PROLIFERATION RESISTANCE ASSESSMENT OF VARIOUS METHODS OF

SPENT NUCLEAR FUEL STORAGE AND DISPOSAL

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TABLE OF CONTENTS

Page

LIST OF TABLES

LIST OF FIGURES

NOMENCLATURE

BWR	Boiling water reactor
C/S	Containment and surveillance
CANDU	CANada Deuterium Uranium reactor
CoK	Continuity of knowledge
DOE	Department of Energy (United States)
FQ	Formula quantity
GW	Giga-watt
HEU	High-enriched uranium
HM	Heavy metal
IAEA	International Atomic Energy Agency
LEU	Low-enriched uranium
LWR	Light water reactor
MAUA	Multi-Attribute Utility Analysis
MOX	Mixed-oxide fuel
MPC	Multi-purpose canister (in dry cask storage)
MRS	Monitored retrievable storage
MT	Mega-ton (metric)
MW	Mega-watt
NDA	Non-destructive assay
NEI	Nuclear Energy Institute (United States)
NNSA	National Nuclear Security Administration (United States)
NNWS	Non-nuclear weapons state
NPP	Nuclear power plant
NPT	Treaty on the Non-Proliferation of Nuclear Weapons
NRC	Nuclear Regulatory Commission (United States)
NWS	Nuclear weapons state
PR	Proliferation resistance
Pu	Plutonium
PWR	Pressurized water reactor
SB	Ultrasonic Sealing Bolt
SQ	Significant quantity
tHM	Metric tons of heavy metal
U	Uranium

ABSTRACT

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Many countries are planning to build or already are building new nuclear power plants to match their growing energy needs. Since all nuclear power plants handle nuclear materials that could potentially be converted and used for nuclear weapons, they each present a nuclear proliferation risk. Spent nuclear fuel presents the largest build-up of nuclear material at a power plant. This is a proliferation risk because spent fuel contains plutonium that can be chemically separated and used for a nuclear weapon. The International Atomic Energy Agency (IAEA) safeguards spent fuel in all non-nuclear weapons states that are party to the Non-Proliferation Treaty. Various safeguards methods are in use at nuclear power plants and research is underway to develop safeguards methods for spent fuel in centralized storage or underground storage and disposal. Each method of spent fuel storage presents different proliferation risks due to the nature of the storage method and the safeguards techniques that are utilized. Previous proliferation resistance and proliferation risk assessments have mainly compared nuclear material through the whole fuel cycle and not specifically focused on spent fuel storage. This project evaluates the proliferation resistance of the three main types of spent fuel storage: spent fuel pool, dry cask storage, and geological repository.

The proliferation resistance assessment methodology that is used in this project is adopted from previous work and altered to be applicable to spent fuel storage. The assessment methodology utilizes various intrinsic and extrinsic proliferation-resistant attributes for each spent fuel storage type. These attributes are used to calculate a total proliferation resistant (PR) value. The maximum PR value is 1.00 and a greater number means that the facility is more proliferation resistant. Current data for spent fuel storage in the United States and around the world was collected. The PR values obtained from this data are 0.49 for the spent fuel pool, 0.42 for dry cask storage, 0.36 for the operating geological repository, and 0.28 for the closed geological repository. Therefore, the spent fuel pool is currently the most proliferation resistant method for storing spent fuel.

The extrinsic attributes, mainly involving safeguards measures, affect the total PR value the most. As a result, several recommendations are made to improve the proliferation resistance of spent fuel. These recommendations include employing more advanced safeguards measures, such as verification techniques and remote monitoring, for dry cask storage and the geological repository. Dry cask storage facilities should also be located at the plant and in a secure building to minimize the proliferation risk. Finally, the cost-benefit analysis of increased safeguards needs to be considered. Taking these recommendations into account, the PR values of dry cask storage and the closed geological would be significantly increased, to 0.57 and 0.51, respectively. As a result, with increased safeguards to the safeguards level of the spent fuel pool, dry cask storage would be the most proliferation resistant method to store spent fuel. Therefore, the IAEA should continue to develop remote monitoring and cask storage verification techniques in order to improve the proliferation resistance of spent fuel.

