helium', T = RIT, P = P_3\right]$

$\gamma = \frac{C_P}{C_V}$

$T_3 - T_2 = DelT_{Blow}$

$$DelT_{Blow} = \left[\frac{T_2 + 273.16}{\eta_{Blow,is}}\right] \cdot \left[\left(\frac{P_3}{P_2}\right)^{\left(\frac{\gamma - 1}{\gamma}\right)} - 1\right]$$

$h_2 = \mathbf{h}\left['Helium', T = T_2, P = P_2\right]$

$s_2 = \mathbf{s}\left['Helium', T = T_2, P = P_2\right]$

$h_3 = \mathbf{h}\left['Helium', T = T_3, P = P_3\right]$

$s_3 = \mathbf{s}\left['Helium', T = T_3, P = P_3\right]$

$W_{Blow} = mdot_{pnm} \cdot \left[h_3 - h_2\right]$

[W]

$$W_{Blow,el} = \frac{W_{Blow}}{\eta_{Blow,el}}$$

$Pressure_{nse} = P_3 - P_2$ *[bar]*

$$volume_{flow} = \frac{mdot_{pnm}}{\rho\left['Helium', T = T_2, P = P_2\right]}$$

==========*Steam Gen*==========

$P_2 = P_1 - dPress_{SG,Pnm}$

$\dot{Q}_{SG,pnm} = mdot_{pnm} \cdot \left[h_1 - h_2\right]$

$\dot{Q}_{SG,pnm} = \dot{Q}_{SG,sec}$

$T_4 = T_{Steam}$

$P_4 = P_{steam}$

$h_4 = \mathbf{h}\left['Water', T = T_4, P = P_4\right]$

$s_4 = \mathbf{s}\left['Water', T = T_4, P = P_4\right]$

$P_4 = P_9 - dPress_{SG,Sec}$

$CP_{SG} = \mathbf{Cp}\left['Water', T = 0.5 \cdot (T_9 + T_4), P = 0.5 \cdot (P_9 + P_4)\right]$

$\dot{Q}_{SG,sec} = mdot_{sec} \cdot \left[h_4 - h_9\right]$

$$\dot{Vol}_{SG,in} = \frac{mdot_{sec}}{\rho['Water', T = T_9, P = P_9]}$$

$$A_{in} = \frac{\dot{Vol}_{SG,in}}{speed}$$

$$A_{tube} = \pi \cdot 0.001^2$$

$$N_{tubes} = \frac{A_{in}}{A_{tube}}$$

$$Length = \frac{785}{N_{tubes} \cdot 2 \cdot \pi \cdot 0.001}$$

$$\dot{Vol}_{SG,out} = \frac{mdot_{sec}}{\rho['Steam', T = T_4, P = P_4]}$$

$$A_{out} = \frac{\dot{Vol}_{SG,out}}{speed}$$

==========*Turbine 1*==========

$$P_{10} = P_{top}$$

$h10_{is} = h['Water', P = P_{10}, s = s_4]$ *assume turbine is isentropic*

$\eta_{T1} = \frac{h_4 - h_{10}}{h_4 - h10_{is}}$ *Turbine isentropic efficiency*

