ASSESSMENT OF PASSIVE DECAY HEAT REMOVAL IN THE GENERAL ATOMIC MODULAR HELIUM REACTOR

A Thesis

by

FRANCOIS GUILHEM COCHEME

Submitted to the Office of Graduate Studies of
Texas A&M University
in partial fulfillment of the requirements for the degree of

MASTER OF SCIENCE

December 2004

Major Subject: Nuclear Engineering

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December 2004

Major Subject: Nuclear Engineering

ABSTRACT

Assessment of Passive Decay Heat Removal
in the General Atomic Modular Helium Reactor (MHR). (December 2004)

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The purpose of this report is to present the results of the study and analysis of loss-of-coolant and loss-of-flow simulations performed on the Modular Helium Reactor developed by General Atomics using the thermal-hydraulics code RELAP5-3D/ATHENA.

The MHR is a high temperature gas cooled reactor. It is a prismatic core concept for New Generation Nuclear Plant (NGNP). Very few reactors of that kind have been designed in the past. Furthermore, the MHR is supposed to be a highly passively safe concept. So there are high needs for numerical simulations in order to confirm the design.

The project is dedicated to the assessment of the passive decay heat capabilities of the reactor under abnormal transient conditions. To comply with the requirements of the NGNP, fuel and structural temperatures must be kept under design safety limits under any circumstances.

During the project, the MHR has been investigated: first under steady-state conditions and then under transient settings. The project confirms that satisfying passive decay heat removal by means of natural heat transfer mechanisms (convection, conduction and radiation) occurs.

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I would like to thank my advisors Dr. Peddicord and Dr. Hassan for their support during the course of my project. I am glad they offered me the opportunity to study these last two years in Texas A&M University. This experience will be a great source of inspiration for my future. I also would like to thank Dr. Marlow from the Nuclear Engineering Department and Dr. Annamalai from the Mechanical Engineering Department for serving on my committee and for providing insights into my research.

Next, I would like to thank the students and staff form the Nuclear Department.

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Finally on a personal note I would like to thank my parents and family. They are the ones that encouraged me to attend Texas A&M for my Masters, and to whom I dedicate this thesis.

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INTRODUCTION

The work presented thereafter has been performed within the framework of the NERI project “Hydrogen Production Plant Using the Modular Helium Reactor”. This project is developed under the U.S. Department of Energy Nuclear Energy Research Initiative. General Atomics is the lead organization for this project and is supported by Idaho National Engineering and Environmental Laboratory, Texas A&M University, and Entergy Nuclear Inc.

Goals / objectives

The purpose of the project is to study and analyze the decay heat removal capabilities of the Modular Helium Reactor (MHR) concept developed by General Atomics. The MHR is an advanced concept reactor that is being designed to provide high reliability and increased safety characteristics as compared to existing power plants. It is a direct descendant of prior High Temperature Gas Reactor research programs that began in the 1950’s. The concept features a helium gas cooled reactor with a graphite moderated prismatic core that contains TRISO fuel.

This study evaluates the passive decay heat removal capabilities of the MHR under abnormal conditions, more specifically under loss-of-coolant and loss-of-flow accident conditions. Ensuring the fuel and structural temperatures under given safety limits is the largest technical concern of this analysis. In order to prevent core reactor damage and avoid release of radio-nuclides, heat removal from the core should adequately match heat generation. During accident conditions, if the reactor is successfully scrammed, heat generation can be reduced at best to decay heat. Therefore, the only things one can address are the heat removal means.

This thesis follows the style and format of *Nuclear Technology.*

The greatest challenge of the MHR project is to prove that the reactor's structural integrity would be ensured at any time during accident. That is, the decay heat removal can be performed with only passive means: natural convection, conduction and radiation transfer.

Background

The world's current situation concerning energy supplies and environment protection is disturbing. The ever increasing consumption of primary energy is at an all-time high, and it is unlikely this trend will change soon. In the past, besides some crisis, fossil fuel production has always provided for the world energy needs, but currently, this production has shown signs of weakness. The findings of new resources are declining, the new mining spots are harder to exploit, and the quality of natural resources is decreasing. Always object of recurrent heated discussions, the fossil fuel shortage is becoming a key political issue.

Less prone to discussion the energy dependence problem is becoming essential to developed countries. Developed countries are the biggest consumers of fossil fuel, yet they do not control the principal resources.

In addition, greenhouse gas emissions, suspected to be a significant global warming factor are becoming more of a public concern.

Fossil fuel shortage, energy dependence and global warming issues are worsened by the emergence of hugely populated countries such as China, India and Brazil, which desire to catch up with western living standards. Thus, they will have tremendous energy needs.

The only acceptable solution is to reduce our fossil fuel consumption. Energy saving policy, alternative and sustainable energies and shifting to hydrogen economy is a set of measures expected to solve the problems.

The hydrogen economy relies on nuclear reactors for the production of electricity and hydrogen for the transportation and chemical industries. It implies developments of new technologies with respect to new requirements such as passive safety, proliferation resistance, fuel utilization and performance; and high temperature output for increased thermal efficiency and hydrogen production.

Technical status of the question

Under abnormal conditions, such as loss of coolant, the main concern for nuclear reactor operators is to preserve the integrity of the core. During a loss of coolant accident, the heat generated by the fuel is not removed from the core as efficiently as during normal operation conditions. Therefore, the core temperature may increase up to fuel melting point, leading to core damage.

Currently, existing power plants are equipped with safety devices that avoid major accidents. The nuclear industry has reached a status where accidents occurring in power plants have no significant impact on public health or the environment. These accidents only affect the economics of the plants because they may involve equipment replacement and loss of production.

However, public safety issues and public opinion are still major concerns to the industry. The next generation of nuclear power plants will have to assure an even greater level of safety. Increasing the safety level of power plants is also an economic concern. It can be economically beneficial for electricity production companies to run safer reactors. It increases their margins by reducing the off the grid period; thus increasing the availability.

The MHR project is incorporated within a more global plan: the rebirth of the nuclear industry, that is, the renewal of the nuclear power plant fleet, which may be tied to the shift towards “hydrogen economy”. For instance, the development of the MHR concept can be understood as a step toward the implementation of the generation IV project. Generation IV goals include the development of new reactors concepts that represent significant advances in economics, safety, reliability, proliferation resistance and waste minimization. It involves the development of new technologies such as high temperatures materials, epithermal and fast-neutrons reactors, gas, lead or molten salt coolant reactors, super-critical water, hydrogen production, as well as other technology.

The development of the MHR would permit early application of some required Generation IV technologies: mainly high temperature gas coolant and hydrogen production. The MHR satisfies the Generation IV goals of improved safety, economics, proliferation resistance, and environmental protection. Furthermore, the MHR can also produce hydrogen [1].

Procedure / methods

The study of the MHR is addressed using the thermal-hydraulic code RELAP5-3D, Athena version. RELAP has been developed by the Idaho National Engineering and Environmental Laboratory (INEEL) for the purpose of analyzing transients and accidents, including both large and small break loss of coolant accidents (LOCA) in Light Water Reactors (LWR). It is based on a non-homogeneous and non-equilibrium model for two-phase flow. The Athena version is the first of the RELAP5 series to provide gas-cooled reactor competence beside other advanced capabilities and improved computational tools.

To accomplish the objectives the following procedure were executed.

- The first step consisted of the core study under normal “steady-state” conditions. During this phase, the MHR model provided by INEEL has been checked, tested and the results have been compared to available information. This part was useful to understand the simplified model of MHR run with RELAP.
- The second step consisted of performing various accident scenarios as LOCA under normal pressure and depression conditions. The runs were set up so that active heat removal devices were shut down. Therefore only passive means (natural convection, radiation transfer and conduction) removed the decay heat.
- The last step is focused on the discussion of the results. The most important concern about the results given by the RELAP simulations is the level of confidence.

BACKGROUND

World energy perspective and energy policy

Energy is essential to human activity. This assertion is all the more true in our modern societies. Energy is involved everywhere, all the time, under many different forms: electricity for household and petrol for transportation for instances. The primary sources of energy are of two kinds: fossil fuels (coal, oil, natural gas, nuclear) and renewable energies (solar photovoltaic, wind power, biomass, hydraulic, geothermic).

It is worth noticing that electricity is not a source of energy. It is produced from a primary source of energy. It is a vector. It is an efficient mean to transport energy, conversely it is ineffective to store.

Energy is intimately linked to the economics of our societies. Affordable, accessible and abundant sources of energy are essential for economic growth. A country has to secure its source of energy to insure the well being of its citizens.

The latest world events have made re-appear the problem of energy supply in the developed countries. War in the Middle East, political instabilities in Russia (Ioukos), economic troubles in Latin America (Venezuela) combined all together have increased the gas prices to consecutive all time records rising concerns about economic recovery.

Developed countries rely very strongly on their energy supply. Numerous publications state that in the next decades, the world will face major challenges concerning energy supplies [2]. The actual world energy trade system will have to undergo drastic changes if the level of economic development is to increase all over the world. The major forecasted issues that will have to be dealt with are the reduction of the fossil fuel reserves, the emergence of very populated countries such as China and India, global climate changes. What if China and India which represent a third of the world population reach the living standards of the USA and West European countries? Can we refuse their right to reach the same level of development? The goal of this paper is not to answer this kind of concerns. However, these topics should be kept in mind when developing future domestic and worldwide energy policies.

The objective of the next paragraphs is to describe the current state of the world energy resources as well as their utilization ratio. In the light of the figures presented, one will then tackle the economic and environmental problems which result from that situation.

Reserves

The existing amount of world primary energy sources is a much debated topic. For political, economic and technical reasons the quantity of fossil fuels remaining in the world is difficult to evaluate.

It is in the interest of exportating countries to minimize as much as possible their claimed reserves in order to keep the prices at a high level. However, they have to keep their proved/declared reserves at a reasonable quantity to avoid conflict with the large-scale consumer countries. These considerations may explain why, to date, there has always been a 30-40 year proven reserve balance between production and consumption [3].

Technical issues also make the evaluation of fossil fuel reserves difficult. For that reason, the reserves are divided into two categories: the proven reserves and the assumed additional reserves also called ultimate reserves. The later are the resources that one knows exist or have a strong probability to exist, but for which the extraction is still debated; the reserves are technologically inaccessible or it is not economically relevant [4].

The next figures present the current state of fossil fuels resources: Oil, Natural Gas, Coal and Uranium. These figures illustrate the primary energy distribution significant disparity in the world. According to Figure 1, 67% of the world oil is located in six Middle-East countries: Saudi Arabia, Iraq, Iran, Kuwait, United Arab Emirates and Libya. According to Figure 2, three countries (Russia, Iran and Qatar), share more than half (57%) of the world natural gas reserves, each one holding over 10 %.

According to Figure 3, two countries, USA and Russia, possess almost half (48.5%) of the world coal reserves. If one adds China, Australia, India and Germany that's a group of six countries holding more than 83% of the reserves.

The same kind of conclusion can be drawn about the world reserves of Uranium. According to Figure 4, three countries (Australia, Kazakhstan and Canada), possess more than half (58%) of the whole reserves. However, one can see that the Uranium reserves are better distributed around the world than other resources.

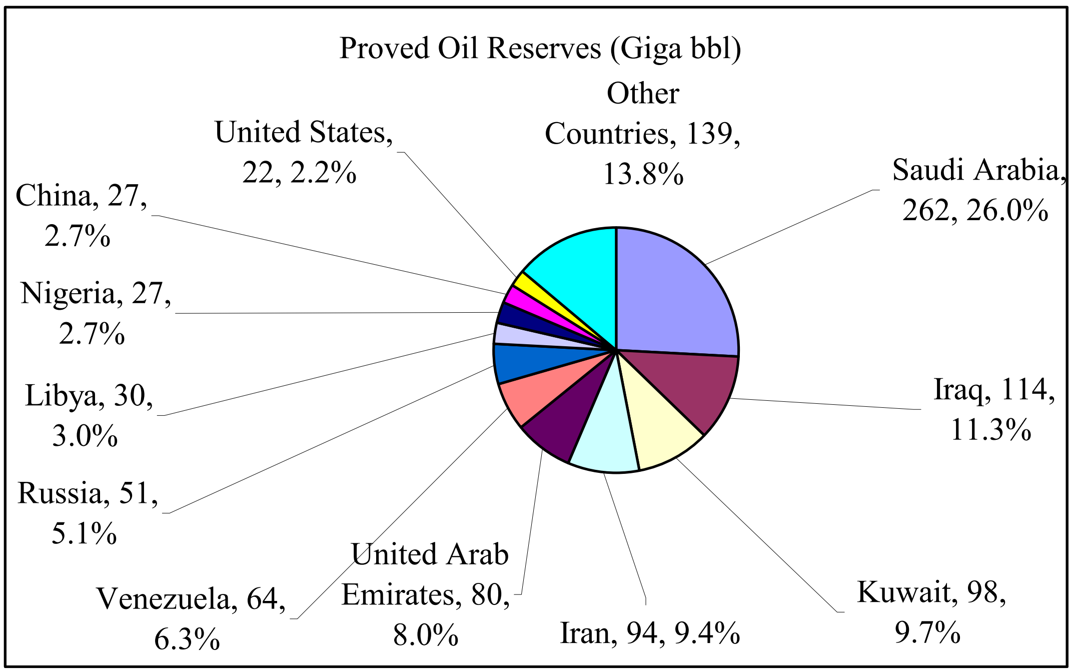


Figure 1: Oil world proved reserves in billions of barrels [5]

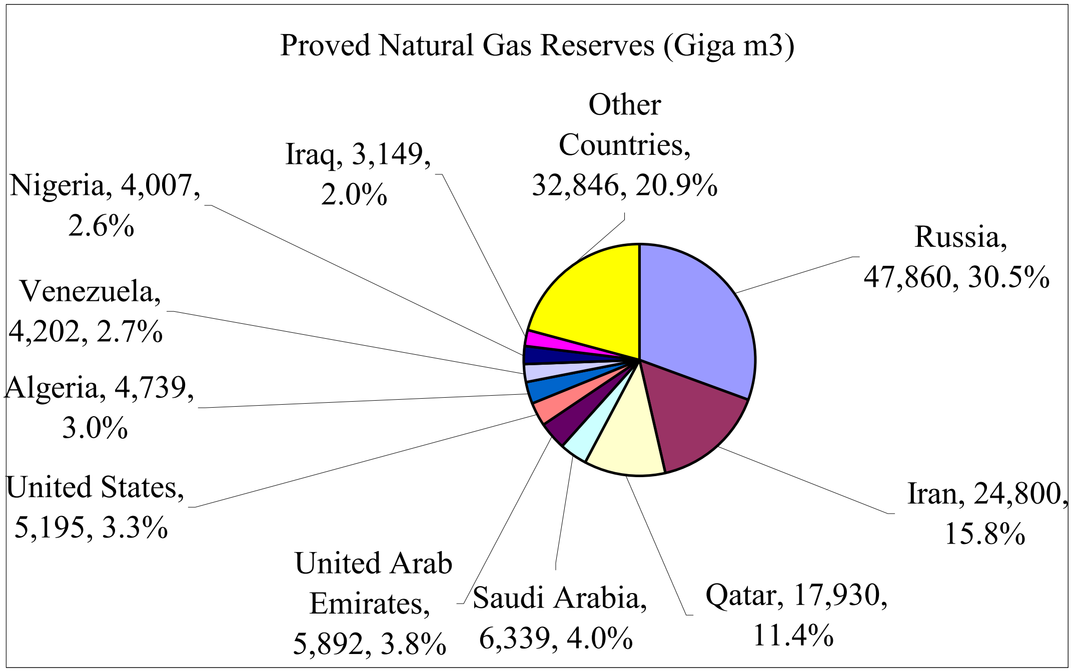


Figure 2: Natural gas world proved reserves in billions cubic meters [5]

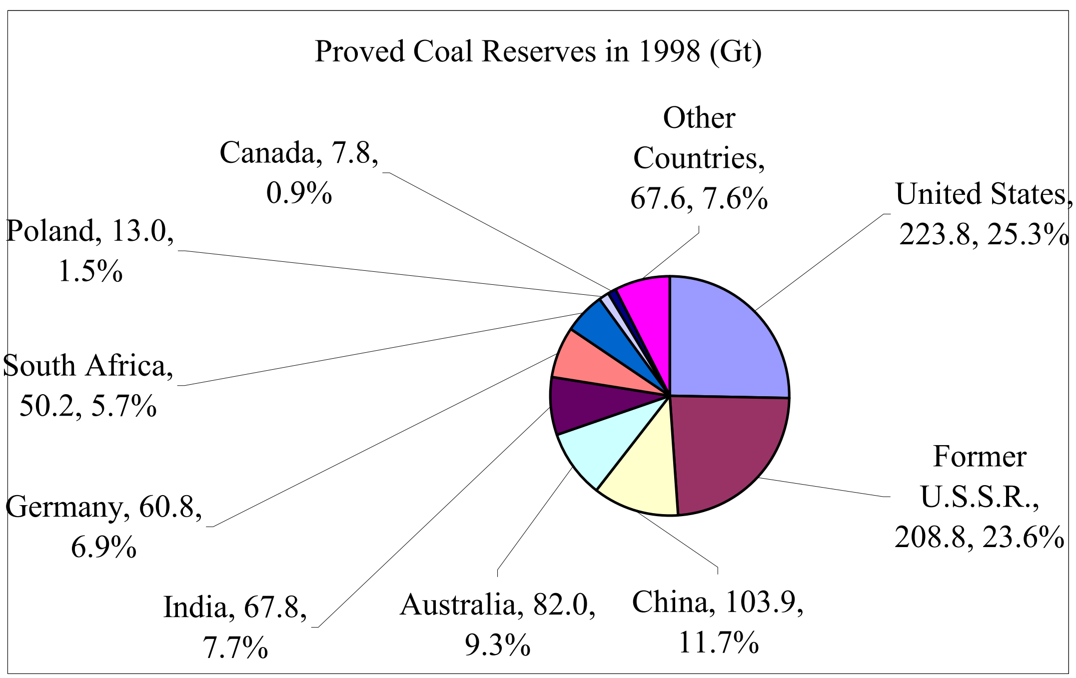


Figure 3: World proved coal reserves in billions tons [6]

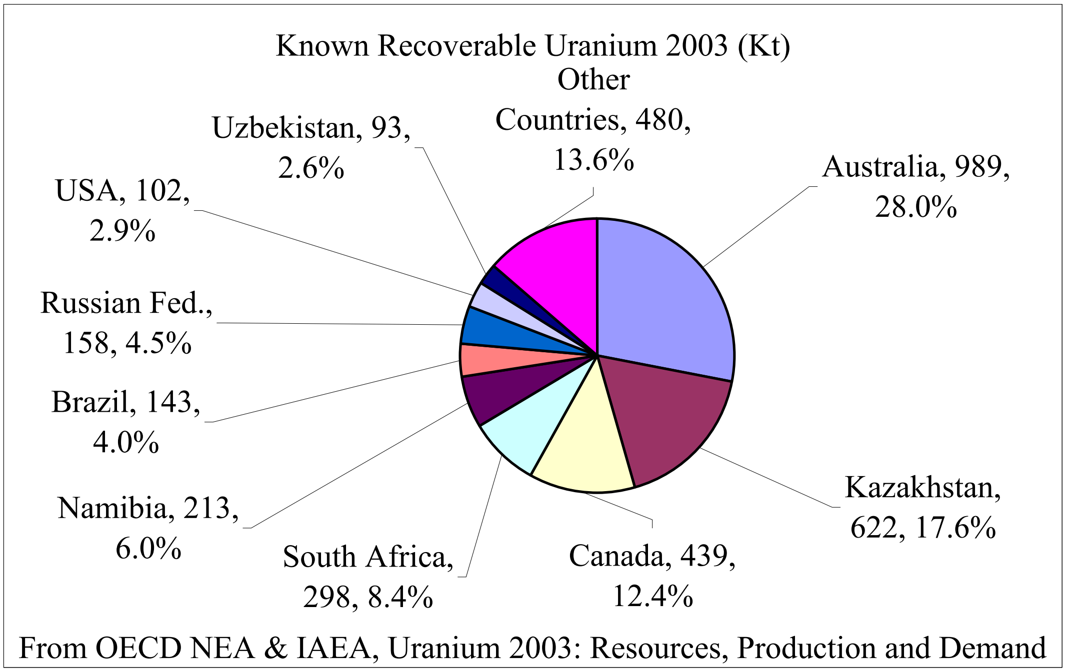


Figure 4: Uranium world reserves in kilo-tons [7]

Energy utilization

The energy consumption of one country depends primarily on its economical wealth and its population. On Figure 5, one can see that the most developed (Western Europe and North America) and most populated (Asia) regions are the largest energy consumers. If China and India were living under modern living standards and assuming the energy needed to reach them was available, their share of the total energy consumption would be around 60%.

Figure 6 shows the utilization ratio of the different primary energies. As one can see, oil, natural gas and coal are the resources of choice. Oil is leading by far, in front of natural gas and coal at the same utilization level. Far behind, one can find renewable and nuclear energies.

Figure 7, Figure 8 and Figure 9 illustrate with more details, the world's oil, natural gas and Uranium consumption distribution. In all cases, amongst the five largest consumers are invariably the USA, Japan, Germany and Russia. The USA are always far ahead first with the three other countries following behind in changing ranks depending on the energy source. One can remark the particular situation of France. It is ranked 9th for the oil consumption, does not appear in the ten first for natural gas consumption (actually ranked around 50th) but appears second on the Uranium consumption. Its very high utilization of nuclear energy (around 80% of its electricity generation) allows the country to save on imported fossil fuels. France decided to rely heavily on nuclear energy during the oil crisis in the seventies.

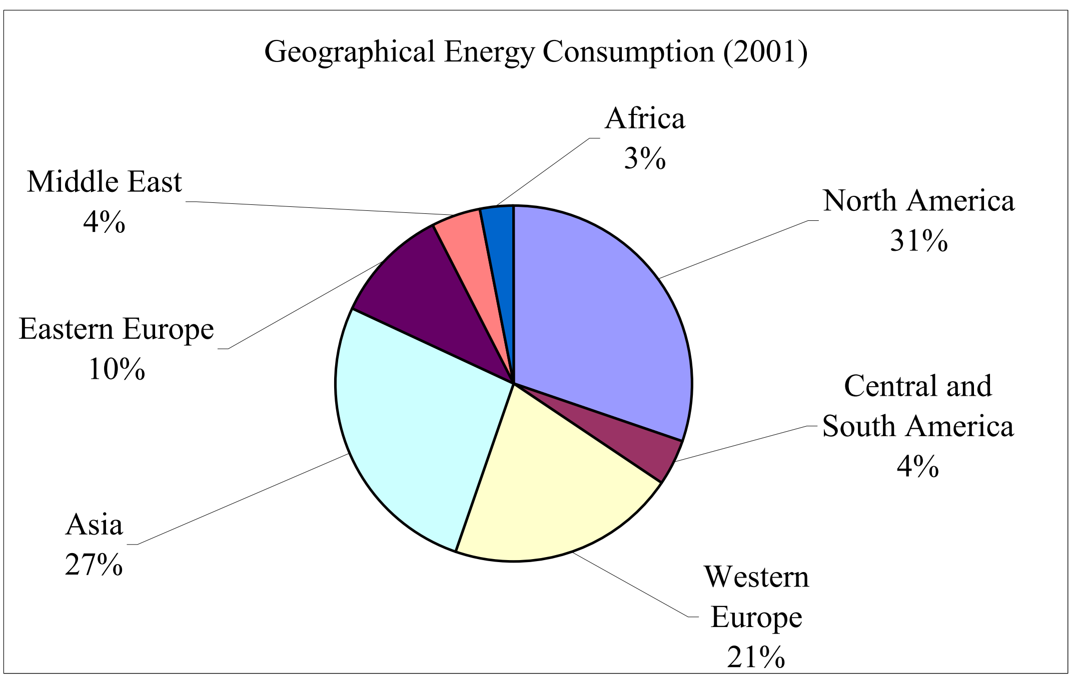


Figure 5: Geographical energy consumption [8]

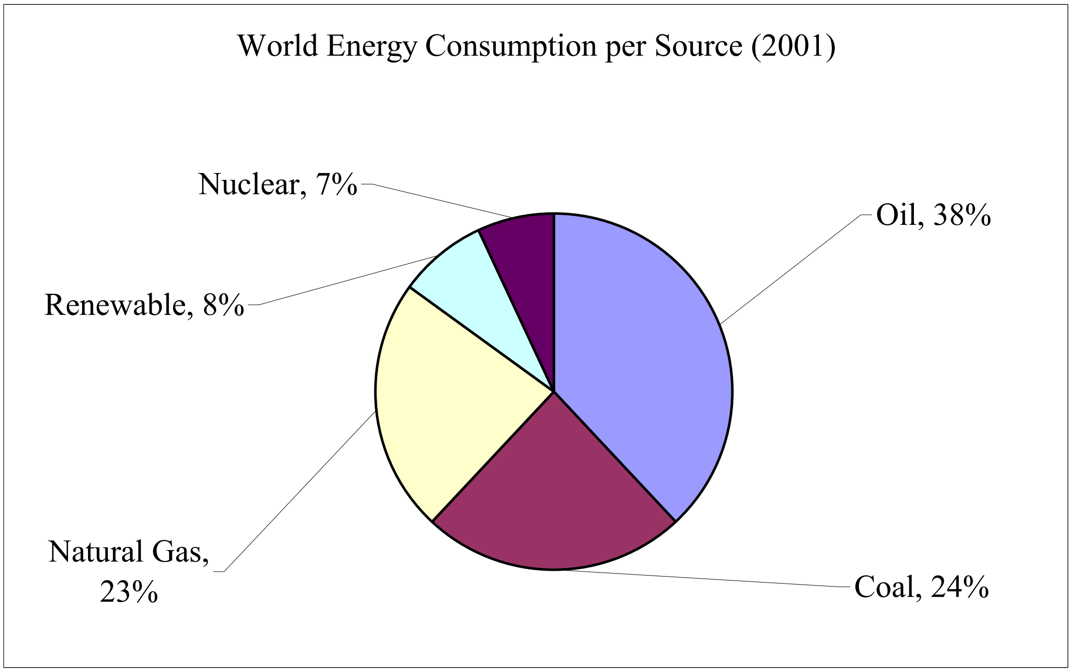


Figure 6: World energy consumption per source [8]