INTRODUCTION

Many countries are planning to build or already are building new nuclear power plants to match their growing energy needs. Since all nuclear power plants handle nuclear materials that could potentially be converted and used for nuclear weapons, they each present a nuclear proliferation risk, especially when built in countries that do not already have nuclear weapons. Special nuclear material requiring safeguards arrives at a nuclear power plant in the form of fresh fuel. It is then burned by fission in the reactor and removed as spent nuclear fuel. Therefore, the largest build-up of nuclear material at a power plant is the spent fuel, which is usually stored in pools inside of the plant or in casks outside of the plant. If a country does not have other nuclear fuel facilities, then spent fuel presents the greatest proliferation risk. Even though spent fuel is extremely radioactive and hard to handle, it contains uranium and plutonium that could be separated out and used in a clandestine nuclear weapons program.

The International Atomic Energy Agency (IAEA) institutes safeguards on most of the special nuclear material in each country that has signed and put into force the Treaty on the Non-Proliferation of Nuclear Weapons. Spent fuel at plants is safeguarded using various methods, and research is underway to develop safeguards methods for spent fuel in centralized or underground storage and disposal. Spent fuel is first stored in the spent fuel pool and then placed in dry cask storage or even a geological repository for permanent disposal.

Each method of spent fuel storage presents different proliferation risks due to the nature of the storage method and the safeguards techniques that are utilized. Previous proliferation resistance/proliferation risk assessments have mainly compared nuclear material through the whole fuel cycle and not specifically focused on spent fuel. This research project will evaluate different methods of spent fuel storage in terms of proliferation resistance, taking intrinsic and extrinsic features into account. The goal of this project is quantify the proliferation resistance of current methods of storing spent fuel and then make recommendations to increase the proliferation resistance.

The first step in this project is to define the spent fuel storage types to be analyzed. Afterwards, existing proliferation-risk evaluation methods will be reviewed and an appropriate method will be identified and modified as necessary. The proliferation-resistant characteristics to be used in the assessment of spent fuel storage will be identified and then used to evaluate the proliferation resistance. From this assessment, the most proliferation-resistant characteristics of spent fuel storage will be discussed. Finally, suggestions for possible ways to make spent fuel storage systems more proliferation resistant will be made.

LITERATURE REVIEW

After an introduction to safeguards and the nonproliferation regime, the various spent fuel storage methods that are examined in this project are presented. To provide a worldwide perspective, the nuclear waste management policies are then summarized for major weapons, and non-weapons states. Since proliferation resistance is mainly increased by safeguards measures, both existing and developing safeguards methods for spent fuel are analyzed. Finally, in order to develop a quantitative proliferation resistance assessment methodology, current and past methodologies are examined.

I. Background

Increasing energy demands have caused many countries to pursue nuclear power because of the large-scale electricity output that it can provide. The World Nuclear Association estimates that nuclear power capacity will increase from the current worldwide capacity of 367 GW to anywhere from 602 to 1350 GW by 2030 (World Nuclear Association 2011). This means that nuclear power capacity will double or even triple within the next 20 years. Each new power plant presents proliferation risk as it adds special nuclear material to the fuel cycle. Both existing and newly-built nuclear power plants will add more and more special nuclear material to the fuel cycle each year.

In order to prevent the diversion of nuclear material to a weapons program, the International Atomic Energy Agency (IAEA) employs safeguards on most of the nuclear material in non-nuclear weapons states (NNWS), as designated by the Treaty on the Non-Proliferation of Nuclear Weapons (NPT). Each NNWS has a specific agreement with the IAEA in regards to safeguarding facilities and material. The goal of IAEA safeguards is to detect and hopefully prevent the diversion of significant quantities of nuclear material from civilian facilities to a nuclear weapons program. International safeguards thus reduces the proliferation risk of a nuclear power program (Doyle 2008).

Nuclear material, or more specifically uranium, is processed at various facilities before it arrives at the plant in the form of fresh fuel. Fuel is removed from the reactor as spent fuel and contains uranium and plutonium, both of which may be diverted to a weapons program. Plutonium in spent fuel has particularly created a proliferation concern because, unlike uranium in spent fuel, it does not necessarily need to be enriched to be used for a nuclear weapon. Since most nuclear material in the fuel cycle eventually ends up as fuel in a power reactor, spent fuel in storage may present the largest build-up of civilian nuclear materials. Spent fuel may also be reprocessed to be re-used for fuel, but this method is only used in one NNWS; Japan.