$$T_{10} = T['Water', P = P_{10}, h = h_{10}]$$

$$W_{turb1} = mdot_{sec} \cdot [h_4 - h_{10}]$$

$$mdot_{bleed} = \alpha \cdot mdot_{sec}$$

$$mdot_{condens} = [1 - \alpha] \cdot mdot_{sec}$$

==========*Turbine 2*==========

$$s_{10} = s['Water', T = T_{10}, P = P_{10}]$$

$$x_5 = x_{turb}$$

$$s_5 = s['Water', x = x_5, P = P_5]$$

$$h_5 = h['Water', P = P_5, s = s_5]$$

$h5_{is} = h['Water', P = P_5, s = s_4]$ *assume turbine is isentropic*

$T_5 = T['Water', P = P_5, s = s_5]$ *Turbine isentropic efficiency*

$$\eta_T = \frac{h_{10} - h_5}{h_{10} - h5_{is}}$$

$$W_{turb2} = mdot_{condens} \cdot [h_{10} - h_5]$$

==========*Condenser*==========

$$x_6 = 0$$

Saturated water

$P_5 = P_{condens}$

$P_6 = P_5$

$s_6 = s['Water', x = 0, P = P_6]$

$T_6 = T['Water', s = s_6, P = P_6]$

$h_6 = h['Water', x = 0, P = P_6]$

$\dot{Q}_{condens} = mdot_{condens} \cdot [h_5 - h_6]$

==========Pump1==========

$P_7 = P_{10}$

$Vol_{in} = v['Steam_{IAPWS}', P = P_6, h = h_6]$

$Vol_{out} = v['Water', P = P_7, h = h_7]$

$h_7 = h_6 + Vol_{in} \cdot 1000 \cdot \left[\frac{P_7 - P_6}{\eta_{pump}}\right]$

P_x in kPa needs to be converted into Pa

$s_7 = s['Water', h = h_7, P = P_7]$

$T_7 = T['Water', s = s_7, P = P_7]$

$W_{pump1} = mdot_{condens} \cdot [h_7 - h_6]$

==========Feedwater heater==========

$P_8 = P_7$

$h_8 = \alpha \cdot h_{10} + [1 - \alpha] \cdot h_7$

$T_8 = T['Water', h = h_8, P = P_8]$

$s_8 = s['Water', h = h_8, P = P_8]$

==========Pump2==========

$Vol_{in2} = v['Water', P = P_8, h = h_8]$

$Vol_{out2} = v['Water', P = P_9, h = h_9]$

$h_9 = h_8 + Vol_{in2} \cdot 1000 \cdot \left[\frac{P_9 - P_8}{\eta_{pump}}\right]$

P_x in kPa needs to be converted into Pa

$s_9 = s['Water', h = h_9, P = P_9]$

$T_9 = T['Water', s = s_9, P = P_9]$

$W_{pump2} = mdot_{sec} \cdot [h_9 - h_8]$

========== System ==========

$$W_{turb} = W_{turb1} + W_{turb2}$$

$$W_{out,gen} = [W_{turb1} + W_{turb2}] \cdot \eta_{Gen}$$

$$W_{prop} = [W_{out,gen} - (W_{pump1} + W_{pump2} + W_{Blow,el})] \cdot \eta_{Motor} \cdot \eta_{Gear}$$

$$\eta_{cycle} = \frac{W_{turb} - [W_{pump1} + W_{pump2} + W_{Blow,el}]}{\dot{Q}_R}$$

$$\eta_{total} = \frac{W_{prop}}{\dot{Q}_R}$$

========*Consistency check*==========

$$err = \dot{Q}_{SG,sec} + W_{pump1} + W_{pump2} - [W_{turb1} + W_{turb2}] - \dot{Q}_{condens}$$

$$percent_{err} = 100 \cdot \frac{err}{\dot{Q}_{SG,sec}}$$

SOLUTION

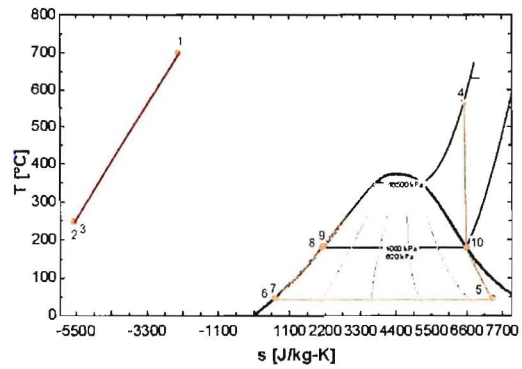
Unit Settings: [J]/[C]/[kPa]/[kg]/[degrees]