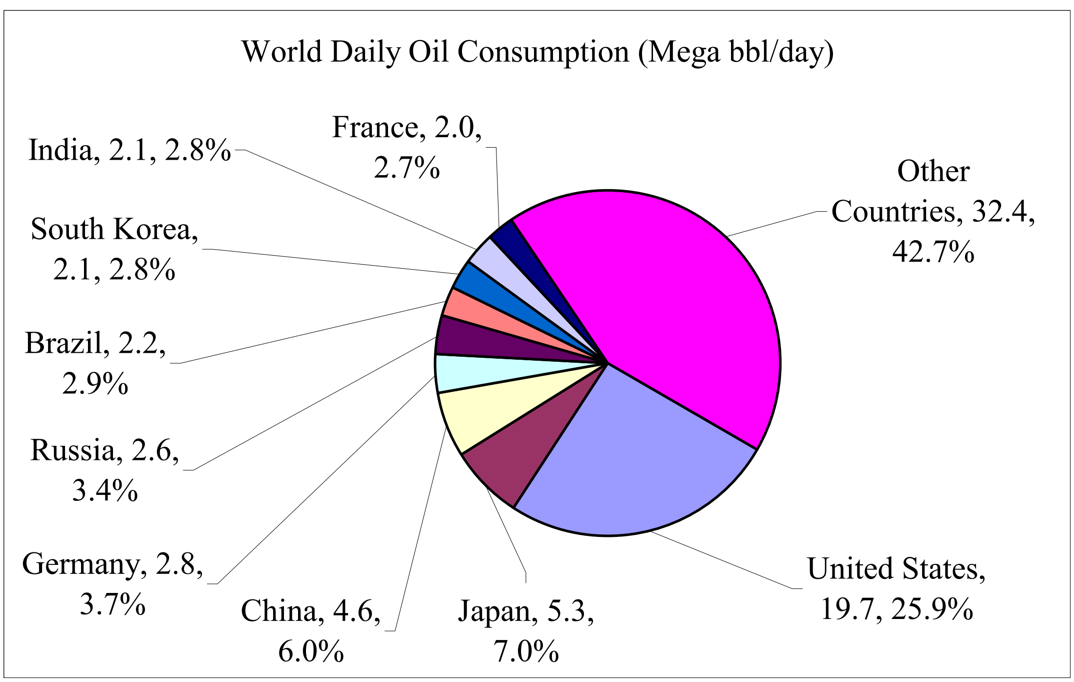


Figure 7: World daily oil consumption in millions barrels per day [5]

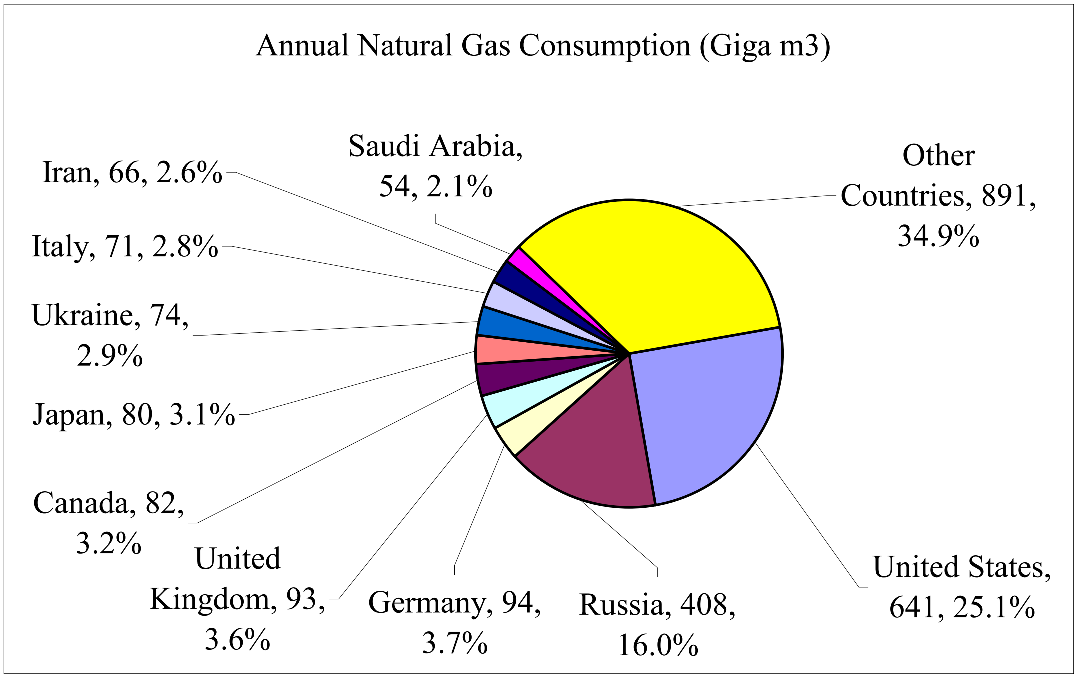


Figure 8: World annual natural gas consumption in billions cubic meters [5]

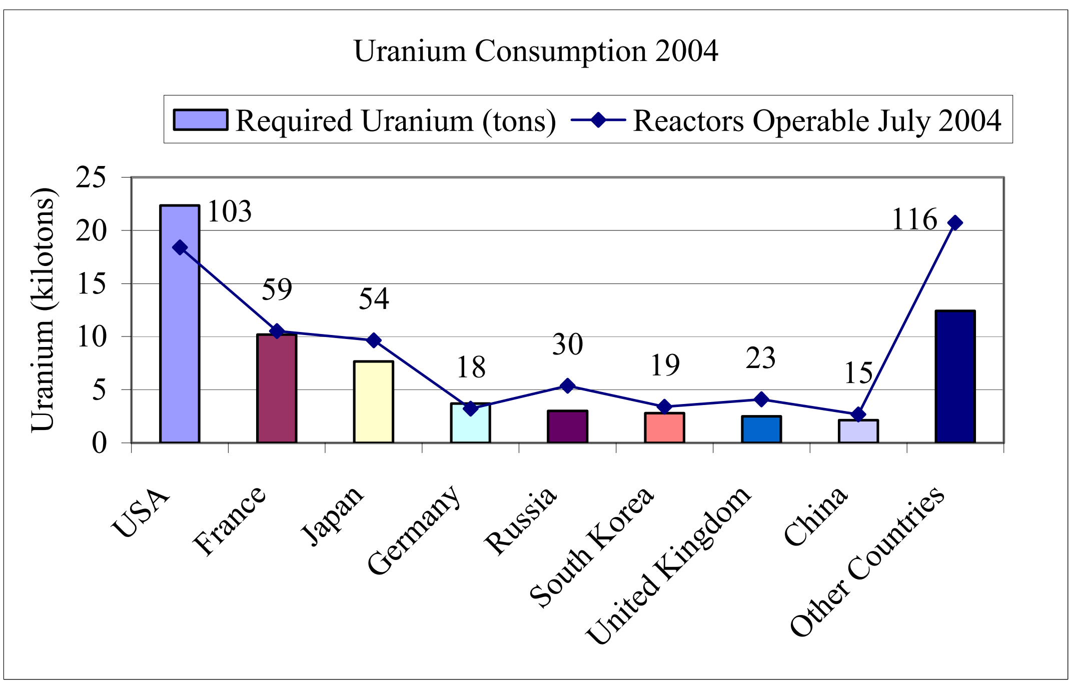


Figure 9: Uranium consumption in 2004 [7]

Energy balance and energy dependence

Comparing the fossil fuel consumption to the reserves information one can draw an obvious conclusion: the largest end-users of primary energy are not the holders of the principal reserves. Countries in North America and Europe are making use of much more primary energy than they produce. As a result they have to import most of their energy needs from foreign countries. The developed countries economies are bonded to the good will of the primary energy owners. North America and Europe are in a situation of energy dependence which inevitably creates geo-political tensions source of global conflicts.

Comparing fossil fuel reserves to current consumption rate raises another concern. Figures given in Table 1 show the remaining years of fossil fuel reserves at present production rate and assuming it will stay constant and no major resources will be uncovered. Fuel shortage is not very far down the road. The values for Uranium and Thorium assume no recycling and no breeding. The fossil fuel shortage depends on the estimations of current reserves and strongly on the forecasted energy consumption rate. The world energy consumption projections depend on the expected evolution of the world energy needs. The energy consumption rate also depends on national energy policy but it is more than unlikely that it will decrease in the near future. On the contrary, the world's use of primary energy is predicted to increase according to most of international agencies. The fuel shortage will have disastrous effects on countries that will not have found solutions to it.

It is worth noticing that the oil dependence issue is strongly linked to the foreseeable fuel shortage. A priori inevitable in a long term, fuel shortage effects may occur earlier than expected for the largest energy consumers. For these countries, there is no problem as long as the production follows their increasing consumption. However, once again, the largest users are not the ones holding the resources' reserves. Therefore, if the producers decide to reduce their production in order to save the shrinking reserves (for most of the large producers, fossil fuel is the main and only income) the consumers will be left with little help.

As of today, consumers rely on relatively low fossil fuel costs. This situation may rapidly change as the resources become more and more difficult, thus expensive to retrieve and exploit.

Table 1: World remaining fossil fuel reserves and time to shortage

	Consumption per year	Reserves (proved & recoverable)	Years of remaining reserves at present production rate
Oil (Mbbl)	27,671	1,025,000	37
Natural gas (Giga m3)	2,555	161,200	63
Coal 2002 (Mt)	4,783	892,876	187
Uranium 2003 (kilotons)	36	3,537	99
Thorium 2003 (kilotons)	6	1,200	212

Modern living standards in North America and Europe have proved to bring a lot of comfort to its beneficiaries but the use of advanced technologies is also very greedy in energy. Population wise, modern countries do not represent the majority of the world. The global situation would become critical if more people were experiencing the same kind of comfort.

The over consumption of modern countries and the emergence of very populated countries such as China or India which assert an access to the same level of comfort is not the only issue. More and more concerns are rising on environmental issues such as global warming and pollution.

Global warming and sustainable development

- Greenhouse gas emissions

In 1995, the Greenhouse Gas Emission Reduction Trading (GERT) association has published the conclusions of the Intergovernmental Panel on Climate Change that stated that "the balance of evidence suggests there is a discernible human influence on the climate system." Possible impacts could include significant effects on regional and local climates/natural ecosystems and social and economic disruption [9].

Whether or not human activity is a major factor in the measured global warming has yet not been scientifically proved in an indisputable way. There are chances, global warming may just be an episode of the natural earth cycle. Nevertheless, it would be irresponsible to consider that human activity will not have any impact on the balance of our planet on the long term.

CO_2 is considered to be the major contributor to the greenhouse effect. It is the most common gas rejection in the world because of the massive use of fossil fuel. Figure 10 shows the 2002 USA greenhouse gas emissions. Carbon dioxide represents the vast majority of the gas emissions. Figure 11 shows the sources of carbon dioxide in the USA. The transportation sector and the industry are the two major producers.

Assuming carbon dioxide and other greenhouse gases are the key factors for the global warming; our efforts should be focused on the reduction of those polluting emissions. These gases are produced from fossil fuels; thus one should try to reduce fossil energy use. Investigations should be pursued in the sectors that use a lot of fossil fuels such as the industry and the transportation. Solution may be found in the development and implementation of alternative sources of energy.

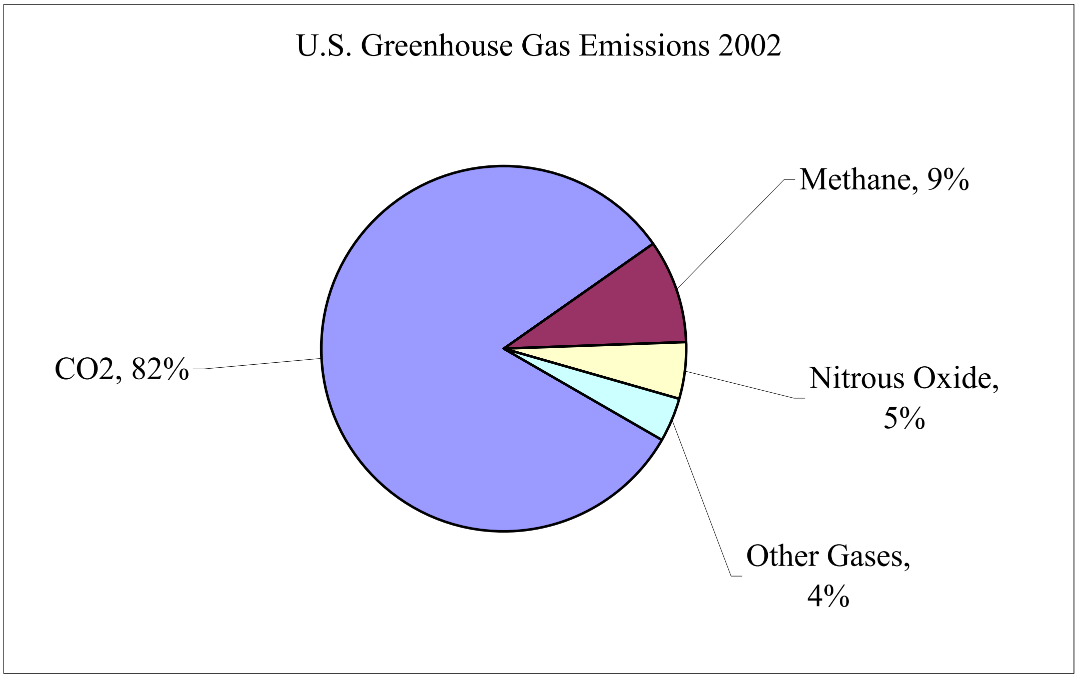


Figure 10: Relative US greenhouse gas emissions in 2002 [10]

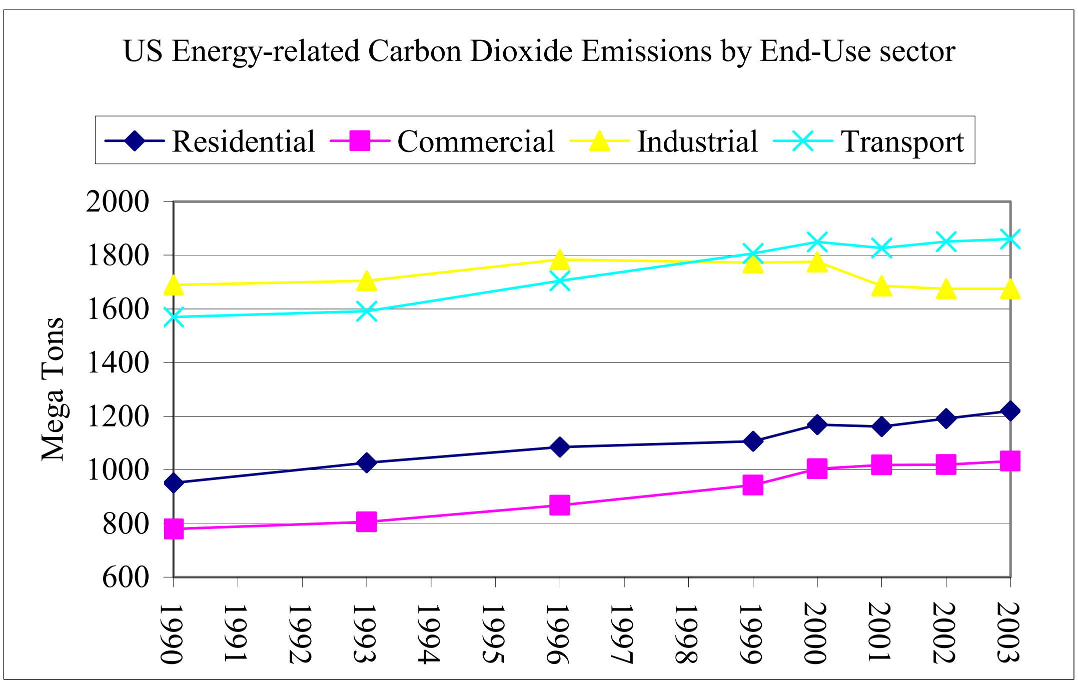


Figure 11: Annual US energy-related carbon dioxide emissions by sector [10]

- Alternative sources of energy

Facing predicted fossil fuel shortage and environmental issues, which may become more and more of a public concern, alternative sources of energy should be investigated to take over the hydrocarbon dependence.

The major alternative sources of energy are wind power, solar panels, bio-mass, hydroelectricity commonly gathered up under the term “renewable energies” and nuclear.

Although nuclear energy is a fossil fuel, it can be classified as an alternative to hydrocarbons energies because its use doesn’t generate gaseous rejections.

Despite the greenhouse effect, the development of nuclear energy has been delayed because of the waste management issue and, to a lesser extent, safety concerns (including proliferation). About 50 years of exploitation of nuclear energy has proved its reliability. Already well established in many countries (North America and Europe) nuclear could be further developed and expanded to fix the flaws of the current power plant fleets. Then, “nuclear power could continue to play a beneficial environmental role, helping to reverse the course of climate change” [11]. Association of nuclear plants with hydrogen production plants could prove very beneficial in the future (see next chapters).

However, it is important to notice that if nuclear was to be widely set in all around the world (let’s say around 20 % of the electricity production), the present technology based on Uranium 235 would not be applicable any more. The Uranium 235 supply would be insufficient. One would face Uranium shortage. The only relevant technology would have to use breeder reactors using Thorium or Uranium 238. However, proliferation wise, breeders using Uranium would not be acceptable because of Plutonium build-up.

Renewable energies are good candidates to partly take over fossil energies. Renewable energies are well distributed around the world. At the human scale they are inexhaustible thanks to the natural cycles. They are a clean source of energy. The pollution is limited to local effects such as dams drowning valleys or windmills blocking the landscape. At first sight, renewable energies could be a perfect substitute to fossil fuels.

Unfortunately they are also diffuse and for the majority, irregular sources of energy. They thus require economy in our consumption. They cannot provide a steady output. Nevertheless, renewable energies are suitable sources of energy at the household level as a complement to the electric grid.

Most of the alternative energy sources are already in use or in an advanced stage of implementation. Hydro-electricity is used all around the world and most of the good spots are already taken. The windmill industry is in a full boom but the space occupation is a limiting factor. Solar technology is also promising but it is still suffering from its low efficiency. Last but not least, geothermal energy, which consists in using the heat of the terrestrial coat, is in its implementation phase.

In fact, the use of renewable energies can be effective on the reduction of fossil fuel needs if energy savings policies are enforced. There the problem shifts from a technical issue to a political one.

In the current global situation, we are facing three major issues: energy dependence, foreseeable fossil energy shortage and global warming due to greenhouse gases emissions. A solution that reduces the fossil fuels utilization would solve the three problems at the same time.

In view of the results presented previously, nuclear energy is obviously part of the solution. Two ways to save on hydrocarbons products would be to produce electricity with nuclear energy (and other alternative sources of energy) and substitute hydrogen for gas as transportation fuel. This procedure has been given the name of hydrogen Economy as one passes from a petrol-based market to a H_2-based one. However, the shift to H_2 economy makes sense only if H_2 is produced from nuclear energy (or any other alternative energy) instead of oil and natural gas.

Hydrogen economy

This chapter addresses the H_2 economy theory. The advantages of shifting to H_2 economy, H_2 production technologies and issues related to the methods of H_2 production from nuclear plants are described in the next paragraphs.

Current hydrogen market

Nowadays, hydrogen is used mainly in two domains at industrial scale. It is used in the chemical industry to manufacture chemicals, mostly fertilizers. It is used in oil-chemistry for the conversion of crude oil into clean liquid fuels: gasoline, diesels, jet fuels and high performance fuels.

The H_2 market is growing rapidly for two reasons: requests for cleaner fuels are increasing and the quality of crude oil is decreasing. More and more H_2 is required for the processing of low-quality heavy crude oil instead of high-quality crude oil.

Crude oil processing involving H_2 includes three operations: increase of the H_2-to-carbon ratio, reduction of the toxicity and cleaning. Cleaning consists of removing sulfurs and other impurities contained in the oil by adding H_2. Likewise, oil toxicity is reduced by destroying variety of carcinogenic compounds such as benzene. More importantly, the H_2-to-carbon ratio has to be increased in order to meet the transportation fuel norms. Ratio for gas is between 1.5 and 2 whereas the ratio for low quality heavy oils, more and more exploited, is as low as 0.8.

The growing need of H_2 in oil industry for the hydrogenation of crude oil is the current driving force in the increasing demand for H_2.

The current hydrogen world consumption is 50 million tons per year, growing at about 10% per year. In the USA, 11 millions tons of hydrogen are produced per year representing a thermal energy equivalent of 48 billions Watts. The US hydrogen production consumes 5% of US natural gas usage, releasing 74 millions tons of carbon dioxide [12].

Shifting to hydrogen economy motivation

Nowadays, hydrogen is produced exclusively from non-renewable natural gas directly at the refineries. Unfortunately, this method contributes to the global warming as it gives rise to quantities of carbon dioxide emissions, and increases the dependence on foreign fossil energy. In fact, we are facing an extravagant situation where natural gas is used as well as oil to produce transportation fuels.

The hydrogen economy concept proposes to produce hydrogen using nuclear power. Therefore saving on natural gas and reducing the amount of greenhouse gas releases.

Nowadays, even though the technologies to produce hydrogen from nuclear power are not ready yet, the hydrogen economy concept would already be relevant. The energy consumed every year in the world to produce hydrogen is equivalent to about 100 large nuclear reactors making the concept economically viable.

In the long term (20-30 years) hydrogen will substitute gas for the propulsion of car equipped with fuel cells and may even be used in the whole transportation domain (liquid hydrogen has low weight and high energy density making it excellent for airplane fuel). The use of hydrogen for all US transport would require some 200 million tons of hydrogen per year representing about 400 nuclear reactors [12].

In the mid-term, hydrogen could be used to produce carbon saver fuels. These fuels are created by increasing the hydrogen to Carbon (H_2-to-C) ratio to values greater than 2. This process increases the energy contained in the fuel thus decreasing the amount of crude oil needed and reduces the carbon dioxide emissions. This process reduces the energy dependence and the gas releases.

Hydrogen production technology

Several processes are considered to produce hydrogen with the use of nuclear reactor: high temperature electrolysis, steam reforming of natural gas and thermo-chemical cycles.

Hot electrolysis is based on the traditional low temperature electrolysis, a proven technology, where some of the energy input is heat, provided by the nuclear reactor, rather than electricity. The typical electrolysis efficiency in an industrial system is about 72%. Assuming efficiencies for typical light water reactors and high temperature reactors to be, respectively, 33% and about 50%, the production of hydrogen by electrolysis associated with a nuclear reactor ranges from 24% up to 36% with advanced reactors [13].

Steam reforming uses natural gas, mainly methane, to produce hydrogen. Hydrogen is actually extracted directly from the poly-carbon chains. Nowadays, industrial processes using steam reforming achieve 80% efficiency and generate almost all the hydrogen produced in the USA. Steam reforming is the most efficient process to produce hydrogen. Nuclear energy may be used to reduce the amount of natural gas needed, although reducing the efficient to 40% (assuming 50% efficiency for advanced reactors). Steam reforming requires heat that is actually provided by a non-negligible amount of natural gas. Heat provided by nuclear reactor could be used to replace that amount of the natural gas. About 20% of gas could be saved. Even though steam reforming is the most efficient process, it is neither adequate nor satisfactory. Natural gas is more expensive than water, the feedstock for electrolysis and thermo-chemical processes, and is bonded to the oil market. More significant, steam reforming still produces greenhouse gases and doesn’t reduce much the oil dependence, two reasons for shifting to hydrogen economy [13].

Several thermo-chemical processes have been identified to produce hydrogen. However, the most promising thermo-chemical means to produce hydrogen is the sulfur-iodine process illustrated on Figure 12. It goes through three chemical reactions where heat and water are the only inputs. Iodine and Sulfur are fully recycled.

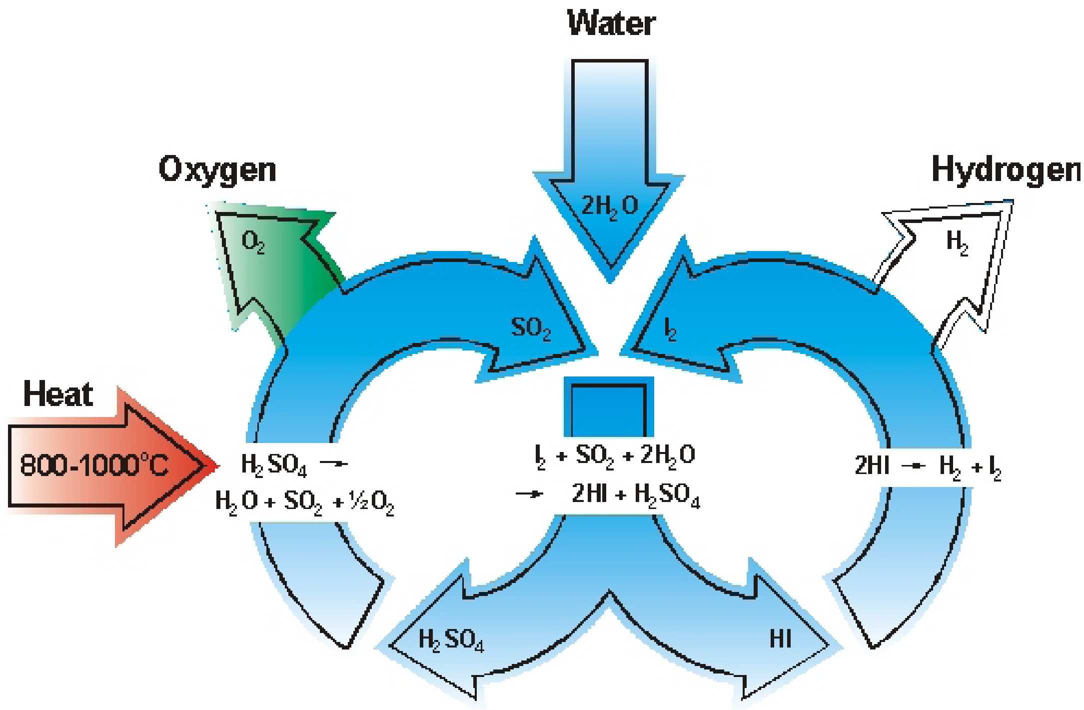


Figure 12: The sulfur-iodine thermo-chemical process

The first reaction uses water as feedstock and two reagents, Iodine and Sulfur. It generates HI and H_2SO_4. The second reaction converts HI into Iodine, which is reused in the first reaction, and the wanted product, hydrogen. This reaction requires rather high temperature input. The third chemical reaction is the decomposition of sulfuric acid. It is catalyzed by temperature. It requires high temperature and low pressure to drive the reaction to the right toward completion.

It is also worth it to notice that in most of the processes, hydrogen is not the only product. Therefore, economics may be improved by commercializing the other valuable products.

Constraints associated to the coupling of nuclear plant and hydrogen generation

In this paragraph the compatibility between nuclear reactors and hydrogen generation plants is addressed. Nuclear plants and hydrogen generation plants have both specific requirements upon which they are technically and economically viable. One of the biggest challenges that the hydrogen economy will have to take up is matching the two technologies. The next paragraphs sum up the principal requirements that will have to be met in order for the generation of hydrogen to be compatible with nuclear energy [14].

- Requirements for nuclear plants

The most important requirement for nuclear plants is the temperature output. All the potential hydrogen generation methods involve very high temperature. Output temperature should be at least 750°C. Temperature over 1000°C would be preferred.

Another condition concerns the output temperature range. Hydrogen production methods require a relatively small temperature range in order to maximize the process efficiency.

Table 2 shows the output temperature and the temperature increase across reactor cores for different reactor coolants. The figures are a little bit confusing. The GT-MHR is the best match temperature output wise, but the worst temperature range wise. The conclusion is the opposite for a typical PWR.