The IAEA measures the amount of special nuclear material in terms of significant quantities (SQs), or the estimated amount of that material theoretically needed to create a nuclear weapon. An SQ of plutonium is 8 kg, and an SQ of low-enriched uranium is 75 kg, measured in terms of the contained U-235 (Doyle 2008).

As of the latest IAEA Annual Report, there are 172,180 SQs in 674 facilities under safeguards worldwide. Of these facilities, 235 are power reactors. There are

132,505 SQs of plutonium contained in safeguarded reactor core fuel and spent fuel, over 75% of the total SQs under safeguards (International Atomic Energy Agency 2010). Furthermore, it is estimated that there are currently over 162,500 SQ of plutonium in spent fuel worldwide (including spent fuel not under safeguards) and that this number is expected to increase to 677,500 by 2050 with the growth of nuclear power (Fukuda et al. 2008). The accumulation of spent fuel presents the largest (in terms of the amount of material) proliferation risk in the civilian nuclear fuel cycle.

Spent fuel in storage will continue to increase the proliferation risk of a nuclear power program as existing plants burn fuel and new plants are introduced into the fleet. A typical 1000 MWe light water reactor (LWR) uses approximately 25 tons of uranium fuel per year (World Information Service on Energy 2009). Of the uranium fuel burned in the reactor, 93% remains U-238, 1% remains U-235, 5% is converted to fission products, and 1% of the converted plutonium remains (World Nuclear Association 2009). This means that a power plant of 1 GWe capacity outputs about 250 kg each of U-235 and Pu in spent fuel. This equates to about 30 SQ of plutonium and 3 SQ of uranium per year.

Due to the delay in making decisions regarding spent fuel management policy in many countries, temporary storage on or off the plan site has become the primary storage method for spent fuel. Storage for spent fuel must be provided either at the reactor or away from the reactor in spent fuel pools or dry storage. Spent fuel storage is considered an item facility in terms of IAEA safeguards because the nuclear material is counted as either an assembly or a cask. The safeguards procedures for items include counting, identification, examination of integrity, non-destructive measurements, and surveillance. However, in the case of dry storage, while casks may be counted themselves, it is difficult to verify the material in the casks because the integrity of the casks should not be compromised. Research and development is still underway to determine more conclusive methods to verify spent fuel in dry storage rather than just using containment and surveillance (Pushkarjov 1986).

While the counting and surveillance of spent fuel assemblies and dry casks is straightforward, research programs continue to develop methods to verify that objects in the casks are indeed spent fuel assemblies. Verification needs to be a part of the safeguards process in order to ensure that no spent fuel assemblies, or even parts of assemblies, have been diverted. Many nuclear power plants were not designed with consideration of international safeguards. Therefore, IAEA inspectors must spend many days in the field and handle various equipment to verify the presence of spent fuel (Doyle 2008). In addition, policies and methods to safeguard the permanent disposal of spent fuel in a geological repository are being investigated. The following section will describe, in more detail, the various methods for storing spent fuel.

II. Spent Fuel Storage Types

Countries around the world have adopted different ways of storing and disposing of spent fuel. These methods mainly include on-site spent fuel pools, on-site dry cask storage, national centralized interim storage, and geological repository. The geological repository is actually a method of disposal of spent fuel. However, in order to aid in comparison, the geological repository will be evaluated as a storage site for spent fuel. The following sections describe each of these general methods of storing spent fuel.

1. On-Site Spent Fuel Pools

A typical light water reactor in the United States discharges and refuels about one-fourth to one-third of the fuel in the core every 12 to 18 months. The spent fuel is then transferred to a temporary wet storage pond. The U.S. Nuclear Regulatory Commission (NRC) requires that spent fuel in the pool be under at least 20 feet of water to allow for adequate radiation shielding (Nuclear Regulatory Commission 2007). Spent fuel pools vary greatly in size both in the U.S. and around the world. The capacity of each spent fuel pool depends on the size which depends on the overall spent fuel management policy at the time the plant was built. Spent fuel pools are either within the reactor building or in a building adjacent to the reactor. The pool is connected to the reactor by a fuel transport canal, and the fuel is usually transported by cranes. Depending on the type of reactor and fuel, the fuel assemblies may sit in baskets or casks in the pool. Table 1 shows the number of spent fuel pools versus their capacities and inventories in tons of heavy metal (t HM) for LWR plants in various countries. LWR plants mainly include pressurized water reactors (PWR) and boiling water reactors (BWR). This data proves that the average capacity per pool varies significantly from country to country (International Atomic Energy Agency 1999).

