RADIATION PROTECTION AND HEALTH PHYSICS IMPLICATIONS ASSOCIATED WITH SMRS FOR POTENTIAL APPLICATIONS IN TURKEY

A Thesis

by

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ABSTRACT

The purpose of this project is to develop and demonstrate the models to assess radiation dose rates to personnel within a reactor facility (or in a public area) using the MCNP Monte Carlo code.

In this project, the performance of proposed Small Modular Reactors (SMRs) facilities is evaluated on the basis of radiation protection and applicable regulatory limits. A health physics and radiation protection analysis of the small modular reactors is carried out. The scope of this project is to construct an assessment of the radiation risks associated with an SMR. To accomplish this, a conservative dose rate calculation is done and compared with the annual limit intake as per regulations. In addition, geographical and safety requirements are considered for this design. First, a review is made of the regulations and amount of radiation that can be released to the public. The basis for this review is the regulations for the protection of radiation from the U.S. Nuclear Reactor Commission. The evaluation of shielding from the ionizing radiation produced in the core of the nuclear reactor is made. The Monte Carlo N Particle (MCNP) simulation software which uses a stochastic algorithmic approach, is used to test the effectiveness of the design model and learn the dose rates in the area of interest.

The objective is to evaluate if this design will be appropriate for use applications in Turkey. Turkey only has one very small research reactor and three power plants under construction. The new plants are located in close proximity to the sea and in comparatively remote areas with low population density.

CONTRIBUTORS AND FUNDING SOURCES

Contributors

This work was supported by a thesis committee consisting of advisor Dr. Kenneth L. Peddicord, co-adviser Dr. John Ford, and Dr. Pavel Tsvetkov of the Department of Nuclear Engineering, and Dr. Karl Hartwig of the Department of Mechanical Engineering.

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DEDICATION

This thesis is dedicated to all people who support me to finish my master's degree. Especially to my love ones who at least did their part to help me finish it; to my parents for their endless love, support, and encouragement, to my boyfriend who’s the best supporter and tried his best to help me in my difficult situations while I did my thesis. A great achievement after being stressed for two years. Those two and a half years of being depressed and losing my path that I never thought I would be coming back again to finish this degree.

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NOMENCLATURE

NRC: Nuclear Regulatory Commission

MCNP: Monte Carlo N-Particle Transport Simulation

IAEA: International Atomic Energy Agency

SMRs: Small Modular Reactors

IEA: International Energy Agency

OECD: Organization for Economic Corporation and Development

PWRs: Pressurized water reactors

HTRs: High-temperature gas cooled reactors

GFR: Gas-cooled fast reactors

MSRs: Molten-salt reactors

(SFRs): Sodium fast reactors

LFR: Lead and lead–bismuth eutectic cooled reactors

MWth: Megawatt

ALARP: (As Low as Reasonably Possible)

VISED: Visual Editor (MCNP)

FA: Fuel Assembly

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CHAPTER 1. INTRODUCTION

The worldwide demand for clean and low-cost energy is ever-increasing, and according to International Energy Agency (IEA) report, an increase of 24-31% in global energy consumption has been predicted by 2040 [1]. Our reliance on unsustainable and harmful fossil fuels is causing catastrophic effects on the environment. This massive demand for energy has driven research in nuclear science to explore new opportunities to utilize nuclear energy as a reliable, cost-effective, safe, and clean source of energy for the future. However, this is not a particularly new concept, as we have been utilizing energy from nuclear since the 1950s; therefore, this is not a new concept. Currently, 10% of the global electricity is being generated from nuclear plants. Several nuclear power plants are also under construction, increasing this share of nuclear energy in the future [2]. Despite concerns following the Fukushima incident, which raised skepticism and disbelief about nuclear safety and adverse reactions in different countries by the public, several countries such as China, India, Malaysia, Indonesia, Philippines, Poland, Turkey, Thailand, and Vietnam are interested in nuclear energy.

Nuclear safety, nuclear proliferation, human health, large capital investment, and significant footprints are the critical issues associated with conventional nuclear power plants. In recent years, several nuclear plant manufacturers have started designing and developing fourth-generation small nuclear reactors that with 300 MWe output or less. These are referred to as Small Modular Reactors (SMR). SMRs are getting attention from both industry and nuclear energy-related regulators because of their potential uses.

The advanced SMRs are being developed using a modular approach that helps to scale the plant size making it more cost-effective and suitable for on-site construction and producing energy at places closer to densely populated sites and industrial areas. The critical design factors and advantages are greater simplicity in design, mass production of modular-based plants, and smaller footprints. Several companies are developing their new designs for SMRs, and there is great competition between the companies to develop reactor designs to meet stringent nuclear regulations and gain safety license approval from authorities. These will be advantageous to playing a leading role in the nuclear industry. Recently, NuScale has announced its unique passive safety advanced SMR design which does not require any external power or coolant in case of any accident and is inherently safe in the case of accidents like Fukushima [3]. The passive safety design increases the safety of SMR without requiring any external supply of water and power. Under normal working conditions, radiation safety is another crucial design factor that is a key to worker safety at the nuclear station. Multiple layers of protection are added to the nuclear power plant to handle worst-case scenarios. These protection layers include the fuel, the reactor vessel, and the primary containment building. The purpose of these layers is to protect the public and people working at the plant from radiation.

A nuclear power plant must receive a license to assure its safety and minimize public health effects before deployment. These licensing laws vary from country to country, and the vendors must meet these regulations to gain approval and credibility. Therefore, nuclear plant designers must demonstrate that the nuclear safety standards have been achieved through the development and design process before a license will be granted and before commencing construction or any commercial operations. It is important to assess what the dose rates will be to determine what radiation zone designation is to be applied to occupational areas. It is imperative designing of a nuclear facility to simulate and test the designs to ensure that standards are met, and that radiation zones can be appropriately assigned.

This project aims to develop and demonstrate the models to assess radiation dose rates to personnel within a reactor facility and to the environment and the public nearby using the MCNP Monte Carlo code.

In this project, the performance of proposed Small Modular Reactors (SMRs) facilities based on radiation protection and the applicable regulatory limit is evaluated.

A health physics and radiation protection analysis of small modular reactors has been carried out. The scope of this project is to conduct an assessment of the radiation risks associated with an SMR. To accomplish this, a conservative dose rate calculation will be done and compared with the annual limits as per regulations. In addition, geographical and safety requirements for this design will be taken into consideration.

First, a review has been made of the regulations and amount of radiation that be released to the public. The basis for this review will be the regulations for the protection from radiation issued by the U.S. Nuclear Reactor Commission. The evaluation of shielding from radiation produced in the core of the nuclear reactor will be made. The Monte Carlo N Particle (MCNP) simulation software which uses a stochastic algorithmic approach, will test the validity of the model and learn the dose rates in the area of interest.

The objective will be to evaluate if this design will be appropriate for use applications in Turkey. Since Turkey has three power plants are under development. The new plants are located in close proximity to the sea and comparatively remote areas with low population density. In this project, the performance of proposed Small Modular Reactors (SMRs) facilities on the basis of radiation protection and applicable regulatory limits is evaluated.

CHAPTER 2. LITERATURE REWIEV

The worldwide demand for clean and low-cost energy is ever-increasing, and according to the International Energy Agency (IEA) report, an increase of 24-31% global energy consumption has been predicted by 2040 [1]. Our reliance on unsustainable and harmful fossil fuels is causing catastrophic effects on the environment. This massive energy demand has driven research in nuclear science to explore new opportunities to utilize nuclear energy as a reliable, cost-effective, safe, and clean source of energy for the future. We have been utilizing energy from nuclear since 50’s; therefore, this technology is not a new concept. Currently, 10% of the global electricity is being generated from these nuclear plants, and several nuclear power plants are under construction that would increase this share of nuclear energy in the future [2]. Despite post-Fukushima incident, and raising skepticism and concerns in public about nuclear safety, several countries such as China, India, Malaysia, Indonesia, the Philippines, Turkey, Thailand, and Vietnam are interested in nuclear energy.

Nuclear safety, nuclear proliferation, human health, large capital investment and large foot prints are the key issues associated with conventional nuclear power plants. In recent years, several nuclear plant manufacturers have started designing and developing fourth-generation small nuclear reactors with output of 300 MW or less and called Small Modular Reactors (SMR). SMRs are getting attention from both industry and nuclear energy-related regulators because of its potential usages. The advanced SMRs are being developed using modular approaches, which help scale the plant size and make it more cost-effective and suitable for on-site build and produce energy for places closer to densely populated sites and industrial areas. Greater simplicity in design, mass production of modular-based plants, and smaller footprints are key design factors and advantages. Several companies are developing their new designs for SMRs and there is great competition between the companies to develop their reactor designs to meet stringent nuclear regulations and receive safety license approvals from authorities. Recently, NuScale announced its unique passive safety advanced SMR design which does not require any external power or coolant in case of an accident and is inherently safe from accidents like Fukushima [3]. The passive safety design increases the safety of SMR without requiring any external supply of water and power. Under normal working conditions, radiation safety is another crucial design factor so that staff can safely work at the nuclear station. Multiple layers of protection are added to the nuclear power plant to meet worst-case scenarios. These protection layers include the fuel, reactor vessel, and the primary containment building. The purpose of using these layers is to protect the public and people working at the plant from high levels of radiation.

A nuclear power plant needs to obtain a license certifying that it meets safety and public health requirements before its deployment. These licensing laws vary from country to country, and the vendors must establish these regulations and gain credibility. Therefore, nuclear plant designers must prove the nuclear safety standards achieved during the development and design process before submitting to the licensing process and before commencing any commercial operations. It is important to assess what dose rates will be to determine what radiation zone designation is to be applied to occupational areas. It is of high importance in the design of a nuclear facility to simulate and test the designs to ensure that standards are met, and that radiation zones meet regulations.

The Paris agreement and the 2030 agenda for sustainable development set up the road map for the global economic transformation plan. The heart of this transformation plan is about the usage and consumption of energy. Energy is the chief contributor to greenhouse gases, global climate changes. Therefore, transforming our global energy system and creating cleaner and secure energy is required for a better future. Nuclear energy has several advantages, such as high energy concentration per unit of mass of nuclear fuel and low greenhouse gas emissions comparable to renewable energy sources still, several Organization for Economic Corporation and Development (OECD) countries are facing challenges in utilizing nuclear energy in their countries due to economic and socio-political reasons. In recent years, the discovery of cheap shale gas, low-cost gas liquefaction, and highly efficient gas combined cycle technology to produce energy has narrowed the economic space for nuclear energy. Therefore, a decline in interest in nuclear energy was observed in OECD countries in spite all its advantages and its potential as a base-load energy.

2.1. Turkey

This particular study focuses on energy needs in Turkey and the potential for small modular reactors to play a role in fulfilling these requirements.

Today, energy generation and trade take place globally. In Turkey also consumption and generation of energy take a form under the influence of the globalization process. Turkey has nearly every kind of primary resource, but around 74% of energy demand is met through imports because these resources do not meet the substantial growth in energy demand, except for lignite and hydropower [4], because the country is rich in rivers, watercourses, and lignite. The estimated total of 10.82 billion tons of lignite reserves has been increased and now reaches 19.32 billion tons. The reserve, which was 8.3 billion tons, increased by 130% [13].

As noted, the primary energy sources in Turkey are oil, lignite, hard coal, natural gas, hydraulic energy, wind energy, geothermal energy, solar and biomass energy. Energy in Turkey is generated from these primary energy resources. The number of major power plants is 669 hydraulic, 68 coal, 262 wind, 52 geothermal, 330 natural gas, 6435 solar, and 253 other power plants [5], [14], [6]. Coal and hydroelectric power in Turkey have had a significant place in domestic energy generation for several years [5]. With improving technology, Turkey plans to decrease its energy imports using domestic resources such as lignite coal, renewables, and nuclear energy [7]. Three nuclear power plants that are not active yet are under construction.

According to the Ministry of Energy and Natural Resources of The Republic of Turkey data, by the mid of 2019, the total installed capacity by the resource is composed of 52.7% thermal and 47.3% renewable power in Turkey. Renewable are sub-classified into hydropower plants (31.4%), wind (8.1%), geothermal (1.6%), and solar (6.2%). Thermals are sub-classified as coal-fired (22.4%), natural gas-fired (28.6%) power plants, (1.7%) other sources [15].

Turkey has several of the largest hydropower plants in the Eastern/Southeastern Anatolia Region. The most important of these plants are Atatürk hydroelectric power plant, comprising six units of 2400 MW, Kara kaya, six units of 1800 MW, and the Keban hydroelectric power plant, which comprises eight units with an installed capacity of 1260 MW [10]. Thermal power plants are located in almost all regions of Turkey, and approximately 300 TW/h of electricity is generated each year from these plants [8].

The grid is synchronized with the European Network of Transmission System Operators for Electricity (ENTSO-E) [11] and linked across most land borders such as Azerbaijan (34.5 kV and 154 kV), Armenia (220 kV), Bulgaria (400 kV), Georgia (220 kV), Iran (154 kV), Iraq (400 kV), Syria (66 kV) without synchronous operations. However, only about 1% of electricity is imported or exported [9].

The old power systems and methods are used for the electricity grid (that is constructed based on these systems came out in the 1800s [12]). Besides, using an extensive amount of fossil fuels for thermal plants causes severe damage to the atmosphere and the environment. In addition, the problems of not meeting the increasing energy demand arise because of the burgeoning population. Turkey's electricity network seems far behind many developed countries with increased orientation to energy sources that pollute the environment less. For Turkey, smart grid concepts are being developed which aim to use renewable energy resources more than fossil fuels and natural gas that are imported from other countries with pipelines [12], [11].

2.1.1. The Major Industries Those That Consume Major Amount of Energy in Turkey

Turkish industries consist of textile (this is most significant in the manufacturing industry) [16], food processing, automotive, electronics, mining (coal, chromate, copper, boron), construction, lumber, paper, and major industries are textile steel/metallurgy, textile, and clothing, petroleum products, food, [17]. The basin of Istanbul-Kocaeli/Izmit, the plain of Çukurova (with Adana as its regional capital), and the region of Izmir are the three crucial traditional industrial areas. In the last 20 years, many other regions such as Kayseri, Gaziantep, Adıyaman, and Denizli have contributed as industrial production centers [18]. According to research about Turkey's top 500 Industrial Organizations in 2018, the most prominent industries with the numbers are tools and devices Industry(7), vehicle industry(7), stone and other soil-based industry(7), iron-steel base metal industry(7), electric machinery, other chemical products industry(5), foodstuffs industry(5), primary chemical industry(3), glass and glassware industry(3), beverage industry (alcoholic and non-alcoholic) (3), weaving industry(3), paper and paper products industry(3), wood, furniture and furnishings industry(2), petroleum products industry(2), rubber products industry(2), essential metal industry outside of iron and steel (2), machinery industry (excluding electric) (2), printing industry(1), plastic products industry not classified elsewhere(1), professional, scientific, health purpose tools and materials industry(1), electricity sector(1) [19].