Table 2: Temperature increases across reactor cores for different reactor coolants [13]

System	Delta T. Inlet to Outlet (°C)	Inlet T (°C)	Outlet T (°C)	Coolant
GT-MHR	359	491	850	Gas (Helium)
Advanced Gas Reactor (Hinkley Point B)	355	310	665	Gas (CO2)
PWR (Point Beach)	20	299	319	Liquid (Water)
Liquid Metal Reactor (Super Phenix)	150	395	545	Liquid (Sodium)

Another condition applies to the reactor operating pressure. It would be optimal if the coolant pressure is kept as low as possible. In fact, the hydrogen generation is maximized at low pressure. Because high temperatures are involved it is preferable to minimize pressure difference to reduce material stress, thus reducing the need for high-strength, high temperature materials. Also, keeping the pressure on the reactor side lower than on the chemical plant side increases the safety.

- Requirements for hydrogen production plant

The compatibility conditions for hydrogen plants are related to their reliability, availability and scale of operation.

Nuclear reactors are economically viable when large units with huge power output are operated. Also, experience shows that nuclear reactors economics are better if base-load operations with continuous output are achieved.

Therefore, the hydrogen plants will have to operate with large energy flows during long period to match the operating scale of nuclear plants.

- Requirements for interface

A fundamental requirement is the isolation between the two plants while allowing high temperature heat transfer. Nuclear plants and hydrogen generation plants are both holding significant inventories of hazardous materials. Accidents on the chemical side must not compromise the nuclear plant safety. On the contrary, the nuclear system must not contaminate with radioactivity the hydrogen generation plant. Therefore the two plants must be isolated and protected from each other. The challenge is to keep the transfer of energy as efficient as possible without increasing the risks too much. Another incentive is to separate the two plants enough for the hydrogen plant to be considered outside of the nuclear plant influence. The radioactivity level on the chemical side must be sufficiently low to avoid classifying the hydrogen production plant as a nuclear system. It simplifies risk analysis and plant licensing.

Primary energy dependence, fossil fuel shortage and global warming could be solved if the modern countries shift to hydrogen economy. The most important point is that it makes no sense, from environmental and fossil energy saving standpoints, to produce hydrogen from a non-sustainable energy source. The point in shifting to hydrogen economy is to get rid of the oil dependence and pollution. Therefore, hydrogen must be produced with methods that are carbon free.

To help reduce the greenhouse gas emissions and the oil dependence modern countries could also implement strong energy savings policies. It is difficult to ask China to take care of its gas emissions whereas western countries are consuming energy without moderation.

Further developments of nuclear energy are necessary in order to consider it as a sustainable solution to our energy needs. But this achievement would only be possible if an acceptable solution is found to the waste problems.

Granted that nuclear energy is part of the solution to solve fossil fuels shortage, oil dependence and global warming, the major role of the nuclear plants would be the production of electricity and hydrogen for transportation and petrochemical industry. However the actual generation of power plants (mainly aging, though reliable, PWR and BWR concepts) cannot meet the requirements. Therefore, new concepts of nuclear plants have to be developed. That is the goal of the Gen IV project presented thereafter.

Generation IV

The concept of generation IV was developed under two major incentives: (1) the need for new extensive researches in nuclear technology in order to render nuclear energy more attractive and (2) the need for an international collaboration, involving the collective skills, the expertise and the resources of many countries to support the effort.

Generation IV project involves ten countries from all over the world: Argentina, Brazil, Canada, France, Japan, the Republic of Korea, South Africa, Switzerland, the United Kingdom and the United States of America. The global project called Generation IV International Forum (GIF) was official launched in September 2002. The goal is too develop new concepts of power plants that represent significant advances in economics, safety, reliability, proliferation resistance and waste minimization. The new concepts are intended for electricity production, hydrogen production for transportation and petrochemical applications, and maybe water desalination. To reach these goals, the development of six energy concepts has been decided: gas-cooled fast reactor system, lead-alloy liquid metal-cooled fast reactor system, molten salt reactor system, sodium liquid metal-cooled fast reactor system, supercritical water-cooled reactor system and very high temperature gas-cooled reactor system [15].

The development of fast reactor complies with the needs for better fuel utilization and minimization of long-lived radioactive isotopes sent to waste repositories. A fast neutron-spectrum makes it possible to utilize available fissile and fertile materials considerably more efficiently than thermal neutron-spectrum.

The increase in reactor temperature comes from the need for better thermal efficiency and hydrogen production requirements.

The following information describes the six energy concepts that will be developed by the GIF. The GFR and the SFR are the two concepts for which the technology may be the most easily available due to previous developments of similar concepts (gas reactors in France and the UK, sodium reactors – Phenix, Monju, in France and Japan).

Gas-cooled fast reactor system

The GFR system is a fast-neutron spectrum using helium as coolant which allows very high temperature outlet. In the electricity production version, the GFR uses a direct Brayton cycle helium turbine for high thermal efficiency. The expected temperature outlet of 850°C makes the development of new fuel essential. The fuel needs to withstand high temperature and must ensure fission products retention. The actual fuel candidates that are potentially available are: composite ceramic, advanced fuel particles, or ceramic clad elements of actinide compounds. Because the GFR will use a closed fuel cycle, fuel candidates must also provide easy reprocessing capabilities.

Lead-alloy liquid metal-cooled fast reactor system

The LFR has a fast neutron-spectrum with a lead or lead/bismuth eutectic liquid metal-coolant. The core is cooled by natural convection. The reactor outlet coolant temperature ranges between 550°C and 800°C. The reactor relies on a closed fuel cycle with long refueling interval (15 to 20 years). The higher temperature concept enables for hydrogen production and water desalination. However, extensive research must be implemented on advanced materials and corrosion issues in order to validate the LFR system.

Molten salt reactor system

The MSR is the only system that features a non-solid fuel. In this concept, the fissionable material is mixed with molten salt that serves as the coolant. The neutron spectrum is expected to be epithermal although epithermal reactors have never been experienced. During operation, the molten salt fuel mixture is continuously filtered allowing complete recycle of actinides although efficient filtering devices must still be developed. The plant features a coolant outlet of 700°C. Molten fluoride salts feature very good heat transfer characteristics and very low vapor pressure that reduces constrains on vessel and piping. The MSR features a complex heat removal piping system (three separate coolant systems which transfer heat to the power conversion equipment) to avoid contamination in case of breaches. This is a major deterrent with respect to capital cost and efficiency.

Sodium liquid metal-cooled fast reactor system

The SFR is a fast neutron-spectrum sodium cooled reactor. On account of the safety issues associate with sodium, the plant features three coolant loops in order to transfer the heat from the core to the power conversion system. However, sodium features interesting characteristics: long thermal response time, large margin to boiling allowing operating near atmospheric pressure. The SFR system is designed for high level waste management, in particular plutonium and higher actinides.

Supercritical water-cooled reactor system

The SCWR is a very innovative system. In this concept, the reactor operates over the thermodynamic critical point of water (374°C, 22.1 MPa or 705°F, 3208 psia). Therefore, the coolant does not change phase and the heat transfer correlations are simpler than in a normal light water reactor. However, the system's behavior under accident condition must be well assessed. The reactor is expected to operate under a pressure of 25 MPa and with a coolant outlet temperature of 550°C. The SCWR may use thermal or fast neutron- spectrum.

Very high temperature gas-cooled reactor system

The VHTR has been designed to more specifically produce hydrogen. It is a graphite-moderated, helium cooled reactor. The core outlet temperature, higher than the GFR's, is 1000°C.

The six Generation IV concepts are presented on Figure 13.

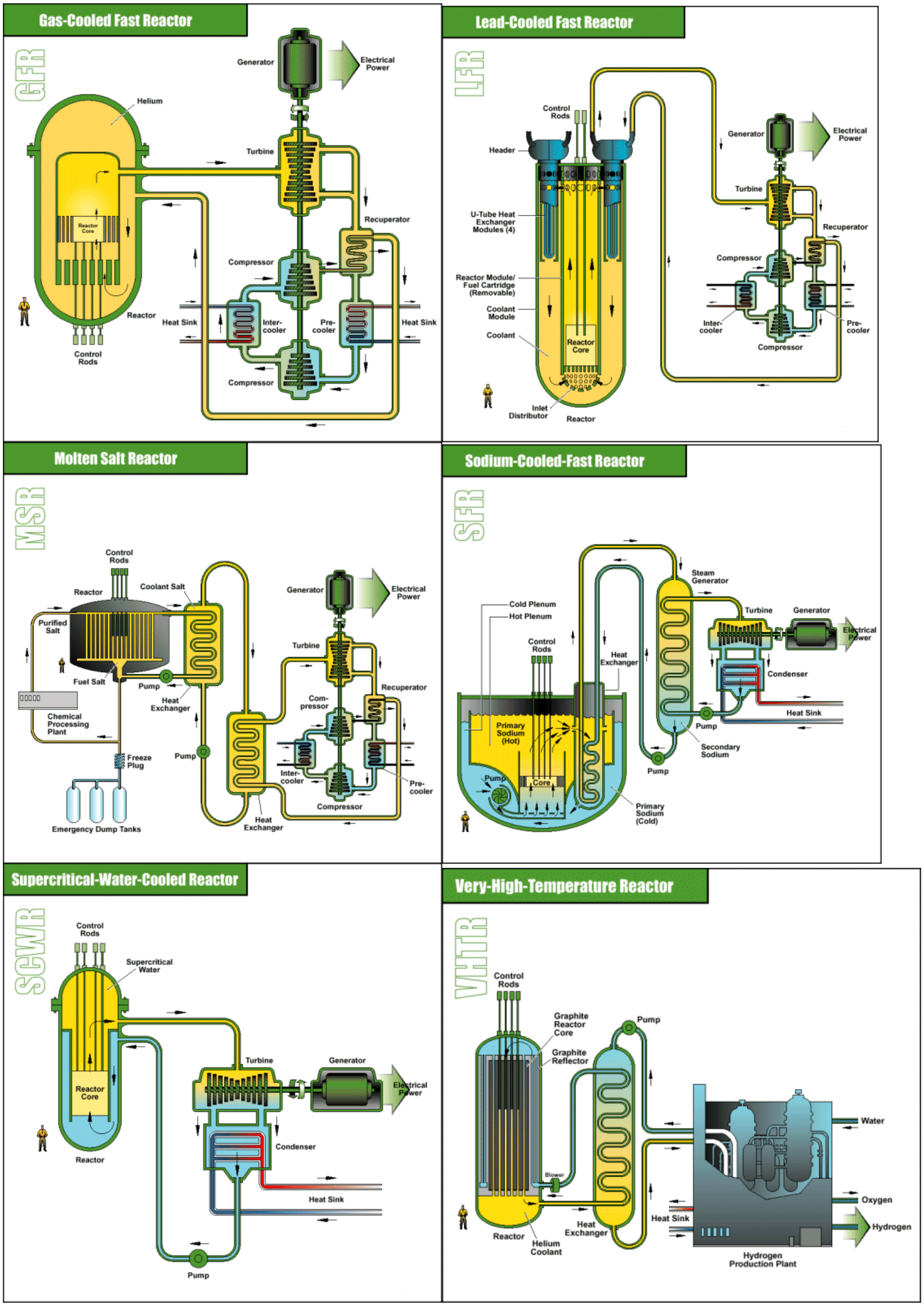


Figure 13: Generation IV concepts

Very ambitious, the Gen IV concepts are still far down the road. The implementation of the different Gen IV concepts is not expected before 2020 (optimistic forecast), more likely around 2030. Many of the technologies proposed are still to be implemented and their feasibility on industrial scale has yet to be proven. By then, it could prove useful to develop intermediate nuclear reactor concepts such as the MHR proposed by GA. On one hand, one needs to fulfill the demand for electricity that keeps on growing. Presumably, the Gen IV reactors will not be available before the current generation is shut down. Also, the licensing renewal process actually carried out in order to extend the lifespan of aging nuclear park will not be sufficient. On the other hand, the technologies involved in Gen IV could be implemented progressively so the research efforts are distributed more evenly. Therefore, the study of one or several intermediate projects seems rather relevant [16].

One of these project concerns the Modular Helium Reactor (MHR) developed by General Atomics (GA). The MHR is especially well suited for producing hydrogen, it has high-temperature capabilities, it is in an advanced stage of development relatively to other high temperature advanced reactor concepts and it has attractive passive-safety features.

GENERAL ATOMIC MODULAR HELIUM REACTOR

The objective of the Modular Helium Reactor (MHR) program is the development of a passively safe, proliferation resistant, economic nuclear power option for commercial power generation and/or hydrogen production.

Genesis

The GA-MHR is directly inspired from the High Temperature Gas-cooled Reactors (HTGR) programs [17].

The HTGR technology has been under development for over forty years and has developed along two distinct paths: pebble bed fuel consisting of ceramic spheres with continuous refueling, and prismatic fuel consisting of hexagonal blocks with periodic batch refueling. Both fuel systems utilize ceramic coated micro-particles of similar design.

The HTGR concept associated with a prismatic core was first investigated in 1956 at the Atomic Energy Research Establishment in Harwell. The construction of several HTGR of various sizes resulted from this initial work: a 20 MW_{th} HTGR called DRAGON-project in the United Kingdom and an 842 MW_{th} reactor at Fort St. Vrain, USA. These various programs were carried out with more or less success. DRAGON was operated from 1964 to 1975, whereas the Fort St. Vrain HTGR was operated from 1974 to 1989.

Although a few, these experiences have demonstrated key elements of the HTGR technology such as successful operation of reactor vessels and reliability of high-quality fuel.

General Atomics has been developing high-temperature, helium-cooled nuclear reactors since the middle of the 60’s for electricity production and a variety of process-heat applications, including hydrogen generation. In more recent years, GA developed the passively-safe, modular-sized design referred to as the Modular Helium Reactor

(MHR). When the MHR is coupled directly to a Brayton-cycle power conversion system to generate electricity it is referred to as the Gas Turbine Helium Modular Reactor (GT-MHR). The power conversion system may be replaced with an Intermediate Heat Exchanger (IHX) for applications that require process heat only. If the heat is used for hydrogen generation the MHR is referred to as the Hydrogen Helium Modular Reactor (H2-MHR). The MHR may also be used for electricity production and heat processing at the same time. This version is referred to as the Hybrid-MHR.

Design

The MHR [18] is a prismatic core type high temperature gas cooled reactor (HTGR). It operates at a thermal power level of 600MWth corresponding to a 280 MWe power output. The power density is 6.6 MW/m^3. The reactor system is located below grade and operates at elevated temperatures with a helium outlet temperature comprise between 850°C (electricity production) and 1000°C (hydrogen production) which leads to high plant thermal efficiency. The reactor's layout is presented on Figure 14.

The MHR utilizes uranium oxy-carbide fuel with TRISO coating as fuel particles.

The MHR features a good conversion ratio and superior fuel economics in comparison with other types of reactors. This is the consequence of a good neutron economy which is obtained from two factors: (1) Nuclear grade graphite constitutes most of the MHR structures (fuel particle coating, core structural material, and moderator and coolant channel walls) and (2) helium is an inert coolant. As a result the MHR experiences low parasitic capture and good neutron economy.

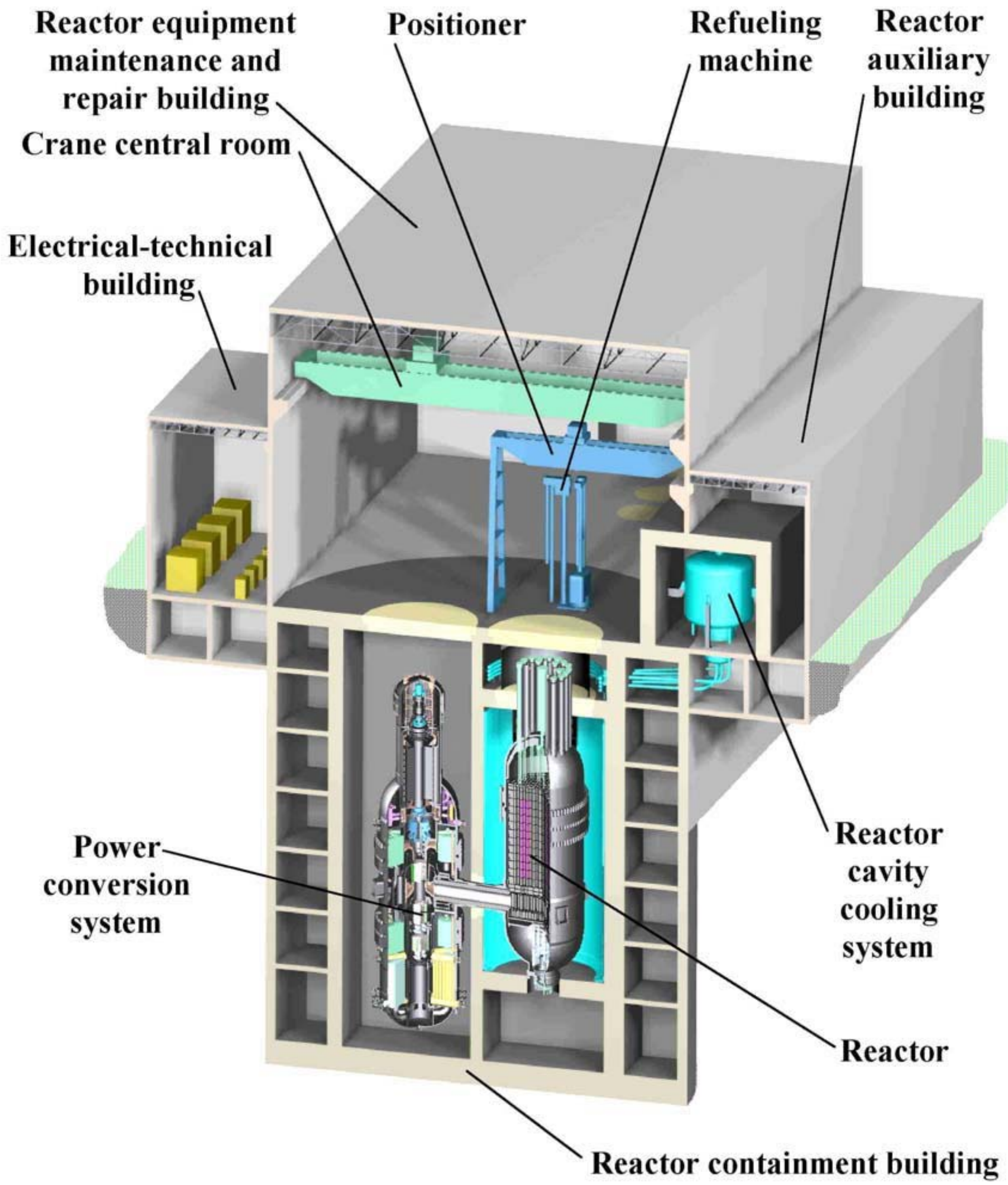


Figure 14: GT-MHR layout

The basic design characteristics of the MHR plant provide a significant reduction in the required plant equipment if compared with current nuclear plants. The MHR features simplified and reduced number of safety systems. In the electricity production version, the MHR allows the removal of the large steam power conversion equipment.

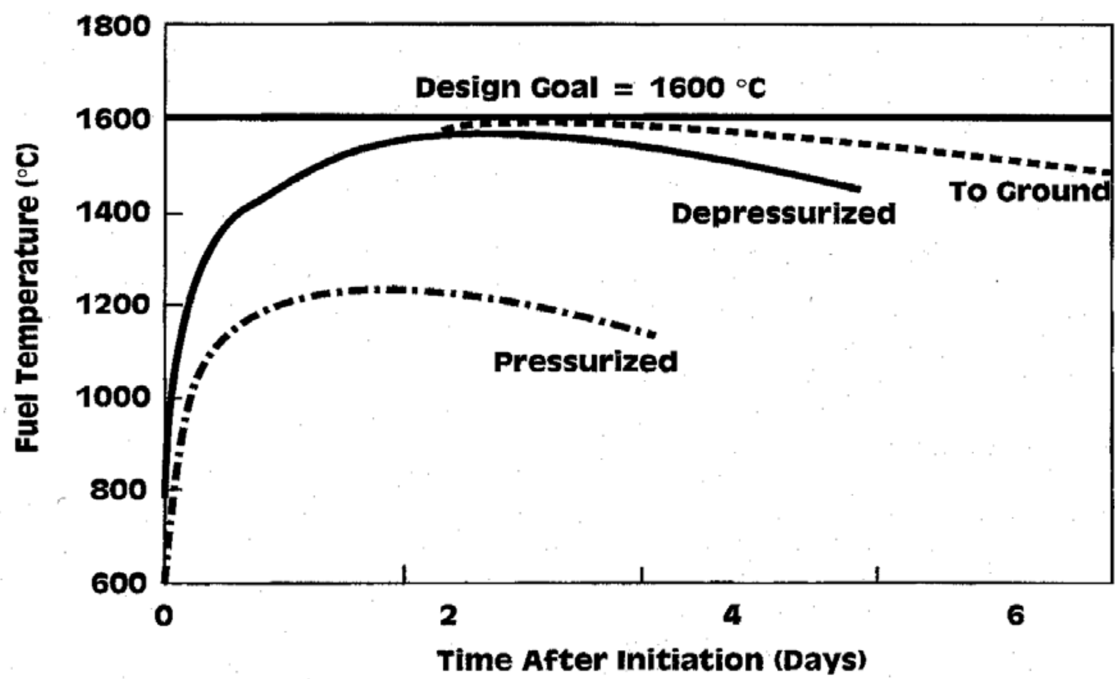
The plant simplification, the reduction in required systems and equipment, the modularization and the reduction in equipment requiring regulatory oversight, all contribute to an increased capacity factor relatively to other nuclear concepts.

Because of the MHR innovative elements, higher operating temperatures and helium coolant, specific technical issues and information needs must be addressed. Extended studies and appropriate development activities have to be pursued in fuel design confirmation and qualification, as well as passive decay heat removal under abnormal conditions.

A big part of the design was motivated by safety issues. In the design of the GT-MHR, the desirable inherent characteristics of the inert helium coolant, graphite core, and coated fuel particles are supplemented with specific design features to ensure passive safety. The release of large quantities of radio-nuclides is precluded by the fuel particle ceramic coatings, which are designed to remain intact during normal operation and off normal events. The integrity of the particles coatings as a barrier is maintained by limiting heat generation, assuring means of heat removal and by limiting the potential effect of air and water ingress on the particles under accident conditions.

For this design, the possibility of a core meltdown is precluded through the use of refractory, coated-particle fuel and nuclear-grade graphite fuel elements with high thermal capacity and conductivity, combined with operation at a relatively low power density with an annular-core arrangement. For instance, during a loss-of-coolant accident, decay heat is removed from the core by convection and radiation, both natural heat transfer mechanisms, to the Reactor Cavity Cooling System (RCCS). On Figure 15 we can check the passive decay heat removal characteristics of the MHR during loss-of-coolant scenario. We can verify the expected slow thermal response during a loss of coolant. The temperature peak occurs around 65 hours (2.7 days) after initiation of the transient.

The MHR fuel is considered to be proliferation resistant and storage proof. Not only the fuel particle coating acts as fission fragments barrier but it is also a highly resistant barrier for ground water during spent fuel storage. It is so resistant that, as of today, no technology exists to retrieve the fuel from the particles. It is anticipated that the spent fuel would contain 30 times less amounts of plutonium than typical light water reactors thanks to higher burnup achievement in MHR.



GT-MHR fuel temperatures during a loss-of-coolant accident (Source: GA)

Figure 15: Peak temperature versus time during loss of coolant scenario

Description

Reactor vessel

The primary components of the MHR are contained within a steel vessel system. The module is located inside an underground concrete silo 25.9 m (85 ft) in diameter and 42.7 m (140 ft) deep, which serves as the containment structure. The reactor vessel is 8.4 m (27.5 ft) in diameter and 31.2 m (102 ft) high. It contains the reactor core, the reactor internals, the control rods mechanisms, the refueling access penetration and the shutdown cooling system. The reactor vessel is surrounded by a reactor cavity cooling system which provides totally passive safety-related decay heat removal by natural draft air circulation. The shutdown cooling system located at the bottom of the reactor vessel provides forced helium circulation for decay heat removal for refueling and maintenance activities. Figure 16 shows the reactor vessel.

The MHR flow in the core vessel is as follows. Pressurized helium (491°C/915°F) enters the reactor through the outer annulus within the cross vessel, flows up the core inlet riser channels located between the reactor vessel inside wall and the core lateral restrain and enters the upper plenum located above the graphite core. Then the coolant flows downwards through the cooling channels located in the active core parts and the reflector parts. A fraction of the coolant by-passes the channels holes and flows into gaps between blocks and into control rods channels (approximately 3%). This fraction is comprised between 10% for the H2-MHR and 20% for the GT-MHR. When leaving the core, the high temperature coolant (850°C/1562°F) is collected in the lower plenum and then exits the core vessel through the inner annulus within the cross vessel.

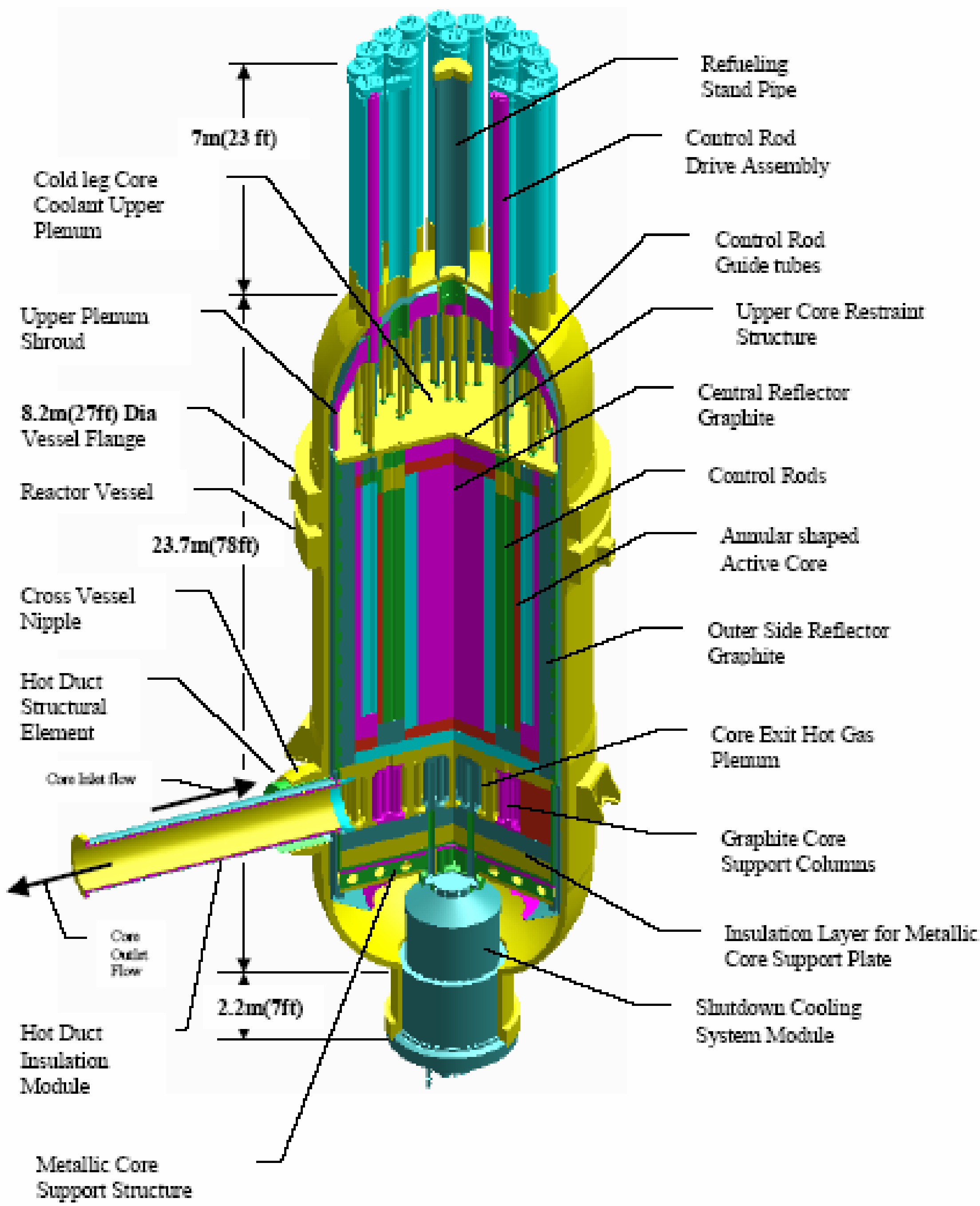


Figure 16: MHR reactor vessel

Core arrangement

The core arrangement is illustrated on Figure 17. The reactor is made of an assembly of hexagonal graphite fuel and reflector elements, also called blocks, arranged into an annular disposition.

