Министерство образования и науки Российской Федерации
Федеральное государственное автономное образовательное учреждение
высшего образования
**«НАЦИОНАЛЬНЫЙ ИССЛЕДОВАТЕЛЬСКИЙ
ТОМСКИЙ ПОЛИТЕХНИЧЕСКИЙ УНИВЕРСИТЕТ»**

Институт Физико-технический
Направление подготовки 14.04.02 Ядерные физика и технологии
Кафедра Физико-энергетические установки

МАГИСТЕРСКАЯ ДИССЕРТАЦИЯ

Тема работы
Обеспечение безопасности при хранении и транспортировании ядерного топливо

УДК 621.039.54:621.039.58:621.311.25

Студент

Группа	ФИО	Подпись	Дата
0АМ5И	Джордж Лиджу		

Руководитель

Должность	ФИО	Ученая степень, звание	Подпись	Дата
Доцент каф. ФЭУ ФТИ	Степанов Б.П.	к.т.н.		

КОНСУЛЬТАНТЫ:

По разделу «Финансовый менеджмент, ресурсоэффективность и ресурсосбережение»

Должность	ФИО	Ученая степень, звание	Подпись	Дата
Доцент каф. МЕН ИСГТ	Рахимов Т. Р.	к.э.н.		

По разделу «Социальная ответственность»

Должность	ФИО	Ученая степень, звание	Подпись	Дата
Ст. преподаватель каф. ПФ ФТИ	Веригин Д.А.			

ДОПУСТИТЬ К ЗАЩИТЕ:

Зав. кафедрой	ФИО	Ученая степень, звание	Подпись	Дата
ФЭУ ФТИ	Долматов О.Ю.	к.ф.-м.н., доцент		

Томск – 2017 г.

ПЛАНИРУЕМЫЕ РЕЗУЛЬТАТЫ ОБУЧЕНИЯ ПО ООП

Код результата	Результат обучения (выпускник должен быть готов)
Профессиональные компетенции	
Р1	Применять глубокие математические, естественнонаучные, социально-экономические и профессиональные знания для теоретических и экспериментальных исследований в области использования ядерной науки и техники
Р2	Способность определять, формулировать и решать междисциплинарные инженерные задачи в ядерной области с использованием профессиональных знаний и современных методов исследования
Р3	Уметь планировать и проводить аналитические, имитационные и экспериментальные исследования в сложных и неопределённых условиях с использованием современных технологий, а также критически оценивать полученные результаты
Р4	Использовать основные и специальные подходы, навыки и методы для идентификации, анализа и решения технических проблем в ядерной науке и технике
Р5	Готовность к эксплуатации современного физического оборудования и приборов, к освоению технологических процессов в ходе подготовки производства новых материалов, приборов, установок и систем
Р6	Способность разрабатывать многовариантные схемы для достижения поставленных производственных целей, с эффективным использованием имеющихся технических средств
Общекультурные компетенции	
Р7	Способность использовать творческий подход для разработки новых идей и методов проектирования объектов ядерного комплекса, а также модернизировать и совершенствовать применяемые технологии ядерного производства
Общепрофессиональные компетенции	
Р8	Самостоятельно учиться и непрерывно повышать квалификацию в течение всего периода профессиональной деятельности.
Р9	Активно владеть иностранным языком на уровне, позволяющем работать в иноязычной среде, разрабатывать документацию, презентовать результаты профессиональной деятельности.
Р10	Демонстрировать независимое мышление, эффективно функционировать в командно-ориентированных задачах и обладать высоким уровнем производительности в профессиональной (отраслевой), этической и социальной средах, а также руководить командой, формировать задания, распределять обязанности и нести ответственность за результаты работы

Министерство образования и науки Российской Федерации
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Институт Физико-технический
Направление подготовки 14.04.02 Ядерные физика и технологии
Кафедра Физико-энергетические установки

УТВЕРЖДАЮ:
Зав. кафедрой
_____ __________О. Ю. Долматов

ЗАДАНИЕ
на выполнение выпускной квалификационной работы

В форме:

Магистерской диссертации

Студенту:

Группа	ФИО
0АМ5И	Джордж Лиджу

Тема работы:

Технологические особенности обращения с отработавшим ядерным топливом на атомной станции	
Утверждена приказом директора (дата, номер)	пр. № 959/с от 16.02.17

Срок сдачи студентом выполненной работы:	

ТЕХНИЧЕСКОЕ ЗАДАНИЕ:

Исходные данные к работе	– активность продуктов деления ОЯТ ВВЭР-1000; – конструкционные материалы: алюминий, свинец; – толщина конструкционных материалов: 2, 5, 10 см.
Перечень подлежащих исследованию, проектированию и разработке вопросов	– проведение анализа этапов обращения ядерного топлива при его использовании на АЭС; – определение характеристик ядерного топлива; – расчет и анализ изменения активности продуктов деления в ОЯТ при хранении; – оценка ослабляющей способности конструкционных материалов; – оценка дозовых характеристик;

	– выделение требований для обеспечения ядерной и радиационной безопасности при транспортировании и хранении ядерного топлива на АЭС.
Перечень графического материала	Схема конструкциияячейки реактора ВВЭР-1000 – обязательный чертеж

Консультанты по разделам выпускной квалификационной работы	
Раздел	**Консультант**
Финансовый менеджмент, ресурсоэффективность и ресурсосбережение	РахимовТ. Р.
Социальная ответственность	Веригин Д. А.

Названия разделов, которые должны быть написаны на иностранном языке:
Реферат
Введение
Теоретическая часть
Практическая часть
Результаты
Финансовый менеджмент, ресурсоэффективность и ресурсосбережение
Социальная ответственность
Заключение

Дата выдачи задания на выполнение выпускной квалификационной работы по линейному графику	

Задание выдал руководитель:

Должность	ФИО	Ученая степень, звание	Подпись	Дата
Доцент каф. ФЭУ ФТИ	Степанов Б.П.	к.т.н.		

Задание принял к исполнению студент:

Группа	ФИО	Подпись	Дата
0АМ5И	Джордж Лиджу		

ЗАДАНИЕ ДЛЯ РАЗДЕЛА
«ФИНАНСОВЫЙ МЕНЕДЖМЕНТ, РЕСУРСОЭФФЕКТИВНОСТЬ И РЕСУРСОСБЕРЕЖЕНИЕ»

Студенту:

Группа	**ФИО**
0АМ5И	Джордж Лиджу

Институт	Физико-технический	**Кафедра**	Физико-энергетические установки
Уровень образования	Магистратура	**Направление/специальность**	14.04.02 Ядерные физика и технологии

Исходные данные к разделу «Финансовый менеджмент, ресурсоэффективность и ресурсосбережение»:
– заработная плата научного руководителя, инженера.
Перечень вопросов, подлежащих исследованию, проектированию и разработке:
– проведение SWOT-анализа; – расчет стоимости проведения научного исследования .

Дата выдачи задания для раздела по линейному графику	

Задание выдал консультант:

Должность	**ФИО**	**Ученая степень, звание**	**Подпись**	**Дата**
Доцент каф. МЕН ИСГТ	РахимовТ. Р.	к.э.н.		

Задание принял к исполнению студент:

Группа	**ФИО**	**Подпись**	**Дата**
0АМ5И	Джордж Лиджу		

ЗАДАНИЕ ДЛЯ РАЗДЕЛА
«СОЦИАЛЬНАЯ ОТВЕТСТВЕННОСТЬ»

Студенту:

Группа	ФИО
0АМ5И	Джордж Лиджу

Институт	Физико-технический	**Кафедра**	Физико-энергетические установки
Уровень образования	Магистратура	**Направление/специальность**	14.04.02 Ядерные физика и технологии

Исходные данные к разделу «Социальная ответственность»:	
1. Описание рабочего места (рабочей зоны) на предмет возникновения:	– вредных факторов работы: микроклимат, освещение, электромагнитные поля, ионизирующее излучение, шум, вибрации, конфигурация рабочего места; – опасных факторов работы: вероятность поражения электрическим током, пожаровзрывоопасность.
2. Знакомство и отбор законодательных и нормативных документов по теме:	– электробезопасность, пожарная безопасность, требования охраны труда при работе на ПЭВМ.
Перечень вопросов, подлежащих исследованию, проектированию и разработке:	
1. Анализ выявленных вредных факторов проектируемой производственной среды в следующей последовательности:	– действие фактора на организм человека; – приведение допустимых норм с необходимой размерностью; – предлагаемые средства защиты (коллективные и индивидуальные).
2. Анализ выявленных опасных факторов проектируемой произведённой среды в следующей последовательности:	– электробезопасность (источники, средства защиты); – пожаробезопасность (причины, профилактические мероприятия, первичные средства пожаротушения).

Дата выдачи задания для раздела по линейному графику	

Задание выдал консультант:

Должность	ФИО	Ученая степень, звание	Подпись	Дата
Ст. преподаватель каф. ПФ ФТИ	Веригин Д.А.			

Задание принял к исполнению студент:

Группа	ФИО	Подпись	Дата
0АМ5И	Джордж Лиджу		

Министерство образования и науки Российской Федерации
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Институт Физико-технический
Направление подготовки 14.04.02 Ядерные физика и технологии
Уровень образования Магистратура
Кафедра Физико-энергетические установки
Период выполнения (весенний семестр 2016/2017 учебного года)

Форма представления работы:

Магистерская диссертация

**КАЛЕНДАРНЫЙ РЕЙТИНГ-ПЛАН
выполнения выпускной квалификационной работы**

Срок сдачи студентом выполненной работы:	

Дата Контроля	**Название раздела (модуля) / вид работы (исследования)**	**Максимальный балл раздела (модуля)**
15.05.2017	*Выдача задания*	
19.05.2017	*Проведение анализа этапов обращения ядерного топлива при его использовании на АЭС*	
22.05.2017	*Расчет и анализ изменения активности продуктов деления в ОЯТ при хранении*	
25.05.2017	*Оценка ослабляющей способности конструкционных материалов, дозовых характеристик*	
29.05.2017	*Выделение требований для обеспечения ядерной и радиационной безопасности при транспортировании и хранении ядерного топлива на АЭС*	
01.06.2017	*Анализ полученных результатов*	
07.06.2017	*Сдача работы*	

Составил преподаватель:

Должность	**ФИО**	**Ученая степень, звание**	**Подпись**	**Дата**
Доцент каф. ФЭУ ФТИ	Степанов Б.П.	к.т.н		

СОГЛАСОВАНО:

Зав. кафедрой	**ФИО**	**Ученая степень, звание**	**Подпись**	**Дата**
ФЭУ ФТИ	Долматов О.Ю.	к.ф.-м.н., доцент		

Summary

The master dissertation consists of 95 pages; 17 figures; 14 tables; 29 references.

Key words: Spent fuel, radioactive material, transportation, safety and security and nuclear fuel cycle.

The primary objective of the thesis:

– analysis of the stages of circulation of nuclear fuel when it is used at nuclear power plants;

– determination of requirements for ensuring nuclear and radiation safety during transportation and storage of nuclear fuel at nuclear power plants;

– determination of nuclear fuel characteristics

This dissertation briefly explains about the methodology of storing and transporting nuclear materials in nuclear confined facility.

The first part of the dissertation deals with the details stage of nuclear fuel circulation at a nuclear power stations.

Then radiation measurement of nuclear materials carried out and analyzed systematically. Step by step the safety of transportation is analyzed by IAEA guidelines Investigation done on spent fuel activities

Future plans: Investigations of advanced tracking of nuclear fuels transportation with implement the advanced radiation safety procedures

Abbreviations

CANDU	Canada Deuterium Uranium (reactor) .
LWRs	Light Water Reactors.
PWR	Pressurized Water Reactor.
IAEA	International Atomic Energy Agency
MOX	mixed oxide
CIRUS	Canadian-Indian Reactor Uranium System
AHWR	advanced heavy water reactor
MSR	Molten salt reactor
TWh	Terra watt hours
KWh	Killow watt hours
GHA	Green gas house
RAPP	Rajasthan atomic Power station
AECL	Atomic energy Canada limited

Contents

Introduction

Spent nuclear fuel is generated from the operation of nuclear reactors of all types and needs to be safely managed following its removal from the reactor core. Spent fuel is considered waste in some circumstances or a potential future energy resource in others and, as such, management options may involve direct disposal (as part of what is generally known as the 'once through fuel cycle') or reprocessing (as part of what is generally known as the 'closed fuel cycle'). Either management option will involve a number of steps, which will necessarily include storage of the spent fuel for some period of time. This time period for storage can differ, depending on the management strategy adapted, from a few months to several decades. The time period for storage will be a significant factor in determining the storage arrangements adopted. The final management option may not have been determined at the time of design of the storage facility, leading to some uncertainty in the storage period that will be necessary, a factor that needs to be considered in the adoption of a storage option and the design of the facility. Storage options include wet storage in some form of storage pool or dry storage in a facility or storage casks built for this purpose. Storage casks can be located in a designated area on a site or in a designated storage building. A number of different designs for both wet and dry storage have been developed and used in different States.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences. States have an obligation of diligence and duty of care, and are expected to fulfill their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

For many years, nuclear materials transportation and storage with proper safety measures are being done with irregular and improper safety measure .The dissertation deals with controls of radiation during storing and transportation nuclear materials in nuclear confined facility.

Topicality: This dissertation presents step by step systematic analysis of safety measures while storing and transportation of nuclear materials in and out of nuclear confined area. Activity of nuclear materials is measured periodically at regular intervals.

In this dissertation safety measures of transporting and storage of nuclear material in nuclear confined facilities. And radiation measurements are carried out.

Purpose of this work: analyzing of nuclear material during transportation and storage.

Tasks:

– Stages of nuclear fuel circulation at nuclear power plants.
– Description of the nuclear fuel cycle.
– Analysis of the main types of nuclear fuel used in different types of nuclear reactors.
– Ensuring radiation safety during transportation and storage of nuclear fuel at nuclear power plants.
– Measuring of nuclear fuel characteristics.

This dissertation primarily deals with critically analyzed various means of radiation safety which as to be practiced during transpiration and storage of nuclear materials in nuclear confined areas. Using the advanced tracing system used nuclear materials while storing and transportation of spent nuclear fuel. Powerful gamma and alpha detectors are used to detect the radiation of nuclear fuels while storing and transportation of nuclear materials.

1 Stages of nuclear fuel circulation at nuclear power station.

India was an exception to the general trend. It has almost no domestic uranium resources (only 1% of world uranium resources) and India's capacity to import enriched uranium fuel on the international market has been historically limited, although this situation has changed in recent years. However, India possesses relatively large amounts of thorium and from the beginning of its nuclear program[1]; it pursued the development of a thorium cycle that could supply indigenously sourced fuel for a long-term nuclear program (Bhardwaj, 2013). The Indian program is structured around three stages, reflecting a strategy aimed at the optimal use of available fissile materials and neutrons within the Indian nuclear power program:

– Initial deployment of heavy water moderated and cooled reactors, which use only natural uranium, do not require enrichment technology and can produce plutonium for later use. The opening of the uranium market to India also facilitates the deployment of LWRs, providing another source of plutonium for the next stages of fuel cycle development.

– The gradual start-up of fast neutron reactors using plutonium-based fuel with thorium blankets to generate 233U. The construction of a 500 MWe prototype fast breeder reactor is underway at the KALPAKKAM site and commissioning is expected in 2015, with other fast reactors (FRs) being scheduled for deployment from the 2020s onwards.

– The construction of a new type of advanced heavy water reactor (AHWR) using thorium/ 233U fuel with recycle and having a very high conversion factor (close to one). In the longer term, dedicated molten salt reactors (MSRs) may also be considered for this stage. Experimental program have been established to prepare for the implementation of the above-mentioned concepts. Thorium oxide pellets have been irradiated in research reactors, including fast neutron reactors, and have been processed using a simplified method for recovering the 233U, which has then been used in other research facilities. Thorium fuels have also been introduced in heavy water power reactors[2]. The Kakrapar 1 and 2 reactors were loaded with 500 kg of thorium and operated for 300 and 100 days respectively with this load. The use of thorium fuel is scenario in the Kaiga 1 and 2 reactors as well as in the Rajasthan 3 and 4 reactors, which are now in commercial operation. A 300 MWe prototype AHWR is also under development. It will be fuelled by a mixture of thorium oxide and plutonium oxide and also a mixture of thorium and 233U (Sinah et al., 2000). In this long-term strategy, reactors entirely fuelled by thorium/233U would be deployed only in the third and final stage. Although time frames for this long-term strategy are rarely announced, deployment of the third phase is foreseen beyond 2070 (Vijayan, 2013), but could be much later. In spite of this thorium-based strategy, a recently signed international agreement has given India access to natural uranium supplies, and India has purchased light water-cooled reactors on the global market[3].

The total installed electrical capacity of India (utilities) was just over 300 giga watts (GW) as of May 2016. Of this, 210 GW (70 percent) constituted thermal power such as coal, gas and diesel. India is thus highly reliant on fossil fuels to meet its energy demands. Hydroelectric power too contributes a significant percentage with a total installed capacity of just over 40 GW. The total installed capacity of grid-interactive renewable power which consists of wind, solar, biomass and small hydro is just less than 43 GW. The installed capacity of nuclear power is 5.78 GW, a mere 1.8 percent of the total capacity[4].