In Turkey, as noted earlier, electrical energy is generated from sources such as thermal, hydraulic, geothermal supports production in industries [20]. Electricity used in manufacturing is at the forefront of the most considerable amount of consumption. According to data from research about the consumption of electricity, 47.2% of the electricity consumed was used in industry in 2014 [21]. The textile and leather sectors consume the most electricity after the iron and steel sector among the manufacturing industry sectors with 15.521 GWh. Natural gas, thermal energy, steam, and electricity are consumed for the textile industry, generally gathered in Marmara Region. According to the Ministry of Industry and Technology data, 1.5 kilowatt/hour of electricity is used for the production of 1 kg of goods [22], [23].

According to the end of 2017 data of the Ministry of Energy and Natural Resources, electricity consumption is 294.9 billion kWh, and according to the Electricity Distribution and Consumption Statistics of Turkey, the highest rate (nearly 50%) belonging to the industry sector. Also, it is observed that consumption is higher in the provinces clustered in developed regions (regions where there are many industries) [24].

2.2.REACTOR TECHNOLOGIES AND APPLICATIONS

2.2.1. Potential in Asia and Newcomers

SMRs are attractive due to their lower unit capital cost, smaller size, and shorter construction time than conventional NPP (Nuclear Power Plants). This would reduce financial risks for investors. These factors have gained attention in recent years. Several potential applications of these SMR plants have been proposed by industry. For example, steam produced from these small reactors can be utilized for oil and natural gas exploration process for hydraulic fracturing, production of hydrogen fuel by electrolysis, biofuels processing, oil refining and desalination of seawater and for heating purposes [25]. Therefore, due to the economic benefits, these SMRs are envisioned to have great potential in the future. These SMRs are small and can be factory-built, transported to the site on the back of heavy trucks or rail, and assembled easily installed below the ground to avoid any impact from missiles or airplanes. They are mounted on seismically isolated bearings to protect them from geological events. Some SMRs are designed to be fueled at the factory and sealed off, providing an extra nuclear safeguard against nuclear proliferation. Multiple units of SMRs can be installed together. This makes it possible to tailor the supply and demand of the electricity generation, allowing control and tailoring of the demand and supply of the total electricity generation according to the region's needs.

Several countries are developing SMRs, and competition is pushing the research and development momentum of this technology. Each country would like to be first in the market as there is a massive demand and potential for export, especially in the Asian market. A list of some of these newly developed SMRs designs proposed by different vendors is shown in Table 1. These designs are at different stages of development, some are at design stage, some at licensing phase with the regulator authority and couple of them are under construction. These designs are at different stages of development, and some are at the design stage, some at licensing phase with the regulator authority, and a couple of them are under construction. These reactors include pressurized water reactors (PWRs), high-temperature gas-cooled reactors (HTRs), gas-cooled fast reactors (GFR), molten-salt reactors (MSRs), sodium fast reactors (SFRs), and lead and lead-bismuth eutectic cooled reactors (LFR).

Table 1. Developed SMRs designs by different vendors

SMR Name	Type	Plant (MW_e)	Developer, Country	Capital Cost Estimate (USD Million)	Status
CAREM	PWR	27	CNEA, Argentina	97	Licensing Stage
SMART	PWR	100	KAERI, ROK	497	Licensed
VBER-300	PWR	295	OKBM Afrikantov, Russia	910	Licensing Stage
KLT-40S	PWR	35	OKBM Afrikantov, Russia	259–294	Under Construction
mPower	PWR	180	BWX Technologies, USA	1007, twin units	Under Development
NuScale	PWR	45	NuScale Power, USA	816	Licensing Stage
Westinghouse SMR	PWR	225	Westinghouse, USA	668	Under Development
AVB-6	PWR	3–10	OKBM Afrikantov, Russia	NA	Under Development
SMR-160	PWR	160	Holtec, USA	NA	Under Development
PRISM	SFR	311	GE-Hitachi, USA & Japan	NA	Licensing Stage
4S	SFR	10	Toshiba, Japan	NA	Under Development
SLIMM	SFR	4–40	University of New Mexico's ISNPS, USA	NA	Under Development
G4M	LFR	25	Gen4 Energy, USA	NA	Under Development
EM^2	GFR-He	265	General Atomics, USA	NA	Under Development
GT-MHR	HTR-He	286	General Atomics. USA – Prismatic	NA	Under Development
GTHTR300	HTR-He	275	JAEA, Japan – Prismatic	NA	Under Development
Xe-100	HTR-He	50	X-Energy, USA – Pebble Bed	NA	Under Development
HTR-PM	HTR-He	2 × 105	Tsinghua Univ. & Shandong Shidaowan Nuclear Power, PRC – Pebble Bed	NA	Under Construction
IMSR	MSR	192	Terrestrial Energy, Canada, USA, & UK	< 1000	Under Development
Stable Salt Reactor	MSR	300	Molex Energy UK	NA	Under Development
ThorCon	MSR	250	Martingale Consortium, International	1200	Under Development
MSTW	MSR	115	Seaborg Technologies, Denmark	NA	Under Development

The clean and abundant supply of fresh water and energy is becoming a global challenge. The codependence of energy water and the shortfall is an emerging concern. D.T Ingersoll et al. [26] have studied the feasibility of NuScale SMR design for both generations of electricity and distillation of seawater. In a detailed report, the NuScale small modular reactor is coupled with various distillation technologies, and its technical and economic viability has been studied. It shows that the NuScale plant coupled with reverse osmosis (R.O.) technology provides the most economical conditions compared to other distillation technologies.

2.3. NUSCALE PLANT OVERVIEW

This new SMR design was proposed by United States vendor company NuScale for the commercial application based on pressurized light water-cooled reactor technology. This design includes several features such as improved safety, reduced complexity, and reduced cost. We will look at some of the design features of the NuScale reactor module.

The NuScale core reactor produces 160 MWt energy in a single module in the nuclear power plant. The reactor core includes of an integral pressure vessel which is surrounded by a steel containment vessel. Individual power modules are put side-by-side in the same pool. The entire assembly of reactors and a model of a single NuScale SMR is shown in Figure 1.

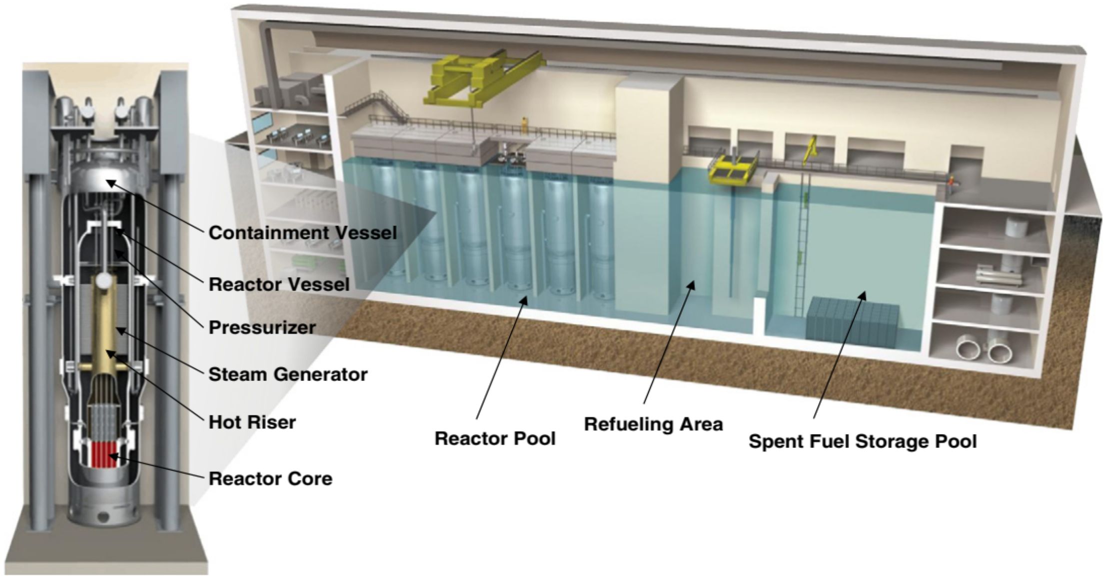


Figure 1. NuScale nuclear reactor design [26].

The core of the NuScale reactor consists of 37 fuel rods and 16 control rods. For the various parts of a single module reactor, the central hot riser above the core and stream generator coil tubes around the hot riser and a pressurizer can be seen in Figure 1.

The steam generator consists of two independent helical coil tube bundles with a separate water feed inlet and steam outlet lines. To manage pressure, a pressurizer is installed at the vessel's top head. The whole reactor vessel is 20 m tall and has 2.7 m in diameter. The reactor is itself enclosed in a steel containment 24.6 m tall and 4.6 m in diameter.

The NuScale design offers several unique features which make it outstanding compared to other plant designs. Its compact size and prefabrication at the factory reduce the overall construction activities at the site. It is easily transported and installed at the site to lessen plant construction costs. A natural circulation cooling system eliminates the need for pumps, pipes, and additional components required for the complex cooling system. This also reduces the need for maintenance and reduces potential failure risks of these components, which improves safety.

The NuScale design can be categorized as an example of the proven and mature pressurized water reactor technology. Thus, there is an abundance of operational data, regulatory laws, and standards. This reduces the uncertainties and risks in plant operations.

The NuScale design follows the modular approach, which is a crucial feature of any SMR, and extends this approach so that each of the NuScale modules works independently and each module has a separate power conversion system. Therefore, the plant can be configured so that some modules generate electricity and other modules can generate steam for the thermal heat applications. Also, NuScale reactor modules can be scaled from 1 to 12 modules providing additional flexibility and deployability according to requirements and economic limitations.

The NuScale design also provides resilience against any plant accident and enhanced safety against events such as Fukushima. The NuScale design containment vessel is a rated pressure of 5.5 MPa, which is twelve times higher than conventional containments. This higher-pressure capability is achieved by decreasing the containment diameter instead of needing costly approaches such as increasing the containment wall thickness. Therefore, the design can withstand the loss of coolant accidents (LOCAs) that may occur in emergencies.

The unique safety design of the NuScale reactor and its emergency core cooling system is very simple. It consists of two independent reactor vent valves (RVVs) and two independent reactor recirculation valves (RRVs). The cooling system removes the excessive heat from the core by steam condensation inside the walls of cold containment vessel and limits the containment pressure. The reactor core heat is removed by recirculating the steam, which is condensed inside the surface of containment vessel by passive conduction and convection using the reactor pool water. Then after removing the heat from steam through the condensation process, the water is reentered into the reactor pressure vessel though the RRVs. The cooling and reactor safety system designed by NuScale reactor is shown in Figure 2.

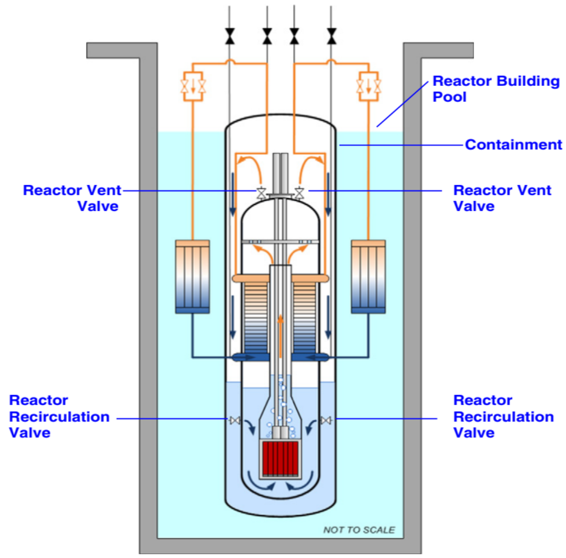


Figure 2. The cooling and reactor safety system [26].

The validation of long-term cooling for the NuScale emergency core cooling system has been carried out. In worst-case scenarios, the simulation results show that in severe accident situations, the 10 MWt reactor power has decayed to 1.1 MWt after one day and after 30 days, the decay heat is generated per module is less than 400 kW. In a report [27], Scott et al. have done a comparative study of the NuScale design safety in case of a severe accident with different light water reactor designs and showed that the NuScale ECCS ensures secure operations of the nuclear reactor and robust response in emergency conditions.

In addition to NuScale, around 39 proposed SMR designs are under development by 15 different countries [28]. These SMR designs are at different development stages. Some are reaching the demonstration phase, while others are at licensing phase.

The decarbonization of our global energy and heat requirements is crucial for the future of the world. SMRs can play essential roles in decarbonizing energy and heat generation if the economic challenges and the need for capital investments for nuclear power generation can be overcome.

The International Energy Agency (IEA) indicates that 22% of the world's fossil fuel electricity is produced by combined heat and power plants (CHP) in 2016. This makes 16% of the global electricity. SMRs used to co-produce electricity and heat energy are expected to have a 5% higher cost compared to reactor plants producing only electricity—a report by Tomi. Et al. [29] has done a detailed study on the economic model for the NuScale and DHR-400 reactors for district heating and cooling grid systems. The report showed that SMRs are cost-efficient for district heating with an average internal return rate from 20% - 7% in the studied reference system. They have shown that large heat pumps with a coefficient of 3.5 had similar economic performance compared to the SMRs, but their potential is limited to available heat sources.

2.4. Licensing Process

The licensing process for a nuclear reactor is an essential step before deployment and any commercial operations to assure that requirements are met for safety, security, and radiation protection. The licensing process can be pretty strict and complex and requires time to complete. This can increase the plant construction and commissioning time and, eventually, the capital cost. Also, the regulatory authorities in each country additionally can have their requirements and licensing laws, and no uniform standard licensing laws exist around the world. This makes the whole process very lengthy.

Ramana et al. [30] have carried out a detailed review of the licensing process of SMRs in several nations, including the United States, China, and Russia, South Korea, and India. These are significant players in the nuclear industry, and vendor companies are focusing on building and licensing in these countries, especially for potential export. Each country has developed a regulatory system that suits its political, legal, and industrial requirements. These regulations are obligations on the licensee to fulfill to get authorization and operate any nuclear facility. The regulatory bodies develop a regulatory framework according to the state's legal system and the guidance provided in the IAEA safety standards. There are three basic regulatory approaches that are followed as a standard defined by IAEA. These are a perspective approach, a performance-based approach, and a goal-setting-based regulatory approach. There are ways to define an acceptance criterion used for judging whether the analysis demonstrates adequate safety.