The active fuel region of the core consists of 102-fuel columns, each made of 10 blocks high. Each block contains blind holes housing the fuel, and full length channels for the helium coolant flow. The columns are arranged in three annular rings.

The active core is surrounded with reflector elements. The inner and outer reflectors are made of respectively 61 and 156 columns. The reflectors above and below the core are composed of two layers.

During normal operations, the core reactivity is controlled with 48 control rods. 36 of them, the operating control rods, are located in the inner annulus of the outer reflector. The 12 remaining control rods, the start-up one, are located in the inner radius of the active core. The startup control rods are fully withdrawn during normal operation of the reactor. They are only inserted during startup/shutdown and refueling operating modes. Only the control rods located outside of the active core are used during reactor operations.

For abnormal situations, eighteen columns in the active core also contain channels for reserve shutdown material. The reserve shutdown material consists of boronated pellets stored above the core. The pellets are released in the active core if reactor scram is required.

The core is designed to preclude exceeding the maximum stress design limits of graphite and metal in the core. The annular core concept associated with low-power density and length to diameter ratio around 2 was chosen because it is able to reject the passive heat passively as show on Figure 18 (RSR stands for replaceable side reflector; PSR stands for permanent side reflector).

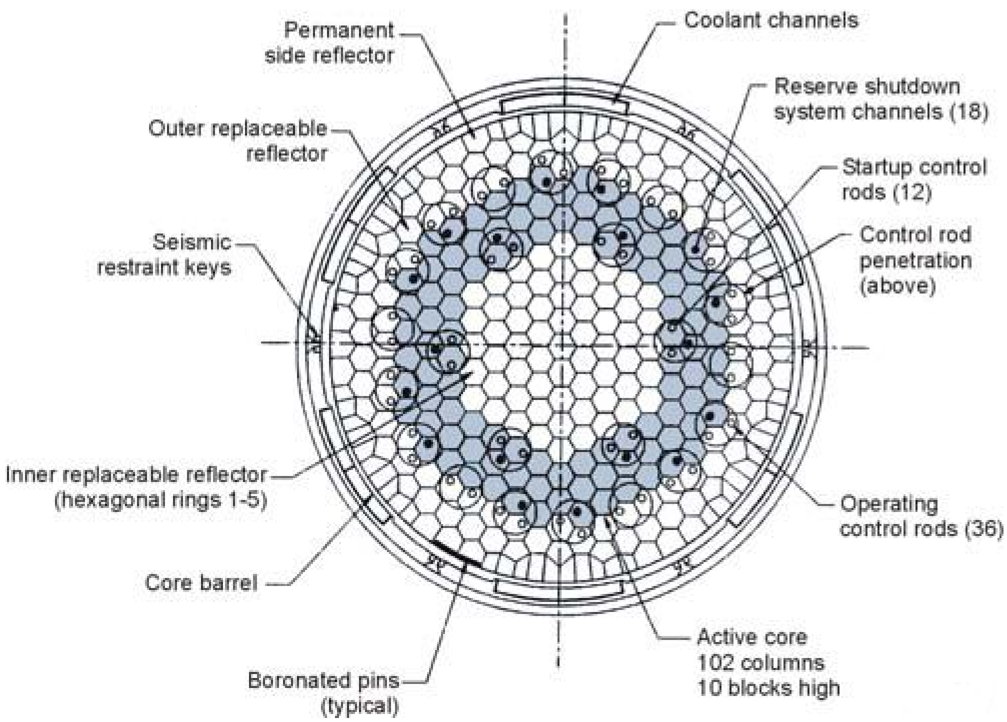


Figure 17: MHR core arrangement

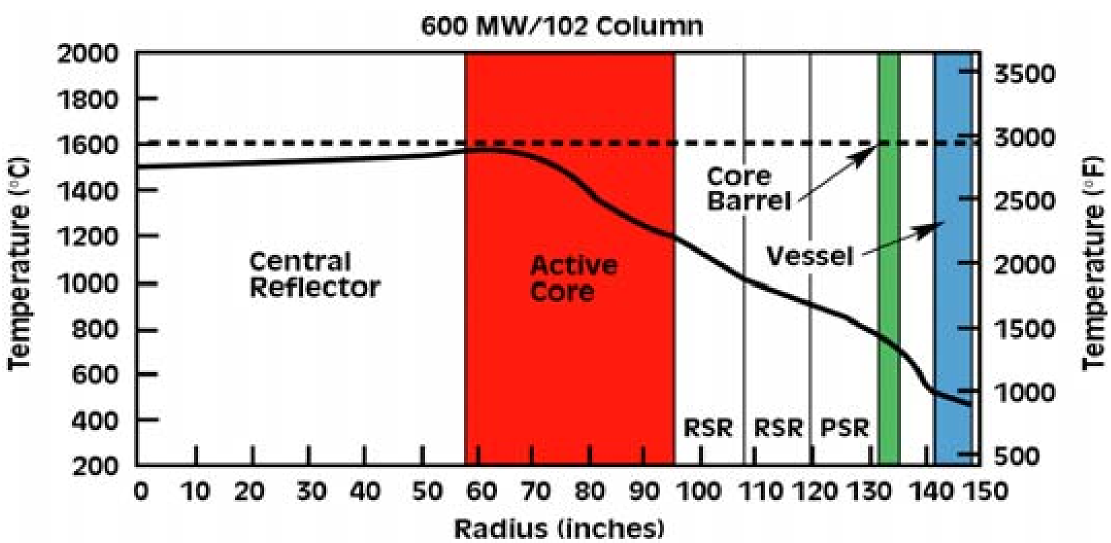


Figure 18: MHR peak temperature profile during a loss-of-coolant accident

MHR Fuel

The MHR fuel is set up like Russian nest dolls as illustrated on Figure 19. The spherical “fuel particles” are blended together to form “fuel compacts” that are stacked in the “fuel elements”.

First, as Figure 20 illustrates, the fissile material is formatted into micro-spheres that are coated with four layers of various varieties of carbon. The fuel kernel is made of uranium oxy-carbide (UCO) which performs well at relatively high burn-up and is highly effective at retaining many fission products. The MHR employs low-enriched uranium (19.8% U235) and natural uranium (0.7% U235 - fertile) fuel particles. The first coating layer is made of low-density, porous pyro-carbon. This layer acts as a buffer as it stops fission fragments escaping the kernel and provides adequate void space to limit fission gas pressure. The second layer, referred to as Inner Pyrolytic Carbon (IPyC), is made of high density carbon and protects the kernel and the buffer from chlorine compounds generated in the third layer. The third layer is made of Silicon Carbide (SiC) and provides metallic fission product retention and mechanical strength to limit dimension changes under irradiation. The forth and last layer, referred to as Outer Pyrolytic Carbon (OPyC), is also made of high-density carbon. It protects the fuel particle from mechanical damage and provides a bonding surface. Table 3 summarizes the fuel particle design parameters.

This particular multi-layer coating is referred to as TRISO coating (TRISO is an acronym for TRI-material, ISO-tropic). This coating system acts like a miniature fission product barrier and is very resistant to oxidation and corrosion.

The fissile and fertile fuel particles mixed with graphite shims are then blended and bonded together with a carbonaceous matrix into a rod-shaped element called fuel compact. This process prevents mechanical interaction between the fuel particles and the graphite moderator. It also maximizes the thermal conductivity in the fuel and provides a secondary fission fragments barrier. The fuel compact design parameters are gathered in Table 4.

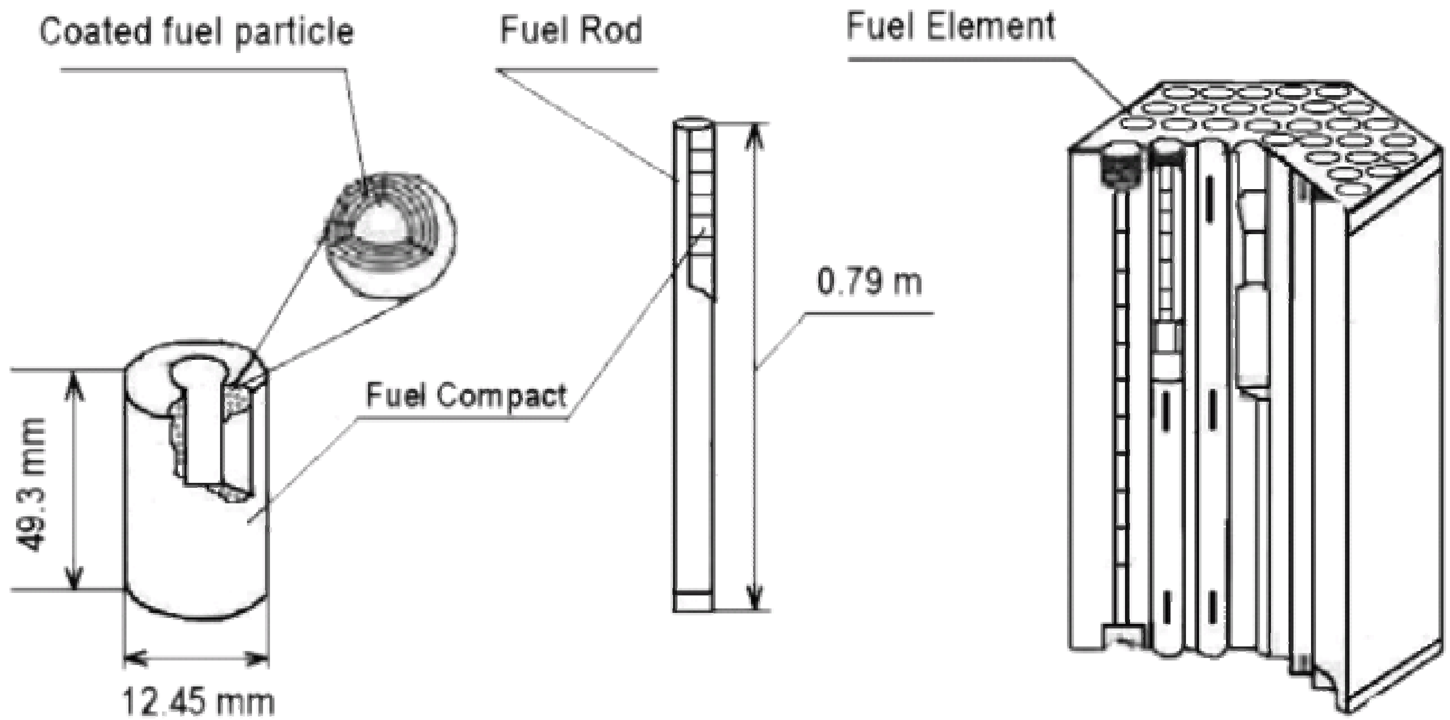


Figure 19: Fuel design

Table 3: Fuel particles design parameters

	Fissile Particle	Fertile Particle
Composition	UC0.5O1.5	UC0.5O1.5
Uranium enrichment, %	19.8	0.7 (Natural Uranium)
Dimensions (µm)		
Kernel Diameter	350	500
Buffer thickness	100	65
IPyC thickness	35	35
SiC thickness	35	35
OPyC thickness	40	40
Particle diameter	770	850
Material Densities (g/cm3)		
Kernel	10.5	10.5
Buffer	1	1
IPyC	1.87	1.87
SiC	3.2	3.2
OPyC	1.83	1.83
Elemental Content Per Particle (µg)		
Carbon	305.7	379.9
Oxygen	25.7	61.6
Silicon	104.5	133.2
Uranium	254.1	610.2
Total particle mass (µg)	690	1184.9

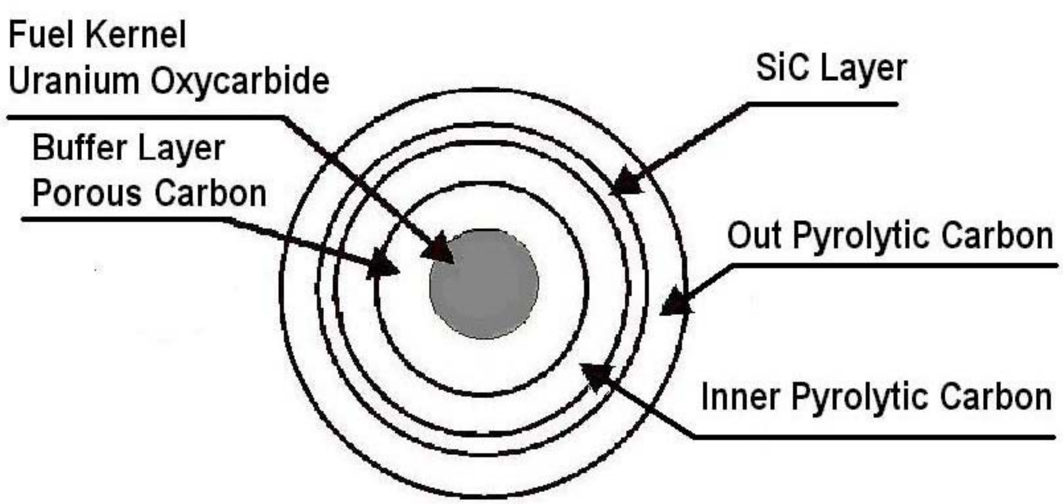


Figure 20: Fuel particles design

Table 4: Fuel compact design parameters

Diameter, mm	12.45
Length, mm	49.3
Volume, cm3	6
Shim particle composition	H-451 or TS-1240 graphite
Shim particle size	99 wt % < 1.19 mm
	95 wt % < 0.59 mm
Shim particle density (g/cm3)	1.74
Binder type	Petroleum pitch
Filler	Petroleum derived graphite flour
Matrix density (g/cm3)	0.8 to 1.2
Volume fraction occupied by matrix	0.39
Volume fraction occupied by shim particles in an average compact	0.41
Volume fraction occupied by fissile particles in an average compact	0.17
Volume fraction occupied by fertile particles in an average compact	0.03
Number of fissile particles in an average compact	4310
Number of fertile particles in an average compact	520
Mass of carbon in an average compact g	6.62

The fuel compacts are then stacked in the blind fuel holes of the hexagonal graphite fuel elements. The fuel compacts are enclosed in the fuel element by cementing the tops of the fuel holes. There are three types of fuel elements: standard fuel elements, reserve shutdown elements that contain a channel for reserve shutdown control, and control elements that also contain a control rod channel. The design of the standard fuel element is summarized in Table 5 and illustrated on Figure 21.

The performance of TRISO coated particle fuel under normal conditions which can be influenced by reactor design and fuel cycle times have been engineered to avoid coating failure in particles. High temperature tests on particles show that significant failure does not take place except under time and temperature conditions far in excess of anticipated accident conditions.

Tests on TRISO fuels have shown the ability to retain integrity after 500 hours in a 1600°C temperature field. Fission gas release at temperature above 1600°C is a function of time at temperature. However, SiC coating failure and loss of fission products become rapid when temperature exceed 2000°C, which is well beyond accident temperatures of the MHR. Figure 22 illustrates the temperature limitations of the TRISO fuel.

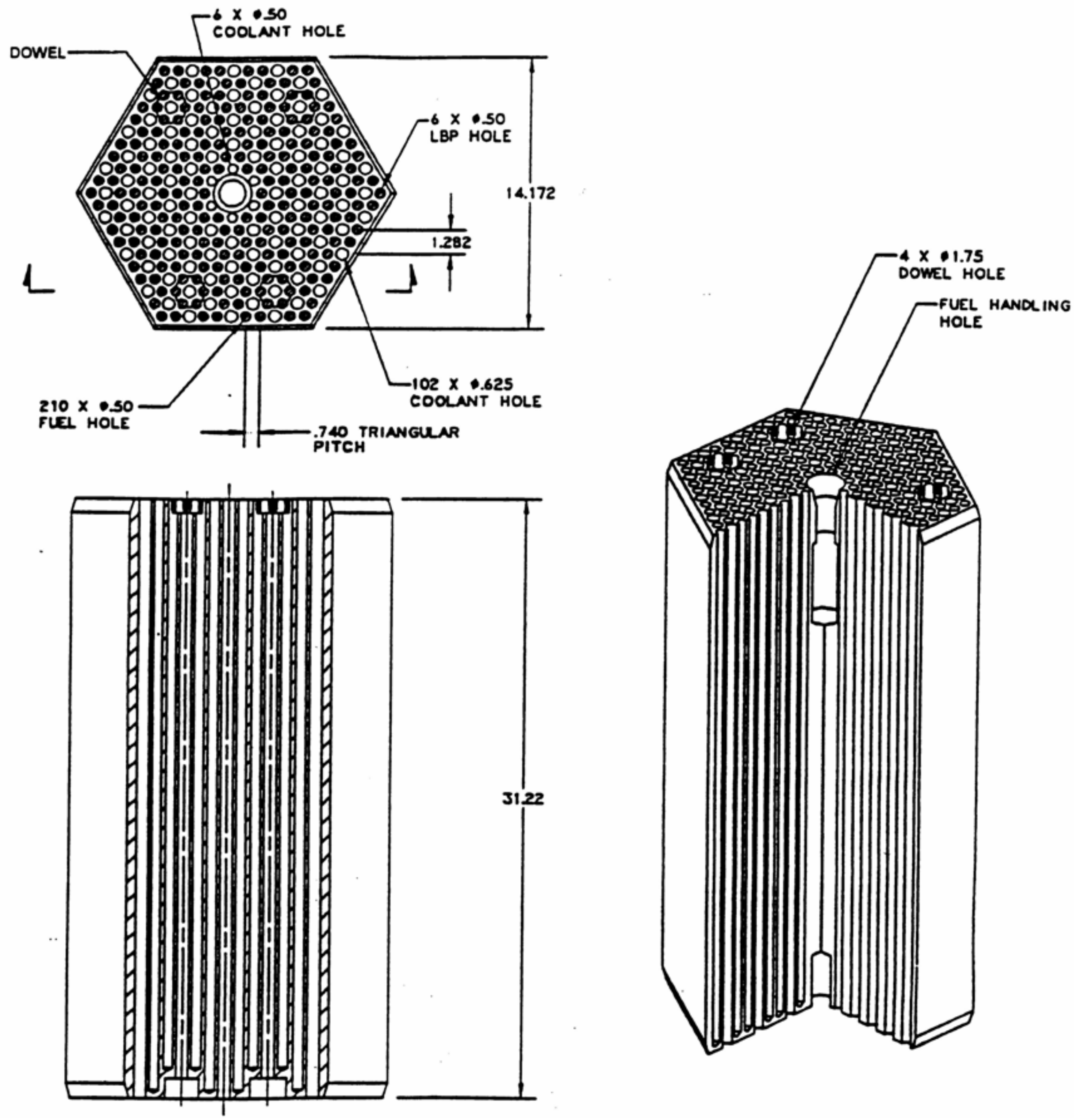


Figure 21: Standard fuel element design

Table 5: MHR Standard fuel element design parameters

Shape	Hexagonal Prism
Type of graphite	Nuclear Grade H-451
Mass of graphite per element	90 kg
Dimensions	794 mm (31.2 in.) in length
	360 mm (14.2 in.) across flats of hexagon
Volume	0.0889 m3
Total number of fuel holes	210
Number of fuel holes under dowels	24
Fuel hole diameter	12.7 mm (0.5 in.)
Fuel hole length	752.6 mm (29.63 in.) under dowels
	781.5 mm (30.77 in.) not under dowels
Number of fuel compacts per fuel hole	14 for holes under dowels
	15 for holes not under dowels
Number of fuel compacts per element	3126
LBP holes per element	6
LBP hole diameter	12.7 mm (0.5 in.)
LBP hole length	781.5 mm (30.77 in.)
Total number of coolant holes	108
Coolant hole diameter	15.88 mm (0.625 in.) for larger holes
	12.7 mm (0.5 in.) for the 6 smaller holes near the center of the block
Pitch of coolant/fuel-hole array	18.8 mm (0.74 in.)
Total mass of an average fuel element	122 kg
Mass of carbon in an average fuel element	110.7 kg
Mass of low-enriched uranium fuel in an average fresh fuel element	3.43 kg
Mass of natural uranium fuel in an average fresh fuel element	0.995 kg
Number of fissile particles in an average fuel element	1.35 E7
Number of fertile particles in an average fuel element	1.63 E6
Electrical energy generated by an average fuel element at discharge	0.637 MWe-yr

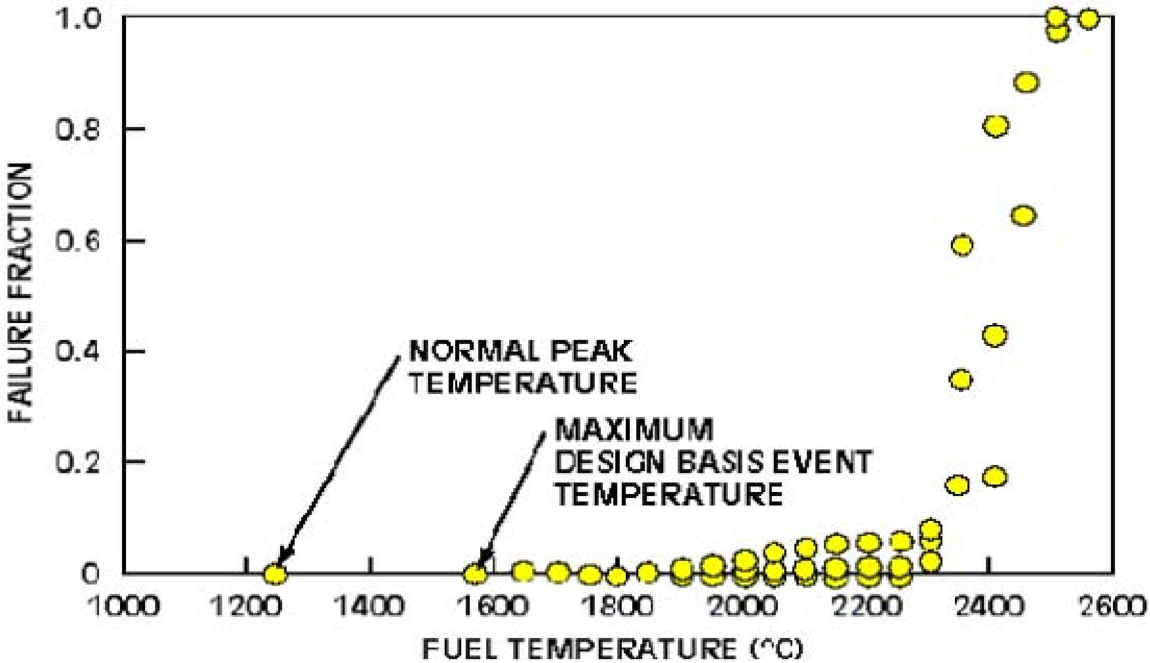


Figure 22: Fuel particles failure fraction

ATHENA: THERMAL-HYDRAULICS ANALYSIS

The two-phase flow thermal hydraulics code simulation RELAP5-3D version ATHENA, specifically intended for gas reactor, has been employed to study the transient behavior of the MHR. The following section is devoted to the presentation of Athena.

RELAP5 series capabilities

The RELAP5 series has been developed by the Idaho National Engineering and Environmental Laboratory (INEEL) to simulate thermal-hydraulics phenomena in light water nuclear reactors. RELAP is the acronym for Reactor Excursion Leak Analysis Program [19].

The code models the coupled behavior of the reactor coolant and the structure core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow.

Past records

The development of the RELAP5 series started in 1976. In almost 30 years, the code has been continuously improved going from the first version RELAP5/MOD1 to successively RELAP5/MOD1.5, RELAP5/MOD2, RELAP5-3D and finally ATHENA. Originally, the code was designed to analyze complex thermal-hydraulic interactions that occur during either postulated large or small break loss-of-coolant accidents in Pressurized Water Reactors. However, as development continued, the code was expanded to include many of the transient scenarios that might occur in thermal-hydraulics systems.

ATHENA code

ATHENA is the latest code in the RELAP5 series [20]. It benefits from the accumulation of experimental data (for instance enhanced thermodynamic properties) and experience in core behavior modeling during severe conditions. It also contains several important numerical enhancements over previous versions of the code such as a new matrix solver and improved time advancement logic.

Moreover, ATHENA besides being able to simulate the behavior of reactor coolant system during transient and a wide variety of hydraulic and thermal transient in both nuclear and non-nuclear systems can be used for simulating space reactors, fast breeder reactor and gas cooled reactor. The latter mentioned capability is the one that has been utilized during the project.

Top level code organization

ATHENA is written in Fortran 77. It is organized in a modular fashion structure where the various procedures are separated in subroutines.

The top level structure of the code, shown on Figure 23, consists of three subroutines. INPUTD and STRIP are dedicated to the interface with users whereas TRNCTL is the computational kernel of the code.

The input block (INPUTD) checks, processes and allocates the input data so that they are usable by the TRNCTL.

The strip block (STRIP) processes the simulation data to make the results compatible with other computer software.

The transient/steady-state block (TRNCTL) is the central part of ATHENA. It handles the steady-state and the transient options by calling the next lower level routines: TRNSET, TRAN and TRNFIN.

The subroutines TRNSET and TRNFIN serve only for the input/output of TRAN which controls the transient advancement of the solution. This last block is the most demanding in term of memory utilization and it is where most of the computational time is spent. Figure 24 shows the main functions associated with TRAN.