In terms of actual power generation, the total electricity generation in India in 2014-15 was 1,278 terawatt hour (TWh), of which nuclear contributed just under three percent. Although India is the fourth largest energy consumer in the world, behind only the US, China and Russia, it continues to remain energy-poor. Its per capita electricity consumption, computed as the ratio of the estimated total electricity consumption during the year to the estimated mid-year population of that year, stood at just over 1,000 kilowatt hours (kWh) in 2014-15.In comparison, developed countries average around 15,000 kWh. China has a per capita consumption of around 4,000 kWh. In 2013, India's population without access to electricity was estimated to be a staggering 237 million, or some 19 percent of the entire population [5].

At the same time, India's total carbon emissions are on the rise, with an estimated 2.3 billion tones in 2014, or an increase of 7.8 percent over 2013 levels. Since 1990, India's GHG emissions have risen by nearly 200 percent. In its NDC, India is committing to reduce the economy's carbon intensity and increase clean energy capacity to 40 percent of the total installed capacity. Nuclear energy—with its massive potential will have to play a key role in the country's future energy mix[6].

1.1Nuclear Power types in India

India's and Asia's first nuclear reactor was the APSARA research reactor. Designed and built in India, with assistance and fuel from the United Kingdom, APSARA reached criticality on August 4, 1956 and was inaugurated on January 20, 1957. A further research nuclear reactor and its first nuclear power scenario were built with assistance from Canada. The 40 MW research reactor agreements was signed in 1956, and CIRUS achieved first criticality in 1960. This reactor was supplied to India on the assurance that it would not be used for military purposes, but without effective safeguards against such use. The agreement for India's first nuclear power scenario at Rajasthan, RAPP-1, was signed in 1963, and followed by RAPP-2 in 1966. These reactors contained rigid safeguards to ensure they would not be used for a military program. The 200 MWe RAPP-1 reactors were based on the CANDU reactor at Douglas Point and began operation in 1972. Due to technical problems the reactor had to be down rated from 200 MW to 100 MW. The technical and design information were given free of charge by AECL to India. The United States and Canada terminated their assistance after the detonation of India's first nuclear explosion in 1974[7].

1.2Description of the nuclear fuel cycle

In the conventional sense, a fuel is a substance which can be combined with oxygen to produce heat, by combustion. By extension, the nuclear fuel is a substance which can produce heat by fission of its heavy nuclei by neutrons. Cars do not burn crude oil straight out of the well, and light bulbs are not plugged directly on hydraulic dams. For fissile nuclei to produce heat by undergoing fission in reactor cores, they must follow "Fuel Cycle," combining many different industrial steps in a sequence One calls the complete chain of industrial operations leading to the supply of a fuel assembly (or element) to a nuclear scenario as the "Front End" of the fuel cycle. Fuel cycles differ from one reactor type to the other and according to the choice of the fissile/fertile nuclei involved. Let us first describe the most usual fuel cycle, the uranium cycle for light water reactors and in India also we have light water reactors. The front end (Fig.1) encompasses the following steps:

- Uranium exploration
- Uranium ore mining
- Ore concentration (milling) as "yellowcake," at the mine mouth
- Shipment of the concentrate
- Conversion to uranium hexafluoride UF_6
- Isotopic enrichment of the uranium
- Fuel fabrication

Between these steps there are, of course, many controls. Each step is, by itself, a complete industrial process which is described in more details hereafter. For reactors using natural uranium and for those recycling plutonium or U^{233}, the enrichment step is bypassed. The fuel fabrication is the most reactor-specific process.

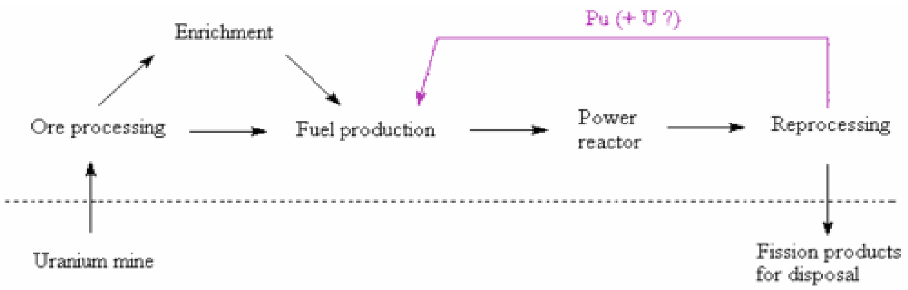

Fig. 1- Front End (Mining and Milling)

The safety of a spent fuel storage facility, and the spent fuel stored within it, is ensured by: appropriate containment of the radionuclide's involved, criticality safety, heat removal, radiation shielding and retrievability. These functions are ensured by the proper site, design, and construction and commissioning of the storage facility, its proper management and safe operation. At the design stage, due consideration also needs to be given to the future decommissioning of the facility [8].

Spent fuel is generated continually by operating nuclear reactors. It is stored in the reactor fuel storage pool for a period of time for cooling and then may be transferred to a designated wet or dry spent fuel storage facility, where it will await reprocessing or disposal (if it is considered to be radioactive waste). The spent fuel storage pools of some reactors have sufficient capacity to accommodate all the spent fuel that will be generated during the lifetime of the reactor.

The basic safety aspects for storage of spent fuel are applicable for the storage of spent fuel from research reactors as well as from power reactors. An approach should be adopted that takes account of the differences between the fuel types (e.g. lower heat generation, higher enrichment and cladding materials that are less corrosion resistant) when considering containment, heat removal, criticality control, radiation shielding and retrievability[9].

Many spent fuel storage facilities at reactors were intended to serve for a limited period of time (a few years) as a place to keep spent fuel between unloading from the reactor and its subsequent reprocessing or disposal. In view of the time being taken to develop disposal facilities and the limited reprocessing program that have been developed, storage periods are being extended from years to decades. This conceptual change in the management of spent fuel has been accompanied by other developments, for example increase in enrichment, increase of burn up, use of advanced fuel design and mixed oxide (MOX) fuel, re-racking, use of burn up credit and, in some cases, extension of storage periods beyond the original design lifetime of the storage facility. Nevertheless, storage cannot be considered the ultimate solution for the management of spent fuel, which requires a defined end point such as reprocessing or disposal in order to ensure safety. The design lifetime of nuclear installations is generally of the order of decades and experience with the storage of spent fuel of up to around fifty years has accrued. While design lifetimes of up to one hundred years have been considered and adopted for certain spent fuel storage facilities, in view of the rate of industrial and institutional change, periods beyond around fifty years are deemed to be ‘long term’ in the context of this Safety Guide. Currently approximately 17% of the world’s electric energy is produced by fission in nuclear power types (NPP). There are more than 430 nuclear reactor units in operation and some 30- 40 are under construction, mainly in Asia. In Sweden ten nuclear reactors are operating and serving the grid as in April 2006, producing about half of the electric energy in the country averaged over the year [10]. The utilization of nuclear power has been a political concern in many countries during the last 20-30 years, as a consequence of accidents/severe events like Three Mile and Chernobyl as well as the long-term storage problems related to the highly radioactive waste produced in the form of spent fuel. In some countries, e.g. Sweden and Germany there are currently scenarios to phase out nuclear power. On the other side there are increased global concerns over the emission of carbon dioxide from fossil fuels (oil, coal and natural gas), possibly with global warming as a consequence, as well as the increased needs for electric energy, especially in developing countries, but also in the industrialized countries as it turns out. These latter issues have driven the construction and scenario of new nuclear power types as mentioned especially in Asia, but recently also in Finland (a 1600 MW pressurized water reactor unit).Recently the debate concerning the future role of nuclear power for the worlds energy needs has also been addressed both by the US and EU. This should also be viewed in light of the concerns over the possible near future shortness of oil, assuming that the production of oil is close to its historical peak and that new, real large oil fields are not discovered anymore the so-called peak oil debate. Furthermore, the geopolitical situation concerning the large scale producers of oil and natural gas complicates the situation. If we consider about india the nuclear fuel cycle is little bit different from others. India as an ambitious nuclear power seeking nation has to a large extent mastered .The components of nuclear fuel cycle with far reaching consequences in both civilian and military sectors. It started as the loudest proponent of non-proliferation of nuclear technology but ended up as a non-signatory of NPT on the grounds of disparity. There were many occasions in the past when it was tried to bring Indian nuclear sector under safeguards. U.S.-India nuclear cooperation agreement was the closet to a credible and sustainable non-proliferation initiative in recent times. On the energy front, India would have to operate its commercial PHWRs at a lower capacity factor without the agreement. This phenomenon would increase the quantity of plutonium produced from the present levels. Further increase in plutonium production by lower capacity factor of PHWRs or a newer facility leads to excessive

Breeders are the route which India sees as the building block for vibrant energy program and also for setting the stage for its third-stage of the nuclear power program. The third-stage reactors are to be fuelled with ^{233}U obtained from the thorium breeding in the fast breeder reactors of second-stage. As evident there is strong opposition to include the breeder program in the U.S.-India civilian nuclear cooperation agreement. The inclusion of FBRs is seen as an effort to cap India's breeder program and eventually diminish it to levels of no significance. Facts suggest that fast breeder reactor design and operation are not new to the world. Fast breeders do not exist in large numbers, purely because of poor economic returns as compared to available nuclear energy alternatives. India's interest in fast breeder program is because of its vast reserves of thorium and availability of ample plutonium from spent fuel .Through this dissertation an alternative to fast breeders has been suggested. A power program with PHWRs followed by thermal breeder reactors constraints proliferation risks, lowers uranium use and establishes thorium breeding cycle. Unlike the FBR the modeled thermal breeder makes reactor-grade plutonium, does not impart huge radiation risk from spent fuel because of lower quantities of ^{228}Th, uses existing core geometry and does not involve the sodium-water heat transfer phase [11].

The suggested thermal breeder design appropriately fits to the nuclear fuel cycle of the nations having CANDU reactor experience and thorium reserves. The EOC fissile content is easy to recover because of negligible production of ^{228}Th. The fuel configuration is proliferation resistant because of lower percentage of ^{233}U and ^{235}U in the fuel and reactor-grade nature of plutonium. The reactor can be brought under safeguards and international monitoring system by involving the international supply of low enriched uranium. Presence of thorium fuel fits the necessities of the fuel cycle that are looking for breeder systems for exploiting the domestic reserves of thorium. By not incorporating any plutonium in the driver fuel, this system undermines the need for reprocessing of thermal reactor spent fuel for extraction of plutonium.

On the long run the TBR cycle stands out to be better positioned with regards to the quantity of uranium used per unit of electricity generated. This cycle can partially be implemented for enhancing the burn up on the EOC fissile content of the CANDU spent fuel and attaining breeder equilibrium state [12]. Through this study a feasible nuclear power program has been suggested to that of the presently pursued three-stage-power program. A viable international nuclear collaboration can be established on the basis of safeguards and verification methods. The closely guarded fuel cycle of India has large scope of improvement by integrating it to the international domain and the objective of military and civilian nuclear sector separation can be more explicitly achieved with proposed thermal breeder nuclear power program.

1.3 Analysis of the main types of nuclear fuel used in different types of nuclear reactors.

Since 1986 growth in nuclear capacity around the world has averaged 1.5% per year. Growth in nuclear electricity generation has been almost twice as fast, at 2.9% per year (IAEA 1986-2006). Much of that increase is due to improved performance and increased capacity factors of existing scenario, which have been aided by, among other things, improvements to the performance and reliability of nuclear fuel.

Nuclear fuel is at the heart of a nuclear reactor, and the safe and economic behavior of this fuel is a key factor in the continuing long term development of nuclear power. This was recognized early, with the key issues determining the economic drivers for fuel development summarized, for example, in 1977 in Oak Ridge National Laboratory's report, "A Survey of Nuclear Fuel Cycle Economics 1970.

It was originally anticipated that there would be a first stage involving pool storage of spent fuel, a second stage where reprocessing would lead to the use of plutonium bearing (MOX) fuels in light water reactors and finally the use of fast breeder reactors. The timescales anticipated are now seen to have been extremely optimistic, and the economics are no longer so clearly driving towards reprocessing and fast reactors. However, the drivers of uranium availability and the costs of enrichment, fuel manufacture and waste management that were identified in 1977 are still valid today, and they have ensured continuous development and improvement in nuclear fuel.

The vast majority of nuclear fuel used today consists of uranium dioxide pellets contained in a sealed tube of zirconium alloy to make a fuel rod. There are many variations in the way the rods are supported in assemblies or bundles for use in the reactor, and improvements in both the fuel rod and assembly structure have been continuous. Table 1 lists typical features of the fuel used in power producing reactors today.

Table 1 – Fuel features

Reactor type	Fuel material	Fuel rod cladding	Typical assembly	Enrichment
AGR	UO2	Stainless steel	Circular array of pins in Graphite sleeves	2-4 %
BWR	UO2	Zircoaloy-2	Square array	Up to 4.95 %
Magnox	U metal	Magnox alloy	-	Natural
RBMK	UO2	E110 , 635	Circular array	Up to 2.8%

Continuation of Table 1

PHWR	UO2	Zircoaloy-4	Circular bundle	Natural
PWR	UO2	Zircoaloy-4	Square array	Up to 4.95 %
WWER	UO2	E110 , E635	Hexagonal array	Up to 4.95 %

The most important determinant of nuclear power's future is cost-competitiveness compared with alternatives. Nuclear power scenarios have a 'front-loaded' cost structure, i.e. they are expensive to build and comparatively cheap to operate. There is, therefore, a strong economic incentive to maximize the utilization of the asset. This means fewer situations outages and, for scenarios with batch reloading, longer operational cycles and shorter outages. The load factor of modern nuclear scenarios with batch reloading is often over 95%, and two year fuel cycles are becoming common. For scenarios with on-load refueling capabilities, maximizing utilization of the asset has meant longer fuel dwell (i.e. increasing the total time a fuel element spends in the reactor)[13]. For all power types there is also a need to minimize waste arising, due to limited on-site storage facilities and the cost of waste removal and treatment.

Because nuclear fuel is the source of the vast majority of the radioactivity produced by a nuclear power scenario, it is imperative that the design and manufacture of the fuel is sufficiently robust not only to allow it to operate normally without incident, but also to withstand any transient or accident that could occur in the scenario in a manner that can ensure that safety is not compromised. This is ensured through a licensing process that oversees, not only operation, but also that the design and manufacture of nuclear fuel is carried out to extremely high standards, and that design needs are codified and performance is demonstrated experimentally.

In the 1970s there was a large program of experimental work to demonstrate how fuel would behave under transient and accident conditions, and safety criteria were defined that provided limits on operation such that fuel would not allow an unacceptable release of radioactivity. The intent was that for normal operation and frequent transient conditions, the fuel cladding would not fail, and that under severe accident conditions any fuel failure would be able to be contained and controlled by the scenario safety systems. Examples were the testing of fuel under transient high power conditions (power ramps) or under severe loss of coolant accident (LOCA) conditions. As burn up has increased it has been necessary to demonstrate that changes in design or materials do not challenge the limits set by these safety criteria.

The need to demonstrate compliance with the safety needs means that improvements to fuel design and operation are carefully considered and implemented incrementally, with experimental demonstration, typically through the use of 'lead test assemblies', following extensive testing and research. The incremental approach to burn up extension has been a feature of nuclear fuel development as limitations on burn up extension have been identified and overcome.

1.4 Regulation of Radioactive Materials Transportation

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled. Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety[14].

DOT regulates the transportation of radioactive materials as part of hazardous materials transport regulations, primarily under Title 49, "Transportation," to CFR Part 173, "Shippers— General Needs for Shipments and Packaging," dated October 1, 2011. Mode specific regulations are given in Parts 174 to 177 and specifications for packaging's are given in

Part 178. In addition, 49 CFR 174.471 allows the use of packaging's certified by the NRC under 10 CFR Part 71. The regulations of 10 CFR Part 20, "Standards for Protections against Radiation," also are relevant. NRC transportation regulations primarily apply to the transportation of packages. DOT regulations include labeling, occupational and vehicle standards, registration needs, reporting needs, and packaging regulations. Generally, DOT packaging regulations apply to industrial and Type A packaging whereas the NRC regulations apply to Type A fissile materials packaging and Type B packaging. Industrial and Type A nonfissile packages are designed to resist the stresses of routine transportation and are not certified to maintain their integrity in accidents, although many do. Type B packages are used to transport very hazardous quantities of radioactive materials. They are designed to maintain their integrity in severe accidents because the NRC recognizes that any transport package and vehicle may be in traffic accidents. This study addresses SNF transportation; therefore, it is only concerned with SNF for Type B packaging. (For the remainder of this report, the term "cask" will be used to refer to the contents plus the packaging.)

Nuclear fuel that has undergone fission in a reactor is extremely hot and radioactive when it is removed from the reactor. To cool the fuel thermally and allow the highly radioactive and short-lived fission products in the fuel to decay, the fuel is discharged from the reactor into a large pool of water. The fuel usually remains in the pool as long as there is space for it. After the fuel has cooled sufficiently, it can be moved to dry surface storage at the reactor or transported to a storage site or other destination. Currently, very little transportation of used commercial power reactor fuel takes place in the United States and there are no scenarios to transport SNF before it has cooled for 5 years. The transportation casks are rated for heat load, which often determines the cooling time needed for the fuel to be transported. Shielding or other considerations may also drive the required cooling time

Type B packages are designed to pass the sequential series of tests for "Hypothetical Accident Conditions." These tests are summarized below.