Maximization of eta_cycle(alpha) 40 iterations: Quadratic Approximations method

α = 0.2203	A_{in} = 0.01671
A_{out} = 0.3168	A_{tube} = 0.000003142
CP_{SG} = 7602	C_P = 5188
$C_{P,H}$ = 5189	C_v = 3120
D = 2	$DelT_{Blow}$ = 2.633
dp = 0.06	$dPress_B$ = 28
$dPress_k$ = 34918	$dPress_{SG,Prim}$ = 40
$dPress_{SG,Sec}$ = 2000	eps = 0.39
err = 0.002081	$\eta_{Blow,el}$ = 0.95
$\eta_{Blow,is}$ = 0.9	η_{cycle} = 0.3718
η_{Gear} = 0.9	η_{Gen} = 0.98
η_{Motor} = 0.96	η_{pump} = 0.9
η_T = 0.6059	η_{T1} = 0.95
η_{total} = 0.3145	FA = 3.142
γ = 1.663	H = 6
$h10_{is}$ = 2.744E+06	$h5_{is}$ = 2.061E+06
Length = 23.49	$mdot_{bleed}$ = 6.592
$mdot_{condens}$ = 23.33	$mdot_{prim}$ = 34.26
$mdot_{sec}$ = 29.92	μ = 0.00003775
N_{tubes} = 5319	$percent_{err}$ = 2.586E-09
$Pressure_{nse}$ = 68	ψ_K = 2.163
$P_{condens}$ = 10	P_{prim} = 6000
P_{steam} = 16500	P_{tap} = 1000
$Qdot_{chk}$ = 8.047E+07	$\dot{Q}_{condens}$ = 5.023E+07
$\dot{Q}_R$ = 8.000E+07	$\dot{Q}_{SG,prim}$ = 8.047E+07
$\dot{Q}_{SG,sec}$ = 8.047E+07	Re = 17334
ρ = 3.788	RIT = 250
ROT = 700	speed = 2
Tav = 475	TavK = 748
T_{Steam} = 565	validity = 28416
vel = 2.879	$volume_{flow}$ = 6.335
$\dot{V}ol$ = 9.045	$\dot{V}ol_{SG,in}$ = 0.03342
$\dot{V}ol_{SG,out}$ = 0.6336	Vol_{in} = 0.00101

Vol_{in2} = 0.001127
Vol_{out2} = 0.001117
$W_{Blow,ol}$ = 500174
W_{prop} = 2.516E+07
W_{pump2} = 655850
W_{turb1} = 2.076E+07
x_{turb} = 0.9

Vol_{out} = 0.00101
W_{Blow} = 475165
$W_{out,gon}$ = 3.031E+07
W_{pump1} = 25928
W_{turb} = 3.093E+07
W_{turb2} = 1.016E+07



13. APPENDIX D: SHIP REACTOR CALCULATIONS

EES Script

```
"============Calculations for ship reactor==============="

Power = 80E6*1.2;
height = 6.12;                                  "30 of the 37 Blocks contains fuel"
N_blocks = 30;                                  "No of blocks at the same height"
N_sleeves = 33;                                 "No of sleeves in each block"
"P_J = 4;"
P=6
T_i = 250
T_o=700
T_av = 0.5*(T_o+T_i)


k_f = 2.5                                       "[W/m.K]"
k_c = 30;

D_sleeve=R_sleeve*2;
R_sleeve=20.5E-3;                               "graphite sleeve radius"
D_GrafRod = R_GrafRod*2;
R_GrafRod = 17E-3;                              "radius of graphite rod around fuel"
D_fuel = R_fuel*2;
R_fuel = 13E-3                                  "fuel outer radius"
R_hole = 5E-3;                                  "Hole in fuel centre"
D_hole = R_hole*2;

m_dot = 34                                      "coolant mass flow rate"
rho=Density(Helium,T=T_av,P=P)
mu =Viscosity(Helium,T=T_av,P=P)
mu_KTA = (3.674E-7)*(T_av+273)^(0.7)            "KTA Equation"
V_dot = m_dot/rho

Flowarea_tube=pi*(D_sleeve^2 - D_GrafRod^2)/4
FlowArea = Flowarea_tube*N_sleeves*N_blocks
vel=V_dot/FlowArea

"---------- determine convection coeff----------"
 Pr=Prandtl(Helium,T=T_av,P=P);
k_HE = Conductivity(Helium,T=T_av,P=P)
Dia_hyd = D_sleeve-D_GrafRod;
Re=rho*vel*Dia_hyd/mu;
Nus=0.023*(Pr^0.4)*(Re^0.8)
Nus2=0.018*(Pr^0.4)*(Re^0.8)*(D_GrafRod/D_sleeve)^(-0.16)
h_conv = Nus*k_HE/Dia_hyd


"---------- Power density----------"
Vol_fuel = height*N_blocks*N_sleeves*pi*(R_fuel^2 - R_hole^2)
PD_F = Power/Vol_fuel
Heat_rate_lin = Power/height;
circ_fuel = N_blocks*N_sleeves*2*pi*R_fuel      "combined circumference of all fuel"
 Heat_rate_lin = h_conv * circ_fuel * (T_surf - T_o);
LD=height/Dia_hyd

"----------Heat Transfer----------"
T_delta_fuel = (PD_F/(2*k_f))*((R_fuel^2 - R_hole^2)/2 + (R_hole^2)*(ln(R_hole) - ln(R_fuel)))
T_delta_clad = PD_F*(R_fuel^2 - R_hole^2)*(ln(R_GrafRod) - ln(R_fuel))/(2*k_c)
T_delta_conv=PD_F*(R_fuel^2 - R_hole^2)/(2*h_conv*R_GrafRod)
T_f_max = T_o+ T_delta_conv + T_delta_clad + T_delta_fuel
"----------total core----------"
A_graptubes = N_blocks*N_sleeves*2*pi*R_GrafRod*height;
```

```
T_del = Power/(h_conv*A_graptubes)

"---------- Pressure drop----------"
f_f=0.079*Re^(-0.25)
```