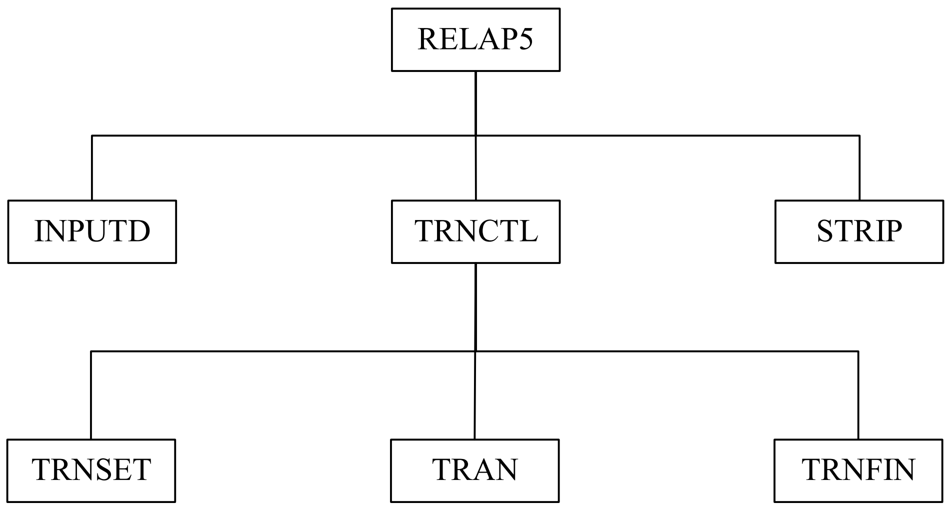


Figure 23: ATHENA top level block structure

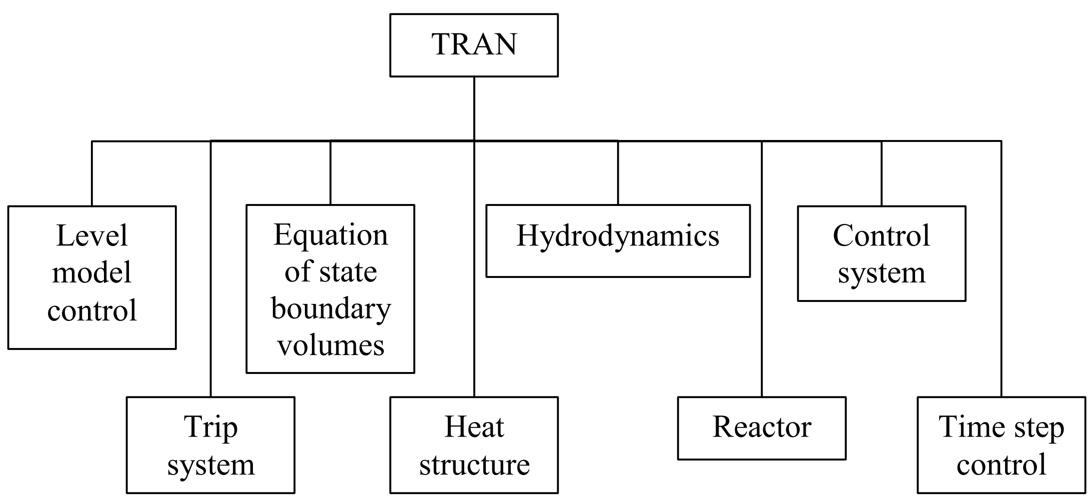


Figure 24: Functional modular structure of transient calculations in ATHENA

Thermal-hydraulics

Hydrodynamic model

The ATHENA hydrodynamic model is based on a transient, two-fluid and two-phase flow model. ATHENA also contains options for simpler hydrodynamic models such as homogeneous flow or thermal equilibrium. The different options can be used independently or in combination.

Equations of motion used in ATHENA hydrodynamic model use volume and time-averaged parameters of the flow. For phenomena transverse to the flow, such as friction or heat transfer, bulk properties associated to empirical transfer coefficient formulations are used. The system model is solved numerically using a semi-implicit finite-difference technique.

ATHENA's structure is based on building blocks that are divided into four fundamental groups: thermal-hydraulic, heat structures, trips, and control variables. The thermal-hydraulic group is composed of components designed to simulate flow paths and fluid-handling equipment. Heat structures are designed to simulate material mass and the interactions between the material mass and the fluid in the fluid passages. Trips are designed to simulate the signals that initiate equipment actions of various sorts (e.g., turning on a pump at a desired time or causing a valve to open at one pressure but close at another pressure). Finally, control systems are designed to give the code modeling added capability by allowing equipment control systems (e.g., proportional- integral-differential controllers and lead-lag controllers).

Field and continuity equations

The ATHENA thermal-hydraulic model solves eight field equations for eight primary dependent variables. The primary dependent variables are pressure, phase specific internal energies, vapor/gas volume fraction (void fraction), phase velocities, non-condensable quality, and boron density. The independent variables are time and distance. The secondary dependent variables used in the equations are phase densities, phase temperatures, saturation temperature.

Closure of the field equations is provided through the use of constitutive relations and correlations for such processes as inter-phase friction, inter-phase heat transfer, wall friction, and wall heat transfer.

The basic two-fluid non-equilibrium differential equations that form the basis for the hydrodynamic model consist of two-phase mass continuity equations, two-phase momentum equations, and two-phase conservation of energy equations. The equations are recorded in differential stream tube form with time and one space dimension as independent variables and in terms of time and volume-averaged dependent variables.

Heat transfer

Heat conduction, heat convection, radiation transfer, natural convection and wall heat transfer are the heat transfer phenomena taken into account in the detailed ATHENA thermal-hydraulics model.

The correlations and methods used in ATHENA to obtain the information necessary for each heat transfer phenomena depend on a lot of parameters including flow regime which determination is mostly empirical.

The ATHENA's heat transfer package decision logic leads to the selection of the appropriate heat transfer correlations and coefficients.

Energy source term and reactor kinetics

The primary energy source for a nuclear reactor is the core. To model the energy generation in the core, volumetric heat sources are placed into appropriate heat structures. Heat structures represent the selected, solid portions of the thermal-hydrodynamic system. Being solid, there is no flow, but the total system response depends on heat transferred between the structures and the fluid, and the temperature distributions in the structures are often important requirements of the simulation.

In ATHENA, the power generated in the core can be specified from a table, or determined by point-reactor kinetics with reactivity feedback. The power is modeled as an internal heat source in heat structures and weighting factors may be used to distribute the energy throughout the active core. The point reactor or space-independent kinetics approximation is adequate for cases in which the spatial power distribution remains nearly constant.

The code provides five reactivity feedback models that can be taken into account for transient analysis. In these models the field equations are coupled to the point kinetics permitting simulation of feedback effects between thermal-hydraulics and neutronics. The point kinetics formulation uses core-average fluid conditions, weighting factors and feedback coefficients to determine a total reactivity to drive the kinetics calculations.

The total reactor power is the sum of immediate fission power and decay heat of activated fragments. The immediate (prompt and delayed neutron) power is that released at the time of fission and includes power from fission fragment kinetic energy, prompt gammas, and neutron moderation. Decay heat is generated as the fission products undergo radioactive decay. There are three options for computing reactor power: fission only; fission and decay heat; or fission, decay heat, and actinide decay power. Actinide decay power is the power resulting from production of $_{239}$U by neutron absorption in $_{238}$U and subsequent two-stage beta decay to $_{239}$Pu.

Users can input fission products decay data or they can use one of the three available sets built into the code: an approximation to the 1973 ANS Proposed Standard (default set), the exact 1979 ANSI/ANS Standard, or the exact 1994 ANSI/ANS Standard.

Run processing

Initial and boundary conditions

In general, the initial conditions are a set of the dependent variables of the problem. The hydrodynamic model requires four thermodynamic state variables in each volume and the velocities at each junction. Heat structures require the initial temperature at each node, control systems require the initial value of all control variables, and kinetics calculations require initial power and reactivity.

Boundary conditions may be required for hydrodynamic models, heat structures, or control components if these parameters are governed by conditions outside of the problem boundaries. The hydrodynamic boundaries of a system are usually modeled using time-dependent volumes and junctions. Examples of these could be mass and energy inflows or an externally specified control parameter.

Obtaining a desired simulation is very dependent upon proper specification of initial and boundary conditions.

Steady-state initialization

- Steady state definition and convergence criteria

The fundamental concept of steady-state is that the state of a reactor system being modeled does not change with respect to time. In the hydrodynamic solution scheme, three terms can be monitored whose variation in time include the variation of all of the other terms. These three terms are the thermodynamic density, internal energy, and pressure. Furthermore, these three terms can be combined into a single-term, enthalpy. Hence, monitoring the time variation of enthalpy is equivalent to monitoring the time variation of all of the other variables in the solution scheme.

- Self-initialization option

ATHENA contains an option to perform steady-state calculations. This option uses the transient hydrodynamic, kinetics, and control system algorithms and a modified heat structure thermal transient algorithm to converge to a steady-state. The differences between the steady-state and transient options are that a lowered heat structure thermal inertia is used to accelerate the response of the thermal transient, and a testing scheme is used to check if steady-state has been achieved. Also, in case of steady-state calculations, the desired core power and other initial conditions are specified through the input table without kinetics package activation.

When steady-state is achieved, the run is terminated, thus saving computer time. The results of the steady-state calculation are saved so that a restart can be made in the transient mode. In this case, all initial conditions for the transient are supplied from the steady-state calculation.

Transient run

All transient analysis problems require initial conditions from which to begin the transient simulation. Usually, the initial conditions will correspond to a steady-state, with the transient initiated from a change of some boundary condition.

RESEARCH

ATHENA model

Nodalization

The Athena model is meant to study the coolant flow and the temperature distribution in the reactor vessel. The input file was provided by General Atomics and is based on their GT-MHR references. These references have been modified to match the H2-MHR expected design (Appendix A).

The input file contains a very simplified 1-D model of the MHR. Components outside the reactor vessel are not taken into account, as well as most of the vessel internals. The gas turbine and the hydrogen production plant are ignored. Inlet and outlet of the reactor vessel are modeled with time dependent volumes and junctions that imitate the H2-MHR version secondary system.

The nodalization of the MHR is shown on Figure 25. The model is made of three structures: the reactor vessel, the containment, and the Reactor Cavity Cooling System (RCCS).

The reactor vessel hosts the inlet plenum, the riser, the upper and lower plenums, and the core which contains the active core, the inner and outer reflectors and the associated coolant channels. The inlet, upper and lower plenums (volume 110, 140 and 160 respectively) are modeled with single volume branches. The riser (volume 130) is modeled by one channel cut into 13 axial nodes. The 12 lowest nodes correspond to the core barrel whereas the top one matches the upper head.

Three parallel channels (volumes 152, 154, and 156) model the three rings of the active annular core. Each channel is sliced into 12 axial nodes. 10 nodes correspond to the active core levels while both ends correspond to the upper and lower reflectors. Similarly, two channels, also sliced into 12 axial nodes, represent the by-pass flows in the inner and outer reflectors (volumes 142 and 145).

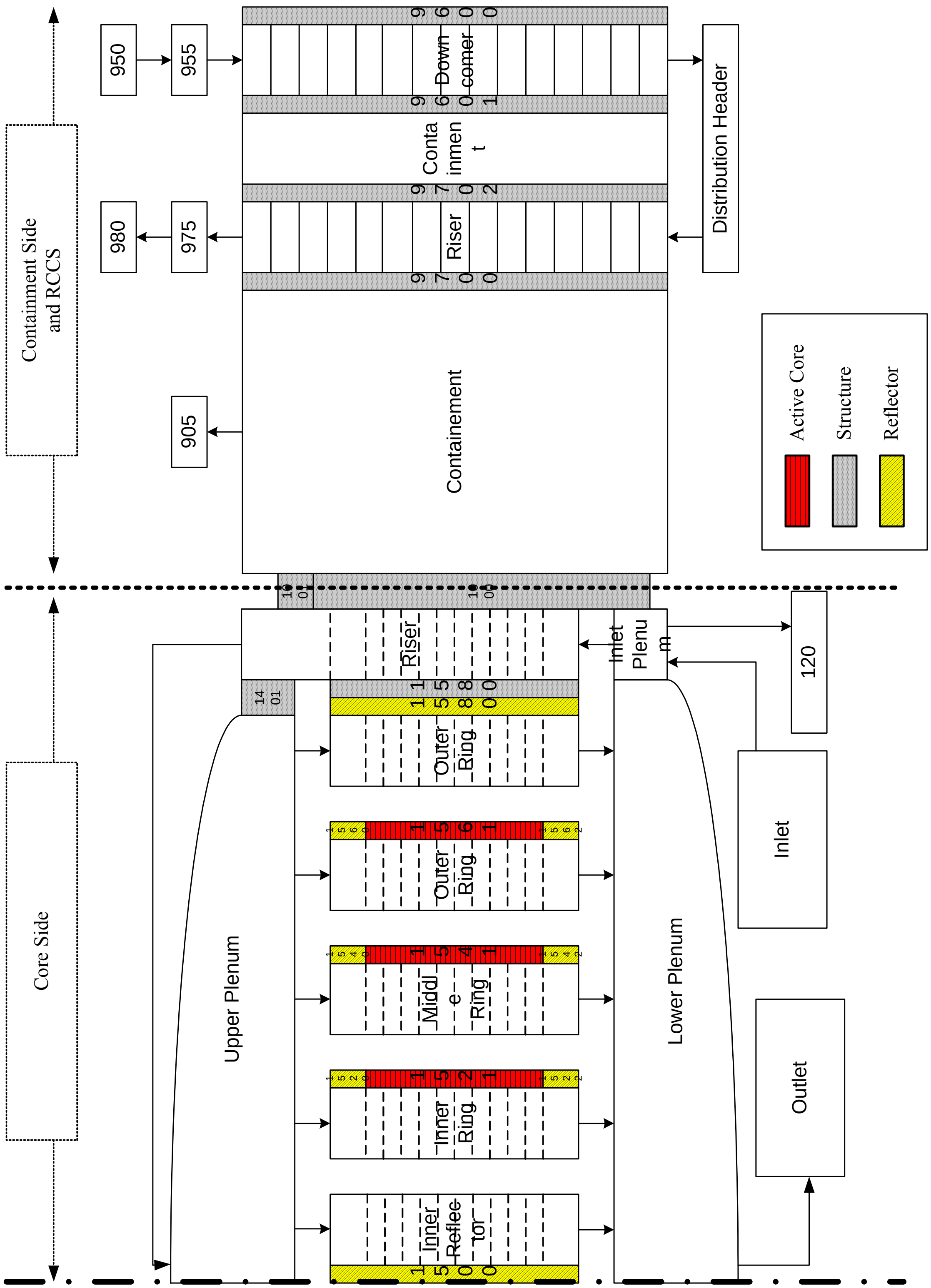


Figure 25: Core nodalization (core vessel, containment and RCCS)

The helium's journey into the reactor vessel starts in the inlet plenum. Then it flows up between the core barrel and the vessel wall, through the riser, and reaches the top of the vessel in the upper plenum. There, the coolant splits into the five core channels, flows downward into the lower plenum where the various flow paths recombine and exits the reactor vessel. The shutdown cooling system (volume 120) located at the bottom of the reactor vessel is outside of the main flow path. Helium there is stagnant during normal operation but remain thermally tied to the main flow.

The containment structures contain the reactor cavity and a pressurizer. They both contain nitrogen which simulates satisfyingly air. The reactor cavity (volume 900) is a large single volume that communicates thermally with the reactor vessel and the RCCS. It is vented through its connection to volume 905. Volume 905 is a time dependent volume that acts like a pressurizer in a PWR. By fixing its pressure, nitrogen can move in and out of the containment volume as the temperature of the gas in the containment changes. This allows the pressure in the containment to remain constant as the temperature changes. Control features ensuring that the containment does not over-pressurize is a requirement in the design of the Next Generation of Nuclear Plant (NGNP).

The RCCS designed to remove heat by natural means is located at the inside periphery of the containment structure. The RCCS is filled with air. The intake and outlet of the system, represented respectively by the time dependent volumes 950 and 980, is the atmosphere. Air circulation in the RCCS is supposed to be natural. The flow paths are as follow. From the intake, air goes through the inlet plenum (955), flows down the down-comer (volume 960) which is attached to the containment wall, gets to the lower distribution header (volume 965) where it is supplied to the riser channels represented by a single channel (volume 970); it is then recombined in the outlet plenum and heated air is discharged back to the atmosphere. The down-comer and the riser volumes are divided into 15 axial nodes to measure the temperature profile in the channels.

The RCCS uses air. The problem is that ATHENA does not have air as a primary working fluid. The solution is to setup the RCCS working fluid as H_2O and define air as a non-condensable gas with a mass fraction of 1.0. For each of the RCCS hydrodynamic volumes, there is a non-condensable option that sets the quality equal to zero and thus treats the fluid in the volumes as a dry non-condensable i.e. air in this case.

Heat structures compose most of the structural components of the model. They represent the fuel elements, the reflectors, the core barrel, the vessel cylinder, the upper head, part of the lower head, the RCCS surfaces and the containment walls. The heat structures match the node cutting arrangement of the adjacent volumes. Heat structures are also provided with radial node array to take into account different material. For instance fuel elements, illustration on Figure 26, are composed of graphite H-451 and fuel compact material (heat structures 1521, 1541 and 1561). The core barrel is made of Incoloy alloy 800 (heat structure 1580). The RCCS walls are made of stainless steel. Heat structure 1960 models the interface between the core containment and the ground (table 960). It is illustrated on Figure 27.

The heat structures govern the heat generation and most parts of the heat transfer mechanisms.

The model simulates radial and axial conduction in the core and reflectors. Radiation heat transfer is modeled from the core barrel to the reactor vessel, from the vessel to the RCCS, and from the RCCS to the containment.

The model also includes space-independent point kinetics calculations using the separable feedback effects option (see power distribution section later).

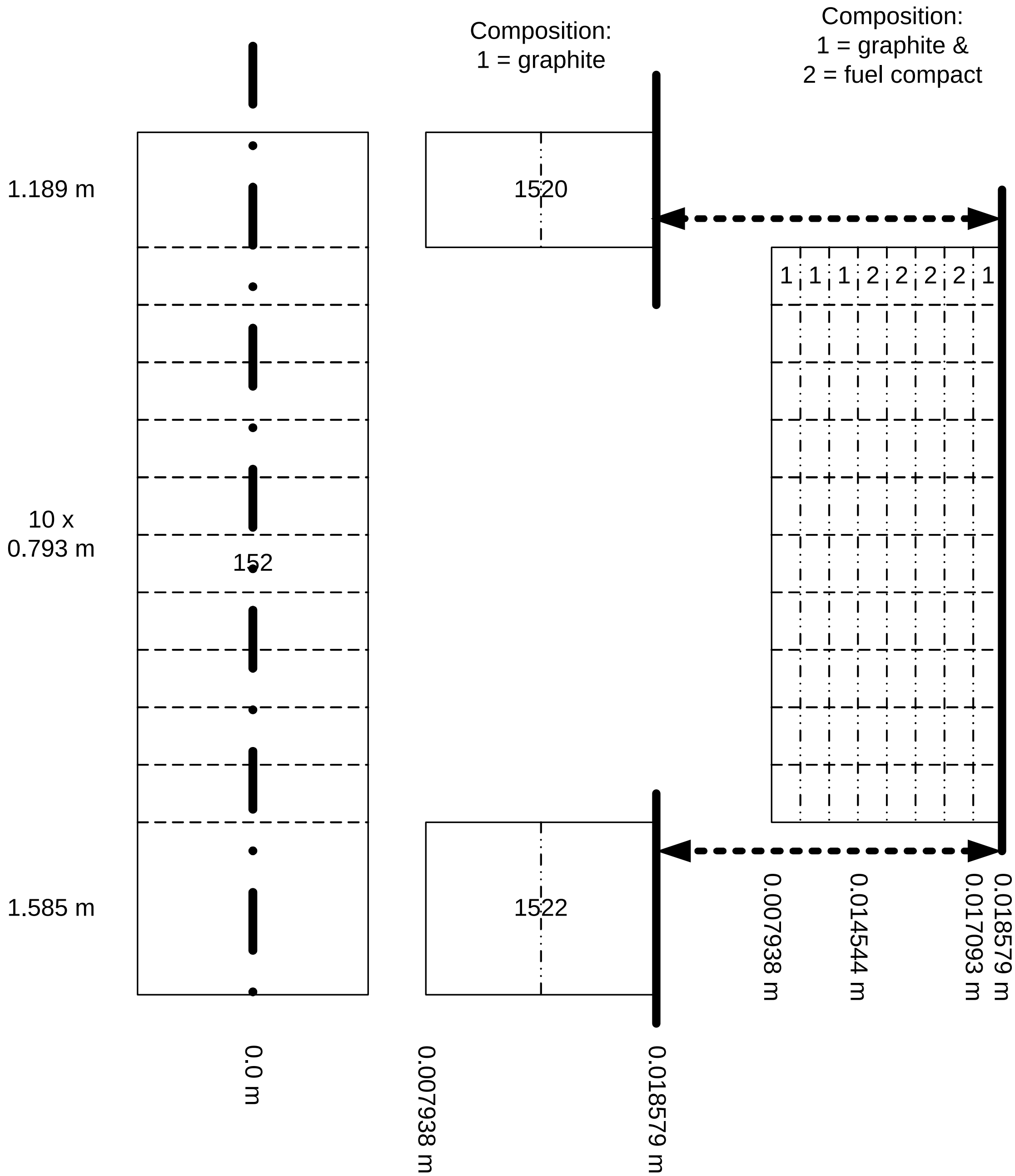


Figure 26: Fuel element heat structure

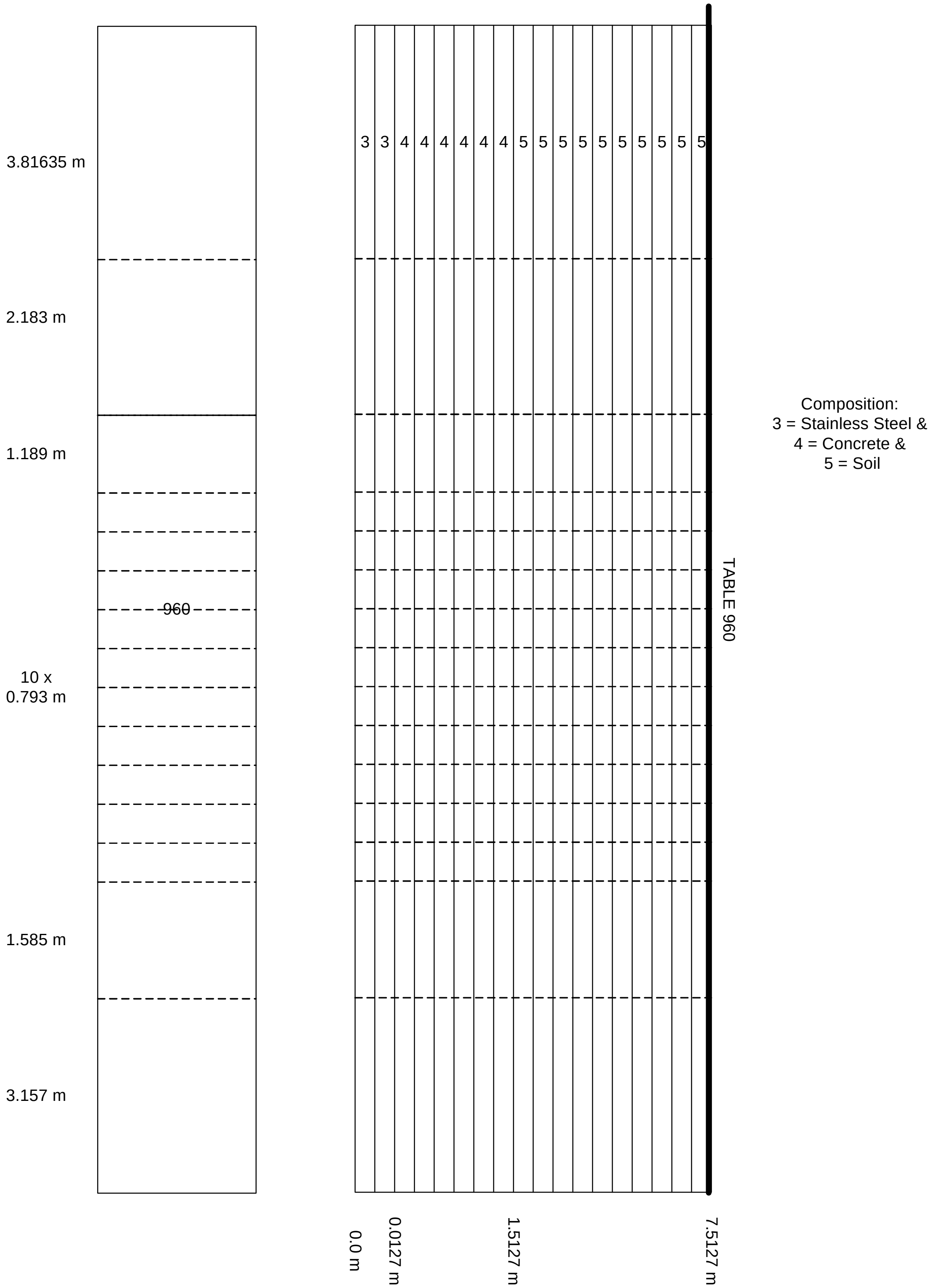


Figure 27: Heat structure modeling interface between reactor and ground

Steady state design conditions

In this section, the initial data sets for thermal-hydraulics volumes, heat structures and neutronics are presented and discussed.

The steady-state conditions for the H2-MHR calculated by General Atomics are presented in Table 6. These figures are compared to the project's results in the results chapter.

The fundamental requirement for the steady state design is the value of the output temperature. It must be 1000°C (1273°K). Most of parameters can be moved to obtain this value. One of the only fixed value is 490 °C (= 763.15 K) for the inlet coolant temperature.

Table 6: GA Steady state conditions for H2-MHR

Core power, MW(t)	600
Primary coolant system	
Reactor inlet temperature, °C	491
Reactor outlet temperature, °C	1000
Mass flow rate, kg/s	226
Bypass flow, %	10
Reactor inlet pressure, MPa	7.07
Reactor differential pressure, kPa	46.2
RCCS	
Pressure, MPa	0.1
Mass flow rate, kg/s	14.4
Inlet air temperature, °C	43
Outlet air temperature, °C	304

Initial data

For a steady state run, when using the initialization option, the initial conditions data are not the most essential part of the Athena input file. Most of the cards require initials values (temperature, pressure) to initiate the calculations but these values need not to be the ones specified in the GA design. Indeed, with the steady-state initialization option, the diverse parameters are forced to convergence. However, inputting values close to steady-state conditions leads to faster convergence.

It would be petty and useless to describe all of the thermal-hydraulics volume initial data. The initial temperature in the active core channel nodes are given in Figure 28 as an example. The initial pressure value is set at 70 bars for all the volumes in the core. The pressure and temperature initial values are the atmospheric values for the containment and the RCCS.

The initial conditions for heat structures and neutronics model are given in the power distribution chapter.