(1) A 9-meter (30-foot) drop onto an essentially unyielding horizontal surface. “Essentially unyielding” in this context means the target is hard and heavy enough that the package absorbs nearly all of the impact energy and the target absorbs very little energy. This test condition is more severe than most transportation accidents.

(2) A 1-meter (40-inch)[6]drop onto a fixed 15-centimeter (cm) (6-inch) diameter steel cylinder to test the package’s resistance to punctures.

(3) An 800 degrees Celsius (C) (1,475 degrees Fahrenheit (F)) fire that fully engulfs the package for 30 minutes.

(4) Immersion under 0.9 meters (3 feet) of water. In addition, a non sequential immersion in 15 meters (50 feet) of water for 1 hour.

In addition to the immersion test of 10 CFR 71.73, an undamaged cask carrying spent fuel is also required by 10 CFR 71.61, “Special Needs for Type B Packages Containing More Than 10^5A_2,” to withstand an external pressure of 2 million Pascal’s (MPa) (290 pounds per square inch (psi)) for a period of not less than 1 hour without collapse, buckling, or in leakage of water. This pressure is equivalent to an immersion in 200 meters (660 feet) of water(Figure-2).

The package tests in 10 CFR 71.73 were developed to envelope real-life accidents. These tests are not intended to represent any specific transportation route, any specific historical transportation accident, or a “worst-case” accident. These tests are intended to simulate the damaging effects of a severe transportation accident in a manner that provides international acceptability, uniformity, and repeatability. All International Atomic Energy Agency Member States use these tests.

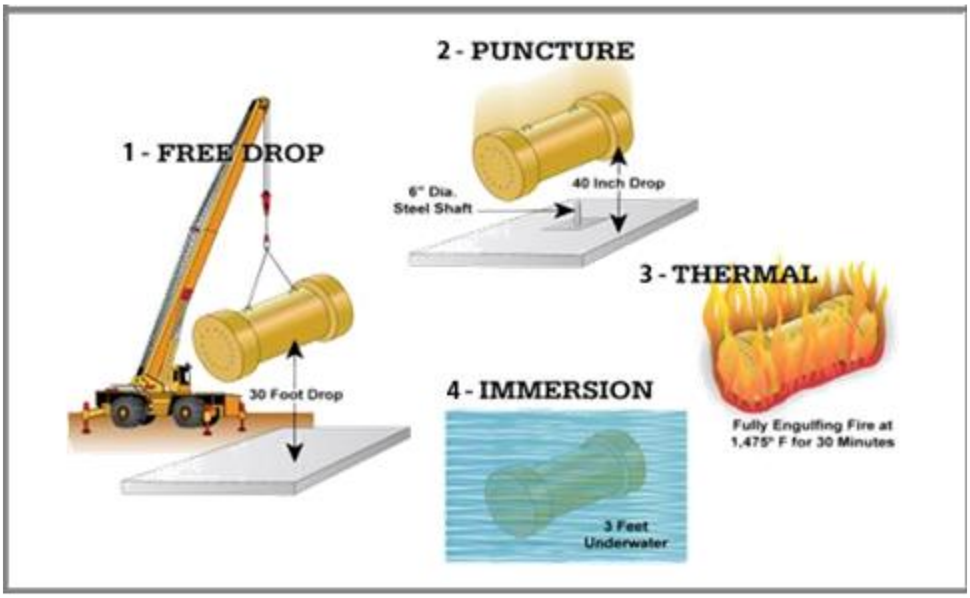

Figure-2 The four tests for Type B packages

The tests are performed on a package design (either physically using a full-scale prototype or sub-scale test unit, or via computational modeling), but not on every package that will be used to transport SNF. A package designer may create computer models to evaluate the performance of a package design or components of the package design, build full-size or scale model packages for physical testing, or incorporate references to previous satisfactory demonstrations of a similar nature. In practice, the safety analysis performed for Type B packages often incorporates a combination of physical testing, computer modeling, and engineering evaluation. The SAR packaging contains information on the package design's performance in the tests and an evaluation against the acceptance criteria in 10 CFR Part 71. The SAR is used to apply for package certification. During the certification process, the NRC reviews the SAR to ensure that the package design meets all criteria specified in10 CFR Part 71.NRC regulations specify that release of material from the package can be no more than the amount allowed to be shipped in a nonaccident resistant Type A package. The regulation also specifies a maximum post-test external radiation dose rate of 0.01 Sv per hour (1 rem/hr) at1 meter (40 inches) from the package surface.

"Standards for Protection against Radiation," prescribe the largest allowable radiation dose that a member of the public may receive from NRC-licensed facilities, exclusive of background radiation, diagnostic or therapeutic radiation, or material discharged to the environment in accordance with NRC regulations. This section of the code does not apply to transport, but provides doses that can be compared to those calculated in this study. These doses are listed below.

– 1 mSv per year (100 mrem per year) total effective dose equivalent (TEDE), including both external and committed internal dose.

– 0.02 mSv per hour (2 mrem per hour) in any unrestricted area from external sources. As shown in Table 2-12, for example, doses from routine, incident-free transportation are considerably below these limits.

– 5 mSv per year (500 mrem per year) from a licensed facility if the licensee can show the need and expected duration of doses larger than 1 mSv (100 mrem) per year.

Although the regulations state clearly that these dose limits do not include background, it can provide a useful comparison to other sources of radiation exposure since it affects everyone. The average background radiation dose in the United States is 0.0036 Sv (360 mrem) per year. 10 CFR Part 20 also regulates occupational doses to 0.05 Sv per year (5 rem per year) TEDE.

2 Ensuring radiation safety during transportation and storage of nuclear fuel at nuclear power scenarios

2.1 General safety considerations for storage of spent nuclear fuel

Spent fuel storage facilities should provide for the safe, stable and secure storage of spent fuel before it is reprocessed or disposed of. The design features and the operation of the facility should be such as to provide containment of radioactive material to sure that radiation protection of workers, members of the public and the nature is optimized within the dose constraints in accordance with the needs established into maintain sub criticality, to ensure removing of decay heat and to ensure retrievability of the spent fuel. These safety functions should be maintained during all operational states and accident conditions.

Various types of wet and dry spent fuel storage facility are currently in operation or under consideration in various States. Spent fuel is keep it in essentially one of three different modes:

I. Wet storage in pools at, or remote from, a reactor place: The spent fuel isstored in standard storage racks or in compact storage racks in which closer spacing of the fuel assemblies or fuel elements is allowed in order to enlarge the storage space.

II. Dry storage in either storage or two purpose (i.e. storage and transport) casks at, or remote from, a reactor place: Casks are modular in nature. Such systems are sealed systems designed to prevent the release of radioactive material during storage. They provide protecting and containment of the spent fuel by physical barriers, which may include a metal or concrete body and metal liner or metal canister and lids. They are usually cylindrical in shape, circular in cross-section, with the long axis arranged either vertically or horizontally [15]. The fuel position is maintained by means of a storage basket which may or may not be an integral part of the cask. Temperature is removed from the stored fuel by conduction, radiation and forced or environmental convection to the surrounding environment. Casks may be enclosed in buildings or stored in an open area.

III. Dry storage in vault type storage facilities: A vault is a huge, radiation shielded facility in which spent fuel is stored. A vault can be either above or below ground level; it may be a reinforced concrete form containing an array of storage cavities. The spent fuel is appropriately contained in order to prevent unacceptable releases of radioactive material. Shielding is provided by the structure surrounding the stored material. Primary heat removal is by forced or natural air convection over the exterior of the storage cavities. This temperature is released to the atmosphere either directly or via appropriate filtration, depending on the system designs. Some systems also use a secondary cooling circuit. However, if natural convection is to be used, the need for active components, e.g. pumps and ventilators, must minimized through higher operational reliability of the system and corresponding cost reduction.

Although designs of spent fuel storage facilities may differ, in general they should consist of relatively simple, preferably passive, inherently safe systems intended to provide adequate safety over the design lifetime of the facility, which may span several decades. The lifetime of a spent fuel storage facility should be appropriate for the envisaged storage period. The design should also contain features to ensure that associated handling and storage operations are relatively straightforward.

Responsibilities of the operating organization

The operating organization is responsible for the safety of all activities associated with the storage of spent for the specification and implementation of the programs and procedures necessary to ensure safety. The operating organization should maintain a high level of safety culture and demonstrate safety [16]. In some instances, the operating organization may be the owner of the spent fuel and in other cases the owner may be a separate organization. In the latter instance, consideration should be given to interdependences, including any activity carried out prior to receipt of the spent fuel at a storage facility, such as its characterization or packaging, or subsequent transport of the spent fuel from the facility, to ensure that conditions for safety will be met.

The responsibilities of the operating organization of a spent fuel storage facility typically include:

(a) Application to the regulatory body for permission to site, design, construct, commission, operate, modify or decommission a spent fuel storage facility;

(b) Conduct of appropriate safety and environmental assessments in support of the application for a license;

(c) Operation of the spent fuel storage facility in accordance with the needs of the safety case, the license conditions and the applicable regulations;

(d) Development and application of acceptance criteria for the storage of spent fuel, as approved by the regulatory body;

Prior to authorization of a spent fuel storage facility, the operating organization should provide the regulatory body with a safety case that demonstrates the safety of the proposed activities and also demonstrates that the proposed activities will be in compliance with the safety needs and criteria set out in national laws and regulations. The operating organization should use. The operating organization may wish to set an operational target level below these specified limits to assist in avoiding any breach of approved limits and conditions. At an early stage in the lifetime of a spent fuel storage facility, the operating organization should prepare preliminary scenarios for its eventual decommissioning. For new facilities, features that will facilitate decommissioning should be taken into consideration at the design stage. Such features should be included in the decommissioning scenario, together with information on arrangements regarding how the availability of the necessary human and financial resources and information will be ensured, for presentation in the safety case.

The operating organization must prepare scenarios and implement programs for personnel monitoring, area monitoring, and environmental monitoring and for emergency preparedness and response [18]. The operating organization should establish a process on how to authorize and make modifications to the spent fuel storage facility, storage conditions, or the spent fuel to be stored, which is commensurate with the significance of the modifications. As part of the work, the potential consequences of such modifications should be evaluated, including consequences for the safety of other facilities and also for the retrieval, reprocessing or disposal of spent fuel.

The operating organization is required to put in place appropriate mechanisms for ensuring that sufficient financial sources are available to undertake all necessary tasks throughout the lifetime of the facility, including its decommissioning. The operating organization should develop and maintain a records system on spent fuel data and on the storage system, which must include the radioactive inventory, location and characteristics of the spent fuel, information on ownership and origin and information about its characterization [17]. An unequivocal identification system should be established, with markings that will last for the duration of the storage period.

Accounting for and control of nuclear materialand physical protection systems.

The operating organization will be required to establish, maintain and implement a system for nuclear material accounting and control as an integrated part of the State system of accounting for and control (SSAC) of nuclear material. In addition, physical protection systems for deterrence and detection of the intrusion of unauthorized persons and for protection against sabotage from within and outside the facility will be designed and installed during the making and operation of the spent fuel storage facility. The implications of such systems and arrangements for the safety of the facility should be assessed and it should be ensured that no safety function would be compromised nor would the overall level of safety at the facility be significantly reduced on account of such systems and arrangements.

Spent fuel management.

National and international policies and principles for spent fuel management that currently constitute an accepted management arrangement can evolve over the lifetime of the facility. Policy decisions (e.g. regarding spent fuel reprocessing) and technological innovations and advances (e.g. in partitioning and transmutation) can lead to fundamental changes in the overall spent fuel management strategy. However, the operating organization retains its responsibility for all activities at all times and continuous commitment by the organization remains a prerequisite to ensuring safety and the protection of human health and the environment.

Resource management

Spent fuel management activities will require financial and human resources and the necessary infrastructure at the site where the spent fuel storage facility is located. Senior management should be responsible for making arrangements to provide adequate resources for spent fuel management activities, to satisfy the demands imposed by the safety, health, environmental, security, quality and economic aspects of the full range of activities involved in the management of spent fuel and the potentially long duration of such activities. Arrangements for funding of future spent fuel management activities should be specified and responsibilities, mechanisms and schedules for providing the funds should be established in due time. The generator of the spent fuel should establish an appropriate funding mechanism.

In the design of facilities for long term spent fuel management, consideration should be given to the incorporation of measures that will facilitate operation, maintenance of equipment and eventual decommissioning of the facility. For long term spent fuel management activities, future infrastructural needs should be specified and scenarios should be made to ensure that these will be met. In such scenario, consideration should be given to the continuing need for support services, spare parts for equipment that may eventually no longer be manufactured and equipment upgrades to meet new regulations and operational improvements, and to the evolution and inevitable obsolescence of software. Consideration should also be given to the need to develop monitoring programs and inspection techniques for use during extended periods of storage [18].

2.2Radiation safety during transportation

Safety security levels for radioactive material in transport.

In order to specify the transport security levels in a manner that is easily understood and integrated into already existing safety and security systems, it was essential to evaluate existing approaches must applied to radioactive material (including nuclear material) and sources.

The Code of Conduct and sort of Radioactive Sources Since IAEA publications are being widely implemented to improve the safety and security of sources, the D-values that were developed to define a dangerous source are suitable for specifying the threshold activity for location changing security levels.

Transport Regulations. These regulations use activity values A_1 and A_2 to specify the amount of radiation level, above which the material must be transported in an accident resistant material. Since the A-values are well understood and used in the transport safety system, with appropriate numerical multipliers they are also suitable for specifying the activity thresholds.

The categorization of sealed sources contained in the Code of Conduct is based on the development of D-values for the Needs in IAEA Safety Standards Series No. GS-R-2.Which specifies needed for emergencies involving a dangerous source. These Safety Needs define a dangerous source as one "that could, if not under control, give rise to exposure sufficient to cause severe deterministic effects". The Safety Needs then go on to define a severe deterministic effect as one that "is fatal or life threatening or results in a permanent injury that decreases the quality of life".

To apply the Safety Needs, an operational definition of a dangerous source was needed. This operational definition of a dangerous source is known as the D-value. The D-value is that quantity of radioactive material, which, if uncontrolled, could result in the death of an exposed individual or a permanent injury that decreases that person's quality of life [19].

For potential exposures, a dose level of 50 mSv has been used on the grounds that, historically, actual accidents involving Type A packages have led to very small exposures. In choosing this reference dose, it is also important to take into account the probability of an individual being exposed as the result of a transport accident, since such exposures may, in general, be considered as ‘once in a lifetime’ exposures.

A scoping model was used to calculate the amount of radioactive material needed to cause resettlement of persons from an area contaminated by an RDD. ICRP 82, Protection of the Public in Situations of Prolonged Radiation impact and an IAEA Safety Guide on emergency response provide recommendations on action levels of dose in respect of actions to be taken following radiological incidents. Details of the scoping model, and its assumptions and parameters, are provided in the Appendix B.

The reverberation of the scoping model were compared with both the A-values and the D-values. This comparison sought to identify multipliers of those values that would approach but not exceed the model results. Given the uncertainties and conservative approaches inherent in the sample, it was unnecessary that a rigorous correlation be found, but only a reasonable one. It was found that a correlation could be made with either set of values. The Appendix B provides the basis for the activity thresholds [20].

As a result, the following activity threshold values are used for the enhanced security level:

– For radioactive sources and other forms of radioactive material containing radio nuclides covered by the Code of Conduct, 10 D per package; or

– For all other radio nuclides, 3000 A_2 per package.

Some radioactive material poses a sufficiently low risk of radiological hazard that it does not present a security concern. Such material includes very small quantities (excepted packages with an activity level not exceeding the level permitted for the radionuclide when it is not in special form), material of low activity concentration and low level contaminated objects that can be transported. No specific security measures for these materials beyond the basic control measures stated in the normal Safety Standards and employed in normal commercial practices are given.

Radioactive material between these two threshold limits should be protected at the normal security level.

2.3 Basic security level

Common security provisions

The competent authority must, at its discretion, provide information to operators regarding the potential change in the threat to radioactive material in transport. Operators should take all threat information into consideration when implementing security measures. For international transport, the threat data's for each State involved in such transport should be considered.

All operators and other persons occupied in the transport of radioactive material must apply security measures for the movement of radioactive material commensurate with their responsibilities and the level of warning.

Radioactive material should be moved only to permit operators. In normal circumstances, it is sufficient that there is an existing business relationship between a carrier and consignee. Where such a relationship does not already exist, a potential carrier's or consignee's apt or capability to receive or transport radioactive material must established by confirmation with applicable national regulatory authorities, or trade and industry associations, that the carrier's or consignee's interests are legitimate [21].

When radioactive material is temporarily stored in transit sites (such as warehouses, marshalling yards, etc.), appropriate security measures should be applied to the radioactive material consistent with the measures applied during use and storage.

In the event that packages need to be transported on open conveyances, it may be necessary for the State to consider in view of the nature of the radioactive material or prevailing threat whether extra security measures should be applied. Such measures may include providing guards, shielding the package to provide for external pre-detonation to prevent or mitigate complaint to the package in the event of a stand-off attack using rocket propelled armor piercing weapons or same as machines that are not easily defended against, and enhancing route surveillance or response capability. Packages should be protected on the basis of advice from safety specialists.

Fundamental security awareness training

Individuals engaged in the transport of radioactive material should receive instruction, including training in the elements of security awareness.

Security awareness teaching should address the nature of security related warnings, with due recognition of security concerns, methods to address such concerns and actions to be undertaken in the event of a security incident. It should include awareness of security ideas commensurate with the responsibilities of individuals and their part in introducing security ideas.