MATLAB Script

```
%Ship Reactor: Fuel temperature

DR=0.1E-3;
height = 6.12;  %30 of the 37 Blocks contains fuel"
N_blocks = 30;  %No of blocks at the same height"
N_sleeves = 33; %No of sleeves in each block"
T_i = 250;
T_o=700;



k_f = 2.5;  %[W/m.K]"
k_c = 30;
h_conv = 1857;

R_sleeve=20.5E-3;   %graphite sleeve radius"
R_GrafRod = 17E-3;  %radius of graphite rod around fuel"
R_fuel = 13E-3; %fuel outer radius"
R_hole = 5E-3;  %Hole in fuel centre"

PD_F = 29.19E6*1.2;
T_delta_fuel = (PD_F/(2*k_f))*((R_fuel^2 - R_hole^2)/2 +
(R_hole^2)*(log(R_hole) - log(R_fuel)))
T_delta_clad = PD_F*(R_fuel^2 - R_hole^2)*(log(R_GrafRod) -
log(R_fuel))/(2*k_c)
T_delta_conv=PD_F*(R_fuel^2 - R_hole^2)/(2*h_conv*R_GrafRod)
T_f_max = T_o+ T_delta_conv + T_delta_clad + T_delta_fuel
TFO = T_o + T_delta_conv + T_delta_clad;
Rc=13E-3:DR:17E-3;
Tc=zeros(size(Rc));
i=0;
for r=13E-3:DR:17E-3;
    i=i+1;
    Tc(i) = TFO -PD_F*(R_fuel^2 - R_hole^2)*(log(r) - log(R_fuel))/(2*k_c);
end
plot(Rc,Tc),grid
Rf=5E-3:DR:13E-3;
Tf=zeros(size(Rf));
i=0;
for r=5E-3:DR:13E-3;
    i=i+1;
    Tf(i) = TFO + (PD_F/(2*k_f))*((R_fuel^2 - r.^2)/2 + (R_hole^2)*(log(r)
- log(R_fuel)));
end
plot(Rf*1000,Tf,Rc*1000,Tc),grid
axis([0;17;700;1150])
xlabel('Radius [mm]')
ylabel('temperature [deg C]')
title('Fuel temperature profile')
```

14. APPENDIX E: CURRENCY EXCHANGE RATES USED IN THIS STUDY

The following currency conversion factors were used throughout this study:

Table 37: Currency conversion factors

Currency	Relative Value
SA Rand [R]	8
USA Dollar [$]	1
Euro [€]	$\frac{1}{1.2}$
UK Pound [£]	$\frac{1}{1.5}$
Australian Dollar [AU$]	$\frac{1}{0.93}$