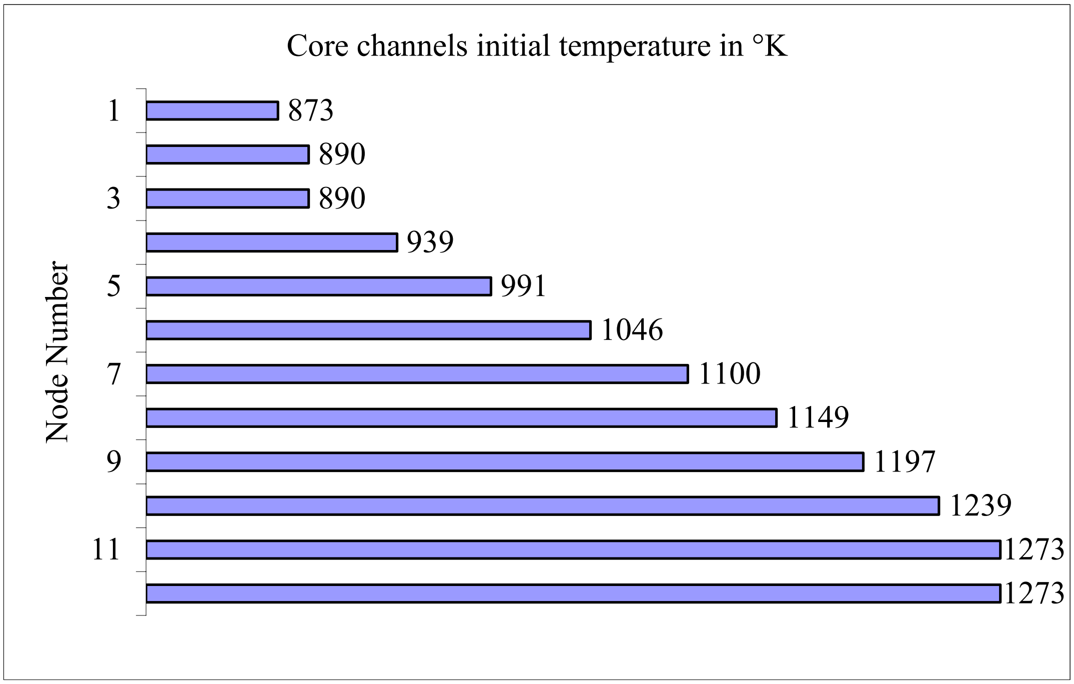


Figure 28: Initial temperature in core channels

Boundary conditions

The boundary conditions issue is more complex. Athena can take into account a large number of cases. General Atomics' input file provides all the boundary conditions for the Steady State conditions.

The ground at the interface with the containment building is assumed to be at a constant temperature: 300°K.

The pressure and temperature conditions in the reactor depend essentially on the set up of the inlet and outlet volumes and junctions simulating the secondary system.

The inlet flow is controlled to achieve the desired helium outlet temperature of 1000 °C.

The outlet pressure is controlled to achieve the desired pressure inlet of 7.0 MPa.

The next two sections show how these two controls are set up.

Outlet temperature controller

The model is supposed to maintain an outlet temperature of 1273°K by controlling the inlet flow. First of all, one needs to understand how the coolant flow is controlled. The coolant flow in the core is set up by a time-dependent junction. This component is located at the inlet of the reactor and is described thereafter:

Inlet time dependent junction

1050000	inflow	tmdpjun		
1050101	100010000	110000000	7.339	
1050200	1	0	cntrlvar	105
1050201	0.0	0.0	0.0	0.0
1050202	2000.0	0.0	2000.0	0.0

The junction connects component 100 to component 110. The junction area is 7.339 m². The mass flow rate is given by control variable 105. If control variable 105 is less than zero, the mass flow rate is 0 kg/s. If control variable 105 is greater than 2000, the mass flow rate is 2000 kg/s

Therefore, the inlet coolant flow in the core is actually controlled by control variable number 105. Control variable 105 is described thereafter:

Inlet flow controller

20501000	temperr	sum	1.0	0.0 0	
20501001	-1273.15	1.0	tempg	160010000	
20501050	inflow	integral	0.3	250.0	0
20501051	cntrlvar	100			

In a first step, RELAP subtracts the outlet temperature (component 160) from the temperature set point (1273.15 °K) and stores the results in control variable 100. This control variable is positive when the outlet temperature is greater than 1273.15°K and vice versa.

Then, RELAP evaluates control variable 105 by integrating control variable 100. The integral is $Y = Y_0 + S \bullet (V_1 + V_{10})\frac{\Delta t}{2}$. Where Δt is the time step, Y_0 and V_{10} are the values at the beginning of the time step and Y and V_1 are the values at the end of the time step. Y and V are respectively the values for control variables 105 and 100 at each time step. According to the definition: S = 0.3, Y_0 = 229 kg/s and V_{10} = 0 (for the first time step).

On Figure 29 one can check that the coolant flow increases when the temperature outlet is greater than the temperature set point and vice versa.

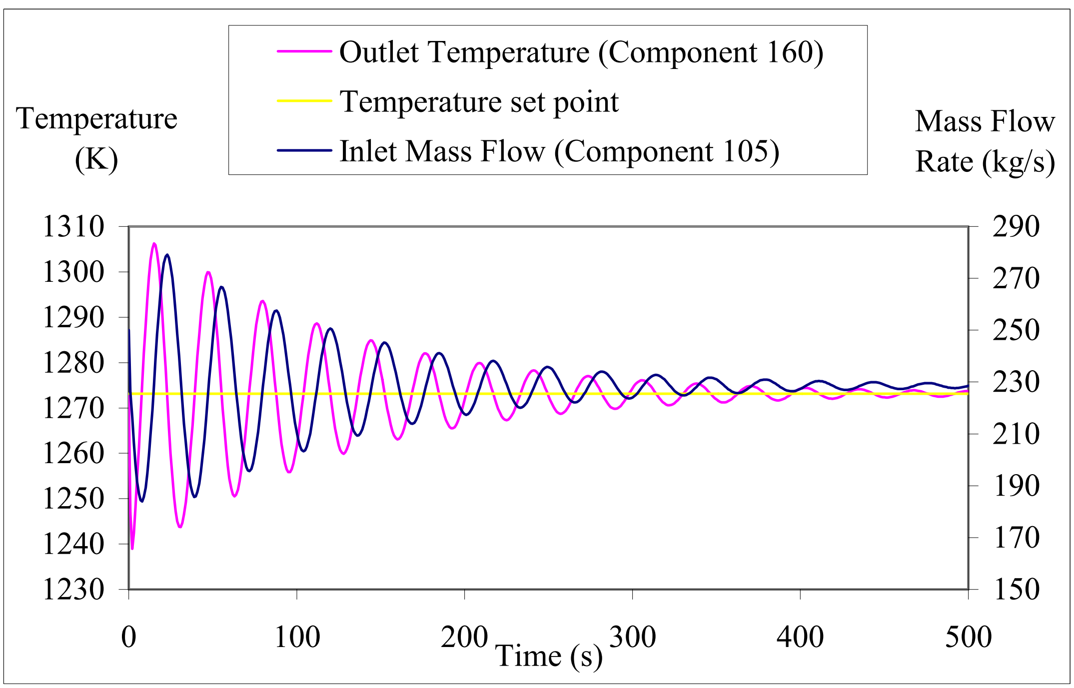


Figure 29: Outlet temperature controller

Therefore, the outlet temperature is controlled by the inlet mass flow rate. On Figure 30 one can check that the mass flow rate of the time dependent junction 105 follows the values of control variable 105.

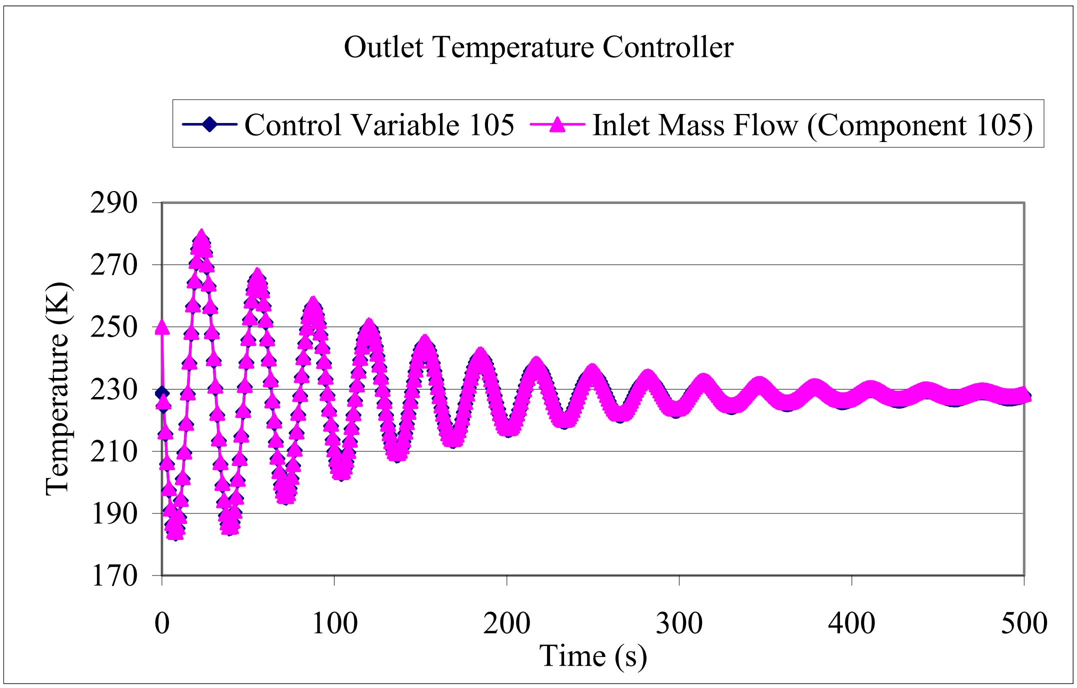


Figure 30: Inlet mass flow rate / control variable 105 relationships

Inlet pressure controller

The model is supposed to maintain the inlet pressure at 7.0 MPa by controlling the outlet pressure.

The inlet pressure controller works pretty much the same as the outlet temperature controller described previously. The core outlet pressure is set up by a time-dependent volume. This component is located at the outlet of the reactor and is described thereafter:

Outlet time dependent volume

1700000	outlet	tmdpvol							
1700101	1.0	0.0	1.0e3	0.0	0.0	0.0	0.0	0.0	10
1700200	003	0	cntrlvar	170					
1700201	5.0e6	5.0e6	1273.0						
1700202	8.0e6	8.0e6	1273.0						

The component volume flow area is 1.0 m² and its volume is 1000 m^3. The component's temperature is constant and equal to 1000 °C, the designed output temperature. The pressure is set up by control variable 170. If control variable 170 is less than 5 MPa, the pressure is 5 MPa. If control variable 170 is greater than 8 MPa, the pressure is 8 MPa.

Therefore, the core outlet pressure is actually run by control variable 170. Control variable 170 is described thereafter:

Inlet pressure controller

20501690	presserr	sum	1.0	0.0	0
20501691	7.0e6	-1.0	p	110010000	
20501700	outpress	integral	0.5	7.0e6	0
20501701	cntrlvar	169			

In a first step, the inlet pressure (component 110) is subtracted from the pressure set point (7.0 MPa) and the result is stored in control variable 169. This control variable is negative when the outlet pressure is greater than 7.0 MPa and vice versa. Then, RELAP evaluates control variable 170 by integrating control variable 169. The integral is $Y = Y_0 + S \bullet (V_1 + V_{10}) \frac{\Delta t}{2}$. Where Δt is the time step, Y_0 and V_{10} are the values at the beginning of the time step and Y and V_1 are the values at the end of the time step. Y and

V are respectively the values for control variables 170 and 169 at each time step. According to the definition: S = 0.5, Y_0 = 7.0 MPa and V_{10} = 0 (for the first time step).

On Figure 31 one can check that the outlet pressure decreases when the inlet pressure is greater than the pressure set point (7.0 MPa) and vice versa.

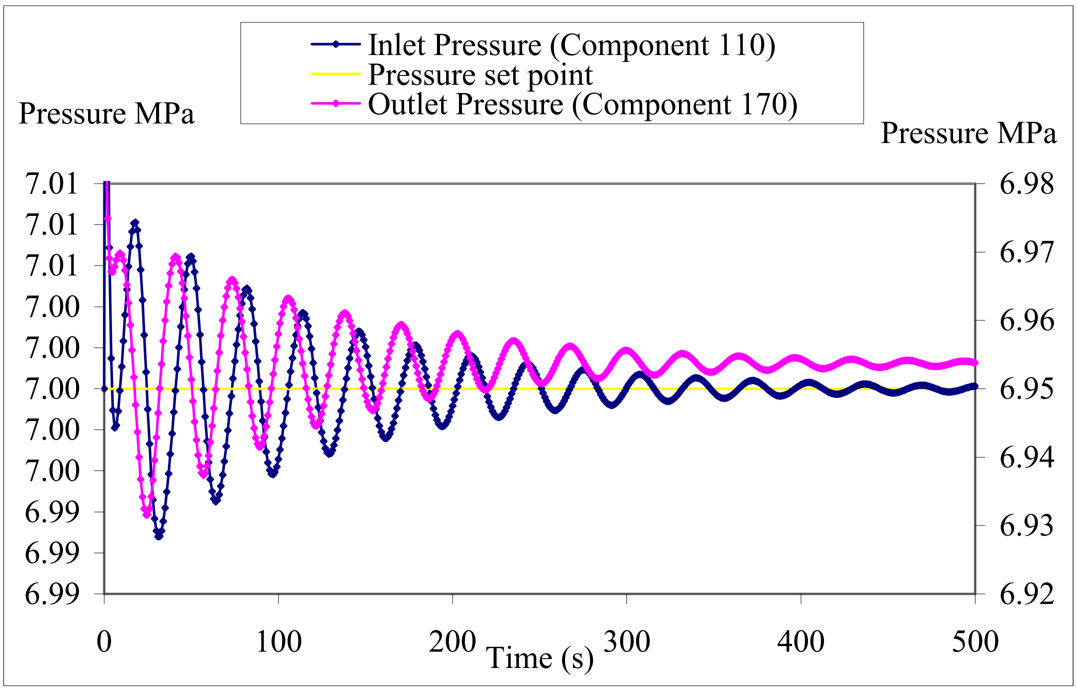


Figure 31: Inlet pressure controller

Therefore, the outlet pressure is controlled by the inlet pressure. On Figure 32 one can check that the pressure in the time dependent volume 170 follows the values of control variable 170.

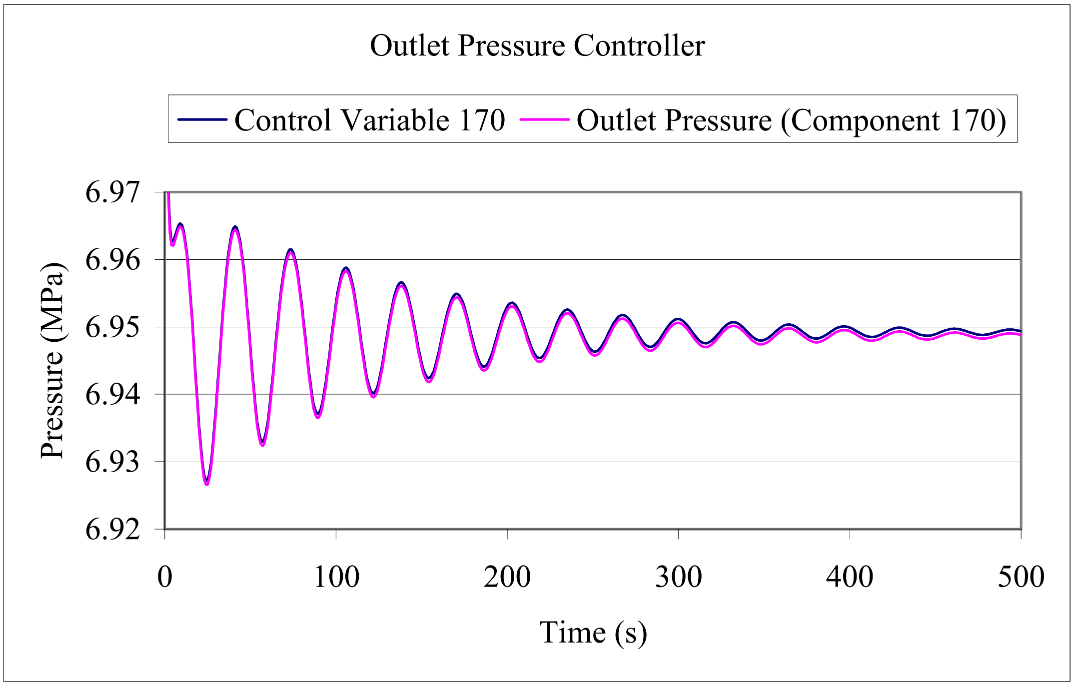


Figure 32: Inlet pressure / control variable 170 relationships

The pressure drop in the core is then set by the pressure difference between components 110 and 170.

Power distribution

The rated power of the MHR is set up to 600 MW_{th}.

The volumetric heat generation or source term, S(t), function of time and position, is defined by the equation: S(t) = Pf * Q(x) * P(t), where Pf is the axial peaking factors distribution, Q(x) is the radial peaking factors distribution and P(t) the rated power.

The input file provides the axial and radial power distribution in the core with peaking factors for the heat structures (see Figure 33 for radial distribution). The active core contains 102 fuel elements on each ten levels. The inner ring holds 30 elements per level with a peaking factor of 0.9832. The middle ring holds 36 elements per level with a peaking factor of 1.103. The outer ring holds 36 elements per level with a peaking factor of 0.911. The axial peaking factors are given in Figure 34. The maximum power is located in the middle ring slightly below core plane.

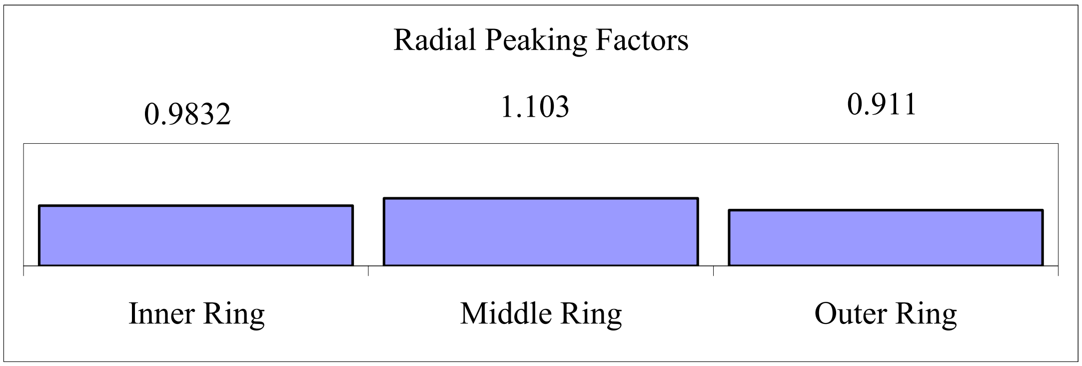


Figure 33: Radial peaking factors

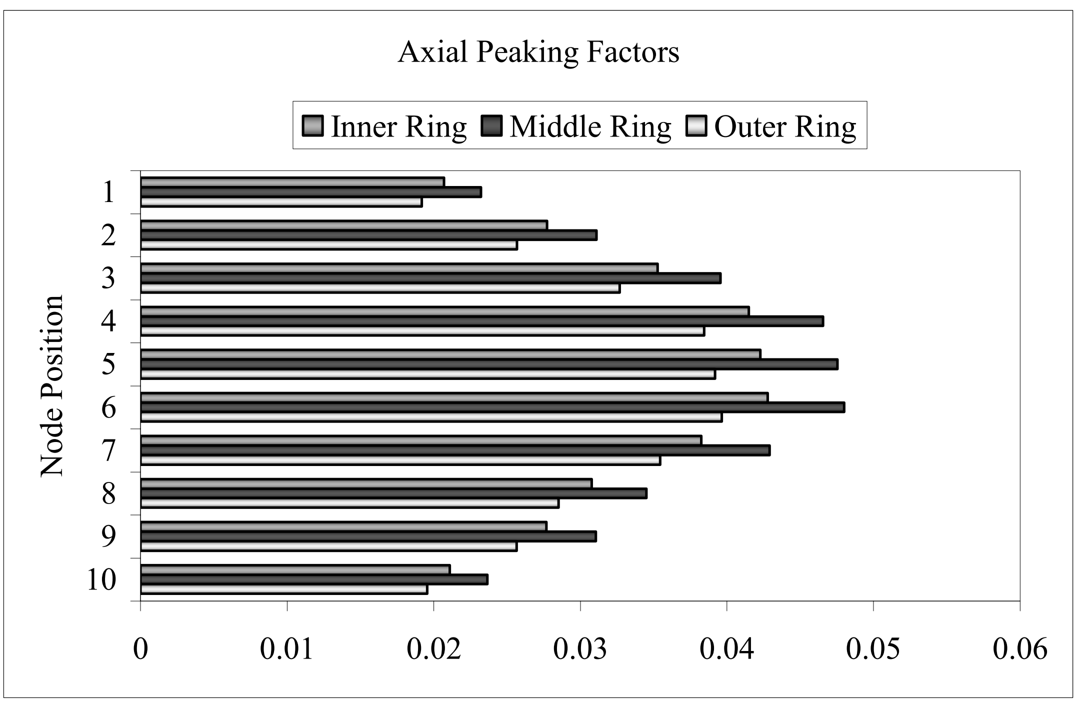


Figure 34: Axial peaking factors

In the input file, a point kinetics model is used. The reactivity feedback effects are taken into account with the "separable" option. This model provides weighting factors assuming each effect is independent (i.e. separability) from each other. Boron feedback is not provided (but there is no boron in the moderator). The model assumes nonlinear feedback effects from moderator and fuel density changes and linear feedback effects from moderator and fuel temperature changes.

The separable option uses the "Volume Weighting Factors" and the "Heat Structure Weighting Factors". The first mentioned factors apply to the feedback effects from thermal-hydraulic volumes whereas the second ones apply to the heat structures. The computes the feedback effects for a volume or for a heat structure first. Then the contribution to the total reactivity is obtained by multiplying the effect by the weighting factor.

The MHR model assumes a fuel Doppler feedback coefficient of $-5*10^{-5}$ Δk/°C. It is based on models developed for the Next Generation of Nuclear Plant.

In the input file the "Volume Weighting Factors" and the "Heat Structure Weighting Factors" have the same values. Similarly to the peaking factors, the volume and heat structure weighting factors have an axial and a radial distribution. The factors are presented in Figure 35 and Figure 36. The volume and heat structure weighting factors follow a cosine shapes.

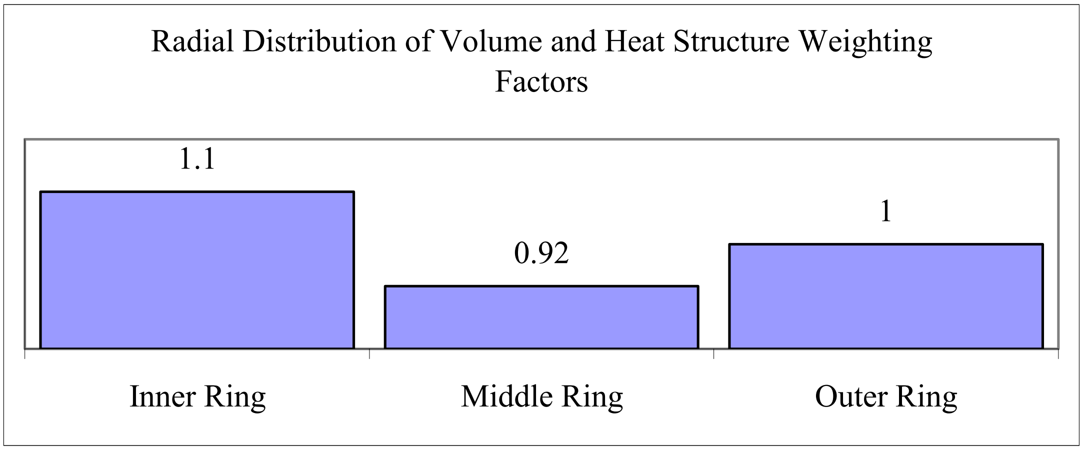


Figure 35: Radial distribution of volume and heat structure weighting factors

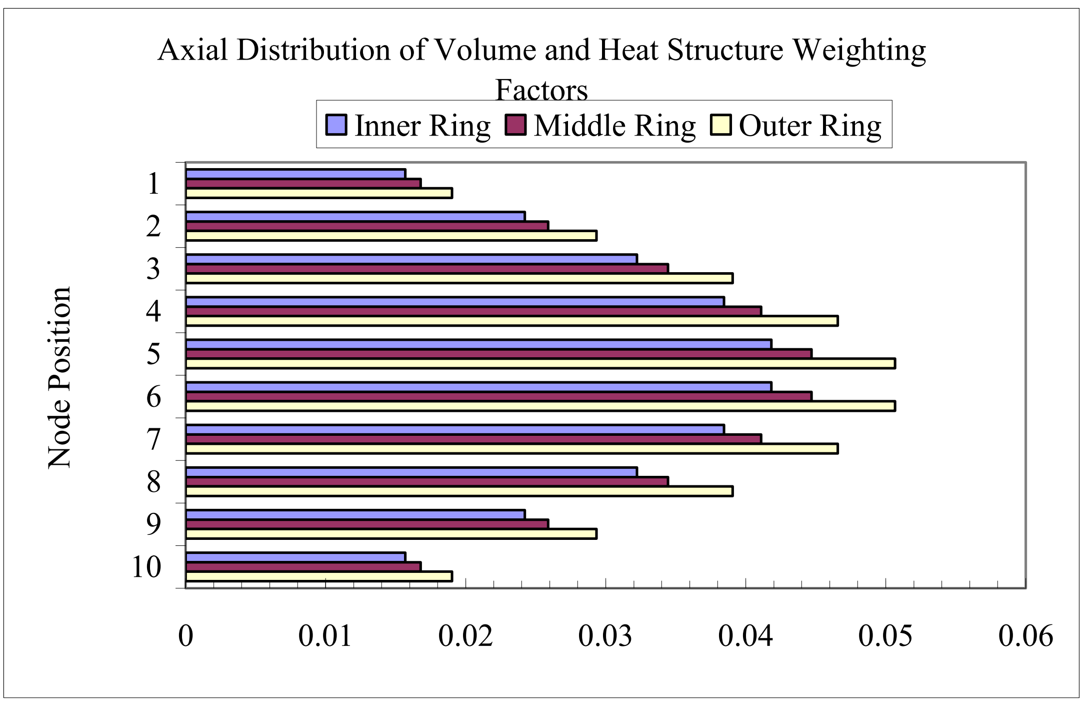


Figure 36: Axial distribution of volume and heat structure weighting factors

Accident scenarios

This section describes designs used for the two accident scenarios considered: loss of coolant and loss of flow. In both cases the reactor experiences a trip situation which conditions initiate the scram.

Scram design

- Scram reactivity change

A scram is a rapid shutting down of a nuclear reactor. This is typically done by the rapid insertion of control rods. It may occur either automatically or manually by the reactor operator. SCRAM stands for Safety Control Rod Axe Man.

An elementary way to set-up a scram consists in starting the scram at the same time as the trip. However, to be realistic, the scram should be set-up by alarming core vital parameter values such as abnormal temperatures and pressures. In both accident cases considered in the project, the scram is controlled by an internal controller.

When the scram occurs, complete insertion of the control rods is simulated. Figure 37 shows the reactivity changes as a function of time. It takes less then 5 seconds to insert almost nine dollars worth of negative reactivity.