Table 1: LWR spent fuel pool data for various countries (International Atomic Energy Agency 1999)

Country	**Type of Reactors**	**Number of Pools**	**Capacity (t HM)**	**Inventory (t HM)**	**Average Capacity per Pool (t HM)**	**% Capacity Full**
France	900 MW PWR	34	5870	4187	173	71%
	1300 MW PWR	20	5420	1608	271	30%
Germany	PWR	13	3176	2011	244	63%
	BWR	6	1385	821	231	59%
Japan	PWR	20	6460	2070	323	32%
	BWR	23	8410	3050	366	36%
South Africa	PWR	2	670	396	335	59%
Spain	PWR/BWR	9	3820	2000	424	52%
Sweden	PWR/BWR	12	1500	730	125	49%
Switzerland	PWR/BWR	5	705	150	141	21%
United States	PWR/BWR	110	59000	38343	536	65%

2. On-Site Dry Cask Storage

Spent fuel pools were originally meant to be temporary storage. However, spent fuel management policies have changed since the construction of many power plants. This has caused many spent fuel pools to come near capacity, as seen in Table 1. Spent fuel pools are able to be re-racked to increase capacity but eventually, interim dry storage may be needed. Dry cask storage is a form of interim dry storage in which the spent fuel is placed in casks either directly outside of the plant or nearby. Several spent fuel assemblies are placed in a sealed metal container with a metal or concrete outer casing to shield the radiation. Casks can be stored upright, usually on a concrete pad, or vertically, in a concrete structure. In addition, the fuel assemblies must have cooled for at least 5 years in the spent fuel pool before they are ready for dry storage (Nuclear Regulatory Commission 2008).

Many other countries, besides the United States, have utilized dry cask storage for spent fuel. Table 2 shows the number of facilities, capacity, and inventory of dry cask storage facilities around the world. Countries that reprocess spent fuel, like France and Japan, have much less dry cask storage capacity than countries that do not reprocess, like Canada and the United States. While the principle of most dry casks is the same, there are many different types of dry cask storage available, as show in Table 3. Older designs hold as few as four PWR assemblies while newer designs may hold as many as 33 PWR assemblies or 61 BWR assemblies. Most of these systems are vertically oriented, with the exception of the NUHOMS storage system, which is horizontal. All of the casks can be loaded in the spent fuel pool and have specific limitations regarding the fuel burnup and cooling time. Many of the casks listed in Table 3 also utilize a multi-purpose canister (MPC). The MPC is the inner metal canister of the cask that can be used for storage in a concrete cask and then be placed in a transport cask and transported elsewhere, by truck or train, for example.

Table 2: Dry cask storage facilities in various countries (International Atomic Energy Agency 1999)

State	**Number of Facilities**	**Design Capacity (t HM)**	**Current Inventory (t HM)**	**Average Capacity per Pool (t HM)**	**% Capacity Full**
Argentina	1	200	64	200	32%
Belgium	1	800	142	800	18%
Canada	8	23067	1930	2883	8%
Czech Republic	1	600	232	600	39%
France	1	180	180	180	100%
Germany	4	8353	58	2088	1%
Hungary	1	162	54	162	33%
Japan	1	73	73	73	100%
Republic of Korea	2	1421	609	711	43%
Lithuania	1	419	0	419	0%
Ukraine	1	50	0	50	0%
United Kingdom	1	958	680	958	71%
United States	16	6855	1270	428	19%

Table 3: Technical specifications for various dry cask types (International Atomic Energy Agency 1999)

Type	Model	Fuel Type	Assembly Capacity	Loaded Weight (t)
CASTOR	la	PWR	4	81
	Ib	PWR	4	65
	Ic	BWR	16	88
	na	PWR	9	121
	nb	PWR	8	85
	V/19	PWR	19	121
	V/21	PWR	21	106
	V/52	BWR	61	121
	X/28F	PWR	28	104
	X/33	PWR	33	106
HOLTEC	MPC-32	PWR	32	unkown
	MPC-24	PWR	24	unkown
	MPC-68	BWR	68	unkown
NAC	STC	PWR	26	113
	I28	PWR	28	113
Sierra Nuclear	VSC-24	PWR	24	unkown
	TranStor	PWR	24	unkown
	TranStor	BWR	61	unkown
TN24	Version 1	PWR	24	93
	Version 2	BWR	52	95
	Version 3	PWR	28	114
	Version 4	PWR	32	115
	Version 5	PWR	24	114
	Version 6	PWR	unkown	125
	Version 7	BWR	unkown	125
NUHOMS (horizontal)	07P	PWR	7	unkown
	24P	PWR	24	unkown
	52B	BWR	52	unkown
Westinghouse	MC-10	PWR	24	103