The perspective regulatory approach focuses on what must be achieved. The regulator body defines essential acceptance criteria and specific acceptance criteria. The licensee must demonstrate that these requirements are met. The goal-setting approach is similar to the performance-based approach in that the regulator sets the safety goals. It is then the licensee's responsibility to demonstrate compliance with these goals and justify that the design codes and specific acceptance criteria demonstrate adequate safety.

Predefined rules and norms are used to judge the reactor design, components, facility performance, and safety criteria. The perspective approach mainly utilizes deterministic considerations because of ensuring safety [31]. Generally, the regulator develops an extensive range of codes that define security requirements and standards to enable technical judgment about the nuclear plant's safety [32]. This approach is beneficial from the licensing point of view because the codes and standards are tailored to the reactor-specific design and construction country. This way, the licensing process's ambiguity, and uncertainty are reduced by leaving subjectivity to the regulatory authority. The other advantage of the perspective approach-based regulation is that the licensing process is efficient and can move more quickly, especially for experienced vendors, contractors, and operators.

The major challenge for SMRs regarding the prospective-based licensing is developing new codes, standards, and practices tailored for each newly developed SMR design. Therefore, this results in a significant burden on the regulatory bodies. In one way, this approach provides licensee more flexibility and can foresee what would be acceptable for the regulatory body to receive authorizations. Specific technical requirements can be taken from the international industry as standards to achieve adequate safety.

The U.S. Nuclear Regulatory Commission (NRC) uses a perspective-based approach for nuclear regulation. Most of the countries considering developing and building SMRs are utilizing a similar approach for SMR licensing.

On the other hand, the focus is more on achieving the desired results in the performance-based regulatory approach, so performance and effectiveness are given more importance. The regulatory body sets the general acceptance criteria and leaves it to the licensee to set its specific acceptance criteria. The licensee must demonstrate that specified safety aspects are met satisfactorily. In this approach, greater involvement is required by the operator in determining how the objectives are achieved. The overall safety is the responsibility of the licensee. As a result, this approach lessens the administrative burden on the regulatory body. The goal-setting approach relates more extensively to risk-informed regulation and ensures that all responsibilities are met to a reasonable possible. Often the term ALARP (As Low as Reasonably Possible) principle is used in this regard [33-35].

2.5. Pressurized Water Reactors

In a pressurized water reactor, light water is used as a coolant and moderator inside the reactor vessel. The high pressure around 150 atmosphere helps to increase the water temperature to 325 °C without it boiling. This high temperature and pressurized water in the primary cooling system transfers heat energy to the secondary water system through the heat exchanger. The water inside the secondary system absorbs that energy and is converted into steam, which drives the turbine unit for electricity production. This way, the water in two primary and secondary units does not mix, thus avoiding any contamination from nuclear radiation. The PWR's closed cycle uses water in an isolated, pressurized water loop flowing between the reactor core and heat exchangers that create steam for the steam turbine power cycle.

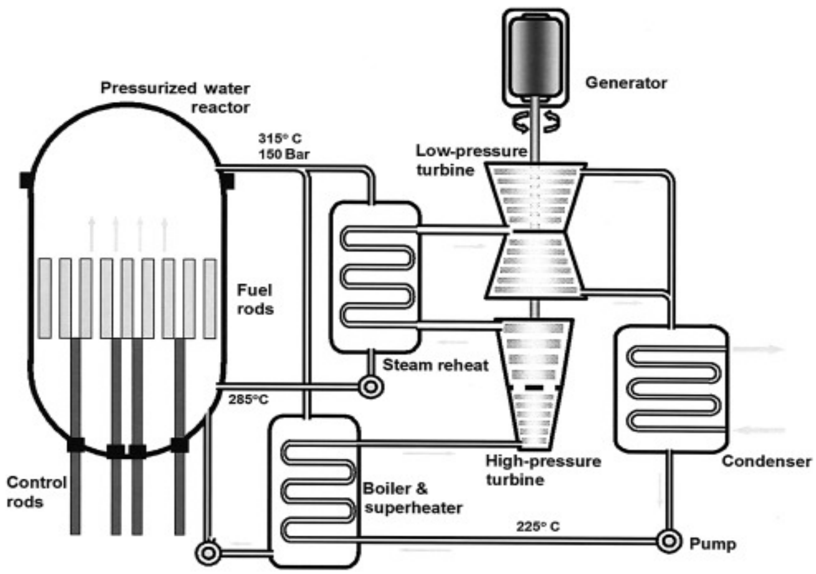


Figure 3. PWR reactor design [36]

Generally, the fuel in pressurized water reactor (PWR) is enriched from 0.7% (natural uranium) to 4% [37]. Compared to boiling water reactors (BWRs), PWRs utilize slightly higher enriched uranium oxide fuel into the core in the form of a pellet. Due to this reason, these reactors have higher power densities than BWRs. Each fuel assembly consists of 200-300 fuel rods. Overall about 100 tons of fuel are loaded into a typical PWR, and a typical reactor can produce 1000 MW of electricity. The fuel assemblies or bundles are loaded vertically inside the core. The fuel rods are pressurized with helium to about 3.4 MPa. The control rods are used to control the reaction rate in the core. In case of any emergency, they fall under gravity and stop the nuclear chain reaction by absorbing the energetic neutrons.

The PWR is different from a boiling water reactor in the sense that it does not turn water into steam in the core. The secondary loop is utilized to produce steam for power generation. The layout of a PWR provides additional shielding from radioactivity since all the primary loop water is circulated within the plant's containment building about the same. Both operate on the Rankine steam cycle. The typical efficiency of these reactors is 33-34% which is determined by the upper limit of the Carnot cycle efficiency of 46% [38].

2.6. Small Modular Reactors (SMRs)

From the beginning, nuclear power plants have been very costly to build, and especially for the developing nations, it has been nearly impossible to invest in nuclear energy to meet their energy demands. In the few decades, the nuclear power plant industry has started to focus on developing small nuclear power reactors. These small-scale modular nuclear reactors started gaining popularity due to their small size, security features, low capital investment, increasing energy needs, and the particular interest of developing nations that do not have nuclear power plants in their countries. Therefore, nuclear plant vendors started investing in these types of plants to capture the billion-dollar industry of the future. The compact size of SMRs allows plant manufacturers to make the entire plant in factories and ship them through railroad networks, cargo planes, and ships to any part of the world. These small reactors can be set up remotely where conventional power stations and the grid are challenging to build. A typical reactor can produce an output of up to 300 MW of electricity. According to IAEA, the KLT-40C is the first Russian-made offshore SMR plant to start its commercial operation and connect with the grid in 2019 [39].

Based on fuel, moderator, and coolant, different SMR designs have been proposed and are at different development phases in the industry in different countries.

Among these different designs, light water-based SMR plants are most attractive due to the lowest technical risks and regulatory issues because light water-based conventional nuclear power plants have been used for decades, and the technology has matured. In these plants, light water is used as a modulator and coolant [40]. In the United States, the Department of Energy encourages and supports private companies to invest and develop light water-based SMRs. The Nuclear Regulatory Commission (NRC) is actively reviewing the proposed designs for licensing and regulatory procedures and helping to develop regulations for SMR plants in the country. Companies such as B&W, Westinghouse, and NuScale are prominent in the race for developing these plants in the U.S. The NuScale designed power plant is hoped to be completed and start its operation in 2027. The estimated cost of the reactor is $4200/kW hour [40], which is considerably lower compared to conventional power plants' cost of $5500 -$8100/kW hour [41].].

2.6.1. Fast Neutron Reactors

The fast neutron reactors are much smaller and simpler than light-water nuclear reactors because they do not have moderators and pressure control units, and they do not need moderators to slow down the neutrons. Usually, a liquid metal such as sodium or lead is used to cool down the reactor.

As a fundamental principle of quantum physics, we know that the energy of electromagnetic radiation depends on its energy, and the relation is described as $E= h\upsilon$. Where E is the energy of the radiation and υ is the frequency of electromagnetic radiation. Therefore, if we look at the electromagnetic spectrum of radiation, we see their radiation which has long-wavelengths or low frequency, such as radio waves. Low energy radiations lie at one end of the spectrum, and radiation with a short wavelength or high frequency are highly energetic radiations such as X-rays and gamma rays on the other end of the spectrum. These high-energy radiations interact with an atom and can remove an inner shell electron, causing an atom's ionization. These are called ionizing radiation. From the wave-particle duality principle, we know that electromagnetic waves may behave both as particles or waves.

Ionizing radiation can be different depending on the phenomenon and particle type taking part in the ionization process. An example of ionization radiation is called particulate radiation. The energy carried by these radiations is in the form of kinetic energy. Alpha and beta radiation are examples of this particulate radiation which causes direct ionization of an atom due to their charged nature and coulombic interaction with the inner shell electrons of the atom. Whereas a neutron is a neutral particle, and it ionizes an atom without involving any coulombic interactions. The collision between fast-moving neutrons and nuclei of an atom causes the indirect ionization of the atom. Gamma and x-rays are examples of indirect ionizing radiation.

Ionizing radiation affects the human body due to the energy carried by this radiation and the transfer to the human cells. To determine the effect of this energy on the biological bodies, we need to quantify the amount of energy imparted per unit mass of a specific body or whole body. The SI unit used to measure this energy absorbed is called the gray (Gy). One gray is defined as being equal to one joule of energy deposited in a body of mass 1kg. The effect of radiation on biological bodies depends not only on the dose absorbed but also on the intensity of the ionization on living cells. For instance, alpha radiation can cause 5-20 times more damage to body tissues than the same amount of absorbed beta or gamma radiation [42]. Therefore, we do not just need to know the amount of radiation absorbed by a body but also measure the potential biological impact due to the exposure to ionizing radiation [43]. This quantification of harmful effects on biological ionizing radiations is measured in Sieverts (Sv). Normal humans receive a 2.4 mSv/yr dose due to naturally occurring radiation in our surrounding environment. This is called background radiation.

Radiation can be dangerous. It can pass through materials and is not be detected, and can cause harm to body cells. Cells can be severely damaged. This can have the potential of altering the DNA, which is the hereditary material in our cells. The cells are affected in different ways. When exposed to radiation, they may die, repair automatically or sometimes mutate, leading to cancer in the body. There are several radiation sources in nature, including building materials around us, space, different natural materials on earth, and radon, which adds up to background radiation. Also, there are artificial sources of radiation as well usually used in medical science for imaging devices for diagnosis purpose of diseases such as X-ray machines, C.T. scan, fluoroscopy, and others.

When radioactive elements, decay different ionizing radiations are emitted. Alpha radiation is emitted when a radioactive element is in an unstable state of decay. The alpha particle is especially helium nuclei, which has a positive charge. All the heavy elements with atomic number $Z \geq 83$ naturally emit alpha radiation. As we know, when the atomic number increases, there is a more significant number of electrons and protons in the nucleus resulting in a lower binding force. Thus, the atom becomes unstable. The unstable atom emits radiation, and the result is called the daughter element. From mass-energy conservation, we know that the total energy should be conserved. The difference in atomic mass of the parent atom to the total atomic mass of daughter atoms is converted into energy. This energy released during the decay appears as kinetic energy. From the mass-energy principle, the energy of alpha particle produced can be found as follows;

$$E_a = \frac{MQ}{(m+M)} \quad \text{Eq. 1}$$

Where; Eα is the alpha particle's energy, Q is the charge, the alpha particle's mass is M, while the daughter element's mass is m.

In a nuclear reactor containing fissile uranium, the alpha decays occur naturally. The alpha decay of uranium is given by the following relation [44];

$$^{238}_{92}\mathrm{U} \rightarrow {}^{4}_{2}\mathrm{He} + {}^{234}_{90}\mathrm{Th} \quad \text{Eq. 2}$$

The parent U^{238} nucleus emits alpha radiation and changes into the daughter element thorium. Beta rays are produced when a radioactive atom decays and the nucleus splits into daughter elements then electrons and antineutrino particles are emitted during the decay process. These negatively charged electrons are also called beta radiations. The antineutrino is produced due to the charge conservation principle. The following reaction shows the production of beta rays when a U^{238} atom decay into a Pa^{234} (protactinium-234) atom [45];

$$^{234}_{90}\mathrm{Th} \rightarrow {}^{0}_{-1}\mathbf{e} + {}^{234}_{91}\mathrm{Pa} + \bar{\nu} \quad \text{Eq. 3}$$

The energy released due to mass difference during the decay is used for recoiling by the emitted particles. Since the daughter nucleus is heavy, most of the recoil energy is taken up as kinetic energy by the beta particle and the antineutrino.

There is another kind of beta decay where a positron is produced instead of an electron. Positron is the same as an electron, just with an opposite charge as an electron. During the positron emission, a neutron is also emitted during the decay process.

2.6.1.1.Neutrons

Neutrons are uncharged particles and have approximately the same mass as a proton. Neutrons are produced during the decay process, and these create a significant risk for the safety of a nuclear power plant. When a U235 (uranium-235) atom is split in a fission process, an average of 2.44 neutrons is produced with other daughter nuclei fission products and beta rays. The reaction chain of uranium-235 is given as below [45,46]:

$$ {}_{0}^{1}n + {}_{92}^{235}U \rightarrow {}_{92}^{236}U \overset{85\%}{\rightarrow} fission + 2.44\,{}_{0}^{1}\bar{n} \qquad \text{Eq. 4} $$

$$ {}_{0}^{1}n + {}_{92}^{235}U \rightarrow {}_{92}^{236}U \overset{15\%}{\rightarrow} {}_{92}^{236}U + {}_{0}^{0}\gamma \qquad \text{Eq. 5} $$

As we can see from the fission reaction, when a neutron is absorbed by uranium-235 and becomes uranium-236, it is unstable and can undergo fission. This results in fission products, and as noted, 2.44 neutrons on average. The fission products are usually krypton-94 and barium-139. These are unstable radioactive elements. Fission occurs about 85% when the uranium-235 atom captures a neutron. The other 15% of the absorptions leads to gammas rays of various energies 0.495 MeV, 0.113 MeV, or 0.171 MeV [35].

These highly energetic neutrons produced during the fission reaction are used to continue the chain fission reaction. The moderator is used to reduce the kinetic energy of these fission neutrons to be used to continue the reaction. These neutrons may also leak out from the core. Therefore, shielding is critical.

Neutrons interact with the atomic nucleus through elastic or inelastic collisions. Generally, the light elements are used to reduce the energy of the neutrons through elastic collisions. The fast-moving neutrons interact with the target nucleus and may get absorbed or scattered by losing their energy. In inelastic collisions with heavy elements, the neutrons can be scattered back after the collision. These neutrons may excite an atom and produce gamma rays. Therefore, heavy elements are not used for neutron shielding. When a neutron collides with a target nucleus, there is a probability that neutron will be absorbed. The neutron capture cross-section is its probability of being absorbed by the target nucleus. It is measured in units of barns (σ). This capture cross-section depends on neutron energy and the target element. Multiple levels of neutron shielding are used to reduce the radiation risk in nuclear power plants.