In the input file the scram is triggered by a control variable monitoring the temperature in a volume located in the middle of the active core. If the temperature of node 8 in volume 154 is greater than 1300°K the scram is initiated. However, the configuration of the scram trigger need not be very selective. Because of the expected length of the transient - at least two days to reach peak temperature, the way the transient is modeled is not too critical. It is actually a conservative measure to delay the scram delay. That said, in real power plants the scram would have to be initiated as soon as possible to avoid core damage.

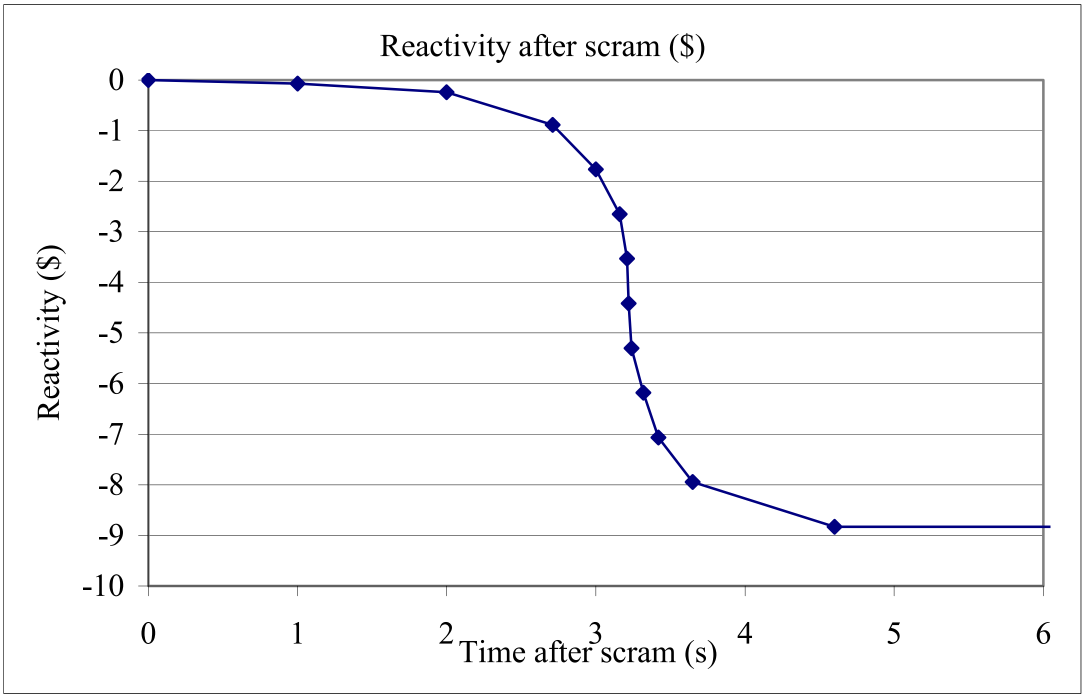


Figure 37: Reactivity versus time during scram

- Power versus time

The massive insertion of negative reactivity resulting from the scram stops the fission process. After that, the power generated in the fuel is basically only decay heat. The decay heat generation is based on GA data for the GT-MHR and is given in Figure 38. The decay heat decreases like an exponential decay. The power decreases very quickly at the beginning but levels off in the long term. After one minute the power is already reduced by a factor of two. However the power is greater than 2 MW until the fourth day.

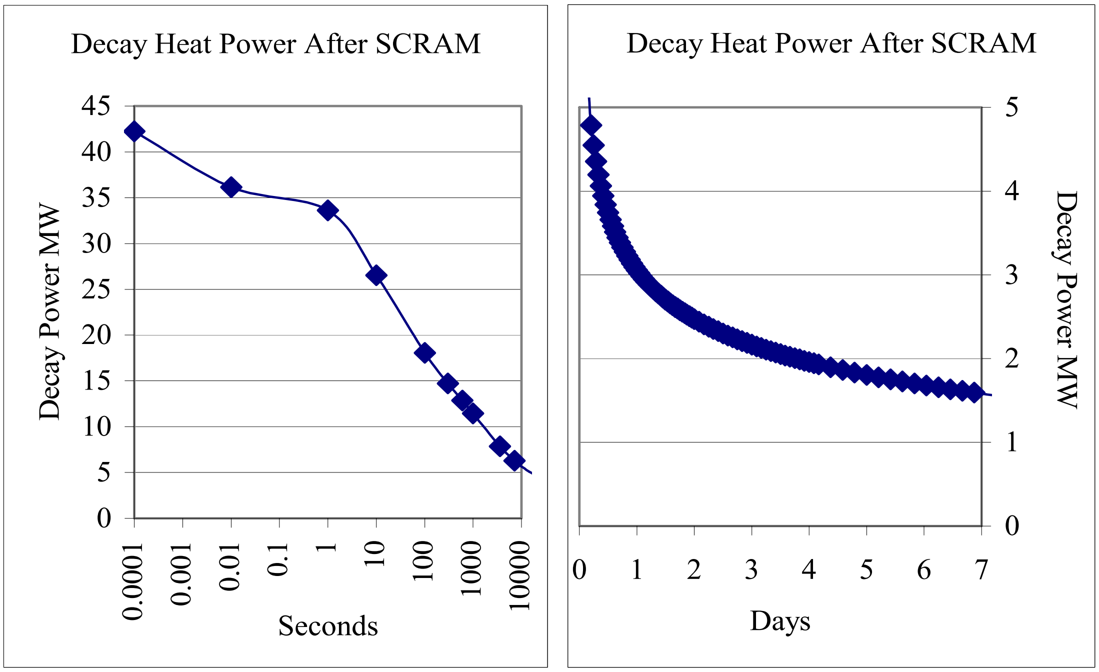


Figure 38: Decay heat power after SCRAM

Loss of coolant

A loss-of-coolant accident occurs when the coolant escape the system. It is usually the consequence of a breach on the piping system. Because of the pressure difference between the interior of the core and the outside, the coolant escapes the reactor until pressure equilibrium. During a loss-of-coolant, the reactor loses the coolant and experiences depressurization.

The loss of coolant is performed by adding two breaks on the coolant path flow. The set-up is illustrated on Figure 39. Considering the geometry of the reactor (compact, no pipes), breaks would most probably occur on the cross-vessel duct. The break has been located on the inlet plenum (volume 110). The break opens on volume 558 which simulates atmospheric conditions. Because the primary system is not a loop, the break has to be simulated on the outlet side as well to balance the sudden sucking effect. The volumes 100 and 170 are closed to avoid sucking the helium out of them.

The loss of flow is repeated for different size of break: 0.0001 m^2, 0.001 m^2, 0.01 m^2 and the maximum break area possible (i.e. the minimum flow area of the adjoining volumes (554 and 110)).

Loss of flow

A loss-of-flow accident occurs when the circulation of the coolant in the core stops. This is usually the consequence of a dysfunction in the circulating pumps. In case of a loss-of-flow, the reactor keeps its coolant and stays pressurized.

The loss of flow is performed by isolating the reactor from the secondary system. As illustrated on Figure 40 the junctions between the inlet volume (100) and the inlet plenum (110) and between the lower plenum (160) and the outlet volume (170) are closed. Therefore, there is no more forced flow in the core. Only natural heat transfer mechanisms are operational. The simulation stops the helium brutally.

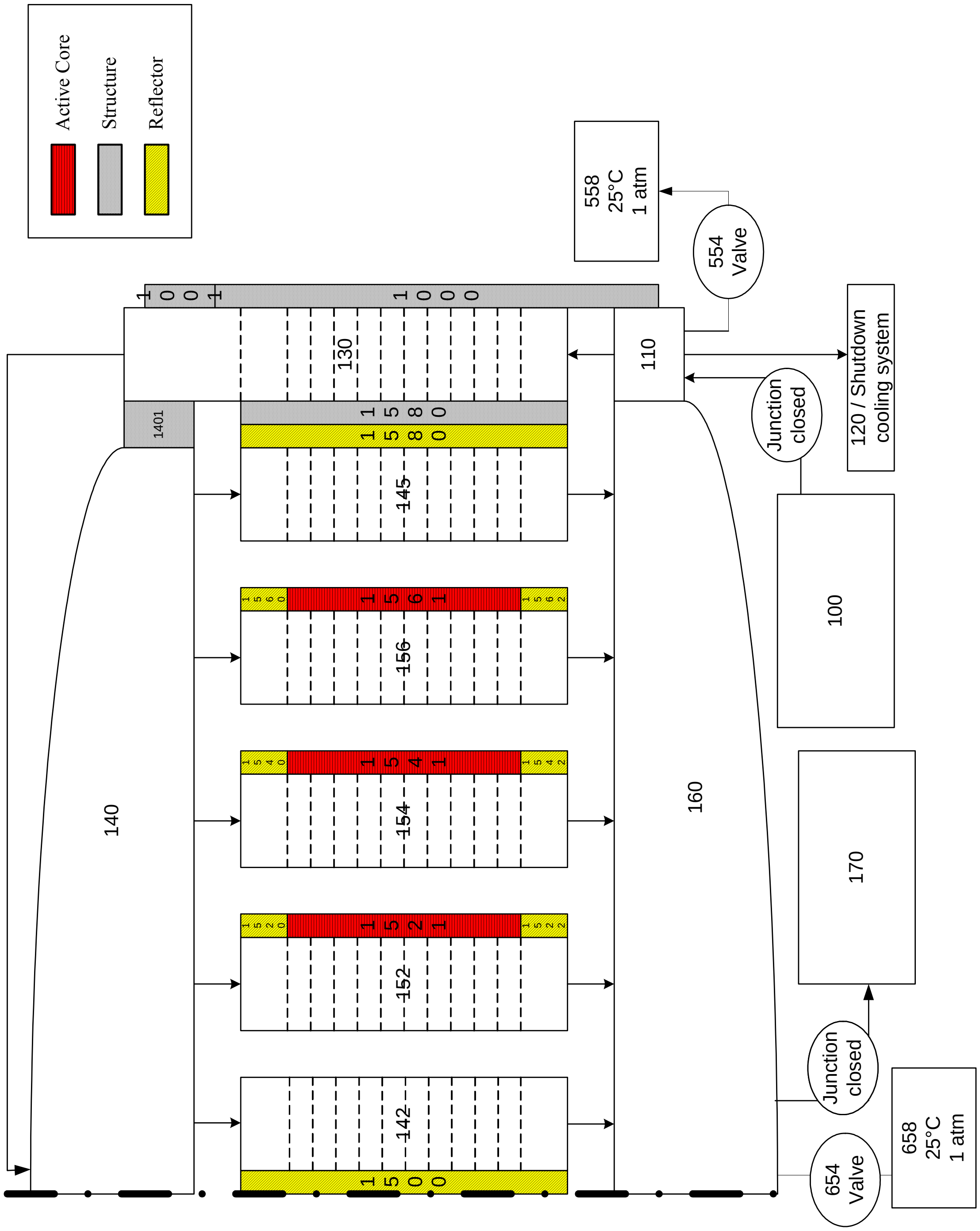


Figure 39: Loss of coolant

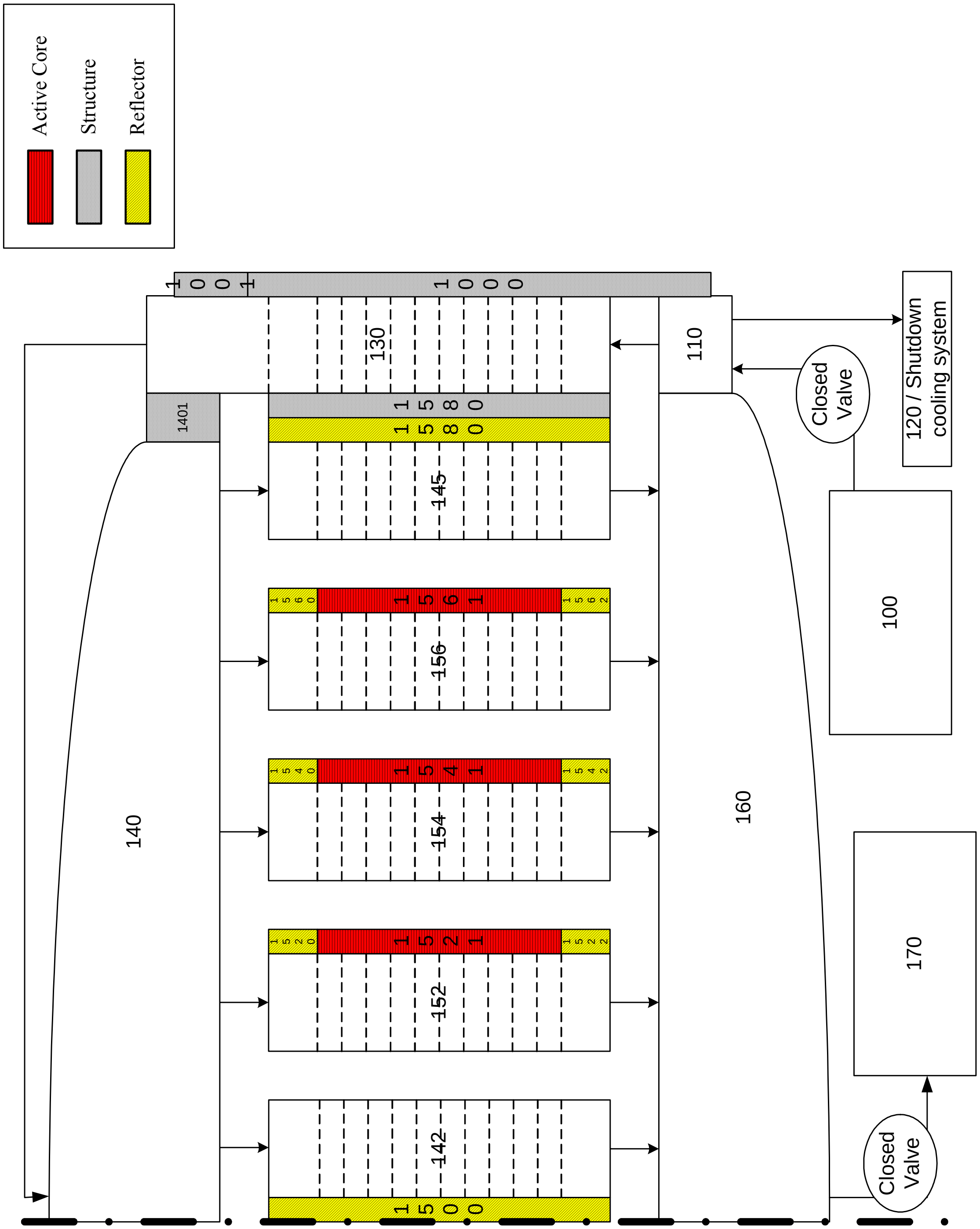


Figure 40: Loss of flow

RESULTS

Steady-state

The ATHENA model presented in the previous chapter was used to obtain steady-state conditions. The results of the calculations are shown in Table 7 where they can be compared to the General Atomics design. The General Atomics design parameters for the GT-MHR are also provided to give another insight.

Table 7: Steady-state conditions for MHR

	Project H2-MHR	GA H2-MHR	GA GT-MHR
Core power, MW(t)	600	600	600
Primary coolant system			
Reactor inlet temperature, °C	490	491	491
Reactor outlet temperature, °C	1000	1000	850
Maximum Fuel Temperature, °C	1251	1276	1218
Mass flow rate, kg/s	228.6	226	320
Bypass flow, %	10	10	20
Ratio of maximum flow to average flow	1.08	1.22	1.07
Ratio of minimum flow to average flow	0.85	0.64	0.89
Reactor inlet pressure, Mpa	7	7.07	7.07
Reactor differential pressure, kPa	51.34	46.2	47.6
RCCS			
Pressure, Mpa	0.1	0.1	
Mass flow rate, kg/s	14.1	14.4	
Inlet air temperature, °C	43	43	
Outlet air temperature, °C	268	304	

The results obtained are very close to the General Atomics H2-MHR design parameters. The main discrepancies concern the maximum and minimum flow to average flow ratios. They are respectively off by 11.5 and 32.8 %. However, there are still some doubts on the General Atomics calculation method. The ratios given by this project are using only the three active channels to calculate the average. If the five channels are used, the results are even more distant.

Figure 41 shows the Steady-State initialization results for the temperatures and the mass flow rates of the coolant in different locations of the core. The parameters start from their initial values and oscillate until reaching constant value. The speed at which the parameters achieve steady-state depends on the initial values and the options chosen for the Steady-State initialization mode. The values obtained will be used as a starting point for the transient runs.

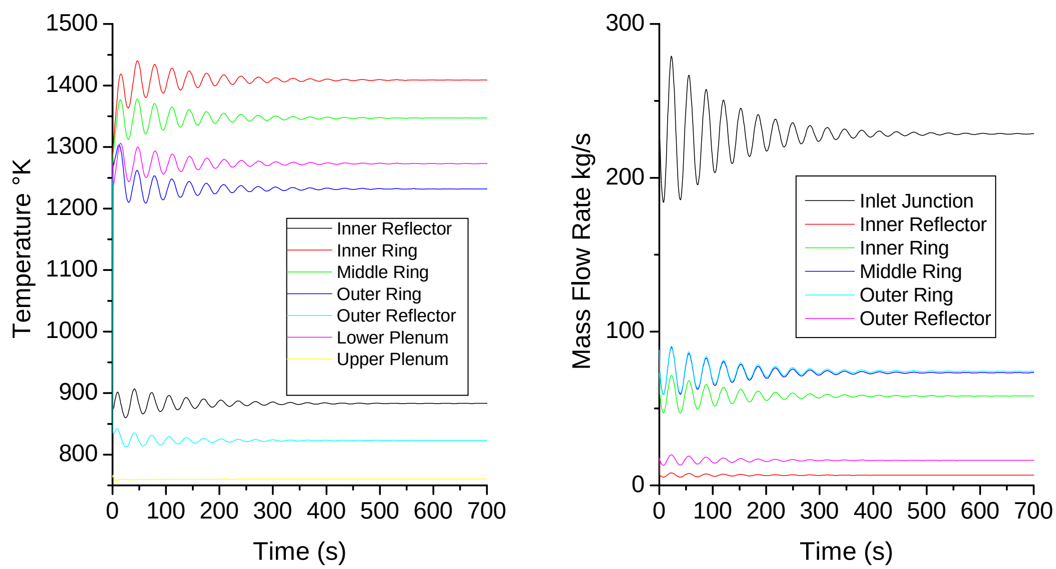


Figure 41: Steady-state initialization

Figure 42 presents the mass-flow rates, the inlet and outlet coolant temperatures as well as the channels' delta temperature for the five core channels. The most noticeable figures are the fuel increase through the three active channels. There are all greater than 450 degrees, the maximum being 650. There is no possible comparison between these values and the common temperature increase in a PWR (around 20°C). Such temperature changes would normally involve dramatic structural issues. Thanks to the graphite structure the core can operate without noticeable deformations.

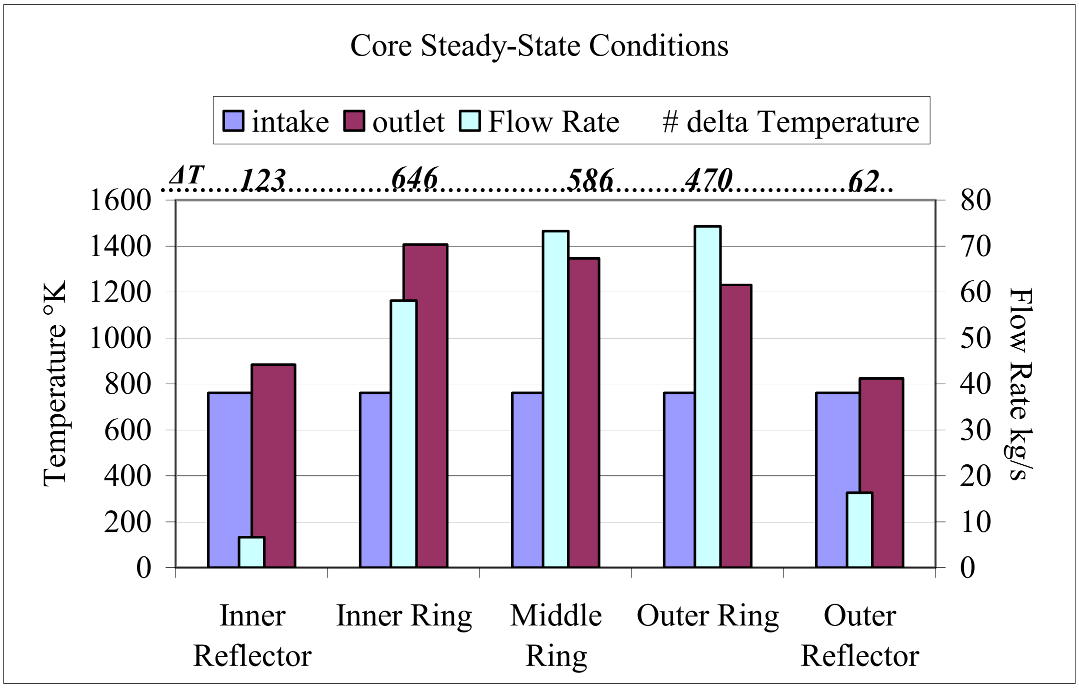


Figure 42: Core steady-state conditions

Figure 43 presents more detailed results of the steady-state conditions. The axial and radial temperature profiles in the coolant, the reflector and the fuel are visible. During steady-state operations the hottest helium locations are obviously at the bottom of the coolant channels because of the heating structures. Surprisingly, the maximum fuel temperature is also located at the bottom of the active heat structures. The maximum fuel temperature recorded, 1489°K, is located on the centerline of the lowest inner ring active structure (heat structure 1521, node 10). The fuel temperature peak is expected to be next to the largest peaking factor (middle ring, node 6). This strange location may be correlated with the flow distribution in the channels. On Figure 42 one can see that the inner active core ring has the lowest coolant flow rate, thus lower heat rate removal. Therefore, even if the heat structures in the inner ring are not generating as much heat as in the middle ring, the temperature centerline can be higher.

The core barrel wall has some interesting issues too. On the top left corner of Figure 43 one can see that the structure separating the riser from the outer reflector is subject to fairly important temperatures (more than 750°K) and temperature gradients (up to 80°K). Cold helium is rising on one side while hot helium is flowing down on the other side. The core barrel walls are made of Incoloy alloy 800 and protected on the inside with H-451 graphite to insure structural stability.

In comparison, the vessel walls (bottom right corner Figure 43) are only subject to relatively small temperature gradients, around 60°K, yet pretty high temperature are expected (up to 750°K). The vessel walls are made of high-temperature resistant T91 steel (T91 stands for 9Cr-1Mo).

The presentation of the steady state would not be complete without the presentation of the RCCS conditions. Figure 44 and Figure 45 show the operating temperature and pressure in the RCCS.