3. Transfer from Spent Fuel Pool to Dry Cask Storage

An important aspect to consider when analyzing dry cask storage is that the spent fuel must be transferred from the pool to the cask. This process may affect the proliferation resistance assessment of dry cask storage. The following steps are taken in a normal transfer of spent fuel from the pool to the canister:

i. Lower canister into loading area in pool by crane
ii. Load spent fuel into canister using crane underwater
iii. Place plug on canister with crane
iv. Remove canister from spent fuel pool by crane
v. Drain canister and fill with inert gas
vi. Weld top plug to canister

While this process can vary among different plants, these steps provide a very general explanation of how the canister is loaded with spent fuel (Office of Civilian Radioactive Waste Management 1994). Once the canister is secure, it is loaded onto the transfer vehicle that moves the canister to the storage area. The canister is then placed in the vertical concrete cask or horizontal concrete structure.

As described previously, the MPC in the concrete cask may be used to transport the spent fuel from a reactor site to a centralized storage site. In the next sections, various off-site spent fuel storage methods are reviewed.

4. National Centralized Interim Storage

Centralized interim storage may be used instead of or in addition to onsite dry storage. This type of storage is usually in the form of a monitored retrievable storage

(MRS) facility. An MRS facility is meant to be used for the temporary storage of spent fuel. The facility may also prepare the spent fuel for final disposal or serve as a central receiving station for nuclear waste. An MRS facility, or series of facilities, may provide an alternative to onsite dry storage for spent fuel. An MRS facility utilizes passive dry storage and thus does not require external power for cooling. The following are examples of some MRS facility designs proposed by the U.S. Department of Energy (Saling and Fentiman 2002):

- *Field Drywell:* Spent fuel is buried in casks underground in an enclosed field where the decay heat is transferred to the surrounding soil and then to the atmosphere.
- *Concrete Cask:* Spent fuel is stored in concrete casks above ground being directly cooled by the atmosphere. The concrete cask is designed to attenuate radiation, thus drastically reducing the radiation dose outside of the cask. This method is the general concept for the previously described onsite dry cask storage.
- *Open Cycle Vault:* Consists of a large, shielded warehouse facility which uses a crane to move the casks containing the spent fuel. Air ducts bring air from the atmosphere to remove the decay heat, which is then released through large ventilation stacks.
- *Closed Cycle Vault:* Follows the same warehouse-type design used for an open cycle vault, but an intermediate cycle of fluid is used to remove the decay heat, thus separating the storage casks from the environment. Large concrete storage modules hold several storage casks and the intermediate fluid flows around the

modules. Heat is removed from the intermediate fluid through a series of pipes exposed to atmospheric air.

- *Concrete Cask in Trench:* Combines the field drywell and concrete cask concepts. The concrete cask is buried and backfilled so the plug of the cask is at ground level.
- *Tunnel Drywell:* Facility uses dry, sealed containers in a mined tunnel that is located near the surface but is well above the underground water table. The major components of the facility follow the design of the field drywell concept.
- *Tunnel Rack Vault:* Uses natural circulation as in the open cycle vault. However, the facility is moved underground to a series of tunnels. All movement of the storage casks is done remotely because the casks are unshielded making the inside of the tunnel unreachable. The decay heat is removed through a series of ventilation tunnels.

Centralized interim storage may utilize any of the aforementioned concepts or variations/combinations of the designs. A state may choose to have a central interim storage facility in order to consolidate on-site dry cask storage sites at numerous plants. A single site may allow for better surveillance and security but may also be more attractive for a terrorist attack, for example. While the central storage facility may be further away from populated areas, safety risks are also introduced with transportation from the plant to the facility. There are many positives and negatives that a state must consider when deciding between on-site or centralized interim dry spent fuel storage (Petroski 2005). Many nations have chosen on-site dry cask storage as interim storage for spent fuel. Few nations are using centralized national storage, as outlined later in this chapter.