Such training should be provided or verified upon employment in a position involving the transport of radioactive material and should be periodically supplemented by retraining as deemed appropriate by the competent authority.

Records of all security training undertaken should be kept by the employer and should be made available to the employee if requested.

Personnel uniqueness verification

Each group member of any conveyance transporting radioactive material should carry means of positive identification during transport (an officially issued photographic identification or biometric record that uniquely identifies the individual). While biometric forms of identification are preferable, some States may not have the capability to confirm biometric details. Therefore, for international transport, a photo identification issued by an officially approved company may be the most appropriate method of identification.

Security confirmation of conveyances

Carriers should perform security identification of conveyances and should ensure that these security measures remain effective during transport. In normal circumstances, and as appropriate to the mode of transport, it will be sufficient for the carrier of the conveyance to carry out aneye inspection to ensure that nothing has been tampered with or that nothing has been affixed to the package or conveyance that might compromise the security of the consignment. Such an inspection will require no more than the carrier's own knowledge of the conveyance.

Printed instructions

Operators must provide appropriate crew members with written instructions on any required security measures, including how to respond to a security incident during transport. At the fundamental security level, it is generally sufficient for these written instructions to contain no more than fundamental details of emergency contacts.

Exchange of security information

Operators should cooperate with each other and with the appropriate authorities to exchange information on applying security measures and responding to security incidents, where the exchange of information does not conflict with necessary for security respect of sensitive information.

Trustworthiness determination

Persons engaged in the transport of radioactive material may be subject to trustworthiness determination by the operator commensurate with their responsibilities. The trustworthiness determination is a determination of the reliability of an individual, including characteristics and details that should be verified, where legally permitted and where necessary, by means of background checks and by checking criminal records. The trustworthiness determination [14] should be based on background checks of previous activities to verify the character and reputation of the individual.

2.4 Improved security level

For packages of radioactive material with contents meeting or exceeding the radioactivity threshold for the enhanced security level as specified in previous section, the following security measures in this section should be applied over and above those for the fundamental level security level.

Identification of carriers

In implementing national security provisions for shipments of radioactive material, the competent authority should establish a program for identifying consignors or carriers engaged in the transport of radioactive material packages requiring the enhanced security level, for the purpose of communicating security related information.

Security situations

All operators and other persons engaged in the transport of radioactive material packages requiring the enhanced security level must develop, adopt, introduce, periodically review as necessary and comply with the provisions of a security scenario.

It is necessary for States to establish clearly responsibility for, and ownership of, the security situation. This will normally be the operator having direct responsibility for the security of the radioactive material in any particular mode or phase of the transport. In the event that transports are subcontracted, it may be appropriate to ensure that contractual arrangements exist to develop and comply with a security situation.

Information required in a security situation under these provisions may be incorporated into scenarios developed for other purposes. However, security situations will, almost invariably, contain information that should be restricted to those who need to know it for the performance of their works. Such information should not be included in scenarios that are developed for other purposes and that may be disseminated more widely.

Advance navigation

The consignor should provide advance notification to the consignee of the scenario shipment, mode of transport and expected delivery schedule.

The consignee should confirm capability and readiness to accept delivery at the expected schedule, prior to the commencement of transport, and should notify the consignor on receipt or non-receipt within the expected delivery time frame.

The consignor, if requested or required, should provide advance shipment notification to the competent authority of any receiving or transit State. At this level, notification that may be required for security purposes may be developed from advance notification already required for other purposes.

Positioning devices

When appropriate, tracking methods or devices may be used to monitor the movement of conveyances containing radioactive material. A simple tracking system will be able to track when a shipment has departed, whether the mode of transport has changed and if the material has been placed in interim storage or the consignment has been received. This information about status changes should be readily available to the appropriate parties (i.e. carriers, shippers and other operators). This tracking system may be as simple as a bar code system that provides information on the package location and status. The tracking system, in conjunction with a communications system and response procedures, will allow the operator and the competent authority to react in a timely manner to a malicious act, including theft of radioactive material.

Additional security system for transport by road, rail and inland waterway

The carrier should ensure, for transport conveyances by road, rail and inland waterway, the application of devices, equipment or other arrangements to deter, detect, delay and respond to theft, sabotage or other malicious acts affecting the conveyance or its cargo and should ensure that these arrangements are operational and effective at all times.

The operator should maintain continuous attendance of the road conveyance during transport where possible. Where non-attendance is unavoidable, the road conveyance should be secured such that it complies with the criteria for protection, detection and response and preferably in a well illuminated area.

2.5 Additional security systems

In certain circumstances, States may consider enhancing the foregoing baseline security measures in view of the design basis threat, the assessment of the prevailing threat or the nature of the material being moved. In such cases, possibly relevant only to certain categories or quantities of radioactive material or to particularly sensitive transports, States may require some or all of the following measures to be applied. This list is not exhaustive.

Additional training, beyond basic security awareness, may be provided to persons engaged in the transport of radioactive material to ensure that they have the proper skills and knowledge for implementing specific security measures associated with their responsibilities. Radioactive material carriers may be subject to a regime whereby their operations are licensed, their security procedures are subject to audit and their security situations are subject to formal approval and periodic review by the competent authority. Automated and real time tracking methods or devices may be required, where feasible, to permit a transport control centre to monitor remotely the movement of radioactive material conveyances and packages and the status of the material.

Persons engaged in the transport of radioactive material may be subject to formal national security clearance commensurate with their responsibilities. Guards may be required to accompany certain transports to provide for continuous effective surveillance of the package and/or conveyance. In such cases it will be important to ensure that guards are adequately trained .suitably equipped and fully aware of their responsibilities. An evaluation of the potential for sabotage and associated radiological consequences for a package design with regard to its mode of transport may be required by the competent authority. This should be done in close consultation with safety specialists. Prior to loading and shipment, appropriately trained personnel may be required to conduct a thorough search of the conveyance to ensure that it has not been tampered with in any way that could compromise security.

Special attention may be given to procedures that address points where responsibility for security is transferred and at intermodal transfer points. Consideration may be given to using conveyances that are specially designed or modified to provide additional security features. The response scenario may be reviewed to ensure that there would be an adequate response to any attempts at theft, sabotage or other malicious acts. In particular, coordination with response forces should be reviewed to ensure an appropriate and timely response to an incident.

2.6 International shipment methods

For air transport, shipment is required to be carried out in accordance with the applicable security provisions .For maritime transport, shipment is required to be carried out in accordance with the applicable security provisions of the International Ship and Port Facility Security Code and of the International Maritime Dangerous Goods Code as required by the International Convention for the Safety of Life at Sea These provisions should be supplemented by the information provided by this guide.

Before an international shipment is undertaken, the originating State may make adequate provisions to confirm that the security needs of the receiving State and any transit States will be met.

3 Measuring of nuclear fuel characteristics.

Each spent nuclear fuel (SNF) assembly contains a fraction of an IAEA significant quantity of plutonium. In the absence of a viable direct measurement technique, the current process can only rely upon operator declarations and modeling to estimate the plutonium content in the fuel. A goal of improved safeguards is to have nondestructive assay (NDA) methods to independently determine plutonium mass in spent nuclear fuel, enabling better techniques to verify declarations, detect pin diversion, reestablish continuity of knowledge, recover after the loss of containment, and determine the mass of material sent to long-term disposition or reprocessing. Multiple measurement techniques will be required to achieve this goal. While several neutron measurement techniques are being developed to assay SNF to determine total fissile material mass, assaying the particular special nuclear material (SNM) isotopes will require high-resolution gamma ray spectroscopy.

The difficulty in making this measurement is that the non-destructive assay (NDA) of a SNF assembly requires a high-resolution measurement in the presence of much larger gamma-ray emissions from the long-lived fission products such as Cs-137. Unlike fresh fuel or separated SNM, spent nuclear fuel has a high fission-product content that makes a direct measurement very difficult, if not impossible. The table below illustrates the problem that the activity of Cs-137 in particular is many orders of magnitude greater than that of the individual fissile isotopes. Given the long half-lives of these isotopes, the values will be relatively stable over decades of cooling time; the only non-negligible impacts will be on Pu-241 ($t_{1/2}$=14.35 years) and Cs-137 ($t_{1/2}$=30.07 years). All of the actinide signatures are many orders of magnitude lower in gamma intensity than the Cs-137 and are at energies where there is significant background from the down-scattering of the dominant 662 keV gamma ray from the decay of Cs-137. The U-235, Pu-240 and Pu-241 gammas are at sufficiently low energy that they will be heavily absorbed in the fuel itself and are unlikely to penetrate shielding (e.g. pool water, lead glass viewport windows, or lead filters). While the Pu-239 gamma-rays are energetic enough to penetrate through the fuel and shielding, they are unlikely to be detectable as they will be buried under the Compton down-scatter of the Cs-137 gammas in the shielding materials and the Compton continuum of the Cs-137 line in the detector itself. For example you can refer the Table 2.

Table 2: Gamma emissions of actinides compared to main fission product emission from Cs-137 for a typical PWR assembly with 3% initial enrichment and 30 GWd/tU burn-up.

Isotope	Mass per Assembly (kg)	Specific activity	Signature gamma	Branching ratio	Y/s assembly
Cs-137	0.33	3.2*1015	661.7	85.1	9.0x1014
U-235	3.6	8.0*106	185.7	57.2	1.6*107
Pu-239	1.9	2.3*1012	413.7	0.001466	6.4*107
Pu-240	1.1	8.5*1012	160.3	0.000402	3.8*107
Pu-241	1.8	3.7*1015	148.6	0.0001855	1.2*1010

Given that the direct passive assay is practically impossible, an alternative approach is using active interrogation techniques to produce short-lived fission products whose delayed gamma spectrum has characteristic gamma rays at energies well above the dominant Cs-137 background, as shown in Figure 1. Delayed gamma assay uses thermal-neutron-induced fission to generate fresh, short-lived (few-second to few-minute half life) fission products and relies on detecting gamma-rays with energies in the 3-6 MeV energy range (orange curve in Figure 1), above the passive gamma-ray background from the long-lived fission products (blue curve in Figure 1). Although the actively induced signal is separated in energy from the passive background, the gamma-ray spectrometer must be able to count at extremely high rates with excellent energy resolution due to the lack of an effective filter at these energies.

Detailed model calculations of the delayed gamma assay technique show that even with a very intense neutron generator the specific gamma rays of interest will be produced at rates 10^6-10^8 below

The passive background rates. Thus, a 3% measurement of a signature isotope will require collection of 10^9-10^{11} gamma-rays, and this must be done is a reasonable period of time (<10^4 seconds). For example, the Clink facility in Sweden requires assay of one complete canister per day and each canister contains 12 BWR or 4 PWR assemblies [9]. Assuming that each assay views ~1/10 of the length of the canister and 24-hour per day operation, each measurement must take ~2 hours including repositioning of either the fuel or the detection system between measurements [22].

Calculations for this scenario show that with the measured passive background and the modeled delayed gamma spectrum as the signal, the gamma-ray spectrometer must still operate at counting rates of order 10^6-10^7 cps in order to collect sufficient statistics for the signals of interest in acceptable times (~1 day per assembly). Note that this is the rate of events processed into an analyzable spectrum and not merely the input rate to the system. Very good energy resolution is still required since the gamma-ray spectrum must be analyzed for a large number of spectral features in which only a relatively small number provide high degrees of discrimination for Pu isotopes and U-235. In addition, the fast neutrons from the fission of U-235, Pu-239 and Pu-241 will induce fission in the dominant U-238 in the fuel at a rate of roughly 10% of the thermal neutron-induced fissions. While the energy resolution of typical laboratory high-purity germanium (HPGe) detectors is not necessary, resolution of better than 10 keV FWHM is still required in the 3-6 MeV energy range, ruling out use of scintillation detectors. The high processing rate and associated fast charge collection needs in the detector element rule out room temperature semiconductors such as CZT that have poor mobility and thus slow charge collection. The most viable technique is to use germanium, the only material available with both the spectroscopic performance and fast charge collection time, as the detector element, and focus on the engineering of high-rate electronics and processing algorithms that can collect and spectroscopic ally process about 1 Mcps of gamma-rays. There is an example of BWR Figure 3.

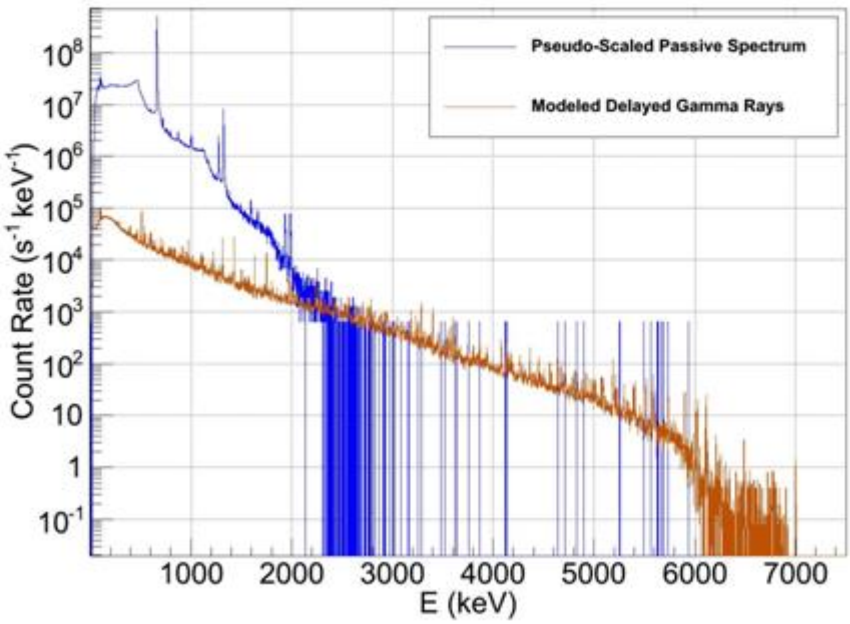

Figure 3: Measured passive background of ~65 GWd/tU burn up, 22.5 year cool down BWR fuel scaled to match equivalent exposure time for simulated delayed gamma assay scenario with 10^{10} n/s (thermal) neutron irradiation with 10 second irradiation/10 sec measurement cycles.

3.1 The UHRGe detector system

The above constraints are unlikely to be met in a standard HPGe detector system that has a coaxial detector diode whose signal is processed using long (approximately 4-16 us) filtering times. Pacific Northwest National Laboratory (PNNL) has developed a new approach to HPGe signal processing that offers an order of magnitude improvement in pulse throughput even with a conventional coaxial germanium diode and promise for an additional order of magnitude or more when these techniques are coupled to a HPGe strip detector [10]. The diode used in these measurements is a 62 mm diameter, 45 mm tall p-type semi-coax from Princeton Gamma Tech (PGT) with a nominal relative efficiency of 37%. For convenience the symmetry axis of the detector was vertical. Even with the side of the detector facing the source, the geometrical acceptance in this configuration is 90% of that of normal-to-the-end-face area that is used for the standard efficiency calculation. The PGT front-end electronics and RG-11 preamplifier have been modified to allow for the high steady-state currents when operating at ~Mcps rates. Modifications include lower feedback resistance (1 GOhm), increased feedback voltage rail (-100 V), and shorter tail time in the second stage of the amplifier (36.5 us). The intrinsic charge collection time for this HPGe diode - and in general for semi-coax detectors - is rather long (700-800 ns). This is a limiting factor in the throughput of the current system and will be addressed in the next evolution of the system (described below) that will employ scenarioar strip diode detectors that will have charge-collection times of 200-250 ns. An additional system-level cost is that more readout channels will be required and the thin detector elements will have less absolute efficiency for full energy deposition of MeV gamma rays. The final system is expected to be a stack of several ~1-2 cm thick scenarioar detectors to recover the stopping power and efficiency for the gamma rays of interest.

The data acquisition system was built around capturing the preamplifier waveforms for subsequent filtering in either an offline computer analysis or a real time online FPGA processor with firmware filters. The waveform digitizer must be high speed, low noise, and have an onboard field programmable gate array (FPGA) for real-time spectral filtering. A Signatec PX14400D (14-bit, 400 MHz sampling) and an AlazarTech ATS9626 (16-bit, 250 MHz sampling) waveform digitizer were evaluated [11]. The Signatec unit demonstrated an unacceptable 350 kHz sinusoidal "noise" generated internally in a DC offset circuit and was subsequently rejected. The AlazarTech ATS9626 has an Altera Stratix III EP3SE260 FPGA dedicated for user programming. At these input rates the data-acquisition system requires continuous (dead-time free) recording of waveforms. A Signatec DR-800 system with 8-lane PCI-e data transfer to a 96 TB RAID array (2.8 MSPS maximum transfer rate) was selected; this system can stream raw digitizer data continuously for ~16 hours, though post-analysis of this volume of data is impractical. For the work presented in this paper, waveforms were recorded to disk and analyzed offline, but the analysis algorithms were required to be implementable in FPGA firmware so that they can be moved to real-time in the future.

As seen in Figure 4, at Mcps rates the waveform has a heavy and continuous pileup where every new pulse is riding on the sum of the exponential tails of 100's of previous pulses that have occurred in the prior several RC time constants of the preamplifier output. The quiescent baseline is at 0. The signals are upward going impulses with a 55.7 us RC decay constant for this data, equivalent to the width of the waveform section shown in the lower half of the figure which contains ~40 individual gamma ray events. The DC level is shifted by 4000-5000 channels in the filter output value (the gain is roughly 4.9 keV per channel) with significant fluctuations due to the Poisson fluctuations of the radioactive decay of the Cs-137 source.