2.6.1.2.Gamma Emission

Gamma rays are produced both during the fission process and through the de-excitation process of an excited daughter nucleus after radioactive decay. Gamma rays are high-energy photons, unlike charges alpha and beta rays, which can penetrate through materials. Gamma can cause ionization, and high exposure can damage cells, and exposure can cause cancer. Therefore, shielding gamma rays is vital for human health and nuclear plant safety. High-density materials or materials with a high atomic number is used for effective gamma shielding. The gamma-ray attenuation through a material is given by,

$$\mathrm{I} = \mathrm{I}_0 . \mathrm{e}^{-\mu x} \qquad \text{Eq. 6}$$

Where I_o is the incident ray intensity, μ is the linear attenuation coefficient (cm^{-1}), and the physical thickness of the absorber (cm). Though gamma rays are far more penetrating, their ionizing power is less compared to alpha particles

As noted, in a nuclear reactor, gamma can be produced by the fission process and radioactive decay of the fission products. Also, when uranium-238 decays, gamma rays are produced along with other alpha and beta radiations. The two gamma rays produced have energies of 0.0496 MeV and 0.114 MeV [46]. The decay reaction is given below [45];

$$^{238}_{92}U \rightarrow {}^{234}_{90}Th + {}^{4}_{2}He + 2{}^{0}_{0}\gamma \qquad \text{Eq. 7}$$

2.6.2. Shielding from Nuclear reactor radiations

Usually, 3-5% enriched uranium is used for this purpose. This radioactive material is a constant source of ionizing radiation. Since the radiations are not visible and some radiations are very powerful, they can pass through thick blocks of materials. Therefore, shielding these radiations is very complex. The radiation shielding is done to protect the workers—the plant's safe operation, the public, and the environment. There are three critical factors for nuclear radiation safety and protection. These are exposure time, distance from the source, and effective shielding.

The exposure time must be minimized to protect from the effects of ionizing radiation on the human body. For this purpose, a maximum time limit is defined for staying in a radiation environment and is called stay time. It is defined as follows;

Stay Time= Exposure limit/dose rate:

Stay time is closely monitored for workers who work closely inside a radioactive zone of the nuclear plant. Another essential component in terms of radiation safety is the distance from the source of radiation. Depending on the radiation type, their penetration from the materials is also different. Alpha and beta radiations can only penetrate from a few inches to a couple of feet, but the gamma rays can travel a long distance through materials. The radiation power is in inverse proportion to distance travel from the source 1/r2.

Shielding is used to protect against harmful effects of ionizing radiation; for this purpose, different shielding materials are used when the radiations pass through the material, these radiations are get absorbed. The thickness of the shielding material is an essential parameter of consideration because different thicknesses of shielding materials can stop different alpha, beta, and gamma rays. Alpha particles, due to relatively heavy mass, can travel only a short distance inside a material so they create a minimum external hazard, but they can pose a severe threat to body organs and tissues if ingested. Alpha radiation is emitted by radon, a naturally occurring element in the earth. Radon gas is present under soil between the cracks present between rocks under the ground. During the decay, alpha emission may cause excitation of an atom leading to gamma rays or X-ray emission from internal conversion, which has a far more penetrating power.

Since alpha radiations have less ability of penetration, therefore these do not create many radiation safety problems during the nuclear power plant operation. The walls of the reactor core and steel walls can quickly stop these radiations from leaking outside. However, there is always concern due to airborne alpha emitters that might be breathed in and lead to radiation dose in the lungs.

Beta particles have enough energy that they can pass through the skin and can cause external radiation damage. Beta radiations may produce secondary radiations called bremsstrahlung radiation when an electrically charged particle like an electron is accelerated or decelerated in the electromagnetic field of the nucleus. This radiative loss or bremsstrahlung radiation effect is more frequently found in heavy atomic number materials. Lead and plastic are more commonly used for shielding beta radiations. For effective shielding from the beta radiation, a light atomic number material is first used, then a heavy atomic number material is used. Wesley et al. [47] have studied beta radiation shielding using plastic and lead in detail and showed that to reduce the transmittance of beta radiation, first plastic is placed and then lead.

Alphas and betas are both charged particles, so the coulombic force of attraction or repulsion plays a role in dissipating the energy of these particles and finally leading to absorption by the shielding material.

Since it is a neutral particle, there is no coulombic interaction in the case of a neutron. Therefore, shielding is more complicated. A direct collision of a neutron occurs with the material. Neutrons are slowed down by losing their kinetic energy, and absorption ultimately occurs. The law of conservation of momentum becomes helpful when considering the collision between particles. An absorbing material whose atomic weight is close to the neutron mass is utilized to shield the neutrons effectively. If a material with heavy atomic weight is used, the neutron will bounce back after the collision, and no attenuation will be obtained. Generally, high-density concrete is used to provide adequate shielding from neutron and proton coming from the nuclear core. Apart from the atomic number, the absorption cross-section of different materials is also an important parameter. The absorption cross-section values for different materials are given in Table 2.

Table 2. The absorption cross-section for thermal and fast neutrons [48]

		Thermal neutron			Fast neutron		
		Scattering	Capture	Fission	Scattering	Capture	Fission
Moderator	H-1	20	0.2	-	4	0.00004	-
	H-2	4	0.0003	-	3	0.000007	-
	C-12	5	0.002	-	2	0.00001	-
Structural materials, others	Zr-90	5	0.006	-	5	0.006	-
	Fe-56	10	2	-	20	0.003	-
	Cr-52	3	0.5	-	3	0.002	-
	Ni-58	20	3	-	3	0.008	-
	O-16	4	0.0001	-	3	0.00000003	-
Absorber	B-10	2	200	-	2	0.4	-
	Cd-113	100	30	-	4	0.05	-
	Xe-135	400	2,000,000	-	5	0.0008	-
	In-115	2	100	-	4	0.02	-
Fuel	U-235	10	99	583	4	0.09	1
	U-238	9	2	0.00002	5	0.07	0.3
	Pu-239	8	269	748	5	0.05	2

2.6.2.1.Absorption cross sections for shielding of neutrons

Boron, cadmium, and lithium are used in nuclear reactors because of their extremely good neutron absorption cross-sections [48]. Fast-moving neutrons may also generate secondary gamma rays. When these neutrons are absorbed and give off their energy by colliding with target nuclei, transitions within the nucleus result in radiative energy loss and produce gammas. These gamma rays have more penetrating power than neutrons. These are high-energy photons, and shielding against them can be difficult. High density, thick, high atomic numbered elements are utilized for these shielding purposes.

2.6.2.2.Removal cross section:

Figure 4 depicts an important experiment at the Oak Ridge National Laboratory a few years ago on the attenuation of fission neutrons in water. The thermal neutrons from a reactor impinge onto a disc of enriched uranium located at the end of a large water tank. The disc becomes a source of fission neutrons in water due to uranium emissions induced by the thermal neutrons.

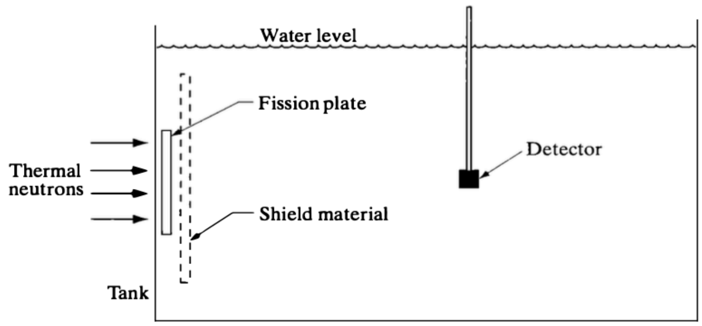


Figure 4. Experimental setup of the attenuation of fission neutrons in water to compute removal cross-section [49].

The probability of a fast or fission-energy neutron undergoing an initial collision, which removes it from the group of penetrating uncollided neutrons, is given by the removal cross-section. The fission neutrons come from a single point source and travel towards the detector. When one of these neutrons collides with hydrogen, its energy is decreased on average by half. The mean free path for the subsequent collision is substantially reduced because of the rapid increase in cross-section with decreasing energy.

As a result, the neutron suffers a second collision near the first, when its energy and mean free path are decreased once again, and so on. As a result, the neutron's energy finally falls below 1 MeV. However, because the mean free path is constantly decreasing, as a consequence, a single hydrogen collision essentially eliminates a neutron from the fast neutrons entering the detector.

Scattering from the oxygen in water is complicated because this nucleus tends to scatter neutrons preferentially through small angles. Neutrons scattered at such angles may continue to the detector. However, neutrons scattered at large angles by oxygen are lost because they suffer collisions with hydrogen and are removed. Thus, a single collision with hydrogen and all but forward scattering collisions with oxygen effectively remove fast neutrons from water.

The fact that the oxygen in water scatters neutrons preferentially through narrow angles complicates scattering by this nucleus. Neutrons scattered at such angles still have a chance of reaching the detector. Neutrons dispersed at large angles by oxygen, on the other hand, are lost because they collide with hydrogen and are therefore eliminated. As a result, a single collision with hydrogen and all collisions with oxygen successfully remove fast neutrons from water, except for forwarding scattering [49.

CHAPTER 3. METHODOLOGY

3.1. Objective

The unique characteristics of SMR's may make them suitable for new and innovative applications. In addition, due to their extreme safety characteristics, SMR's may be located closer to electrical load centers in populated communities or adjacent to industrial complexes where the output can be used directly in the form of process heat or hydrogen generation or desalination of water. SMRs can be appropriately designed for use in urban areas; they can minimize the dose exposure on the public compared with current reactor designs. If the public realizes the true potential of nuclear energy and SMRs, they can significantly reduce CO2 emissions. NuScale core design has been used to determine if this core design is convenient for the public areas.

3.2. Data Collection

The MCNP code has been used to carry out the work. The Monte Carlo particle analysis program is considered the most advanced and capable analysis tool for this type of application. This code has been used to calculate the radiation dose for the model design, represent major facility structures and components with an appropriate level of detail using data for the NuScale design.

The outcome of this research is an evaluation of the NuScale small modular nuclear reactor and its radiation effects and health physics implications. Results have been dose rates at the reactor core and the pool and then have been compared with a PWR reactor. When the model design was calculated with MCNP code, the results highlighted this system's functions, advantages, and disadvantages for an appropriate geographical area.

This study demonstrates the value of the reactor design that has a lower radiation dose and why the SMR (NuScale) may be an attractive option. The audience for this design would be developing countries that are considering SMR designs.

3.2.1. Monte Carlo N-Particle Transport Simulations (MCNP)

MCNP6 is a neutral particle Monte Carlo simulation code that may simulate neutron, photon, and coupled neutron/photon transport. Monte Carlo N-Particle Transport Simulations (MCNP) have been used for analyzing neutron radiation at the core surface and in the pool outside the reactor. The code works with any 3D material arrangement in geometric cells. MCNP6 has been used for the NuScale dose calculation methodology by performing fixed source transport calculations. For the core and pool dose calculations, the neutron flux has been calculated, and an F5 tally, that is, a tally card to measure flux at a point or point detector, has been used to calculate dose at the core surface and the point detector outside the reactor. The dose conversion cards have been used for the F5 tally to obtain the dose amount for the points of interest. F4 (the average cell flux tally) was used to compute dose for the regions with dose conversion card.

3.3. Research Design Model

The NuScale reactor design model has been simulated for this project by using MCNP code. Dose calculations have been performed utilizing data and numerical information from MCNP, compared with the PWR reactor, and then used to assess the SMR design. As mentioned above, the MCNP calculation was done for the NuScale reactor. The core is arranged in 17×17 square lattice arrays with 264 fuel rods, 24 guides for reactor control, and one central instrumentation tube [51]. The NuScale reactor generates 160 MW of thermal power using four UO2 fuel enrichment. The coolant and moderator are light water (H2O). The enrichment of the fuel in the core ranges from 4.29 w/o % (with Gd2O3) to 4.33 w/o % (without Gd2O3). The values are shown in Table 3. In Figure 5, the numbers represent fuel assemblies of four different compositions of Gd2O3 and UO2 in the reactor's core. 1: 4.33 w/o UO2 and 0% Gd2O3, 2: 4.32 w/o UO2 and 2% Gd2O3, 3: 4.30 w/o UO2 and 6% Gd2O3, 4: 4.29 w/o UO2 and 0% Gd2O3. The 17×17 square lattice arrays with the four different compositions of the fuel are distributed across 37 assemblies. Shielding layers around the core, including a heavy reflector, water reflector, barrel, outer water reflector, and reactor vessel. As shielding materials, water and SS-316 have been used in these layers. Stainless steel heavy reflector varies in radial thickness nominally ten to thirty centimeters surrounding the reactor core [54]. For the attenuation calculations, the material of iron with a high mass percentage of the composition of SS-316 has been taken.

Table 3. The compositions of Gd_2O_3 and UO_2 in the fuel assemblies.

Gd	**UO_2**
0 %	4.33 w/o
2 %	4.32 w/o
6 %	4.30 w/o
8 %	4.29 w/o

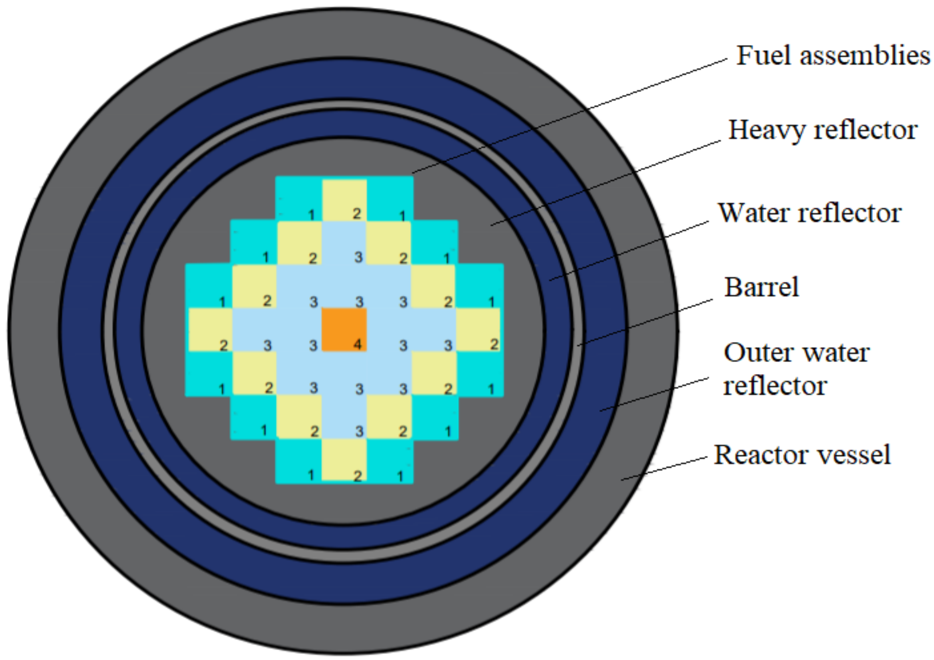


Figure 5. NuScale Core design. 1: 4.33 w/o UO_2 and 0% Gd_2O_3, 2: 4.32 w/o UO_2 and 2% Gd_2O_3, 3: 4.30 w/o UO_2 and 6% Gd_2O_3, 4: 4.29 w/o UO_2 and 0% Gd_2O_3.