5. Geological Repository

For countries that do not plan to reprocess their spent fuel, a deep geological repository remains the preferred option for the final disposal of most high level waste forms. A geological repository is first sited on a stable landform, preferably in an unpopulated area. There should be no major groundwater flow in this area. Tunnels or caverns are built into the landform at depths between 250 and 1000 meters. A receiving and handling facility is usually located outside of the entrance tunnels. Spent fuel, and other waste forms, are packaged into metal casks and sent down the tunnels to their final location. There can be several engineered barriers to keep the radioactive waste from entering the accessible environment, including the waste form itself, waste package, tunnel, and surrounding rock. These barriers help prevent the leaching of the radioactive materials into the environment. While some countries around the world are in the process of citing or licensing a deep geological repository, none have been officially opened (World Nuclear Association 2009).

6. International Centralized Interim Storage

The concept of an international fuel cycle has been proposed in order to promote nuclear energy expansion while reducing proliferation risk. This model utilizes reliable fuel cycle service arrangements to ensure that states get fuel without the need to develop enrichment, fabrication, and reprocessing technologies. This may also involve creating international fuel cycles facilities with multinational investment and operation, while being safeguarded by the IAEA. This model reduces the need for countries without fuel cycle capabilities to develop the infrastructure (Mathews, Kessler, and Elkhamri 2006).

An international fuel cycle would most likely require countries which borrow nuclear fuel to return it when it is used. Whether the country of origin decides to reprocess the returned spent fuel or not, some variation of an international spent fuel storage facility may arise. This facility may be for interim storage before reprocessing or a geological repository for final disposal.

Besides in the case of an international fuel cycle, countries may also need to invest in a multinational spent fuel storage site due to a lack of resources to build their own site. A country with unfavorable geological characteristics or simply a small land mass could find it difficult to site a national interim storage facility or geological repository. More specifically, a country with a small nuclear power infrastructure could find it uneconomical to have their own repository. While multinational repositories and interim storage sites have been discussed, it is still illegal in many countries to accept nuclear waste from other countries (World Nuclear Association 2009).

7. Alternative Spent Fuel Disposal Methods

While the aforementioned spent fuel storage and disposal methods are the most widely used, some other methods have been discussed in the past and even implemented. Long-term above ground storage was investigated by several countries, including the United States, but is not currently planned anywhere. However, some dry cask storage or centralized storage facilities may inevitably become long-term since policymaking regarding the disposal of high level waste is slow. These above-ground packages would have to be replaced about every 200 years. Permanent structures above ground could also be built to house spent fuel for tens of thousands of years. The advantage to above-ground storage over a deep repository is that the fuel could be retrieved more easily and possibly reprocessed and re-used (World Nuclear Association 2009).

Some other alternative high level waste disposal methods have also been proposed. The idea of disposal in outer space proposed for highly-concentrated waste was abandoned due to the high cost and potential risks of launch failure. Sea disposal was implemented by various countries in the past, primarily for low level radioactive waste, but is no longer permitted by international agreements. Subduction zone and sub seabed disposal was not implemented by any country and is also not permitted by international agreements. Disposal in ice sheets was proposed for wastes that are heat-generating but was rejected by countries that have signed the Antarctic Treaty (World Nuclear Association 2009).

III. Nuclear Waste Management Policies around the World

While every nation has its own radioactive waste management policy, in general, the three spent fuel policies are direct disposal, reprocessing, and export. Of the 32 states with operating nuclear power plants in 2009, thirteen followed a direct disposal policy, six followed the reprocessing policy, six exported spent fuel, and the remaining states were undecided. Military ambitions, namely developing a nuclear weapons program, have driven many states to reprocess spent fuel in order to extract plutonium to be used for weapons. The United States' declaration against reprocessing in 1977 for nonproliferation reasons drove many other states to not reprocess (Högselius 2009).

Once large-scale nuclear power plants started being built in the 1960s, many countries had ambitions of internally developing the infrastructure for the entire fuel cycle, including reprocessing and fast breeder reactors. However, these fast breeder reactors have yet to be fully developed, and thus many states, especially smaller states, have turned away from reprocessing for technological reasons (Högselius 2009).

The political culture and civil society have shaped nuclear energy policy as a whole. Countries with authoritarian structure have been able to sustain reprocessing-oriented spent fuel policy more than democratic nations where anti-nuclear groups have had much influence. This is particularly true after the Three-Mile Island and Chernobyl accidents and with the rising fear of nuclear proliferation (Högselius 2009).