The signal processing technique used is based on conventional trapezoidal filtering techniques. A fast triangular filter (80 ns rise time) is used as a trigger filter to identify pulses and to determine event times. In addition, the time over threshold for this filter can be used to identify accidental coincidences (Figure-5). Energy filters are trapezoidal with gap times fixed to 800 ns (slightly longer than the longest charge collection time in the HPGe diode) and rise times from 32 ns to 4 us. The ideal trapezoidal filter rise time at low rates is found to be 2 us; this should be considerably longer for this HPGe diode and preamplifier but the digitizer used is introducing some additional low frequency noise that degrades resolution for long filter lengths. This is not a concern for high rate operation where one typically gets maximum usable filter rise time around 512 ns [23]. We employ a variable length filter where the longest filter that can be evaluated between the previous and next events is used to evaluate the current event. Our method, developed independently. The resulting energy spectrum for the Cs-137 source is shown in Figure 4. One can readily see the large increase in throughput achieved by using a variable length filter as opposed to the fixed length (512 ns) that represents the optimum at this count rate (1.1 Mcps input). One can see the single 661.7 keV line as well as the 2X and 3X 661.7 keV lines from accidental coincidences. The right-hand panel in Figure-4 shows how one can suppress accidental coincidences (trigger pile-up) using a cut on the time-over-threshold of the trigger filter output [24].

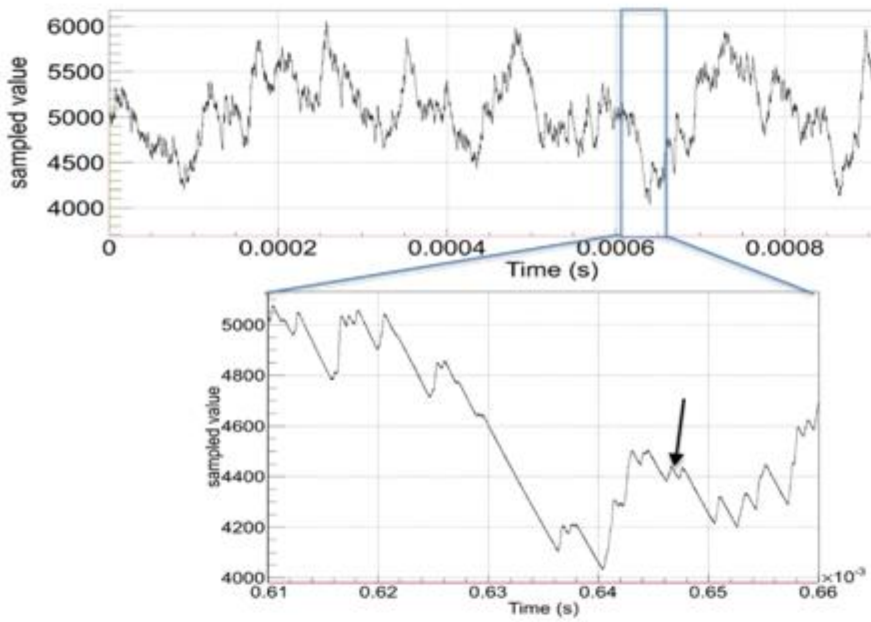

Figure-4: Waveform collected at input count rate of ~1 Mcps using a 722 uCi Cs-137 source. Above: 1 ms trace. Below: 50 us section of waveform.

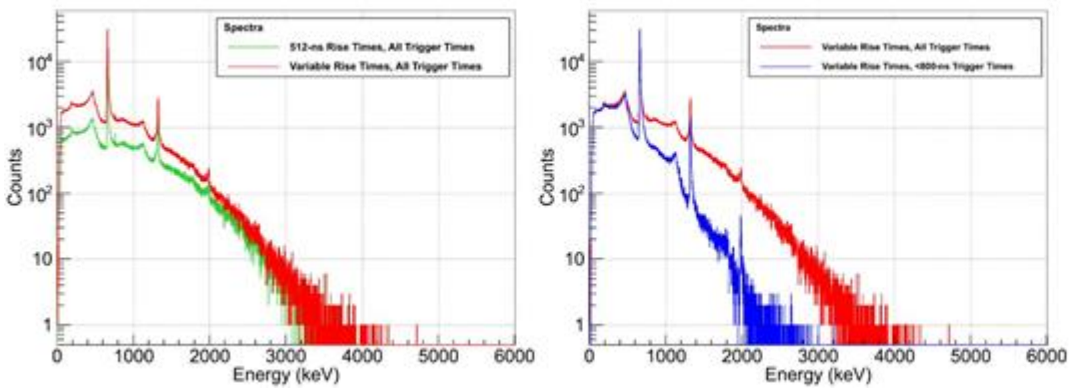

Figure-5: Energy spectrum from 722 uCi Cs-137 source at ~1.1 Mcps input rate.

3.2 The Spent Fuel Measurement

Passive gamma-ray measurements were made on a 58.2 g sample of ATM-109 fuel shown in Figure 4. ATM-109 is high-burn up-BWR fuel with an initial enrichment of 3% U-235 that originates from an assembly irradiated in the Quad Cities I reactor from Feb 1979 through September 1987 to a burn up of 43 GWd/tU and again from November 1989 through September 1992 at an average exposure of ~70 GWd/tU, though the burn up estimates for this fuel range from 60-75 GWd/tU [15]. The cool down time from last irradiation to the measurements described in this paper is 21.5 years. The sample used in the measurements is 58.1993±0.0006 g, including the cladding, and measures about 5.5 cm in length. Measurements were made with the detector viewing the fuel through a large diameter (~6”) open port in a hot cell at the Radiological Processing Laboratory at PNNL. The detector was located in a waste storage area behind the hot cells at distances of ~403 cm and ~803 cm from the fuel. Measurements were taken with no shielding and with up to 3/4” lead shielding to study the impact of “light” shielding on the detector rate and performance.

The gamma-ray energy spectrum shown in Figure 5 was obtained at ~803 cm standoff with no shielding. A triangular filter with rise time l=80 ns was used as a “trigger” to identify individual pulses in the recorded waveforms. The distribution of time between triggers was fit for an exponential to estimate the rate of triggered events; the rate was 413 kHz for this running condition. A trapezoidal filter with L=512 ns (rise time) and G=800 ns (flat top) was used to determine the pulse height (energy). Several event quality cuts are made based on the trigger filter output to select the events that are displayed in the energy histogram. Most important of these is a cut on the time over threshold of the trigger filter, which was restricted to the range 300-700 ns; the lower bound removes spurious events from baseline fluctuations while the upper bound eliminates “trigger pileup” events (accidental coincidence) where the trigger filter does not return below threshold between gamma ray interactions. The resulting rate of events recorded in the energy histogram is 167 kcps, or 40% of the triggered events.

Figure 5 clearly illustrates that the passive spent fuel spectrum has a large amount of trigger pileup remaining; in fact, most of the events above the 661.7 keV Cs-137 peak are trigger pileup events with one or more Cs-137 gamma-rays (full energy peak or Compton) with the tail of this distribution extending up to the region where delayed gamma signatures of interest reside. The 700 ns bound on rejecting this pileup is determined by the (longest) charge collection times in the HPGe diode. As mentioned above, this also sets the lower limit for the gap time of the energy filter (800 ns) used in the analysis. Both of these can be reduced to ~200-300 ns using a scenarioar (strip) detector configuration and this is being pursued as the next evolution of the system development. This should greatly improve throughput and reduce accidental coincidence events due to trigger pileup.

3.2.1 Comparison of measurements with known fuel history.

At long cooling times, the gamma ray emissions of spent fuel are dominated by Cs-137. The mass of Cs-137 can be estimated based on the sample mass, burn up and cool down time. Production is ~6.25 g Cs-137 per GWd/tU burn up. This equates to 22 mg Cs-137 in 50 g ATM-109 (for 70 GWd/tU). The average production time of the Cs-137 is about 25 years (see irradiation history above) so the current Cs-137 mass is estimated to be 1.2 mg with an activity of 40 GBq and 661.7 keV gamma-ray production rate of $\sim 3.6\times10^{10}$/s. The observed rate of 661.7 keV gammas is 2.8×10^{4}/s unshielded at ~800 cm standoff. Random coincidences in the detector will shift Cs-137 gamma rays out of the photo peak (to higher energies). The rate of events in the 2×661.7 keV peak ($3.8\text{x}10^{2}$/s) and the peak-to-total ratio for the detector (~0.2) can be used to estimate the magnitude of this effect; correcting for this ~7% loss, the 661.7 keV photo peak event rate is 3.0×10^{4}/s [24]. With a 5cm × 1 cm source at this range a point source approximation is appropriate. We can scale the observed rate at 803 cm to the expected rate at the nominal 25 cm range used for the ASME standard for efficiency measurements; the result is 3.1×10^7/s or an absolute photo peak efficiency of 0.086%. This is consistent with, but somewhat lower, than expected for this HPGe detector indicating that the assumed 70 GWd/tU burn up estimate is too high and that the burn up is closer to 65 GWd/tU. In addition to Cs-137, both Cs-134 and Eu-154 are observed in the spectrum. For Cs-134 the 563, 569, 604.3, 795.6 and 802 keV lines were observed. For Eu-154 the 247.2, 591.2, 723, 757.8, 873.1, 994.1, 1004, 1275, 1493 and 1597 keV lines were observed. The 757.8 keV feature in the spectrum includes a significant contribution of accidental coincidence summing of the 661.7 keV Cs-137 line with 96.5 keV uranium x-rays. The Cs-137/Eu-154 ratio can be used to estimate the burn up [16] with a result of 67 GWd/tU, in reasonable good agreement with the pedigree of the sample (Figure-6).

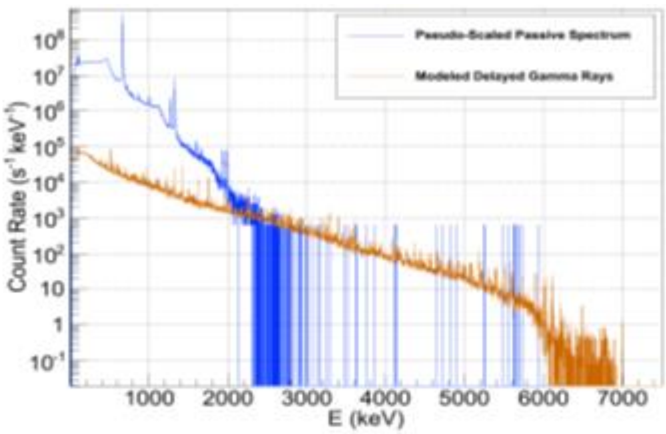

Figure 6: Energy spectrum obtained at 803 cm with no shielding.

Measurements were made at the near (403 cm standoff) position with no shielding and lead shielding of ¼, ½ and ¾ inch thickness. The results are shown in Figure 6. The bare case has an input rate of 1.4 Mcps. The ½” lead shielding case at 403 cm has a very similar count rate (~500 kcps input) and spectral shape as the data collected at 803 cm with no shielding. The bare fuel data include the ~100 keV uranium x-rays while the shielded data do not. The shielded data has roughly the same yield for the 661.7 keV Cs-137 peak but lower Compton continuum at low energy. At higher energy (above the 2X 661.7 keV accidental coincidence peak) the continuum is nearly identical while the photo peaks for the higher energy (1274.4 keV, 1596.5 keV) Eu-154 lines are slightly larger.

3.2.2Current and future development

This measurement was made using stored waveforms and offline analysis. Since this measurement, the UHRGe system has been further developed and the real time processing implemented with the offline analysis algorithms implemented in the digitizer FPGA and real-time data-collection in the host CPU. The FPGA performs the filter algebra on the data stream in real-time while the host CPU performs the final energy calculations that require floating point division. A total of (at least) 8 filters can be evaluated concurrently at full data rate. At present, the energy resolution from the real-time analysis is slightly worse than from the offline code. The filters being used have a minimal gap time (flat top) so the output is extremely sensitive to precise event timing and it is suspected that this may not yet be getting calculated correctly in the real-time processing [25].

To further improve the rate capability it will be necessary to speed up the charge collection time in the detector diode and go to a strip design that distributes the gamma rays over multiple readout channels. A scenario strip detector is being developed by Lawrence Berkeley National Laboratory for the next iteration of the system. The scenario geometry will result in much shorter drift times (~200 ns vs. ~700 ns), which will directly and dramatically increase the throughput and reduce the accidental-coincidence (trigger-pileup) rate. This new detector is segmented into 10 strips, and assuming the demonstrated UHRGe rates with the coaxial detector, the total input-rate capability for the system will be in excess of 5 Mcps and the usable spectroscopic event rate should be well over the target 1 Mcps.

As part of the larger SNF assay project, gamma-ray mirrors are being developed by the Lawrence Livermore National Laboratory. Those multilayer mirrors have high reflectivity even at energies up to 600 keV. It may be possible to use a mirror assembly to deflect the dominant 661.7 keV gammas out of the line of sight to the detector. Four consecutive 80% reflectivity elements would be required to suppress the dominant gamma line by a factor of 625. Since these mirrors have very shallow reflection angles it is necessary to place them as close as possible to the source (fuel assembly). The manufacturing limitation for these mirrors might dictate different collimation and scanning geometry such as a longer slit collimation rather than viewing a ~22 cm × 22 cm square length of an assembly or similar section of a cask.

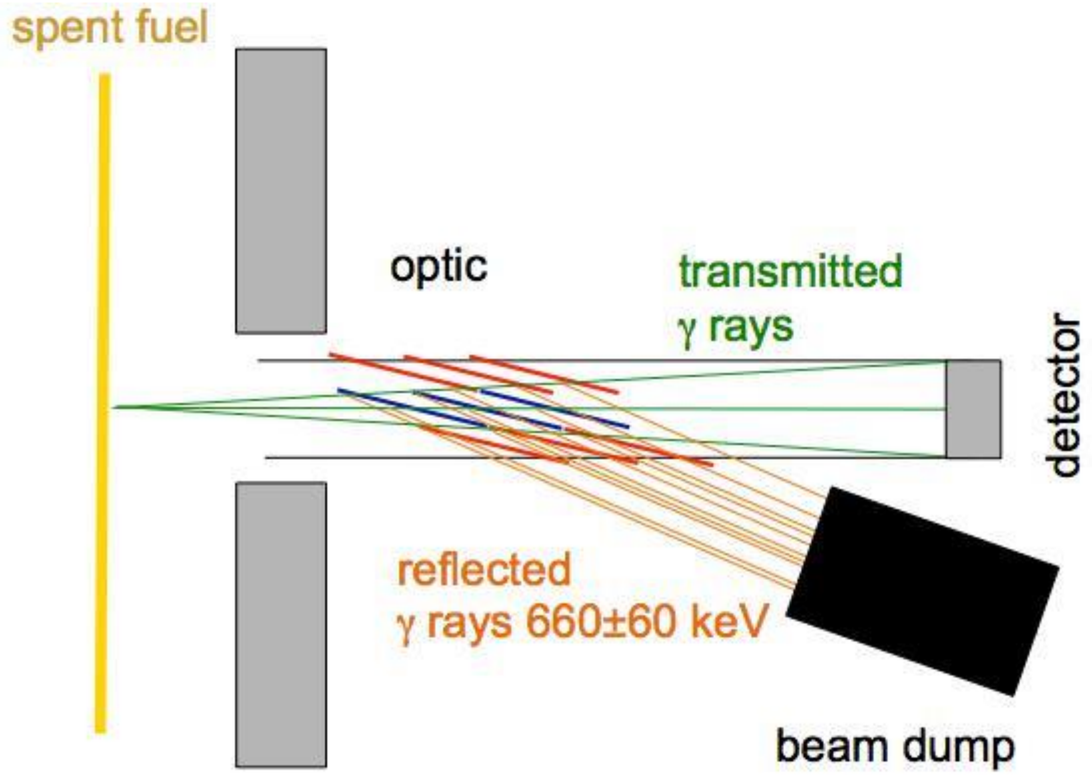

Figure-7: Layout of gamma-ray detector.