The reactor core was modeled with radially heavy reflector, water reflectors, barrel, and reactor vessel without control rods and nozzles. The primary technical parameters for the NuScale reactor design basic are shown in Table 4 and additional information in Table 5.

Table 4. Basic technical data of the NuScale reactor design (IAEA Aris Database, 2020).

Parameter	**Value**
Reactor thermal power	160MW$_{th}$ (45 MWe)
Coolant and moderator	Light water
Active core height	200 cm
Fuel pellet material	Uranium dioxide (UO_2)
Fuel assembly type	Square array 17×17
Fuel cycle Length	24 months
Number of pins in a FA	17x 7 (264 fuel rods+24guide tubes+1 measurement tube)

Table 5. General information and major parameters about the NuScale reactor design [59],[60].

Parameter	Value
Fuel assembly pitch	21.5cm
Water reflector radius	105.45 cm
Heavy reflector radius	92.875 cm
Barrel outer radius	111.15 cm
Reactor vessel inner radius	130.79 cm
Reactor vessel outer radius	154.15 cm
Active fuel length	200cm
Core active radius	150.5 cm
Number of fuel assembly	37
Fuel assembly pitch	21.503 cm
Fuel assembly geometry	Square (width = 21.4 cm)
Fissile enrichment	<4.95w/o
Gap width	0.0082 cm
Clad thickness	0.0609 cm
Overall fuel radius (fuel pellet + gap + clad)	0.4751 cm
Clad material	M5
Fuel rod pitch	1.259 cm
Core reflector width	6.35–30.988 cm
Coolant type	Light Water
Core barrel inner diameter/outer diameter	187.96 cm/198.12 cm
Average coolant temperature	284 ^{o}C
Inlet temperature at full power	258.3 ^{o}C
Burnable Absorber material	Gd_2O_3
Fuel pellet density	10.53gr/cm3
Gd_2O_3 concentration	Up to 8%

The Monte Carlo method has the advantage of allowing an accurate 3D representation of the reactor vessels and key components without introducing extra geometric error into the meshing process. The geometric model of the NuScale SMR includes the explicit reactor core, reflectors, reactor vessel, and reactor internals. The fuel assemblies include the fuel rods. The configuration of horizontal and vertical views of the NuScale MCNP model are shown in Figures 6 a and b, respectively. NuScale design dimensions and material composition data are used to develop the input parameters of the geometric model.

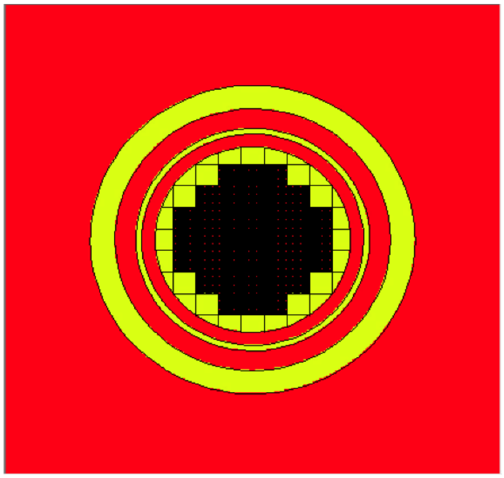
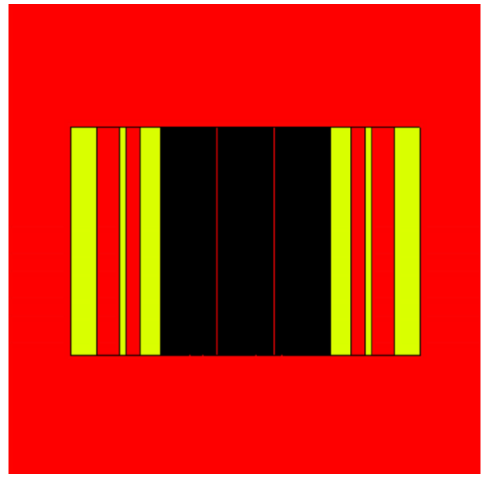



Figure 6. a) XY horizontal views of the reactor core b) XZ vertical views of reactor core

The reactor core has been modeled for a steady-state condition that is independent of time.

All of the heat produced by the fission process in the system must be removed as it is produced. The design and operation of the coolant system are, of course, critical considerations in a nuclear reactor [55].

For the model design, the core structure was designed without control rods and other reactor components. The material of many components in the core structure, such as barrel, heavy reflector, and reactor vessel, is steel. However, the iron material, which has a high percentage in the steel composition, was processed for the analytical calculations.

The reactor was at a steady-state that was at full power conditions and was independent of time. Also, the reactor was assumed at critical conditions, but for the MCNP result, it was at a supercritical state.

The information about fuel assemblies, fuel rods, cladding material, moderator, and different material densities have been inserted into the MCNP6 model to perform a fixed source neutron transport calculation. The temperature-dependent cross-section data libraries have been created using the MCNP6 code package's Doppler broadening utility function MAKXSF. The code output has obtained the neutron dose in the core surface and core region, and the pool. The VISED simulator code has been used for a core design for illustrations. Figure 7 illustrates that outside of the reactor with a point inside the reactor pool.

3.3.1. NuScale Reactor Geometry (Reactor Geometry, Source information for MCNP)

The NuScale core design for this project is shown in Fig. 7. The core is surrounded by the reflectors, barrel, and water for shielding. A point detector has been placed into the pool that is surrounding the reactor. The distance of this point detector is 215 cm from the core. All radius information for these materials has been given in Table 5. The source term of neutrons per fission in the core and the pool have been obtained from MCNP. According to Eq. 8, the intensity of neutrons has been calculated.

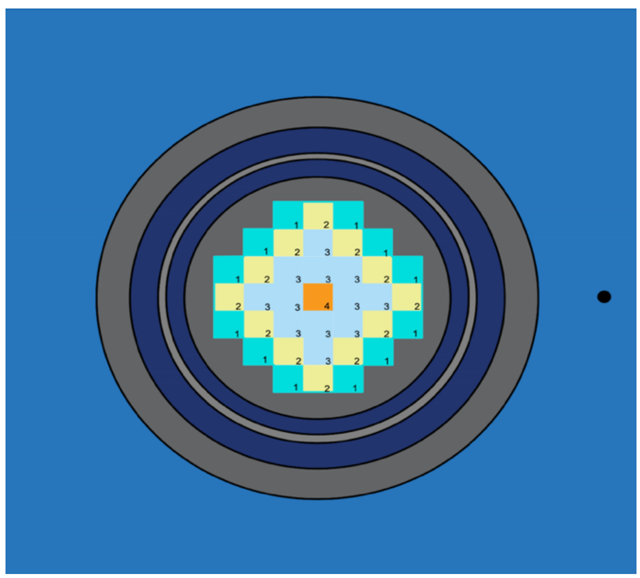


Figure 7. The point at which dose should be calculated is in the reactor pool.

The following scaling factor in units of fission neutrons per unit time should be used to normalize an F4 (the average cell flux tally) tally by the steady-state thermal power of a critical system:

$$S = \frac{P[W]\bar{\nu}\left[\frac{neutron}{fission}\right]}{\left(1.6022.10^{-13}\frac{J}{MeV}\right)w_f\left[\frac{MeV}{fission}\right]} \quad \text{Eq. 8}$$

Where; $\bar{\nu}$ is the average number of neutrons released per fission. The value of $\bar{\nu}$, which reflects the value averaged over fissile isotopes and neutron energies, is given in the MCNP output file showing the final k_{eff} result. w_f is average prompt energy released per fission is ~ 200 MeV [56]. P is the power of the reactor, which is 160 MWth [57].

Eq.9 shows the macroscopic cross-section: $\Sigma(E)= n\sigma(E)$ is expressed units per centimeter travel of the neutron in a medium. Where n is nuclei density, and σ is the microscopic cross-section. The microscopic cross-section is the parameter for a single nucleus interacting with a neutron, but the macroscopic cross-section determines the properties of the entire medium concerning the interaction with a neutron. Since all interactions tend to remove energy from the fast neutron beam; as a result, the Σ_r value is not too different from the total (scattering and capture) macroscopic cross-section Σ_T (Σ_T= NσnT) of the material, but is generally lower [50].

$$\sum(E) = n\sigma(E) \qquad \text{Eq. 9}$$

$$I(x) = I_0 e^{-\Sigma_t x} \qquad \text{Eq. 11}$$

The microscopic cross-section σ(E) measures the probability of occurrence of a particular nuclear reaction. The macroscopic cross-section, Σ(E) for a combination of materials, may be determined using the equation below, which can be represented as a linear sum of cross-sections of different materials.

$$\Sigma(E) = n_1\sigma_1(E) + n_2\sigma_2(E) + n_3\sigma_3(E) \qquad \text{Eq. 12}$$

Eq. 11 represents the shielded intensity calculation, where; I_o is the unshielded intensity or dose rate (obtained from MCNP for the regions of interest), I(x) is the shielded intensity or dose rate with attenuation, and Σ_t is the total cross-section. Σ_r has been used for our calculations.

Table 6 shows the macroscopic removal cross-section, Σ_r, for materials used for shielding. In this research, the removal cross-sections for iron and water have been used to calculate the neutron intensity for the core region and pool.

Table 6. Macroscopic Removal cross sections for the shielding materials.

Material	**Macroscopic Removal Cross Section Σ_r (cm^{-1})**
Water	0.103
Iron	0.1576
Uranium	0.174
Lead	0.118

3.4. Dose Rates for PWR and NuScale

For NuScale, I_o values have been calculated for I_o (dose rate or unshielded intensity [51]) values obtained from MCNP. Eq. 13 shows the ratio of the NuScale and PWR volume to compute I_o for PWR. To calculate this value, the parameters have been given in Table 7.

$$I_o \text{ (PWR reactor)} = I_o \text{ (NuScale reactor)} \times \frac{\text{Volume of PWR core}}{\text{Volume of Nuscale core}} \qquad \text{Eq. 13}$$

Table 7. Some parameters and values for calculation the mass of fuel for NuScale and the result of the mass ratio PWR/NuScale

Fuel density	10.42 g/cm^3
Fuel pellet volume	103.569513 cm^3
Fuel pellet radius	0.406 cm
Total Fuel assembly #	37
2% Gd2O3 assembly #	12
6% Gd2O3 assembly #	12
8% Gd2O3 assembly #	12
Uranium # in FA	12
Total # rod with Gd2O3 in FA	32
Total Fuel Rod Without Guide tube	264
Total Uranium rods in FA	232
Total Uranium mass in FA	11.9761355 tons
The ratio of mass PWR/NuScale	7.347946284

3.5. Dose-Equivalent Rate Calculation

For these reactor cores of varying forms and other features, a variety of estimated neutron shielding formula based on removal cross-sections have been established. For radioactive neutron sources, Eq. 16 is a straightforward and useful formula. Because the intensities are low in comparison to fission sources, relatively thin shields are required. As a result, beyond the shield, scattered neutrons contribute considerably to the dose, and their influence may be represented by a buildup factor B.

$$H = \frac{BSqe^{-\Sigma_r T}}{4\pi R^2} \qquad \text{Eq. 14}$$

Where; Σ_r is the removal cross-section, and q is the dose-equivalent rate per unit neutron fluence rate (Sv/hr per neutron cm^{-2} s^{-1}) for neutrons of the source energy. B is a buildup factor usually assumed to be 5 for neutrons. During shielding calculations, the buildup factor is a correction factor that considers the impact of scattered radiation and any secondary particles in the medium [58]. The factor q and approximate total intensity corresponding to the neutron energy can be obtained from Table 8.

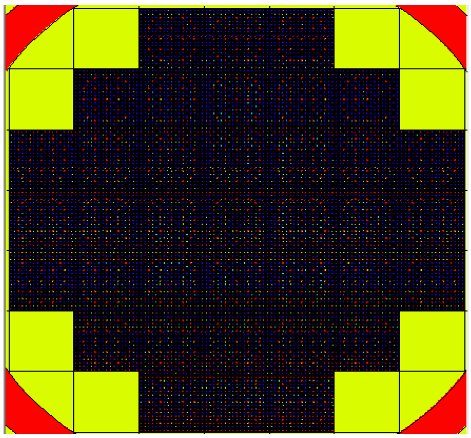
Table 8. Mean Quality Factors Q, for monoenergetic neutrons that give dose-equivalent rate of 1 mSv in 40 hr.

Neutron Energy (MeV)	Q (quality factor)	Intensity from MCNP (n/s)
0.1	2	3.82×10^{-2}
1	11	3.2×10^{-2}
10	6.5	2.13×10^{-3}

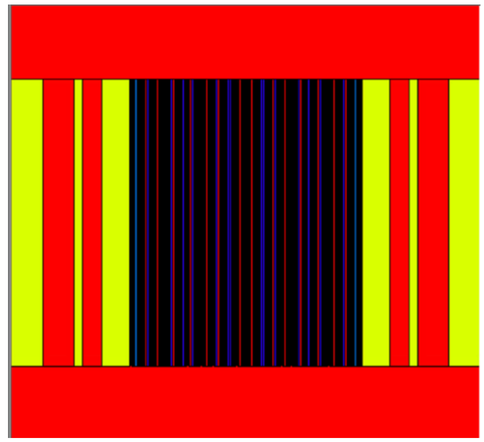
CHAPTER 4. CONCLUSION AND RESULTS

4.1. MCNP Results

4.1.1. Reactor core configurations results from VISED



a) Top view of the reactor core



b) axial views of the reactor core

Figure 8. a) Horizontal view on NuScale MCNP explicit core configuration. b) Vertical view of NuScale MCNP explicit core configuration.

VISED results are shown in Figure 8 and Figure 9. Figure 8 has horizontal and axial views of the reactor core that includes fuel rods inside the fuel assemblies, and also, it is surrounded by water and heavy reflectors, barrel and reactor vessel. The core has 37 fuel assemblies that are shown as the black part of the horizontal view. When the assemblies are zoomed, tubes can be shown clearly, as in Figure 9.