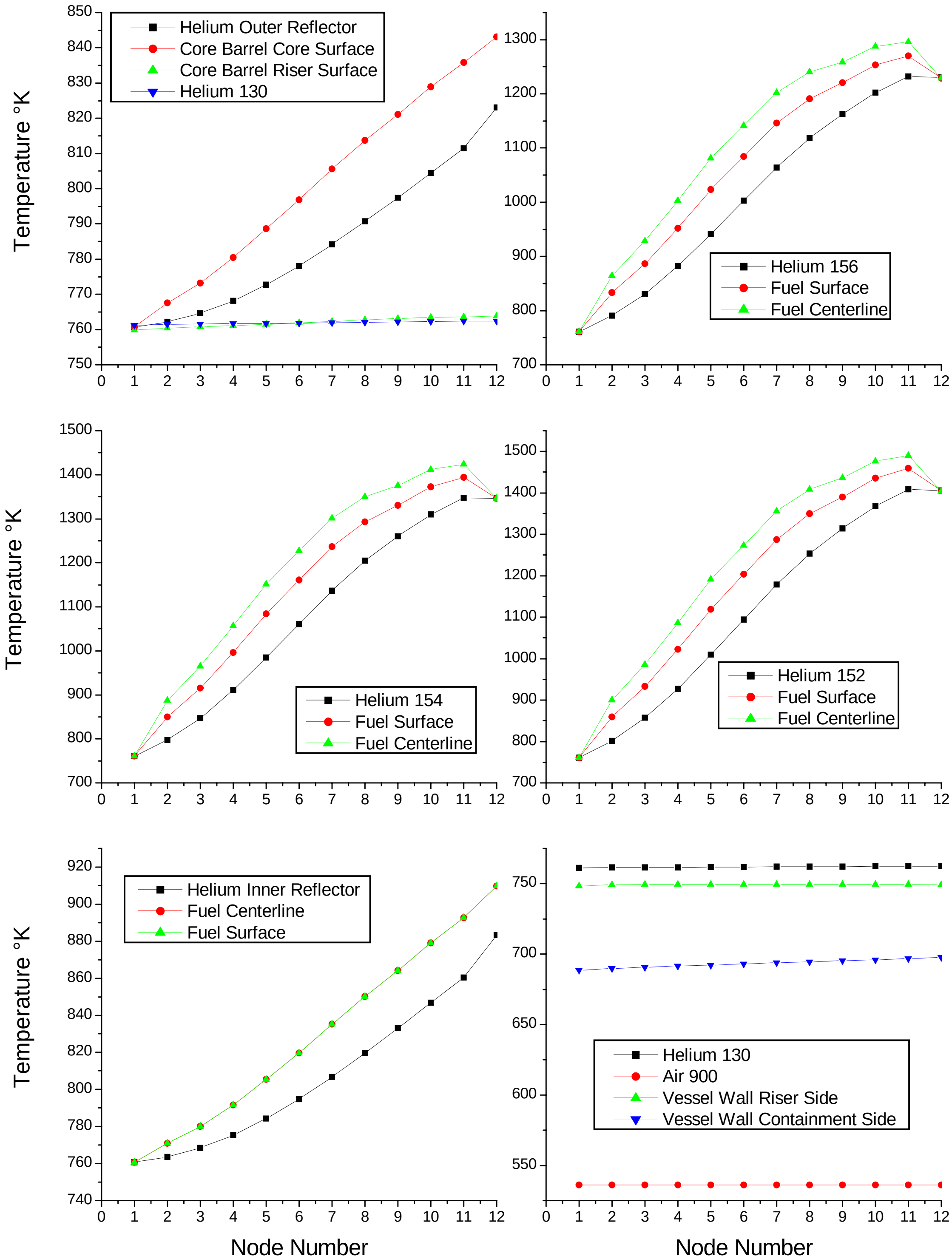


Figure 43: Core axial and radial temperature gradient

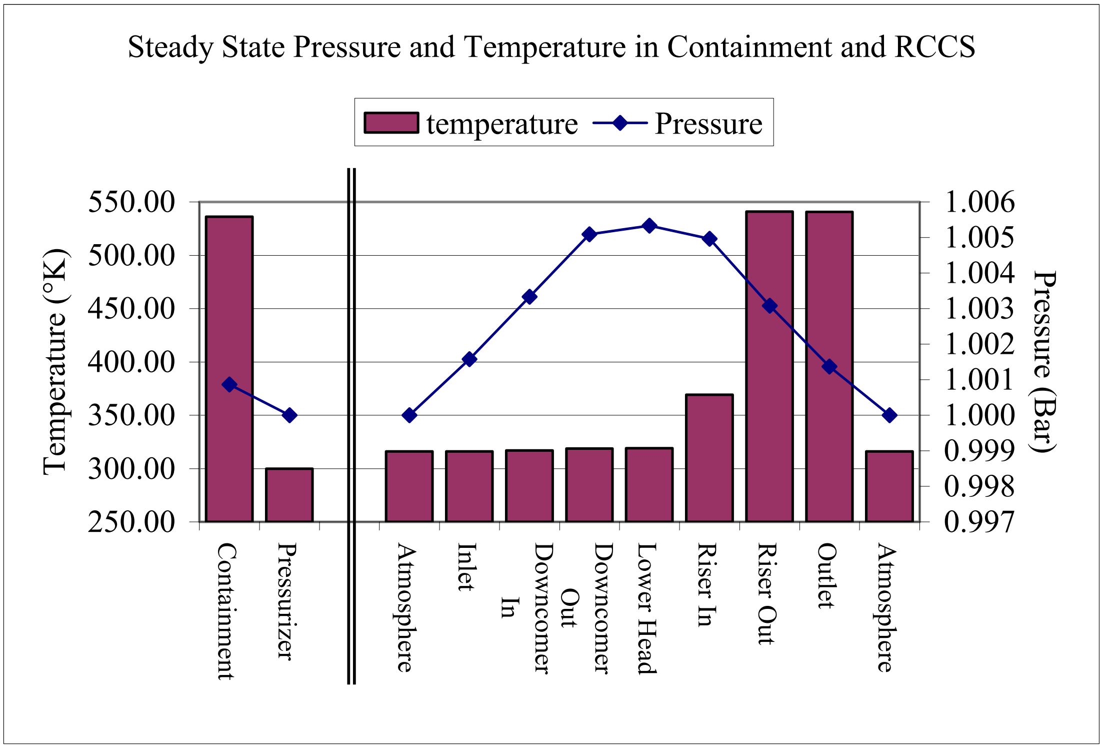


Figure 44: Containment and RCCS steady-state conditions

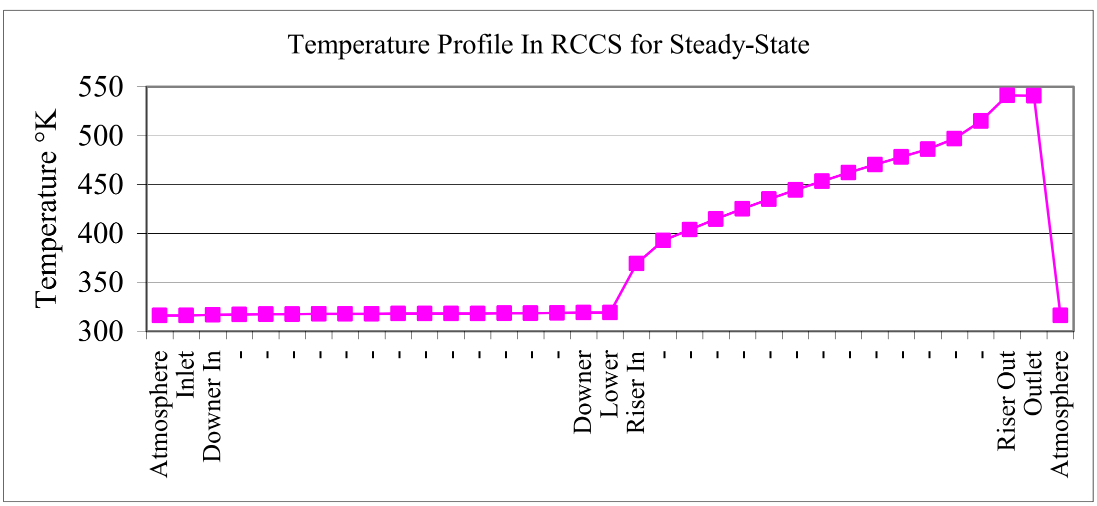


Figure 45: Temperature profile in RCCS

SCRAM sequence

The SCRAM initialization needs not to be too rapid after the beginning of the transients. Figure 46 shows the temperature increase in the fuel and the coolant if no scram occurs. The increase rate is constant and equal to 3.5 degrees per second. Considering that it takes about 4 seconds to lower the control rods in the reactor (Figure 37) there is an average temperature increase of 14°K that cannot be avoided. But this is affordable safety wise. Even if the scram is delayed for a short period of time, the temperature increase is not penalizing. Therefore, the rapidness of the accident detection needs not be too accurate.

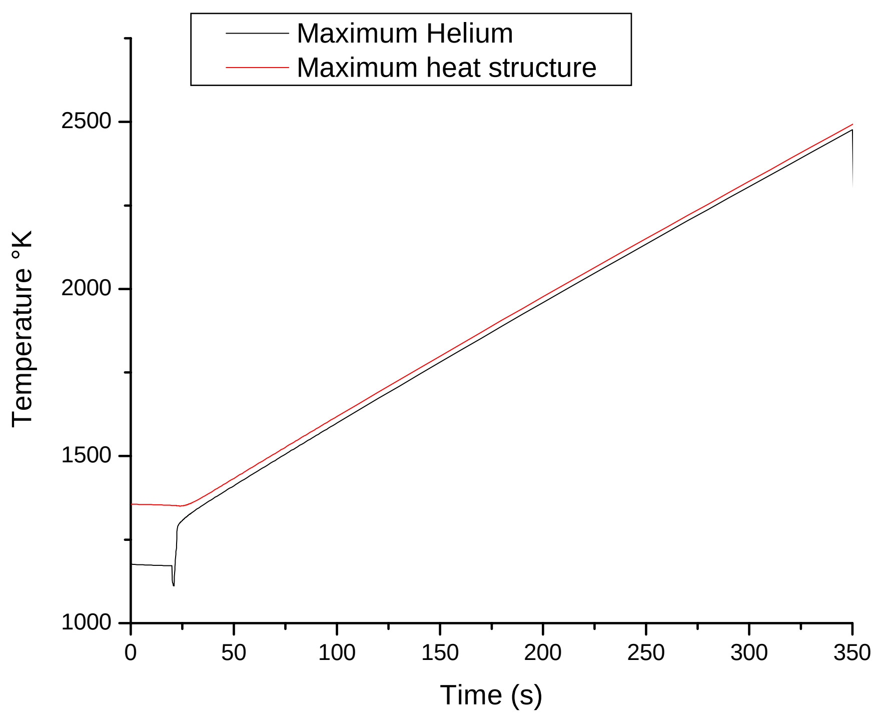


Figure 46: Temperature increase if no SCRAM

Figure 47 shows the evolution of the power in the core after the SCRAM occurs. When the SCRAM is initiated, the massive addition of negative reactivity stops promptly the fission process. The fission contribution to the total power decreases by two orders of magnitude in less than one hundreds seconds. The main contribution to the power is then the decay heat. The power generated in the core just after the SCRAM is an order of magnitude less than during normal operation. After that, the heat produced decreases slowly, stays very significant and must be removed to preclude core melting. For instance, after 4 days, the core is still generating more than 2 MW. This represents the electrical consumption of 25,000 80 W light bulbs.

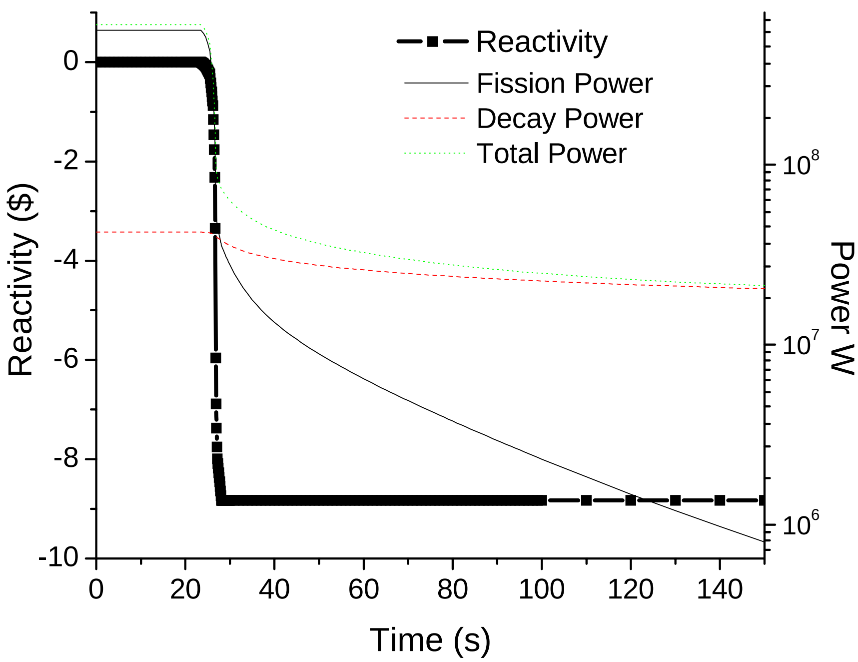


Figure 47: Power vs reactivity during SCRAM

Accident scenarios

The main concerns during the studied accident scenarios are the measurements of the peak temperature in the reactor and the time at which it occurs after initialization. According to General Atomics design (Figure 15) the peak for a loss of coolant scenario occurs around 70 hours, that is to say almost 3 days, after initialization and the temperature remains under the design limits (1600°C/1873°K).

Loss of coolant

The temperature profiles for different size of break are presented on Figure 48. Results of the loss of coolant simulations show that the temperature of the fuel in the hottest spots reaches a maximum and then decreases slowly. The whole phenomenon happens over a rather long period of time. Whatever the size of the break, the temperature peaks in the fuel occur 2.5 days after initialization. As one can see, the temperature in the core remains under the design limits at all time. The peak temperature in the core is 1752 °K or 1480 °C.

The temperature profiles obtained at the beginning of the transients depend on the break size. Figure 49 and Figure 50 present, respectively, the core pressure and the breach mass-flow rate as a function of time for the different break size. The results for the largest break area are suspicious; the mass flow rate reaches a peak at 4000 kg/s which correspond to supersonic speed. For the four different sizes, one can notice that the mass flow rates are always going to zero after a certain period of time and that the pressures go to the atmospheric pressure after the same amount of time. After the initial depressurization, the pressure in the core is set to the atmospheric pressure and there is no more noticeable coolant flow. The temperature heat transfer is then performed by natural mechanisms.

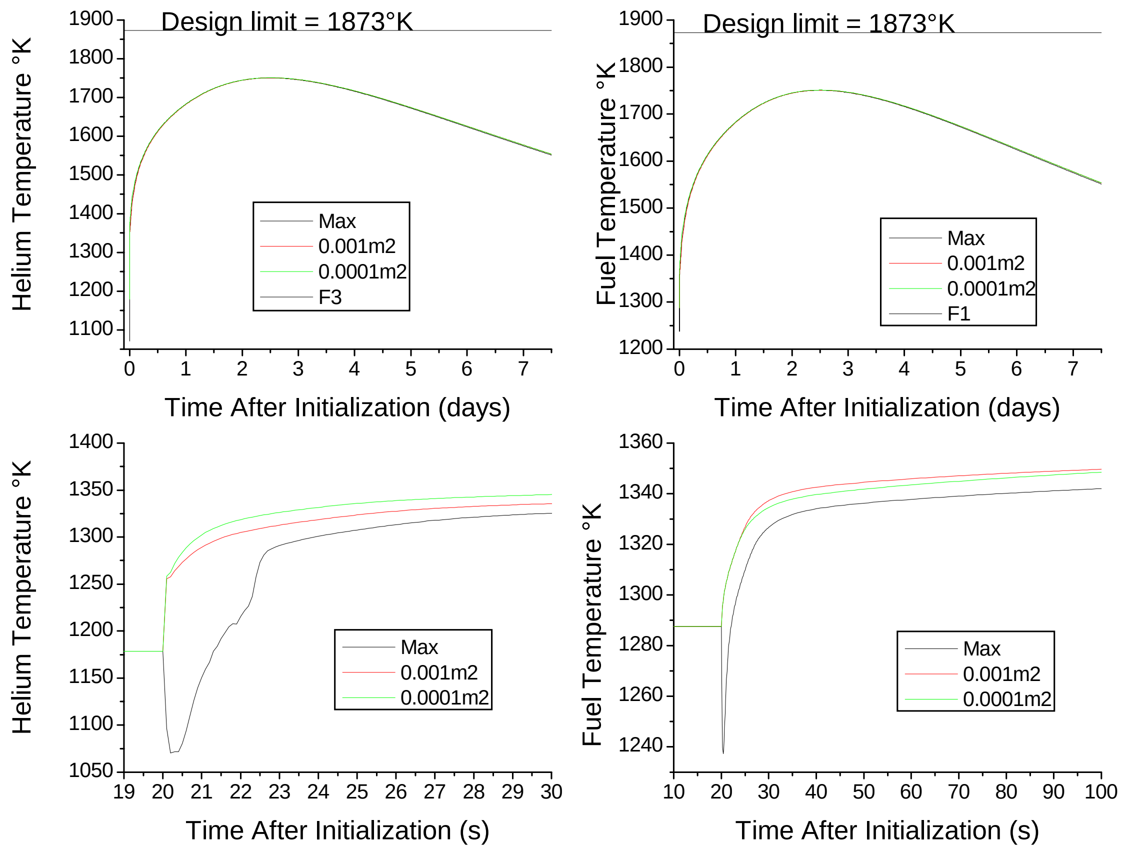


Figure 48: Peak temperatures

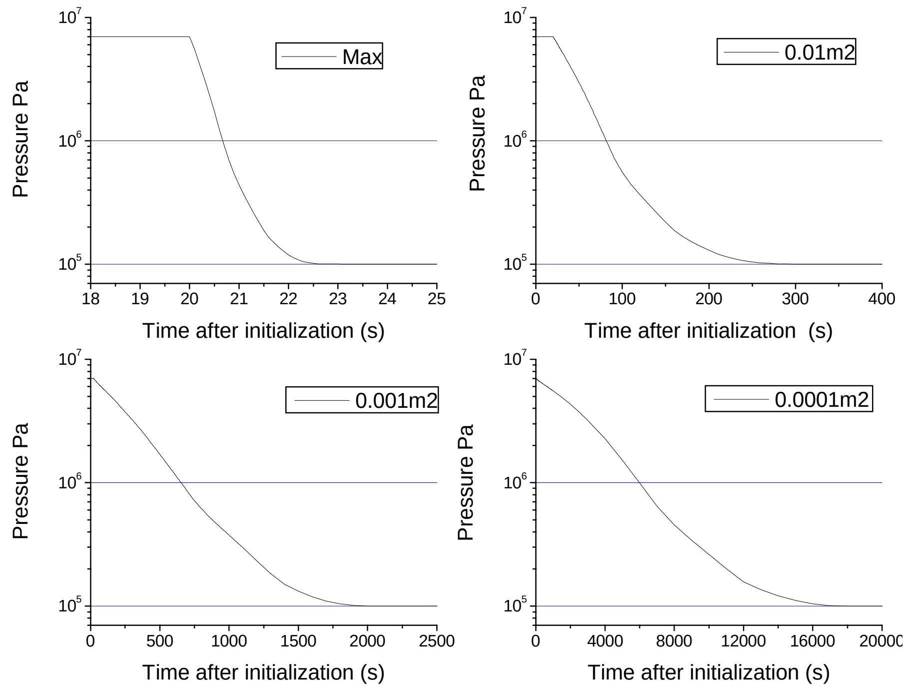


Figure 49: Pressure in the core during loss-of-coolant

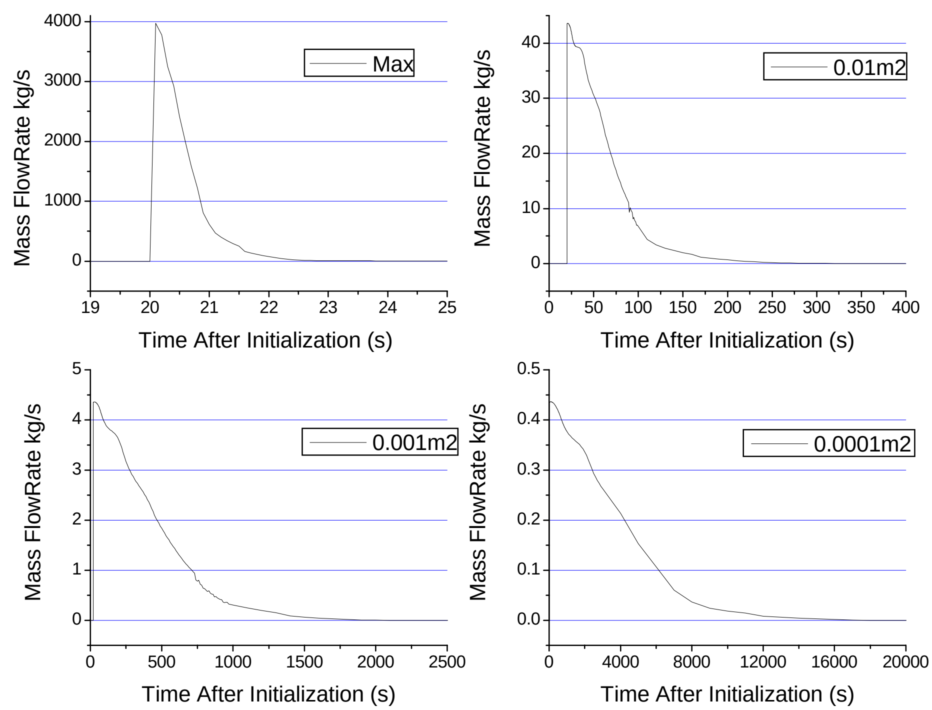


Figure 50: Break mass-flow rate during loss of coolant

Loss of flow

The temperature profile for the loss-of-flow is presented in Figure 51. Results of the loss-of-flow simulations show that the temperature of the fuel in the hottest spots reaches a maximum and then decreases slowly. The whole phenomenon happens over a rather long period of time. The temperature peaks in the fuel occur 1.4 days after initialization. As one can see, the temperature in the core remains under the design limits at all time. The peak temperature in the core is 1522°K or 1250°C.

During a loss-of-flow, the temperature peak occurs sooner and is lower than for a loss-of-coolant accident. It takes only 3 hours an a half to reach 1522°K during a loss-of-coolant accident.

Figure 52 shows the core pressure and the mass flow rate variations at the beginning of the loss-of-flow. The loss-of-flow is performed by instantaneously stopping the flow in the core. Both inlet and outlet volumes (100 and 170) are closed instantly withdrawing the reason for the flow: the pressure difference between the inlet and outlet. It takes less than a second for the flow in the core to stop completely. The result of the loss-of-flow is a temperature increase at constant volume. The pressure in the core increases.

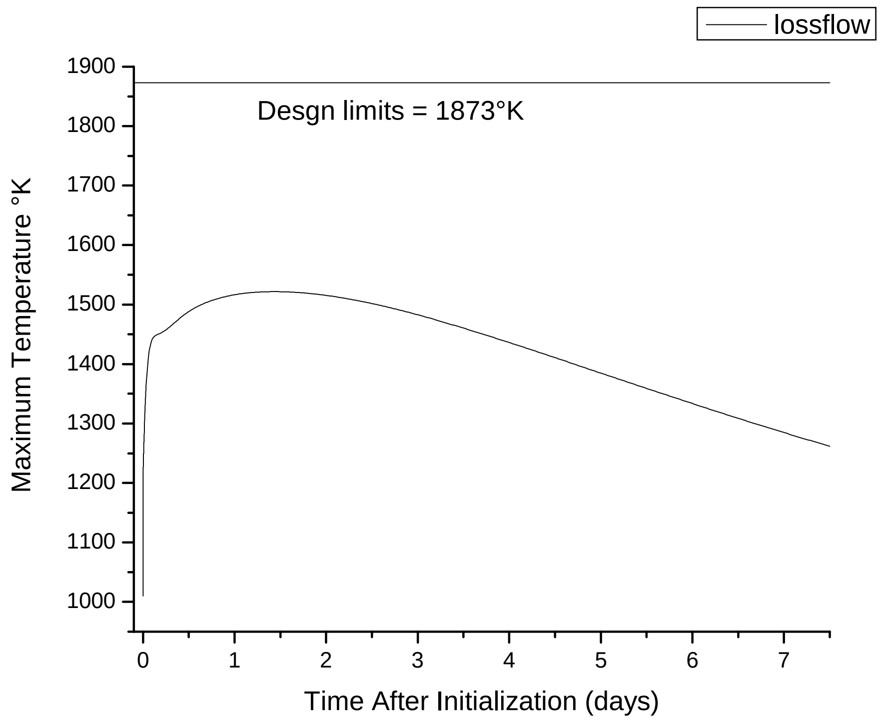


Figure 51: Maximum core temperature during loss of flow

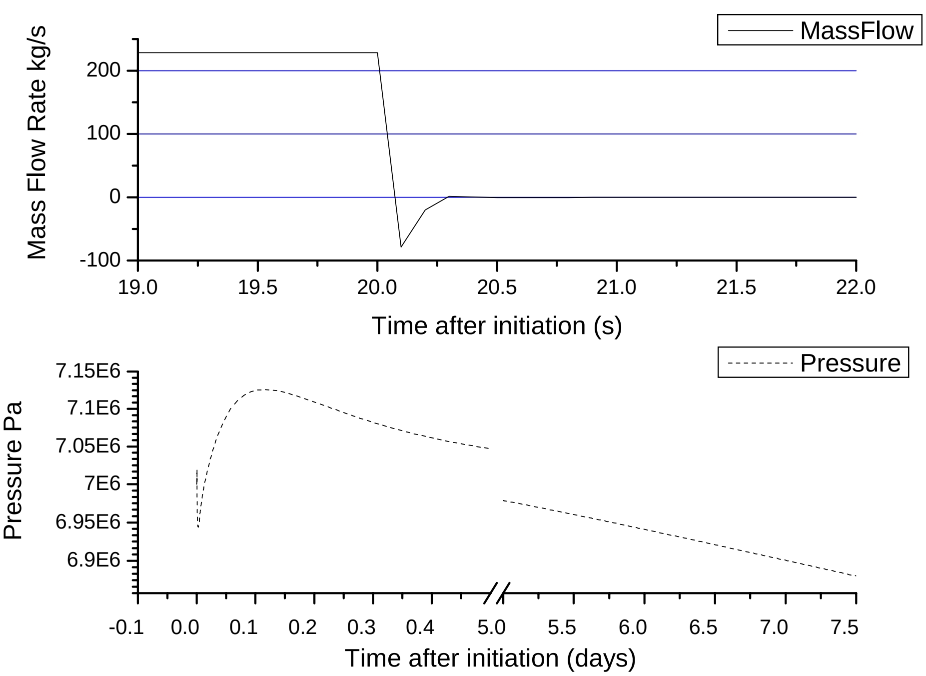


Figure 52: Mass flow rate and pressure in volume 105 during loss of coolant

The behavior of the core under the two studied scenarios is very original compared to light water reactor. The MHR parameters are varying very slowly. This is due to the high heat capacity of the graphite, the low power density of the core, the core geometry and the shift of heat transfer toward radiative heat transfer. Thanks to the huge quantity of graphite that constitute the structure of the core, most of the heat generated in the core after an accident can be stored in the graphite. Graphite acts like a heat sink. The graphite also provides a lot of inertia to the heat transfer because of its low heat transfer coefficient. Therefore the thermal response is slower than in a PWR. In a PWR, the water is the major component. It has a huge heat capacity. So, in case of a loss of coolant in a PWR, not only the coolant is lost but also the main heat receptacle is gone. The heat has no other option than accumulating in the fuel. In that respect, the loss of the helium coolant in the MHR is not that dramatic. Helium is not comparable to the water heat capacity wise. Also noticeable during a loss-of-coolant accident, the peak temperature occurs at the same time whatever the size of the break.

CONCLUSION

The assessment of the world primary energy resources and their utilization shows that we may face major problems in the near future: fossil fuel shortage, energy dependence and global warming. The world is consuming non-renewable energies at a faster rate than it is discovering new supplies. At present consumption rate, and if new reserves detection rate stays low, oil shortage is only 40 years ahead. The principal consumers, USA and Western Europe, are relying on cheap and available fossil fuels although they do not hold the main reserves. And last but not least, greenhouse gas emissions caused by human activities are more and more suspected to have a significant impact on the measured global warming. This last issue is a rapidly growing concern in the public opinion (Kyoto protocol).

The massive use of fossil fuels is the key problem. Their utilization must be reduced either by reducing energy needs or by finding alternative energies (or both, ideally). One of the proposed solutions is to shift to hydrogen economy, system in which hydrogen is used instead of fossil fuel for transportation and chemical industries. The best way to produce hydrogen is to couple advanced nuclear reactors to hydrogen generation plants. Nuclear energy is the only relevant method to produce hydrogen on an industrial scale. It is carbon free.

Unfortunately, as of today, there are no operating nuclear reactors capable of producing hydrogen. There are technical issues to be solved and the system has to be economically viable. However, the economics problem may disappear when fossil fuels become rarer, thus more expensive.

The technical issues are already starting to be tackled thanks to the Generation IV initiative program. Generation IV goals are the development of new reactors concepts that represent significant advances in economics, safety, reliability, proliferation resistance and waste minimization. It implies developments of new technologies such as high temperature output for increased thermal efficiency and hydrogen production.

Because the implementation of the Generation IV technologies is going to be a long process, development of intermediate nuclear reactors will probably be done in the mean time. For instance, the MHR concept that has been developed by General Atomics can be considered as a step toward the implementation of the generation IV project. The development of the MHR would permit early applications for some required Generation IV technologies: mainly high temperature gas coolant and hydrogen production.

Because very few reactors similar to the MHR have been designed in the past and because the MHR is supposed to be a highly passively safe concept there are high needs for numerical simulations in order to confirm the design. The project was dedicated to the thermal-hydraulics assessment of the passive decay heat capabilities of the reactor under abnormal transient conditions: loss-of-coolant and loss-of-flow.

The study of the MHR was addressed using the thermal-hydraulic code RELAP5-3D, Athena version. RELAP has been developed by the Idaho National Engineering and Environmental Laboratory (INEEL) for the purpose of analyzing transients and accidents, including both large and small break loss of coolant accidents (LOCA) in Light Water Reactors (LWR). It is based on a non-homogeneous and non-equilibrium model for two-phase flow. The Athena version is the first of the RELAP5 series to provide gas-cooled reactor competence.

The simulations were performed successfully. The results of the loss-of-flow and loss-of-coolant simulations showed that the peak fuel temperature in the core reached a maximum below the design limits and then decreased slowly. The whole phenomenon happened over a rather long period of time compared to current light water reactor. The passive decay heat removal capability of the MHR is mainly the consequence of the huge quantity of nuclear grade graphite constituting most of the core. Graphite is the central component of the MHR: it is used as structure, moderator, and heat tank in case of accident.

The project has confirmed the great capability of the MHR regarding the decay heat removal. However, the model used was very simplified. Therefore, more simulations have to be carried out on a more complex model which would include the secondary system and more detailed core internals. On the paper, the MHR looks like a great concept. The main obstacle to its practical development will be the development of materials sustaining the very high temperature involved during operations.

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