3.3 Reduction of activity in decay products in SNF

1.Calculation of reduction of activity in decay products in SNF. By use the data given in Table 3 (activity of isotopes in SNF,Ci) and Law of radioactive decay for calculation activity of isotopes through time Perform calculations for times: 1, 5, 10, 30, 50, 100 years. By create a graph based on calculated data (activity/time) Fig.8

Table 3 Activity of isotopes in SNF,Ci

Isotops	T(1/2) years	Activity(Ci/ton of U)(.5 years)
Kr 85	10.8	11000
Sr 90	29	80000
Y 90	0.00712329	80000
Y 91	0.156164	130000
Zr 95	0.178082	220000
Nb 95	0.0958904	430000
Ru 106	1.01096	370000
Rh 106	0.0821918	730000
Sb 125	2	7300
Te 125	0.158904	3000
Te 127	0.287671	5100
I 129	17000000	0.033
Cs 134	2.3	19000
Cs 137	26.6	100000
Ba 137	0.427397	100000
Ce 144	0.778082	790000
Pr 144	3.23E-05	790000
Pm 147	2.6	110000
Eu 154	8.8	6000
Eu 155	1.7	5600

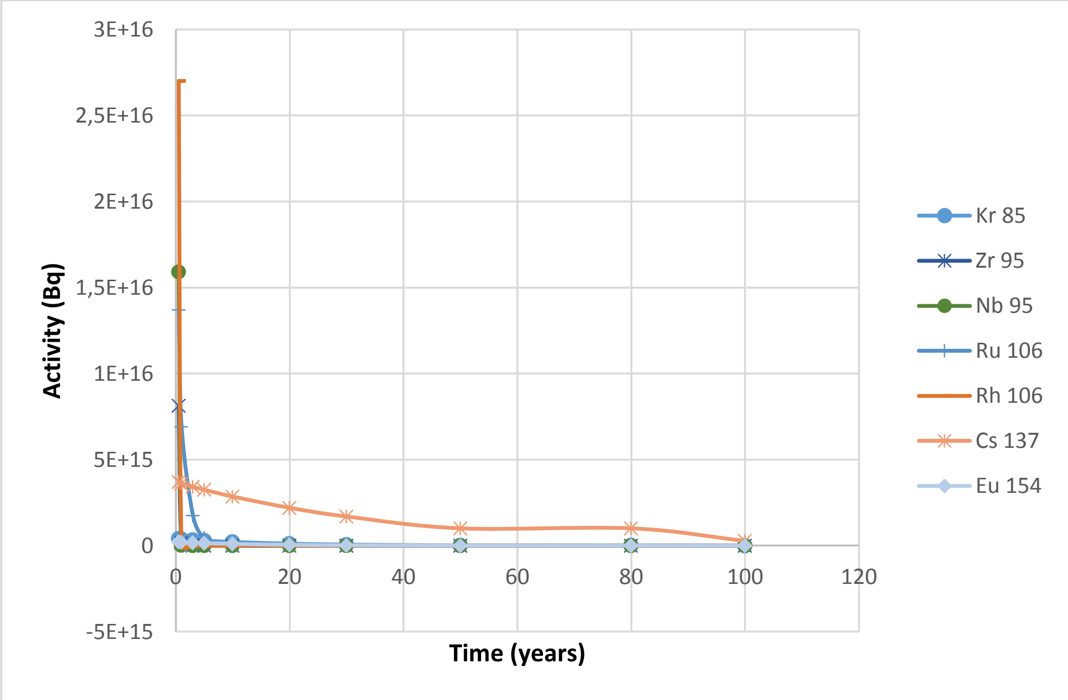

Figure -8 Activity(Bq)/time(years)

2.Calculation of doses on the surface of Casks

Detailed calculation is performed in order to obtain the intensity and doses of gamma-radiation while transporting SNF.

1) By choosing the elements that emit gamma-radiation And finding the following data for each isotope: n_γ – number of gamma-particles emitted in one decay; E_γ- energy of emitted gamma-particles.

2) Calculate the initial intensity of gamma-radiation:

$$I_0 = \frac{n_\gamma A \varepsilon_\gamma}{4\pi r^2}$$

Where r = 0,05 m – distance from radiation source;

3) Calculate the reduction of intensity for gamma-radiation when it passes through the material of the cask(Table-4):

$$I = I_0 \exp(-\mu * x)$$

Where μ – coefficient of attenuation of gamma-radiation (find it in the table below. It is determined depending on the energy of gamma-particles and material in which gamma-radiation moves); x – thickness of material (use three cases: x = 2, 5 and 10 cm).

4) Calculate the activity depending on calculated intensity (for all data Table-6):

$$A = \frac{I_0 4\pi r^2}{n_\gamma \varepsilon_\gamma}$$

5) Calculate the dose on the surface of the cask(Table-7):

$$\dot{H} = \frac{A E_\gamma n_\gamma w_\gamma}{m}$$

Where m = 70 kg – average mass of human; w_γ = 1 – quality coefficient for gamma-radiation.

Table 4 Radiation isotopes details

Isotope	A0, Bq	T1/2, years	E_γ, MeV	E_γ, J	n_γ	I_0, J/(s*m2)
Cs-137	3.7*1015	26.6	0.6617	1.06*10-13	1	1429.38123
Nb-95	1.591*1016	0.0958904	0.765	1.23*10-13	1	1.0756*10-90
Kr-85	4.07*1014	10.8	513.997	3.57*10-14	1	16.89462595
Rh-106	2.701*1016	0.0821918	622.2	4.30*10-11	1	1.793*10-79
Zr-95	8.14*1015	0.178082	724.2	5.00*10-14	1	6.448*10 -48
Ru-106	1.369*1016	1.01096	622.2	9.77*10-15	1	1.2479*10-6
Eu-154	2.22*1014	8.8	996.8	1.64*10-14	1	2.738408604

Table 5 – Intensity of isotopes in Lead and Aluminum.

Isotope	I_0, J/(s*m2)	Lead			Aluminum		
		I(2), J/(s*m2)	I(5), J/(s*m2)	I(10), J/(s*m2)	I(2), J/(s*m2)	I(5), J/(s*m2)	I(10), J/(s*m2)
Cs-137	1429.38123	152.1697879	5.285656988	0.019545639	965.838778	536.4626401	201.340383
Nb-95	1.075*10 -90	1.48509E-91	7.61902E-93	5.39686E-95	7.44451E-91	4.28653E-91	1.70826E-91
Kr-85	16.89463	9.51478*10 10	4.02137E-25	9.57193E-51	8.855117853	3.360183482	0.668309146
Rh-106	1.793*10 -79	1.31534*10 -83	8.26631*10 -90	3.8116*10 -100	1.02814*10 -79	4.4653*10 -80	1.1122*10 -80
Zr-95	6.448*10 -48	6.86434*10 -49	2.38434*10 -50	8.81698*10 -53	4.35687*10 -48	2.41997*10 -48	9.08241*10 -49
Ru-106	1.248*10 -6	8.05779*10 -08	1.3221*10 -09	1.40071*10 -12	8.19934*10 -07	4.3669*10 -07	1.52814*10 -07
Eu-154	2.7384	0.564043981	0.052727241	0.001015247	1.964773458	1.194081117	0.520678219

Table 6 – Activity of isotopes in Lead and Aluminum.

Isotope	A_0, Bq	Lead			Aluminum		
		A(2), Bq	A(5), Bq	A(10), Bq	A(2), Bq	A(5), Bq	A(10), Bq
Cs-137	1.69*10 +15	1.80*10 +14	6.26*10 +12	2.32*10 +10	1.14*10 +15	6.36*10 +14	2.39*10 +14

Continuation of Table 6

Nb-95	1.10*10 -78	1.52*10 -79	7.81*10 -81	5.53*10 -83	7.63*10 -79	4.39*10 -79	1.75*10 -79
Kr-85	5.94*10 +13	3.34*10 +05	1.41*10 -12	3.36*10 -38	3.11*10 +13	1.18*10 +13	2.35*10 +12
Rh-106	3.79*10 -94	3.85*10 -72	2.42*10 -78	1.11*10 -88	3.01*10 -68	1.31*10 -68	3.25*10 -69
Zr-95	1.62*10 -35	1.72*10 -36	5.99*10 -38	2.21*10 -40	1.09*10 -35	6.08*10 -36	2.28*10 -36
Ru-106	1.60*10 +07	3.30*10 +05	5.41*10 +03	5.74*10 +00	3.36*10 +06	1.79*10 +06	6.26*10 +05
Eu-154	2.09*10 +13	1.37*10 +12	1.28*10 +11	2.47*10 +09	4.78*10 +12	2.90*10 +12	1.27*10 +12

Table 7 – Dose rate of isotopes in Lead and Aluminum.

Isotope	Lead			Aluminum		
	$\dot{H}$(2), mrem/h	$\dot{H}$ (5), mrem/h	$\dot{H}$ (10), mrem/h	$\dot{H}$ (2), mrem/h	$\dot{H}$ (5), mrem/h	$\dot{H}$ (10), mrem/h
Cs-137	9.83*10 +07	3.41*10 +06	1.26*10 +04	6.24*10 +08	3.47*10 +08	1.30*10 +08
Nb-95	9.59285*10 -86	4.92*10 -87	3.48606*10 -89	4.81*10 -85	2.77*10 -85	1.10344*10 -85
Kr-85	0.06146005	2.59758*10 -19	6.18292*10 -45	5719900.126	2170486.519	431689.519
Rh-106	8.496*10 -76	5.34*10 -82	2.46*10 -92	6.6412*10 -72	2.88433*10 -72	7.184*10 -73
Zr-95	3.80947*10 -40	1.32323*10 -41	4.89312*10 -44	2.41791*10 -39	1.343*10 -39	5.04042*10 -40

Continuation of Table 7

Ru-106	0.016576025	0.000271976	2.88146*10 -07	0.168672229	0.089833446	0.031436114
Eu-154	116031.9047	10846.74672	208.8508733	404181.9685	245639.5441	107110.9479

Finally, if the source is too intensive and time or distance do not provide sufficient radiation protection, the shielding must be used. Radiation shielding usually consist of barriers of lead, concrete or water. There are many many materials, which can be used for radiation shielding, but there are many many situations in radiation protection. It highly depends on the type of radiation to be shielded, its energy and many other parametres. For example, even depleted uranium can be used as a good protection from gamma radiation, but on the other hand uranium is absolutely inappropriate shielding of neutron radiation.After calculating and comparing between the results it was shown that Lead has excellent shielding properties against gamma rays.

3.4 Selection of Casks

Past risk assessments of spent fuel transportation have used generic cask designs with features similar to real casks but generally without all of the conservatisms that are part of real cask designs, such as assumptions on material strength and energy-absorbing capabilities of impact limiters. In the current study, the risk assessment was performed using actual cask designs with all of the design margins that contribute to their robustness [26]. Because it is too costly and timeconsuming to examine all casks, a subset of casks was selected for the risk assessment. Tthe various NRC-certified spent fuel casks at the time the study began, provides options for choosing the casks, describes some important features of the various cask designs, and finally concludes with the casks chosen (Appendix B).

The casks chosen for detailed analysis were the NAC-STC (Figure 8) and the HI-STAR 100 rail casks. The GA-4 truck cask (Figure 9) was used to evaluate truck shipments, but detailed impact analyses of this cask were not performed because previous analyses of both truck and rail casks have shown that truck casks have significantly lower probability of release of radioactive material in impact accidents. The impact analyses from Sprung et al. were used to assess the response of the GA- 4 cask. The NAC-STC cask was chosen because it is certified for transport of spent fuel either with or without an internal welded canister. For transport of spent fuel without an internal canister, the NAC-STC's COC allows the use of elastomeric or metallic O-rings. Although five casks in the group use lead for their gamma shielding, only the NAC-STC cask can transport fuel not contained within an inner welded canister. As noted in the analyses of previous chapters the inclusion of spent fuel without an inner welded canister ensures that the potential pathway for radioactive material release into the environment was considered.

The Russian abbreviation VVER stands for 'water-water energy reactor' (i.e. water-cooled water-moderated energy reactor). The design is a type of pressurized water reactor (PWR). The main distinguishing features of the VVER compared to other PWRs are:

- Horizontal steam generators;
- Hexagonal fuel assemblies;
- No bottom penetrations in the pressure vessel;
- High-capacity pressurizes providing a large reactor coolant inventory.

Reactor fuel rods are fully immersed in water kept at 15 MPa pressure so that it does not boil at the normal (220 to over 300 °C) operating temperatures. Water in the reactor serves both as a coolant and a moderator which is an important safety feature. Should coolant circulation fail, the neutron moderation effect of the water diminishes, reducing reaction intensity and compensating for loss of cooling, a condition known as negative void coefficient [27]. Later versions of the reactors are encased in massive steel pressure shells. Fuel is low enriched (ca. 2.4–4.4% ^{235}U) uranium dioxide (UO_2) or equivalent pressed into pellets and assembled into fuel rods.

Reactivity is controlled by control rods that can be inserted into the reactor from above. These rods are made from a neutron absorbing material and, depending on depth of insertion, hinder the chain reaction. If there is an emergency, a reactor shutdown can be performed by full insertion of the control rods into the core. The corresponding VVER-1000 design for cask is shown in APPENDIX A.

The HI- STAR 100 rail cask was chosen because it was the only all-steel cask in the group certified for transport of fuel in an inner welded canister. The GA- 4 truck cask was selected because it has a larger capacity than the NAC-LWT; therefore, it was more likely to be used in a large spent fuel transportation campaign. The chosen casks included all three of the most common shielding options: lead, depleted uranium (DU), and steel [28].

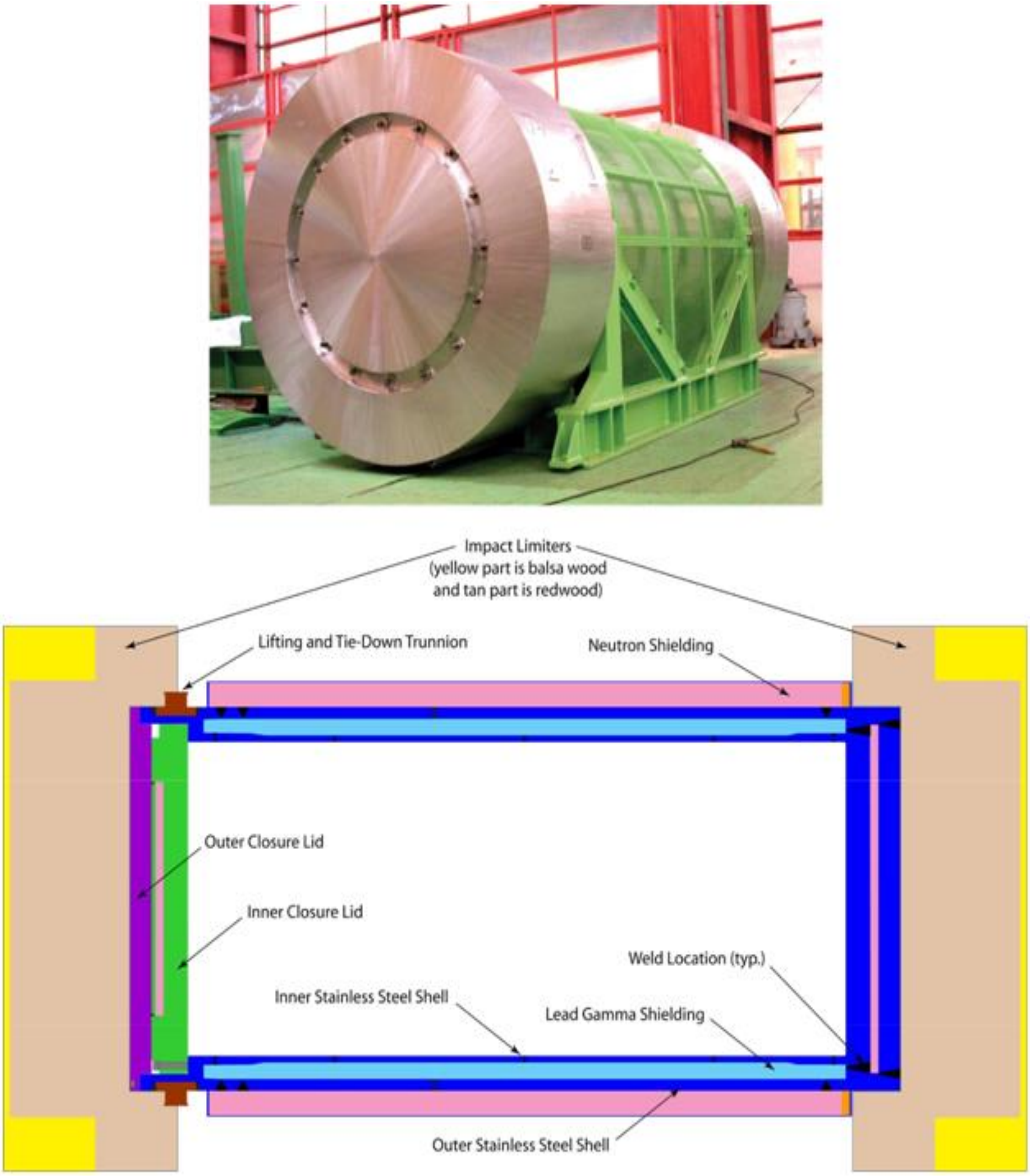

Figure-8 Photograph and cross-section of the NAC-STC cask Figure source: (courtesy of NAC International)

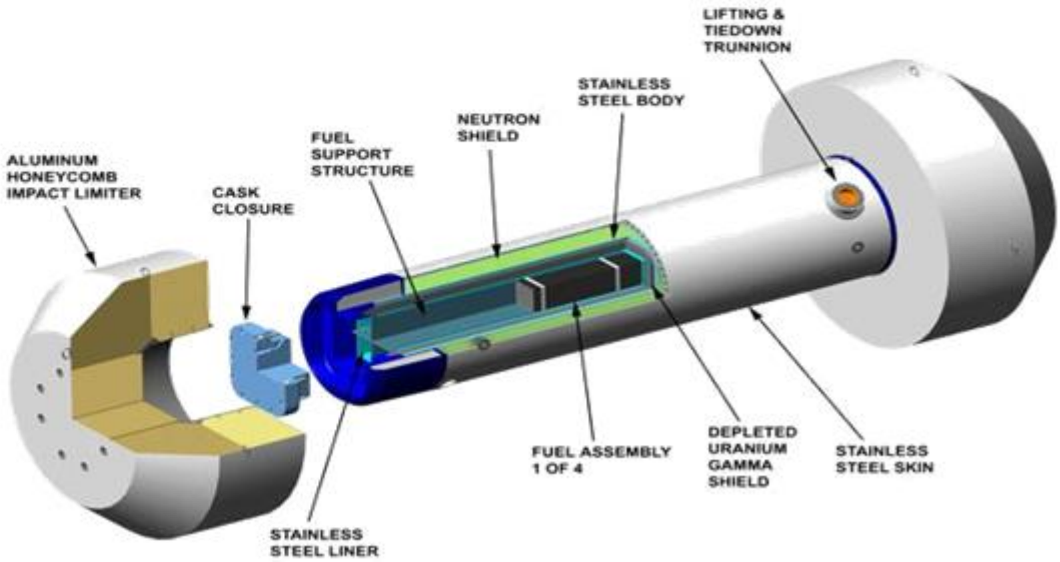

Figure-9 GA-4 cask

Table-8 Casks Chosen and Reasons for Selection

Cask Chosen	Type of Cask	Reason for Consideration in this Study
HI-STAR 100 Rail Cask4	Rail-Steel Cask	This was the only all-steel cask in the group that was certified for transport of fuel in an inner welded Canister
NAC-STC Rail Cask7	Rail-Lead Cask	Only the NAC-STC cask of this group can transport fuel that is not contained within an inner welded canister, thus ensuring the maximum potential for radioactive material released into the environment was considered.
GA-4 Truck Cask	Truck-DU	The GA-4 truck cask was chosen because its large capacity made it more likely to be used in any large transportation campaign.