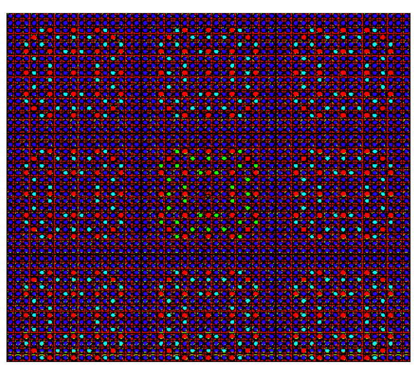
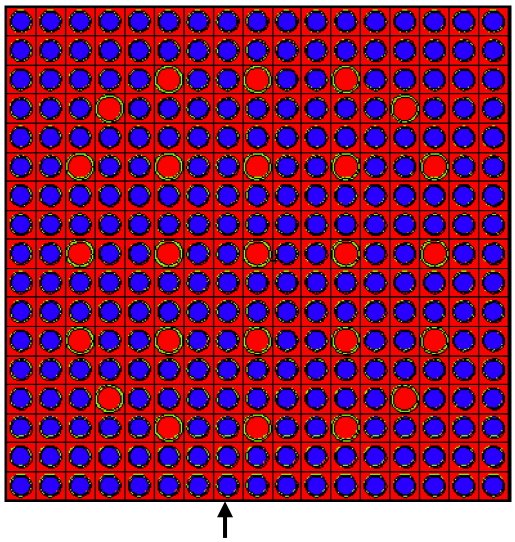
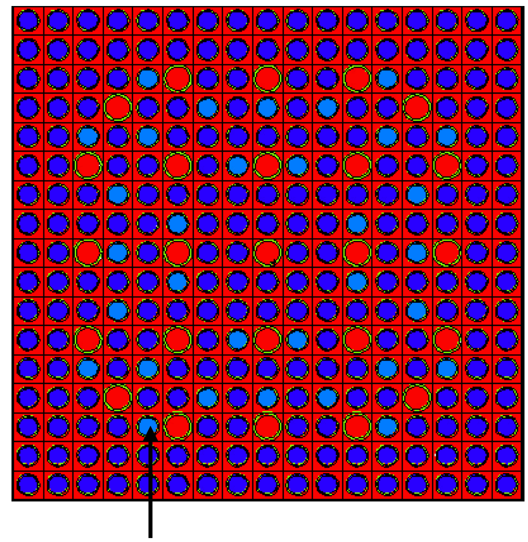


Figure 9. Zoomed in fuel assemblies that are 1 of 8% Gd_2O_3 and 8 of 6% Gd_2O_3 around the center tube.

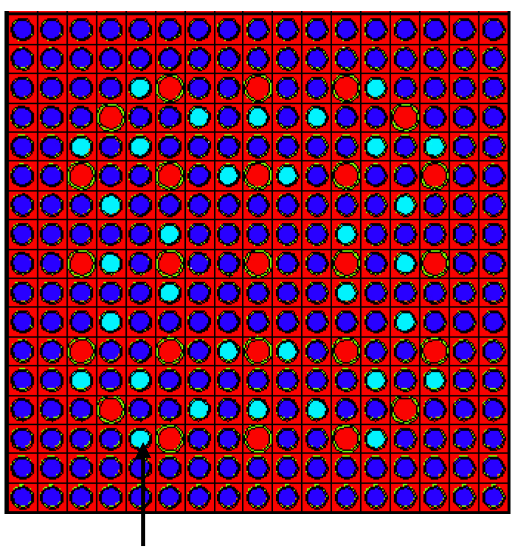
4.1.2. NuScale Fuel Assemblies illustration's results from MCNP



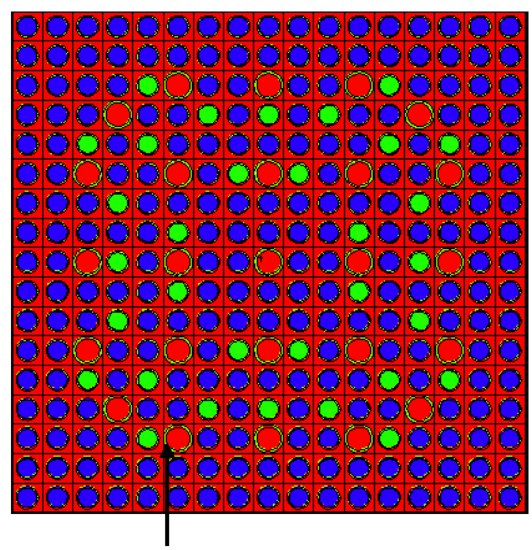
a) Blue points are 4.33 w/o UO_2, red points are guide tubes



b) 2% Gd_2O_3 (light blue points)



b) 6% Gd_2O_3 (turquoise points)



d) 8% Gd_2O_3 (green points)

Figure 10. Fuel assembly's configuration results from VISED

Figure 10 illustrates the fuels assemblies consisting of one uranium fuel rod and three different Gd_2O_3 additives. Red rods are guide tubes, and other rods are all fuel rods. NuScale has 24 guide tubes, and they are filled up with water. Fuel rods have enriched uranium dioxide (UO_2) that are cylindrical ceramic pellets. The fuel cladding material is M5^{TM} that improves corrosion resistance, and is zirconium-based material [54]. Fixed burnable neutron absorber rods were added in selected fuel rod assemblies to reduce the beginning-of-life moderator coefficient. The burnable neutron absorber rod is made of gadolinium oxide Gd_2O_3 mixed with enriched uranium and is mechanically similar to fuel rods [53].

According to the MCNP result, the effective multiplication factor (k_{eff}) is 1.16502 with a standard deviation of 0.00008. It was assumed that the reactor was in critical condition, and then the k_{eff} was taken 1. Also, the average number of neutrons produced per fission is 2.456 from MCNP. This value was used in Eq. 8 as $\bar{\upsilon}$.

Table 9. The MCNP results from the core surface.

Fast neutrons (n/s)	$8.46931 \times 10^{11} \pm 9.8 \times 10^{9}$
Thermal neutrons (n/s)	$4.472 \times 10^{10} \pm 9.84 \times 10^{9}$
Total (n/s)	$8.91647 \times 10^{11} \pm 9.84 \times 10^{9}$

Table 10. The MCNP F5 tally results from the core surface.

Point detector location	**Dose (mSv/s)**
core surface	0.76702677±4804.38
pool	$2.44 \times 10^{-11} \pm 7821.57$

Table 11. The MCNP F4 tally results from the core surface and pool.

Location	**Dose (mSv/s)**
core region	5728150±0.23
pool	37121.31±1.38

Table 9 shows the MCNP result of thermal and fast neutrons per second for the reactor core surface. As a result, the fast neutrons were produced more than the thermal neutrons.

According to Table 10, F5 tally results gave the dose at the core surface and pool. The point detector had a 215 cm distance from the core for the pool region. Since the core surface radius of 92.875 cm, the point detector for this region was put far as radius. Moreover, the result shows that the dose in the pool is lower than in the core surface. This means the dose can be reduced by shielding materials that are around the core. The neutron intensity by region is shown in Table 11.

The produced neutrons were higher in the core region than in the reactor pool region. These regions were not taken as a point. For this reason, the neutrons produced per second were higher than the F5 tally result.

4.2. Analytical Results

According to Eq. 8, the neutron source strength was 1.23×10^{13} neutrons/s. This value has been multiplied by all results from MCNP.

Using Eq. 9, the volume of the PWR reactor/volume of the NuScale reactor has obtained at 7.347946284, and I_o (PWR reactor) was 9.01089×10^{13}. It was expected that the source term for PWR would be greater than the NuScale source due to the larger PWR core volume.

Neutron intensities in the reactor pool at the point of 215 cm distance from the core surface were determined for each option by using Eq. 11 for dose assessments. Table 12 shows the neutron intensity at the pool outside for each reactor. For NuScale, it was 8.64×10^{6} n/s at the point that is in the pool. The intensity was 1.50×10^{8} n/s for PWR at the point that is in the pool. The result shows that the NuScale small modular reactor results in a reduced neutron intensity outside of the reactor than for the PWR.

Dose rates were calculated with Eq.14 as a representative analytical formula. Accordingly, the dose at the NuScale core surface was 7.4548×10^{-5}mSv/s.

Table 12. The result of the neutron intensity for each reactor at a point outside of the reactors

Reactor type	**The neutron intensity at the point outside**
NuScale	8.64×10^{6}
PWR	1.50×10^{8}

4.3. MCNP F4 Tally Intensity results

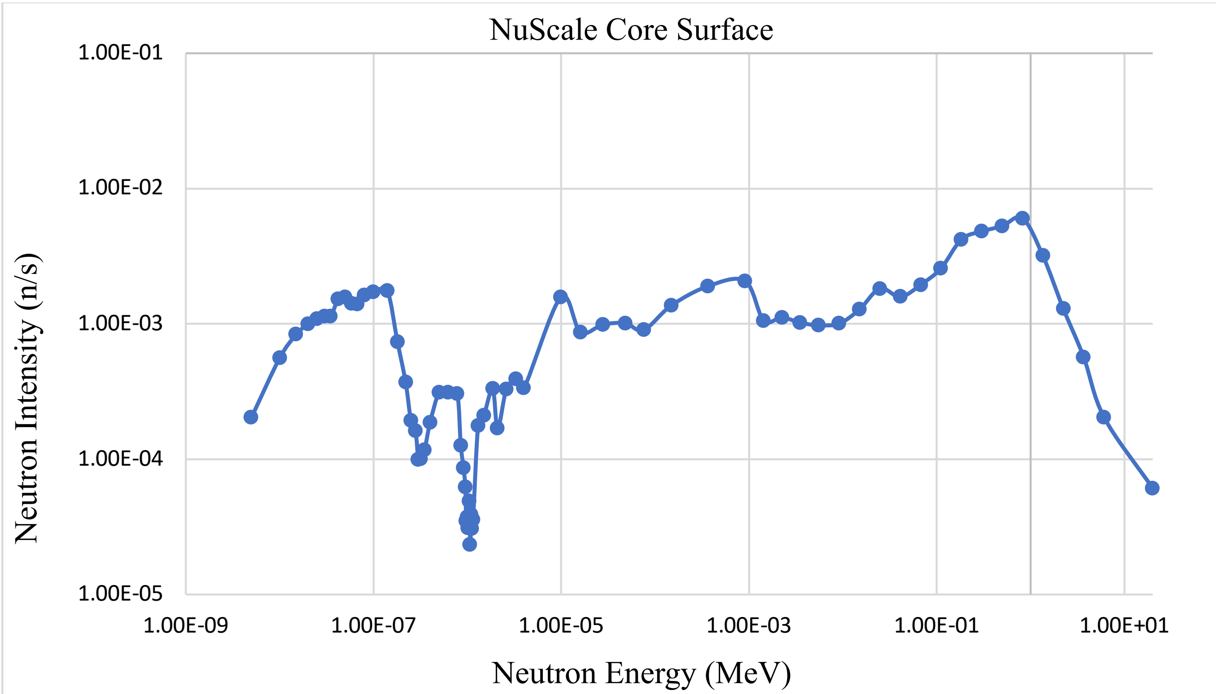


Figure 11. Neutron Intensity versus Neutron Energy for the core surface.

The neutron production on the core surface, core region and on the pool has been given as the neutron intensity versus neutron energy graphs. For Figure 11 and Figure 12, the spectrum taken from the reactor core surface shows the average energy in the reactor is around 1 MeV. The peak is at 1 MeV shows the fast neutron intensity from the fission. This region is fast neutron region. The thermal neutrons were mostly in $1x10^{-7}$-$1x10^{-8}$ MeV range. The thermal neutrons intensity at that region shows that the reactor is thermal. According to the core region graph, the intensity of fast neutron was higher than the surface because the region is bigger than the surface. This intensity decreased in the pool region. The consideration is focused on the neutrons above 0.1 MeV because if the fast neutrons were shielded, the thermal neutrons would be eliminated due to their low energy.

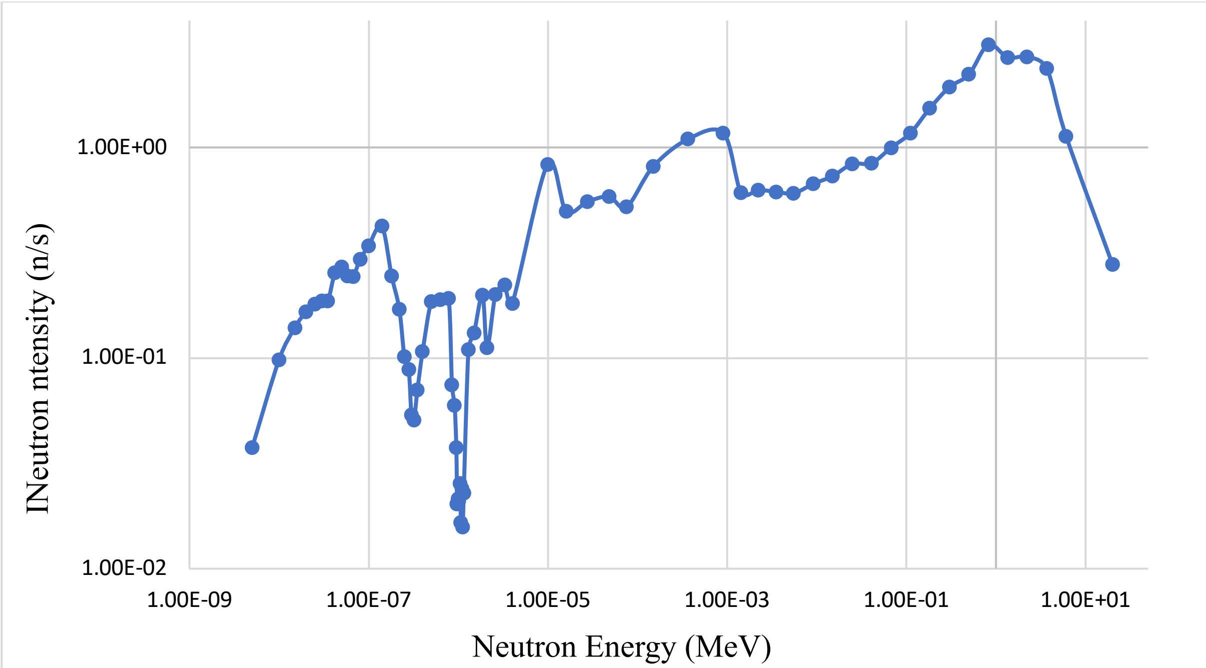


Figure 12. Neutron Intensity versus Neutron Energy for the core region.