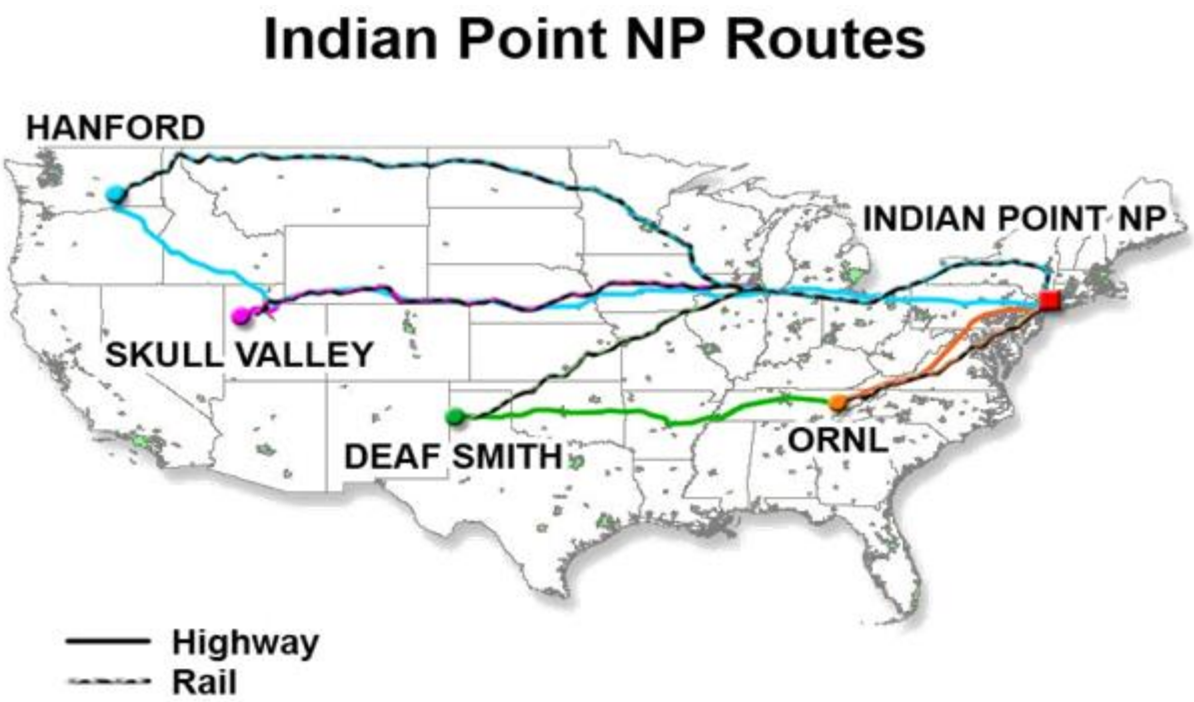

Figure 15 -Highway and rail routes from Indian Point Nuclear plant

3.5 Transportation of spent Nuclear Fuel from Shutdown Reactors

Both the Blue Ribbon Commission on America's Nuclear Future (BRC) and DOE have recommended that spent nuclear fuel being stored at shutdown reactor sites be first in line for transfer to consolidated storage or permanent disposal in a geologic repository. Transporting spent fuel from shutdown reactor sites raises a number of issues, including the potential that some of the fuel assemblies are damaged and may require re-analysis and possible repackaging; the need to upgrade certificates of compliance for some canisters before transport casks or impact limiters can be fabricated; the need for changes to the existing Standard Contract for reasons noted previously; and the need to upgrade or build new infrastructure to provide rail, barge, or heavy-haul truck access to the shutdown sites. In addition, several shutdown sites have not maintained the capability to transfer storage canisters to transportation casks. This is likely to be an issue for a growing number of sites as utilities decommission storage pools at shutdown reactors [29]. At present, only two of the nation's 14 shutdown reactor sites have transport casks that have been fabricated and are available; however, these casks do not have impact limiters. Vendors would need to update Nuclear Regulatory Commission (NRC) certificates of compliance for transport casks, as needed, and manufacture the casks—this process is likely to take several years.

3.6 Conclusion

The following is a synopsis of the review of the individual safety related criteria and issues of special concern, together with recommendations for further action.

To some extent, the current framework of fuel safety criteria remains applicable, being largely unaffected by the “new” or modern design changes; the numeric values of the individual safety criteria may, however, change in accordance with the particular fuel and core design features. Some of these values have already been or are continuously being – adjusted.

Assessment of fuel safety criteria, the following process is recommended: Continue to further develop best-estimate analysis methods, together with a suitable uncertainty analysis, in all areas of safety analysis. Continue to perform experimental studies for benchmarking of best-estimate codes and extending the verification validation basis for safety criteria and the codes Review, and adjust or change where necessary, safety criteria based on the above codes and test data; define or quantify necessary margin to safety limits.

A HPGe spectrometer has been developed and evaluated for high rate operation using Cs-137. A 50 g sample of high burn up BWR spent fuel has been assayed bare and with light shielding up to ¾” of lead. Data have been collected using high burn up BWR spent nuclear fuel at input count rates as high as

1.4 Mcps with throughput as high as 420 kcps and 247 kcps after rejecting trigger pileup events. Analysis of the Cs and Eu observed is consistent with the known fuel history. Evaluation of potential delayed gamma signal rates and the measurement speed needs at nuclear facilities confirm that the required rate of usable events in the energy spectrum must be in excess of 1 Mcps. Achieving rates of a few Mcps will require use of segmented scenario HPGe detectors, currently under development.

In this study, the risk associated with the transportation of spent nuclear fuel (SNF) was estimated by examining the behavior of three NRC-certified casks during routine transportation and in transportation accidents. Two casks are designed for transport by railroad: (1) a cask with steel gamma shielding and an inner welded canister for the spent fuel and (2) a cask with lead gamma shielding that can transport spent fuel within an inner welded canister or without an inner canister A third cask with depleted uranium (DU) gamma shielding is designed to transport directly loaded spent fuel by highway. The response of these casks is typical of other cask designs. The use of certified cask designs means this risk assessment includes the factors of safety typically included in cask designs but not specifically considered in previous risk assessments.

The risks associated with routine shipments and shipments where an accident occurs are calculated separately. During routine transportation, the risk and the consequence are the same. In this case, the dose to residents living along a transportation route, to people sharing the highway or railway, people at stops, and transportation workers are all calculated. Regulations allow limited external radiation from the cask. The dose of radiation to members of the public during routine transportation is a small fraction of the naturally occurring background radiation that individuals experience. After calculating and comparing between the results it was shown that Lead has excellent shielding properties against gamma rays.

Thus the detailed investigation has been performed and results are being tabulated and recommendations are given for future research.

4 Financial management, resource efficiency and resource conservation

4.1 Introduction to SWOT

SWOT analysis (alternatively SWOT matrix) is an acronym for strengths, weaknesses, opportunities, and threats and is a structured planning method that evaluates those four elements of an organization, project or business venture. A SWOT analysis can be carried out for a company, product, place, industry, or person. It involves specifying the objective of the business venture or project and identifying the internal and external factors that are favorable and unfavorable to achieve that objective. Some authors credit SWOT to Albert Humphrey, who led a convention at the Stanford Research Institute (now SRI International) in the 1960s and 1970s using data from Fortune 500 companies. However, Humphrey himself did not claim the creation of SWOT, and the origins remain obscure. The degree to which the internal environment of the firm matches with the external environment is expressed by the concept of strategic fit.

- Strengths: characteristics of the business or project that give it an advantage over others;
- Weaknesses: characteristics of the business that place the business or project at a disadvantage relative to others;
- Opportunities: elements in the environment that the business or project could exploit to its advantage;
- Threats: elements in the environment that could cause trouble for the business or project.

Identification of SWOTs is important because they can inform later steps in planning to achieve the objective. First, decision-makers should consider whether the objective is attainable, given the SWOTs. If the objective is not attainable, they must select a different objective and repeat the process.

Users of SWOT analysis must ask and answer questions that generate meaningful information for each category (strengths, weaknesses, opportunities, and threats) to make the analysis useful and find their competitive advantage. SWOT analysis aims to identify the key internal and external factors seen as important to achieving an objective. SWOT analysis groups key pieces of information into two main categories:

1. Internal factors – the strengths and weaknesses internal to the organization
2. External factors – the opportunities and threats presented by the environment external to the organization

Analysis may view the internal factors as strengths or as weaknesses depending upon their effect on the organization's objectives. What may represent strengths with respect to one objective may be weaknesses (distractions, competition) for another objective. The factors may include all of the 4Ps as well as personnel, finance, manufacturing capabilities, and so on.

The external factors may include macroeconomic matters, technological change, legislation, and socio-cultural changes, as well as changes in the marketplace or in competitive position. The results are often presented in the form of a matrix.

SWOT analysis is just one method of categorization and has its own weaknesses. For example, it may tend to persuade its users to compile lists rather than to think about actual important factors in achieving objectives. It also presents the resulting lists uncritically and without clear prioritization so that, for example, weak opportunities may appear to balance strong threats.

It is prudent not to eliminate any candidate SWOT entry too quickly. The importance of individual SWOTs will be revealed by the value of the strategies they generate. A SWOT item that produces valuable strategies is important. A SWOT item that generates no strategies is not important.

SWOT stands for: **S**trength, **W**eakness, **O**pportunity, **T**hreat. A SWOT analysis guides you to identify your organization’s strengths and weaknesses (S-W), as well as broader opportunities and threats (O-T). Developing a fuller awareness of the situation helps with both strategic planning and decision-making.

The SWOT method was originally developed for business and industry, but it is equally useful in the work of community health and development, education, and even for personal growth.

SWOT is not the only assessment technique you can use. Compare it with other assessment tools in the Community Tool Box to determine if this is the right approach for your situation. The strengths of this method are its simplicity and application to a variety of levels of operation.

Strengths	**Weaknesses**
• Potential series production of reactors (« off-the-shelf product ») • Reducition of construction period of low-power units (reduction of civil works) • More accessible financing for reactors (lower global construction cost than high-power stations)	• Ill-suited certification process of new reactors for spatial multiplication of units • Necessary adaptation of international safety controls (e.g.: prescriptive character of recommendations based on peer reviews)
Opportunities	**Threats**
• Emergence of new markets (electro-intensive industry, isolated sites, replacement of low-power production stations, etc.) • Aftereffects on the whole nuclear industry and especially on the downstream sector (dismantling and waste treatment)	• Nuclear proliferation with multiplication of units • Acceptability by populations and politics (difficulty to perceive nuclear energy as an energy of the future)

Fig-16, SWOT analysis schematic representation.

1. Listing Your Internal Factors: Strengths and Weaknesses (S, W)

– Human resources - staff, volunteers, board members, target population

– Physical resources - your location, building, equipment

– Financial - grants, funding agencies, other sources of income

– Activities and processes - programs you run, systems you employ

– Past experiences - building blocks for learning and success, your reputation in the community.

Don't be too modest when listing your strengths. If you're having difficulty naming them, start by simply listing your characteristics (e.g.., we're small, we're connected to the neighborhood). Some of these will probably be strengths.

Although the strengths and weakness of your organization are your internal qualities, don't overlook the perspective of people outside your group. Identify strengths and weaknesses from both your own point of view and that of others, including those you serve or deal with.

How do you get information about how outsiders perceive your strengths and weaknesses? You may know already if you've listened to those you serve. If not, this might be the time to gather that type of information. See related sections for ideas on conducting focus groups user surveys and listening sessions.

2. Listing External Factors: Opportunities and Threats (O, T)

Cast a wide net for the external part of the assessment. No organization, group, program, or neighborhood is immune to outside events and forces. Consider your connectedness, for better and worse, as you compile this part of your SWOT list.

3. Forces and facts that your group does not control include:

- Future trends in your field or the culture
- The economy - local, national, or international
- Funding sources - foundations, donors, legislatures
- Demographics - changes in the age, race, gender, culture of those you serve or in your area
- The physical environment (Is your building in a growing part of town? Is the bus company cutting routes?)
- Legislation (Do new federal requirements make your job harder...or easier?)
- Local, national or international events

The most common users of a SWOT analysis are team members and project managers who are responsible for decision-making and strategic planning.

An individual or small group can develop a SWOT analysis, but it will be more effective if you take advantage of many stakeholders. Each person or group offers a different perspective on the strengths and weaknesses of your program and has different experiences of both.

Likewise, one staff member, or volunteer or stakeholder may have information about an opportunity or threat that is essential to understanding your position and determining your future

4.2 Raw materials, purchased products and semi-finished products for the research project

This item includes the cost of all kinds of purchasing materials, components and semi-finished products necessary for the implementation of works on the subject. Number of required material values determined by the norms of consumption.

Where:

Additional salary = basic salary × (10-14%);

Benefit expenses = (basic salary + additional salary) × 35, 5%;

Overhead cost = basic salary × (50-65%).

Calculating the expenses of material costs based on the current price list or negotiated prices. The expenses of material costs include transportation and procurement costs (4 – 6% of the price). In the same item includes the expenses of paperwork (stationery, copying materials). The results of this term are presented in the Table-9 and Table-10 below.

Table 9 – Raw materials, components and semi-finished products

Title	Mark, size	Quantity	Price per each, Ruble	The sum, ruble
Computer	SONY	4	700	2100
Bar code reader	SIMENSE	4	100	400
Radiation Detector	NUCLIONIX	3	200	600
Printer	SAMSUNG	2	300	600

Total of materials				3700
Transportation and procurement expenses (2%)				74
Total items $C_м$				3774

Table 10 – The calculation of the basic salary

Performers by Category	Laboriousness, Number of people × Hours	Salary per one person × days, Ruble	Total salary under the tariff (wage), Ruble
Supervisor 1/Docent	1×30	200×30	6000
Supervisor 2/Scientific staff	1×28	100×28	2800
Total: 8800			

4. Social Insurance

Social insurance, public insurance program that provides protection against various economic risks (*e.g.*, loss of income due to sickness, old age, unemployment) and in which participation is compulsory. Social insurance is considered to be a type of social security, and in fact the two terms are sometimes used interchangeably. Usually it is calculated 30% of the labor wages.

5. Overhead Expenses

Overhead is an accounting term that refers to all ongoing business expenses not including or related to direct labor, direct materials or third-party expenses that are billed directly to customers (APPENDIX C). A company must pay overhead on an ongoing basis, regardless of whether the company is doing a high or low volume of business. It is important not just for budgeting purposes but for determining how much a company must charge for its products or services to make a profit. For example, a service-based business that operates in a traditional white-collar office setting has overhead expenses such as rent, utilities and insurance (table 12).

Table.11 - Estimation of overall expenses.

S.No	Type of expenses	Budget
1	Labor wages	2340
2	Raw materials and resources	4130
3	Social insurance	200
4	Over head	250
5	Total	6920

4.3 Conclusion

The process of completion of the project started with doing a systematic SWOT analysis the step by step analysis of strength weakness and opportunities and threats were critically analyzed and after performing these analysis the methodology how to overcome the weakness and threats and how to achieve the objective.

The payment for achieving the project is evaluated and total number of days required are found out and along with the expenses to carry out the project the overload expenses and social funding is also evaluated.

The dissertation was done in a period 632 days totally with the participation of master's student and the support of two supervisors. In which, the researcher, i.e. the student, who carried out this thesis spending 567 days for working on, the supervisor 1 is 30 days and supervisor 2 is 28 days.

The amount of money had been spent for doing this dissertation is 19494 Ruble totally. Particularly, which had to pay for the supervisor 1; 6000 Ruble is the wage, which had to pay for the supervisor 2. Overhead cost – 2800 Ruble.

In addition, there are some money also paid for some necessary equipment, which had been used for doing this research, such as computer, Bar code reader, printer, radiation detector. Totally, it costs 3774 Ruble for all.