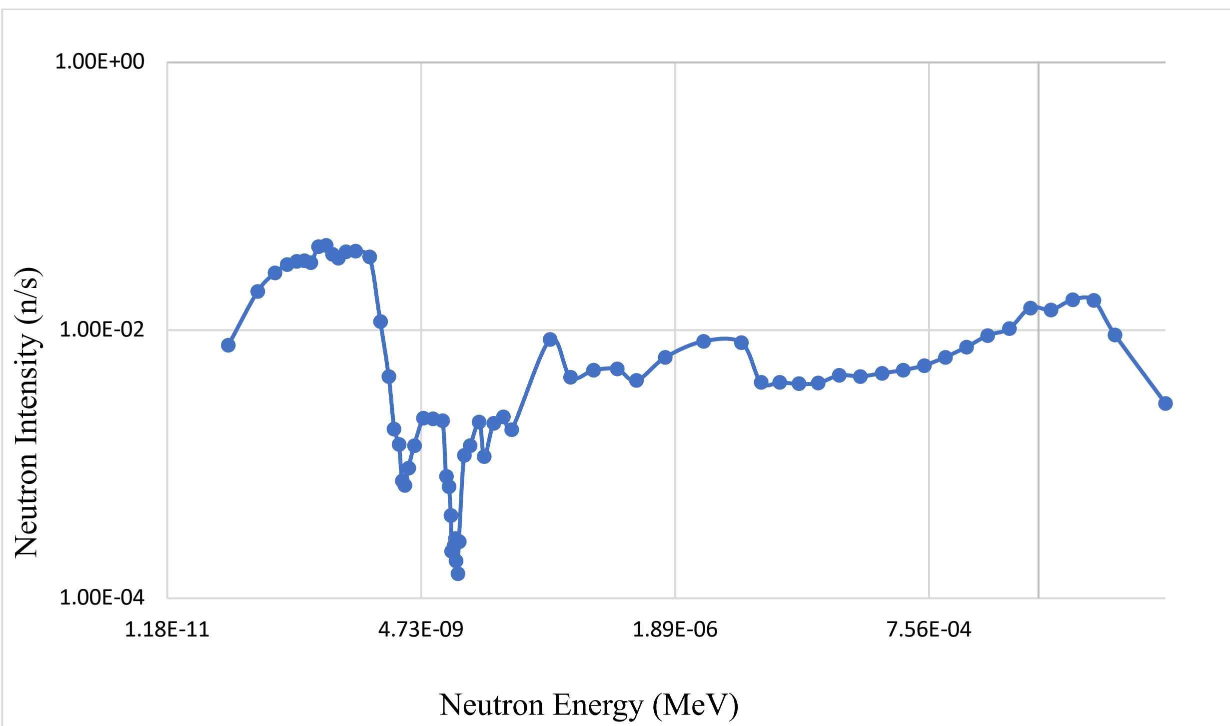


Figure 13. Neutron Intensity versus Neutron Energy in the pool outside the core.

4.4. The Shielding Thicknesses versus Intensity Graphs for NuScale reactor

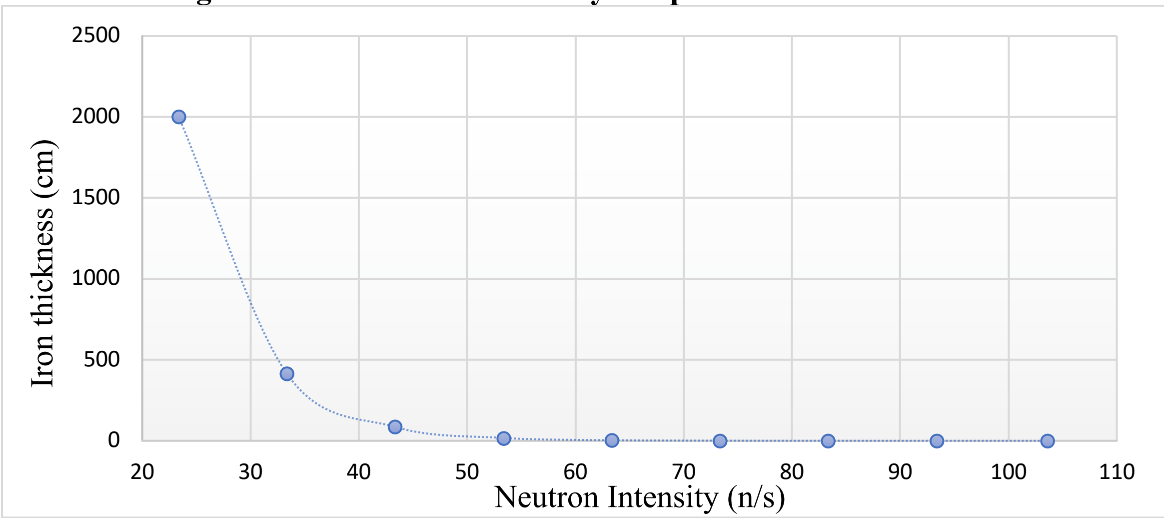


Figure 14. Intensity graph according to iron thickness change

A neutron shield moderates fast neutrons to thermal energy, mainly by elastic scattering, before absorbing them. The light elements, leading hydrogen, are the most efficient in slowing down neutrons. Water and paraffin are also excellent neutron shielding materials. Figure 14 indicates the change in neutron intensity by increasing the iron (stainless steel) material thickness. When the thickness was increased by 10 cm, neutron intensity decreased in the region. The result shows the shielding materials thickness effect on the neutron population at a region of interest.

Table 13. Neutron Intensity values according to increased thickness of iron for NuScale reactor

Reactor vessel (Iron) Thickness (cm)	Neutron Intensity (n/s)
23.36	2003.233658
33.36	414.2700199
43.36	85.6713088
53.36	17.71688223
63.36	3.663862735
73.36	0.75768919
83.36	0.156690616
93.36	0.032403721
103.36	0.006452381

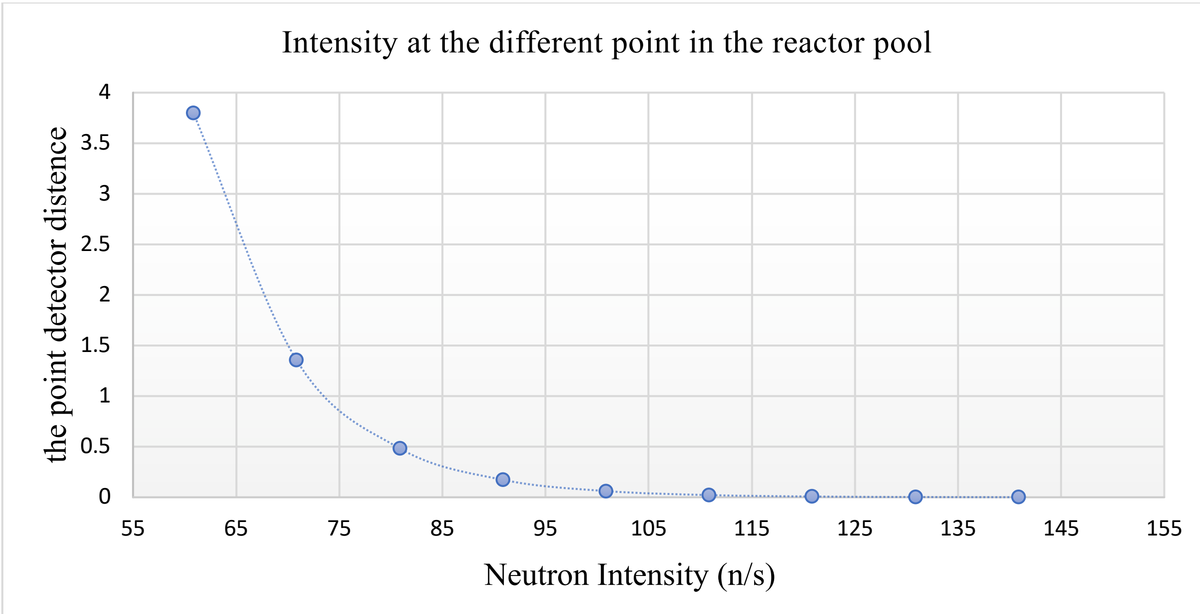


Figure 15. Intensity graph according to the point detector distance change.

Table 14. Intensity values according to increased point detector distance for NuScale reactor

Point detector distance from the reactor (cm)	Neutron Intensity (n/s)
60.85	3.799874283
70.85	1.356581568
80.85	0.484309063
90.85	0.172901706
100.85	0.061727113
110.85	0.022037009
120.85	0.007867366
130.85	0.002808704
140.85	0.001002727

According to Figure 15, when the distance of the point detector was changed, the neutron intensity reduced substantially. In conclusion, it can be seen that water is an efficient material to shield neutrons from the reactor.

CHAPTER 5. CONCLUSION

According to the MCNP simulation results, the shielding provided by the heavy reflector, water reflector, barrel, reactor vessel, and reactor pool are sufficient to protect the public from core radiation. According to US NRC regulations, the radiation exposure limit to the general public is 100 mrem (1mSv)/year.

MCNP results indicate that the maximum absorbed dose calculated in the simulations from the F4 tally result was 1.0557 mSv/s per unit of the core region volume and 1.5964×10^{-4} mSv/s per unit of the pool region volume. However, in the results from F5 tally point detector, the dose is 0.76702677 mSv for the point at the reactor surface and 2.44×10^{-11} mSv for the point at the pool. A dose equivalent of 7.694784×10^{-4} mSv would be received for core radiation dose if a public member was exposed to this on an annual basis. This result is far below the permissible limits, and a NuScale reactor would offer enough neutron radiation shielding based on the simplified MCNP calculations. Based on this analysis, the radiation exposure for a small modular reactor such as the NuScale design is sufficiently low and would be an attractive option for electricity and process heat generation in Turkey.

APPENDIX

MCNP Input File

```
c      Analysis of NuScale reactor
c      Cell Cards
1    1      0.0754              -205 +217 -12        u=1 imp:n=1 imp:p=1 $ 4.33 w/oU235
2    16     2.502E-05           -205 +217 +12 -13  u=1 imp:n=1 imp:p=1 $ gap
3    5      0.042834624.        -205 +217 +13 -14  u=1 imp:n=1 imp:p=1 $ clad
4    13     0.100302291      +14                   u=1 imp:n=1 imp:p=1
5    13     0.100302291      +205             u=1 imp:n=1 imp:p=1
6    13     0.100302291      -217             u=1 imp:n=1 imp:p=1
c     Gd2O3   (4.32 w/o U-235 % with 2% Gd2O3 universe)
7        2     0.0703              -205 +217 -12        u=2 imp:n=1 imp:p=1
8       16     2.502E-05         -205 +217 +12 -13  u=2 imp:n=1 imp:p=1 $ gap
9        5     0.04283462369 -205 +217 +13 -14  u=2 imp:n=1 imp:p=1 $ clad
10     13     0.100302291     +14                     u=2 imp:n=1 imp:p=1
11     13     0.100302291   +205                    u=2 imp:n=1 imp:p=1
12     13     0.100302291   -217                    u=2 imp:n=1 imp:p=1
c   Gd2O3    (4.30 w/o U-235 % with 6% Gd2O3 universe)
18      3      0.0700                 -205 +217 -12        u=5 imp:n=1 imp:p=1
19    16      2.502E-05           -205 +217 +12 -13  u=5 imp:n=1 imp:p=1 $ gap
20       5      0.04283462369   -205 +217 +13 -14  u=5 imp:n=1 imp:p=1
21      13    0.100302291       +14                      u=5 imp:n=1 imp:p=1
22      13    0.100302291       +205                    u=5 imp:n=1 imp:p=1
23      13    0.100302291        -217                u=5 imp:n=1 imp:p=1
c   Gd2O3    (4.29 w/o U-235 % with 8% Gd2O3 universe)
24       4       0.0698               -205   +217   -12     u=7 imp:n=1 imp:p=1
25     16     2.502E-05            -205   +217  +12 -13 u=7 imp:n=1 imp:p=1 $ gap
26       5     0.04283462369    -205   +217  +13 -14 u=7 imp:n=1 imp:p=1 $ clad
27     13     0.100302291        +14                         u=7 imp:n=1 imp:p=1
28     13     0.100302291       +205                        u=7 imp:n=1 imp:p=1
29     13     0.100302291        -217              u=7 imp:n=1 imp:p=1
c   Central Tube Universe  (guide tube)
13    13    0.100302291     -205 +217 -15        u=3 imp:n=1 imp:p=1
14    5      0.04283462369 -205 +217 +15 -16   u=3 imp:n=1 imp:p=1 $ CLAD
15    13    0.100302291     +16            u=3 imp:n=1 imp:p=1
16    13    0.100302291    +205           u=3 imp:n=1 imp:p=1
17    13    0.100302291    -217           u=3 imp:n=1 imp:p=1
c   FA #1 lattice (fuel lattice)
101   13   0.100302291  -9   lat=1 u=9  imp:n=1 imp:p=1
         fill=-9:9   -9:9   0:0   $ square lattice pin
```

```
9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9
9   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   9
9   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   9
9   1   1   1   1   1   3   1   1   3   1   1   3   1   1   1   1   1   9
9   1   1   1   3   1   1   1   1   1   1   1   1   1   3   1   1   1   9
9   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   9
9   1   1   3   1   1   3   1   1   3   1   1   3   1   1   3   1   1   9
9   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   9
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9   1   1   1   3   1   1   1   1   1   1   1   1   1   3   1   1   1   9
9   1   1   1   1   1   3   1   1   3   1   1   3   1   1   1   1   1   9
9   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   9
9   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   9
9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9   9
c    FA #2 lattice (fuel lattice)
201  13  0.100302291 -9 lat=1 u=4  imp:n=1 imp:p=1
      fill=-9:9  -9:9  0:0 $ square lattice pin
4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4
4   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4
4   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4
4   1   1   1   1   2   3   1   1   3   1   1   3   2   1   1   1   1   4
4   1   1   1   3   1   1   2   1   2   1   2   1   1   3   1   1   1   4
4   1   1   2   1   2   1   1   1   1   1   1   1   2   1   2   1   1   4
4   1   1   3   1   1   3   1   2   3   2   1   3   1   1   3   1   1   4
4   1   1   1   2   1   1   1   1   1   1   1   1   1   2   1   1   1   4
4   1   1   1   1   1   2   1   1   1   1   1   2   1   1   1   1   1   4
4   1   1   3   2   1   3   1   1   3   1   1   3   1   2   3   1   1   4
4   1   1   1   1   1   2   1   1   1   1   1   2   1   1   1   1   1   4
4   1   1   1   2   1   1   1   1   1   1   1   1   1   2   1   1   1   4
4   1   1   3   1   1   3   1   2   3   2   1   3   1   1   3   1   1   4
4   1   1   2   1   2   1   1   1   1   1   1   1   2   1   2   1   1   4
4   1   1   1   3   1   1   2   1   2   1   2   1   1   3   1   1   1   4
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4   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4
4   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4
4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4   4
c    FA #3 lattice (fuel lattice)
300  13  0.100302291 -9 lat=1 u=6  imp:n=1 imp:p=1
        fill=-9:9  -9:9  0:0 $ square lattice pin
```

```
6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6
6 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 6
6 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 6
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6 1 1 1 3 1 1 5 1 5 1 5 1 1 3 1 1 1 6
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6 1 1 1 3 1 1 5 1 5 1 5 1 1 3 1 1 1 6
6 1 1 1 1 5 3 1 1 3 1 1 3 5 1 1 1 1 6
6 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 6
6 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 6
6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6 6
c    FA #4 lattice (fuel lattice)
301  13  0.100302291 -9 lat=1 u=8  imp:n=1 imp:p=1
fill=-9:9  -9:9  0:0 $ square lattice pin
8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
8 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 8
8 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 8
8 1 1 1 1 7 3 1 1 3 1 1 3 7 1 1 1 1 8
8 1 1 1 3 1 1 7 1 7 1 7 1 1 3 1 1 1 8
8 1 1 7 1 7 1 1 1 1 1 1 1 7 1 7 1 1 8
8 1 1 3 1 1 3 1 7 3 7 1 3 1 1 3 1 1 8
8 1 1 1 7 1 1 1 1 1 1 1 1 1 7 1 1 1 8
8 1 1 1 1 1 7 1 1 1 1 1 7 1 1 1 1 1 8
8 1 1 3 7 1 3 1 1 3 1 1 3 1 7 3 1 1 8
8 1 1 1 1 1 7 1 1 1 1 1 7 1 1 1 1 1 8
8 1 1 1 7 1 1 1 1 1 1 1 1 1 7 1 1 1 8
8 1 1 3 1 1 3 1 7 3 7 1 3 1 1 3 1 1 8
8 1 1 7 1 7 1 1 1 1 1 1 1 7 1 7 1 1 8
8 1 1 1 3 1 1 7 1 7 1 7 1 1 3 1 1 1 8
8 1 1 1 1 7 3 1 1 3 1 1 3 7 1 1 1 1 8
8 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 8
8 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 8
8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
202  6   0.0869099782 -17 lat=1 u=10 imp:n=1 imp:p=1
fill=-4:4  -4:4  0:0 $ square lattice pin
```

```
10  10  10   10  10 10  10  10  10
10  10  10   9   4   9  10  10  10
10  10   9   4   6   4   9  10  10
10   9   4   6   6   6   4   9  10
10   4   6   6   8   6   6   4  10
10   9   4   6   6   6   4   9  10
10  10   9   4   6   4   9  10  10
10  10  10   9   4   9  10  10  10
10  10  10   10  10 10  10  10  10
400    6    0.0869099782     -100 -205 +217 fill=10   imp:n=1 imp:p=1
401   13   0.100302291       +100 -101 -205 +217      imp:n=1 imp:p=1
402    6    0.0869099782      +101 -102 -205 +217      imp:n=1 imp:p=1
403   13    0.100302291      +102 -103 -205 +217  imp:n=1 imp:p=1
404    6     0.0869099782   +103 -104 -205 +217   imp:n=1 imp:p=1
205   13    0.100302291      -105 (104:205:-217)    imp:n=1 imp:p=1
206    0   +105                                       imp:n=0 imp:p=0

c     Surface Cards
9      rpp   -0.6295   0.6295   -0.6295   0.6295   -100   +100
205   pz    100
217   pz    -100
12     cz     0.406       $ fuel pellet radius
13     cz     0.4142     $ gap
14     cz     0.4751     $ clad
15     cz     0.50419   $ guide inner
16     cz     0.61214   $ $guide outer
17     rpp    -10.7515 10.7515 -10.7515 10.7515 -100 100
100   cz     92.875
101   cz    105.45
102   cz    111.15
103   cz    130.79
104   cz    154.15
105   rpp   -312 312 -298.5 298.5 -1143 1143
106   rpp   210   220   -10   10   -10   10 $ the point distance from core

mode n p
kcode 500000 1 100 250
ksrc 0 0 0
c      Material Card
m1   92235       0.0010      $ 4.33 U-235
     92238       0.0225
     8016        0.0470
m2   92235       0.001       $ Gd2O3 %2, 4.32 U-235
     92238       0.0220
     8016        0.0461
```

```
     64157      0.00049315
m3   92235      9.6151e-04   $ Gd2O3 %6, 4.30 U-235
     92238      0.0211
     8016       0.0442
     64157      0.0015
m4   92235      9.3887e-04   $ Gd2O3 %8, 4.29 U-235
     92238      0.0207
     8016       0.0432
     64157      0.0020
c Clad (Zr 98.87% + Nb 1% + Oxigen 0.13%)
m5   40090.80c   0.02201284
     40091.80c   0.0047470
     40092.80c   0.00717020
     40094.80c   0.00711852
     40096.80c   0.00112294
     41093.80c   0.00041878
     8016.80c    0.00031567
     8016.80c    0.00000011
     8016.80c    0.00000058
c    Steel, Stainless 316 (tot=0.0869099782)
m6   6000.80c    1.64000E-04
     14028.80c   8.02398E-04
     14029.80c   4.07430E-05
     14030.80c   2.68586E-05
     15031.80c   3.60000E-05
     16032.80c   2.18339E-05
     16033.80c   1.74800E-07
     16034.80c   9.86700E-07
     16036.80c   4.60000E-09
     24050.80c   6.84381E-04
     24052.80c   1.31976E-02
     24053.80c   1.49650E-03
     24054.80c   3.72511E-04
     25055.80c   8.89000E-04
     26054.80c   3.37627E-03
     26056.80c   5.29526E-02
     26057.80c   1.22354E-03
     26058.80c   1.61599E-04
     28058.80c   6.70557E-03
     28060.80c   2.58298E-03
     28061.80c   1.12280E-04
     28062.80c   3.57998E-04
     28064.80c   9.11716E-05
     42092.80c   1.86242E-04
     42094.80c   1.16088E-04
     42095.80c   1.99796E-04
```

```
       42096.80c   2.09334E-04
       42097.80c   1.19853E-04
       42098.80c   3.02832E-04
       42100.80c   1.20857E-04
c        Air
m15    7014.80c    0.806059555
        8016.80c    0.193940445
c       Helium
m16    2004.80c      1
c        Moderator
m13    1001.80c     0.665924458
        8016.80c     0.333630217

F34:n  205
SD34 1
      de34 log 1.0E-09 1.0E-08 2.5E-08 1.0E-07 2.0E-07
               5.0E-07 1.0E-06 2.0E-06 5.0E-06 1.0E-05
               2.0E-05 5.0E-05 1.0E-04 2.0E-04 5.0E-04
               0.001 0.002 0.005 0.01 0.02
               0.03 0.05 0.07 0.1 0.15
               0.2 0.3 0.5 0.7 0.9
               1.0 1.2 1.5 2.0 3.0
               4.0 5.0 6.0 7.0 8.0
               9.0 10.0 12.0 14.0 15.0
               16.0 18.0 20.0
     df34 log 1.29 1.56 1.76 2.26 2.54
              2.92 3.15 3.32 3.47 3.52
              3.54 3.55 3.54 3.52 3.47
              3.46 3.48 3.66 4.19 5.61
              7.18 10.4 13.7 18.6 26.6
              34.4 49.4 77.1 102 126
              137 153 174 203 244
              271 290 303 313 321
              327 332 339 344 346 347 350 352
F14:n 400
SD14 1
     de14 log 1.0E-09 1.0E-08 2.5E-08 1.0E-07 2.0E-07
              5.0E-07 1.0E-06 2.0E-06 5.0E-06 1.0E-05
              2.0E-05 5.0E-05 1.0E-04 2.0E-04 5.0E-04
              0.001 0.002 0.005 0.01 0.02
              0.03 0.05 0.07 0.1 0.15
              0.2 0.3 0.5 0.7 0.9
              1.0 1.2 1.5 2.0 3.0
              4.0 5.0 6.0 7.0 8.0
              9.0 10.0 12.0 14.0 15.0
              16.0 18.0 20.0
```

```
     df14 log 1.29 1.56 1.76 2.26 2.54
          2.92 3.15 3.32 3.47 3.52
          3.54 3.55 3.54 3.52 3.47
          3.46 3.48 3.66 4.19 5.61
          7.18 10.4 13.7 18.6 26.6
          34.4 49.4 77.1 102 126
          137 153 174 203 244
          271 290 303 313 321
          327 332 339 344 346 347 350 352
F1:n 100
E1 0 2.5e-8 10
F5:n 92.875 0 0 1 $ point detector r=1
F5:n 92.875 0 0 1 $ point detector r=1
de5 log 1.0E-09 1.0E-08 2.5E-08 1.0E-07 2.0E-07
     5.0E-07 1.0E-06 2.0E-06 5.0E-06 1.0E-05
     2.0E-05 5.0E-05 1.0E-04 2.0E-04 5.0E-04
     0.001 0.002 0.005 0.01 0.02
     0.03 0.05 0.07 0.1 0.15
     0.2 0.3 0.5 0.7 0.9
     1.0 1.2 1.5 2.0 3.0
     4.0 5.0 6.0 7.0 8.0
     9.0 10.0 12.0 14.0 15.0
     16.0 18.0 20.0
df5 log 1.29 1.56 1.76 2.26 2.54
     2.92 3.15 3.32 3.47 3.52
     3.54 3.55 3.54 3.52 3.47
     3.46 3.48 3.66 4.19 5.61
     7.18 10.4 13.7 18.6 26.6
     34.4 49.4 77.1 102 126
     137 153 174 203 244
     271 290 303 313 321
     327 332 339 344 346 347 350 352
F15:n 215 0 0 1
de15 log 1.0E-09 1.0E-08 2.5E-08 1.0E-07 2.0E-07
     5.0E-07 1.0E-06 2.0E-06 5.0E-06 1.0E-05
     2.0E-05 5.0E-05 1.0E-04 2.0E-04 5.0E-04
     0.001 0.002 0.005 0.01 0.02
     0.03 0.05 0.07 0.1 0.15
     0.2 0.3 0.5 0.7 0.9
     1.0 1.2 1.5 2.0 3.0
     4.0 5.0 6.0 7.0 8.0
     9.0 10.0 12.0 14.0 15.0
     16.0 18.0 20.0
df15 log 1.29 1.56 1.76 2.26 2.54
     2.92 3.15 3.32 3.47 3.52
     3.54 3.55 3.54 3.52 3.47
```

```
     3.46 3.48 3.66 4.19 5.61
     7.18 10.4 13.7 18.6 26.6
     34.4 49.4 77.1 102 126
     137 153 174 203 244
     271 290 303 313 321
     327 332 339 344 346 347 350 352
F11:n 100
E11   0   1.00E-11
          5.00E-09
          1.00E-08
          1.50E-08
          2.00E-08
          2.50E-08
          3.00E-08
          3.50E-08
          4.20E-08
          5.00E-08
          5.80E-08
          6.70E-08
          8.00E-08
          1.00E-07
          1.40E-07
          1.80E-07
          2.20E-07
          2.50E-07
          2.80E-07
          3.00E-07
          3.20E-07
          3.50E-07
          4.00E-07
          5.00E-07
          6.25E-07
          7.80E-07
          8.50E-07
          9.10E-07
          9.50E-07
          9.72E-07
          9.96E-07
          1.02E-06
          1.05E-06
          1.07E-06
          1.10E-06
          1.12E-06
          1.15E-06
          1.30E-06
          1.50E-06
```

```
            1.86E-06
            2.10E-06
            2.60E-06
            3.30E-06
            4.00E-06
            9.88E-06
            1.60E-05
            2.77E-05
            4.81E-05
            7.55E-05
            1.49E-04
            3.67E-04
            9.07E-04
            1.43E-03
            2.24E-03
            3.52E-03
            5.50E-03
            9.12E-03
            1.50E-02
            2.48E-02
            4.09E-02
            6.74E-02
            1.11E-01
            1.83E-01
            3.03E-01
            5.00E-01
            8.21E-01
            1.35E+00
            2.23E+00
            3.68E+00
            6.07E+00
            2.00E+01
F44:n 400
SD44 1
E44  0  5.00E-09
            1.00E-08
            1.50E-08
            2.00E-08
            2.50E-08
            3.00E-08
            3.50E-08
            4.20E-08
            5.00E-08
            5.80E-08
            6.70E-08
            8.00E-08
```

1.00E-07
1.40E-07
1.80E-07
2.20E-07
2.50E-07
2.80E-07
3.00E-07
3.20E-07
3.50E-07
4.00E-07
5.00E-07
6.25E-07
7.80E-07
8.50E-07
9.10E-07
9.50E-07
9.72E-07
9.96E-07
1.02E-06
1.05E-06
1.07E-06
1.10E-06
1.12E-06
1.15E-06
1.30E-06
1.50E-06
1.86E-06
2.10E-06
2.60E-06
3.30E-06
4.00E-06
9.88E-06
1.60E-05
2.77E-05
4.81E-05
7.55E-05
1.49E-04
3.67E-04
9.07E-04
1.43E-03
2.24E-03
3.52E-03
5.50E-03
9.12E-03
1.50E-02
2.48E-02

```
            4.09E-02
            6.74E-02
            1.11E-01
            1.83E-01
            3.03E-01
            5.00E-01
            8.21E-01
            1.35E+00
            2.23E+00
            3.68E+00
            6.07E+00
            2.00E+01
F54:n 205
SD54 1
E54   0   5.00E-09
            1.00E-08
            1.50E-08
            2.00E-08
            2.50E-08
            3.00E-08
            3.50E-08
            4.20E-08
            5.00E-08
            5.80E-08
            6.70E-08
            8.00E-08
            1.00E-07
            1.40E-07
            1.80E-07
            2.20E-07
            2.50E-07
            2.80E-07
            3.00E-07
            3.20E-07
            3.50E-07
            4.00E-07
            5.00E-07
            6.25E-07
            7.80E-07
            8.50E-07
            9.10E-07
            9.50E-07
            9.72E-07
            9.96E-07
            1.02E-06
            1.05E-06
```

1.07E-06
1.10E-06
1.12E-06
1.15E-06
1.30E-06
1.50E-06
1.86E-06
2.10E-06
2.60E-06
3.30E-06
4.00E-06
9.88E-06
1.60E-05
2.77E-05
4.81E-05
7.55E-05
1.49E-04
3.67E-04
9.07E-04
1.43E-03
2.24E-03
3.52E-03
5.50E-03
9.12E-03
1.50E-02
2.48E-02
4.09E-02
6.74E-02
1.11E-01
1.83E-01
3.03E-01
5.00E-01
8.21E-01
1.35E+00
2.23E+00
3.68E+00
6.07E+00
2.00E+01

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