5 Social responsibility

In modern conditions, one of the main directions of radical improvement of all preventive work to reduce occupational traumatism and occupational morbidity is the widespread implementation of an integrated OSH management system. That means combining isolated activities into a single system of targeted actions at all levels and stages of the production process. Occupational safety is a system of legislative, socio-economic, organizational, technological, hygienic and therapeutic and preventive measures and tools that ensure the safety, preservation of health and human performance in the work process [16].

Rules for labor protection and safety measures are introduced in order to prevent accidents, ensure safe working conditions for workers and are mandatory for workers, managers, engineers and technicians. A dangerous production factor is a manufacturing factor whose impact under certain conditions leads to trauma or other sudden, severe deterioration of health [16]. A harmful production factor is a production factor, the effect of which on a worker under certain conditions leads to a disease or a decrease in working capacity.

1.Analysis of hazardous and harmful production factors

The working conditions in the workplace are characterized by the presence of hazardous and harmful factors, which are classified by groups of elements: physical, chemical, biological, psycho physiological. The main elements of the production process that form dangerous and harmful factors are presented in Table 13

Table 13 - The main elements of the production process, forming hazardous and harmful factors

Table 12- Parameter of working process.

Name of the types of work and the parameters of the working process	FACTORS GOST 12.0.003-74 Occupational safety standards system		Normative documentation
	harmful	Dangerous	
Work with PC	Chemical Toxic		GOST 12.1.007-76 Occupational safety standards system. Harmful substances.
		Electricity	GOST 12.1.038-82 Occupational safety standards system. electrical safety
	The impact of radiation (HF, UHF, SHF, etc.)		SanPiN 2.2.2 / 2.4.1340-03 Sanitary-epidemiological rules and regulations. "Hygienic requirements for personal computers and organization of work"

The following factors effect on person working on a computer:

Physical:

– temperature and humidity;
– noise;
– static electricity;
– electromagnetic field of low purity;
– illumination;

– presence of radiation;

Human:

– physical overload (static, dynamic)

– mental stress (mental overstrain, monotony of work, emotional overload).

2.Organizational arrangements

All personnel are required to know and strictly observe the safety rules. The training of personnel in occupational safety and industrial sanitation consists of introductory briefing and briefing at the workplace by the responsible person.

The knowledge of safety rules is checked by the qualification commission or by the person responsible for the workplace after training at the workplace. After that, commission assign the qualification group on safety corresponding to the employee’s knowledge and experience of work and issue a special certificate.

Persons serving electrical installations must not have injuries and illnesses that interfere with production work. The state of health is established by medical examination before being employed.

3.Technical Activities

The rational layout of the workplace provides for a clear order and permanent placement of objects, means of labor and documentation. Object, what is required to perform the work more often, should be located in the easy reach of the workspace, as shown in Fig. 16

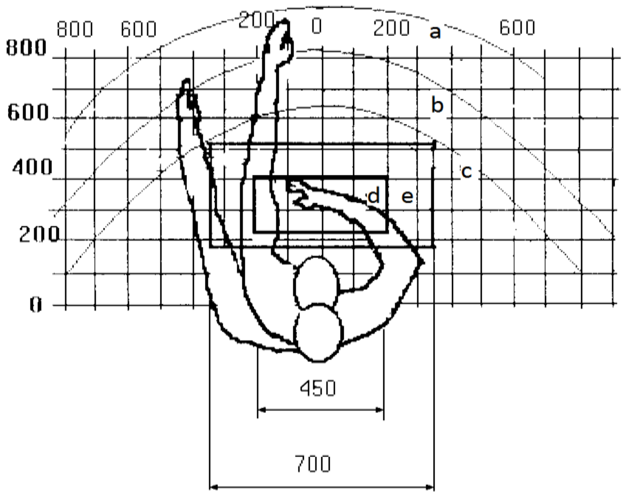

Figure 17 - Hand reach zones in the horizontal plane

a - Zone of maximum reach of hands;

b - Reach zone of fingers with outstretched arm;

c - Easy reach zone of the palm;

d. Optimum space for fine handmade work

e. The optimum space for rough manual work;

Optimal placement of objects of labor and documentation in the reach of hands: the display is located in zone a (in the center);keyboard - in the area of e / d; the system unit is located in zone b (on the left);the printer is in zone a (right);The documentation is placed in the easy reach of the palm - in (left) - literature and documentation necessary for work; In the drawers of the table - literature that is not used constantly. When designing a desk, the following requirements must be taken into account.

The height of the working surface of the table is recommended within 680-800 mm. The height of the working surface, on which the keyboard is installed, should be 650 mm. The working table must be at least 700 mm wide and at least 1400 mm long. There should be a legroom of not less than 600 mm in height, a width of at least 500 mm, a depth at the knee level of at least 450 mm and at the level of elongated legs - not less than 650 mm. The work chair must be lift able and adjustable in height and angle of inclination of the seat and backrest, as well as the distance of the backrest to the front edge of the seat. It is recommended that the height of the seat be above the floor level of 420 to 550 mm. The design of the working chair should ensure: the width and depth of the seat surface is not less than 400 mm; Seat surface with recessed front edge.

The monitor should be located at the eye level of the operator at a distance of 500 - 600 mm. According to the norms, the viewing angle in the horizontal plane should be no more than 45º to the normal of the screen. It is better if the viewing angle is 30 °. In addition, it should be possible to select the level of contrast and brightness of the image on the screen. It should be possible to adjust the screen: Height +3 cm; on a slope from 10 to 20 degrees with respect to the vertical; in the left and right directions. The keyboard should be placed on the surface of the table at a distance of 100 - 300 mm from the edge. The normal position of the keyboard is its placement at the elbow level of the operator with an angle of inclination to the horizontal plane of 15º. It is more convenient to work with keys that have a concave surface, a quadrangular shape with rounded corners. The key design should provide the operator with a click sensation. The color of the keys should contrast with the color of the panel. It is recommended to choose soft, low-contrast floral shades that do not disperse attention (low-saturated shades of cold green or blue colors) in the case of monotonous mental work requiring considerable nervous tension and great concentration. Shades of warm tones are recommended at work, which requires intense mental or physical tension, due to they excite human activity.

4.Safe work conditions

The main parameters characterizing the working conditions are microclimate, noise, vibration, electromagnetic field, radiation, illumination.

The air of the working area (microclimate) is determined by the following parameters: temperature, relative humidity, air speed. The optimum and permissible values of the microclimate characteristics are established in accordance with [17] and are given in Table 14.

Table 13- Optimal and permissible parameters of the microclimate

Period of the year	Temperature, ^{C}	Relative humidity,%	Speed of air movement, m / s
Cold and changing of seasons	23-25	40-60	0.1
Warm	23-25	40	0.1

The measures for improving the air environment in the production room include: the correct organization of ventilation and air conditioning, heating of room. Ventilation can be realized naturally and mechanically. In the room the following volumes of outside air must be injected: at least 30 m^3 per hour per person for the volume of the room up to 20 m^3 per person; natural ventilation is allowed for the volume of the room more than 40 m^3 per person and if there is no emission of harmful substances. The heating system must provide sufficient, constant and uniform heating of the air. Water heating should be used in rooms with increased requirements for clean air. The parameters of the microclimate in the laboratory regulated by the central heating system, have the following values: humidity 40%, air speed 0.1 m / s, summer temperature 20-25 ° C, in winter 13-15 ° C. Natural ventilation is provided in the laboratory. Air enters and leaves through the cracks, windows, doors. The main disadvantage of such ventilation is that the fresh air enters the room without preliminary cleaning and heating.

Noise and vibration worsen working conditions, have a harmful effect on the human body, namely, the organs of hearing and the whole body through the central nervous system. As a result, attention is weakened, memory deteriorates, the response decreases, and the number of errors in work increases. Noise can be generated by operating equipment, air conditioning units, daylight illuminating devices, as well as penetrating from the outside. When working on a PC, the noise level in the workplace should not exceed 50 dB.

The screen and system blocks produce electromagnetic radiation. Its main part comes from the system unit and the video cable. According to [2], the intensity of the electromagnetic field at a distance of 50 cm around the screen along the electrical component should be no more than:

- In the frequency range 5 Hz - 2 kHz - 25 V / m;
- In the frequency range 2 kHz - 400 kHz - 2.5 V / m.
- The magnetic flux density should be no more than:
- In the frequency range 5 Hz - 2 kHz - 250 nT;
- In the frequency range 2 kHz - 400 kHz - 25 nT.
- There are the following ways to protect against EMF:

Increase the distance from the source (the screen should be at least 50 cm from the user);

The use of pre-screen filters, special screens and other personal protective equipment. When working with a computer, the ionizing radiation source is a display. Under the influence of ionizing radiation in the body, there may be a violation of normal blood coagulability, an increase in the fragility of blood vessels, a decrease in immunity, etc. The dose of irradiation at a distance of 20 cm to the display is 50 μrem / hr. According to the norms [2], the design of the computer should provide the power of the exposure dose of x-rays at any point at a distance of 0.05 m from the screen no more than 100 μR / h. Fatigue of the organs of vision can be associated with both insufficient illumination and excessive illumination, as well as with the wrong direction of light.

5.Electrical safety

Depending on the conditions in the room, the risk of electric shock to a person increases or decreases. Do not operate the computer in conditions of high humidity (relative air humidity exceeds 75% for a long time), high temperature (more than 35 ° C), the presence of conductive dust, conductive floors and the possibility of simultaneous contact with metal components connected to the ground and the metal casing of electrical equipment. The computer operator works with electrical devices: a computer (display, system unit, etc.) and peripheral devices. There is a risk of electric shock in the following cases: with direct contact with current-carrying parts during computer repair; when touched by non-live parts that are under voltage (in case of violation of insulation of current-carrying parts of the computer);when touched with the floor, walls that are under voltage; short-circuited in high-voltage units: power supply and display unit. Measures to ensure the electrical safety of electrical installations: disconnection of voltage from live parts, on which or near to which work will be carried out, and taking measures to ensure the impossibility of applying voltage to the workplace; posting of posters indicating the place of work; electrical grounding of the housings of all installations through a neutral wire; coating of metal surfaces of tools with reliable insulation; inaccessibility of current-carrying parts of equipment (the conclusion in the case of electro orating elements, the conclusion in the body of current-carrying parts) [18].

6.. Fire and explosive safety

According to [18], depending on the characteristics of the substances used in the production and their quantity, for fire and explosion hazard, the premises are divided into categories A, B, C, D, E. The room belongs to category B according to the degree of fire and explosion hazard. It is necessary to provide a number of preventive measures.

Possible causes of fire: malfunction of current-carrying parts of installations; work with open electrical equipment; short circuits in the power supply; on-compliance with fire safety regulations; presence of combustible components: documents, doors, tables, cable insulation, etc. Activities on fire prevention are divided into: organizational, technical, operational and regime. Organizational measures provide for correct operation of equipment, proper maintenance of buildings and territories, fire instruction for workers and employees, training of production personnel for fire safety rules, issuing instructions, posters, the existence of an evacuation plan. The technical measures include: compliance with fire regulations, norms for the design of buildings, the installation of electrical wires and equipment, heating, ventilation, lighting, the correct placement of equipment. The regime measures include the establishment of rules for the organization of work, and compliance with fire-fighting measures. To prevent fire from short circuits, overloads, etc., the following fire safety rules must be observed: elimination of the formation of a flammable environment (sealing equipment, control of the air, working and emergency ventilation);use in the construction and decoration of buildings of non-combustible or difficultly combustible materials; the correct operation of the equipment (proper inclusion of equipment in the electrical supply network, monitoring of heating equipment);correct maintenance of buildings and territories (exclusion of the source of ignition - prevention of spontaneous combustion of substances, restriction of fireworks);training of production personnel in fire safety rules;

The publication of instructions, posters, the existence of an evacuation plan; compliance with fire regulations, norms in the design of buildings, in the organization of electrical wires and equipment, heating, ventilation, lighting; the correct placement of equipment; well-time preventive inspection, repair and testing of equipment.

In the case of an emergency, it is necessary to:

- Inform the management (duty officer);
- Call the Emergency Service or the Ministry of Emergency Situations tel. 112;
- Take measures to eliminate the accident in accordance with the instructions.

Conclusion

This dissertation is carried out by step by step systematic analysis of the objectives. To some extent, the current framework of fuel safety criteria remains applicable, being largely unaffected by the “new” or modern design changes; the numeric values of the individual safety criteria may, however, change in accordance with the particular fuel and core design features. Some of these values have already been or are continuously being adjusted.

Assessment of fuel safety criteria, the following process is recommended: Continue to further develop best-estimate analysis methods, together with a suitable uncertainty analysis, in all areas of safety analysis. Continue to perform experimental studies for benchmarking of best-estimate codes and extending the verification validation basis for safety criteria and the codes Review, and adjust or change where necessary, safety criteria based on the above codes and test data; define or quantify necessary margin to safety limits.

Finally, if the source is too intensive and time or distance do not provide sufficient radiation protection, the shielding must be used. Radiation shielding usually consist of barriers of lead, concrete or water. There are many many materials, which can be used for radiation shielding, but there are many many situations in radiation protection. It highly depends on the type of radiation to be shielded, its energy and many other parametres. For example, even depleted uranium can be used as a good protection from gamma radiation, but on the other hand uranium is absolutely inappropriate shielding of neutron radiation.After calculating and comparing between the results it was shown that Lead has excellent shielding properties against gamma rays.

The process of completion of the project started with doing a systematic SWOT analysis the step by step analysis of strength weakness and opportunities and threats were critically analyzed and after performing these analysis the methodology how to overcome the weakness and threats and how to achieve the objective.

The payment for achieving the project is evaluated and total number of days required are found out and along with the expenses to carry out the project the overload expenses and social funding is also evaluated.

The dissertation was done in a period 632 days totally with the participation of master's student and the support of two supervisors. In which, the researcher, i.e. the student, who carried out this thesis spending 567 days for working on, the supervisor 1 is 30 days and supervisor 2 is 28 days.

The amount of money had been spent for doing this dissertation is 19494 Ruble totally. Particularly, which had to pay for the supervisor 1; 6000 Ruble is the wage, which had to pay for the supervisor 2. Overhead cost – 2800 Ruble.

In addition, there are some money also paid for some necessary equipment, which had been used for doing this research, such as computer, Bar code reader, printer, radiation detector. Totally, it costs 3774 Ruble for all.

The final part of the dissertation is social responsibility the project has to be done as per the rules and regulation in-order to achieve that it is necessary to produce a systematic analysis of the safety factors the numerous amount of threat is involved in achieve the goal of the project fire safety radiation safety electrical safety in order to achieve the success of the project the people who are involved in the project as to trained properly and so ultimate safety is achieved and among this is most important is radiation safety the personnel's who are involved in the project as to be shielded from radiation proper radiation shielding methods as to be used regarding the fire safety all inflammable materials should be kept away from the fire reachable area and thus fulfilling all these safety measure the project can be executed safely and smoothly thus allowing us to perform all necessary safety precaution during course of the project and regarding the electrical safety proper voltage as to maintained properly and smooth function of the project as to be ensured.

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Appendix A

Appendix B

Table B – Lists the casks that were NRC-certified for the transportation of irradiated commercial light-water power reactor fuel assemblies.

Cask	Package ID	Canister	Contents (Number of assemblies)	Type
IF-300	USA/9001/B( )F	No	7 PWR, 17 BWR	Rail
NLI-1/2	USA/9010/B( )F	No	1 PWR, 2 BWR	Truck
TN-8	USA/9015/B( )F	No	3 PWR	Over weight
TN-9	USA/9016/B( )F	No	7 BWR	Over weight
NLI-10/24	USA/9023/B( )F	No	10 PWR, 24 BWR	Rail
NAC-LWT	USA/9225/B(U)F-96	No	1 PWR, 2 BWR	Truck
GA-4	USA/9226/B(U)F-85	No	4 PWR	Truck
NAC-STC	USA/9235/B(U)F-85	Both	26 PWR	Rail
NUHOMS®-MP187	USA/9255/B(U)F-85	Yes	24 PWR	Rail
HI-STAR 100	USA/9261/B(U)F-85	Yes	24 PWR, 68 BWR	Rail
NAC-UMS	USA/9270/B(U)F-85	Yes	24 PWR, 56 BWR	Rail
TS125	USA/9276/B(U)F-85	Yes	21 PWR, 64 BWR	Rail
TN-68	USA/9293/B(U)F-85	No	68 BWR	Rail
NUHOMS®-MP197	USA/9302/B(U)F-85	Yes	61 BWR	Rail

Appendix C

(Obligatory)

The planned schedule of research and development (R&D) by item

Table A.1 – The planned schedule of research and development

Code of work (from ИСР)	Type of work	Performer	$T_{к}$, sche, days.	Durationofwork													
				Feb		March			April			May			June		
				2	3	1	2	3	1	2	3	1	2	3	1	2	3
1	Selection of research topic	Supervisor 1	4														
2	Study of scientific literatures	LIJJU GEORGE	120														
3	Practice	LIJJU GEORGE	100														
4	Searching material and study in library	LIJJU GEORGE	90														
5	Data processing	LIJJU GEORGE	50														
6	Split the general topic into detail sub-	Supervisor 1 and 2	10														

	topics																
7	Writing papers	LIJJU GEORGE	50														
8	Check the correction of paper	Supervisor 1 and 2	20														
9	Team working	LIJU GEORGE	2														

In which:

	Supervisor 1
	Supervisor 2
	LIJU GEORGE