Geological conditions have also shaped spent fuel policy in a few cases. For example, Japan had turned to reprocessing due it its highly unstable geology and inability therefore to have a safe deep geologic repository.[1] Energy policy as a whole has shaped spent fuel management policy because states with thriving nuclear industries have generally been more optimistic about reprocessing to reduce total spent fuel and high-level waste inventory (Högselius 2009).

The past can be used to predict the future of spent fuel management policy. For example, changing energy policies due to the environmental movement have made nuclear power become more favorable. Also, new technologies, such as pyroprocessing, may make reprocessing more favorable in some countries (Högselius 2009). The following sections will examine the spent fuel management policies of some major nuclear and non-nuclear weapons states and then outline and summarize the current policies in all 32 nuclear power states.

1. Spent Fuel Management in Nuclear Weapons States

While the IAEA only safeguards a small fraction of spent fuel storage in nuclear weapons states, it is still important to explore spent fuel management policies in those states because they may shape non-nuclear weapons states' policies. The weapons states, China, France, Russia, the United Kingdom, and the United States, have generally been leaders in the nuclear power industry and also in spent fuel management. As mentioned before, many NNWS followed the United States' decision to ban reprocessing in 1977. Still, some countries plan to follow France's policy of reprocessing and reusing all fuel.

[1] Japan's nuclear energy policy is currently experiencing changes due to the accidents at the Fukushima Daiichi nuclear power plant in March 2011.

The Nuclear Waste Policy Act of 1982 gave the U.S. Department of Energy (DOE) the responsibility of disposing of civilian spent nuclear fuel. In 1987, the Act was amended to designate Yucca Mountain, Nevada, as the country's first deep geological waste repository. The DOE submitted the license application for the proposed Yucca Mountain repository in 2008. However, President Barack Obama's administration withdrew the license application for the Yucca Mountain project in 2009 (World Nuclear Association 2009). The DOE has since appointed a commission to explore the options for managing the back end of the fuel cycle and make recommendations for the storage, processing, and disposal of spent fuel (NEI 2010).

France has a very ambitious nuclear power program with 59 power plants generating 76% of the country's total electricity. All spent fuel is placed in interim storage before some of it is reprocessed into new fuel. In order to dispose of the remaining radioactive waste and possible spent fuel in the future, the French government plans to license a deep geological repository in 2015 and have it operational by 2025. The United Kingdom, Russia, and China all have policies of reprocessing spent fuel, but it is unclear whether all fuel will be reprocessed or whether part of it will be directly disposed of (World Nuclear Association 2009), (Högselius 2009).

2. Spent Fuel Management in Selected Non-Nuclear Weapons States

While most NNWS plan to directly dispose of their spent fuel after a period of interim storage, a geological repository has yet to be built anywhere in the world. Canada, Finland, Germany, and Sweden all plan to license geological repositories soon, and those repositories are expected to be operational in the 2020 to 2035 timeframe.

Japan and Switzerland are still in the process of reviewing and selecting sites and numerous other countries have only begun feasibility studies. Meanwhile, Japan plans to start the operation of a large reprocessing facility in 2012 in order to recycle and re-use civilian fuel. The Netherlands, Belgium, and Italy export their high level waste to France to be vitrified while Bulgaria and Ukraine export theirs to Russia. The vitrified waste is usually returned to the country for final disposal and not kept in France or Russia. In Japan has a large interim dry storage facility that will hold spent fuel until it is reprocessed (World Nuclear Association 2011).

3. Comparison of Spent Fuel Management Policies around the World

Table 4 provides a compressed summary of spent fuel management policies in countries with civilian spent fuel. As explained before, policies are generally grouped into three categories, direct disposal, reprocessing, and export. Many countries are also undecided on official policy or are considering a combination of the two or three options. In addition, as examined in the previous section on onsite dry storage, many countries have spent fuel in dry cask storage at the plants. In order to estimate the percent of fuel that is experiencing each policy, the total nuclear capacity for each country is also listed.

Figure 1 divides the nuclear capacity into each general policy category. This data indicates that 44% of spent fuel in the world is or will be reprocessed, 3% is or will be exported, 51% will be directly disposed of, and 2% is unknown. When examining just the NNWS, where most of the spent fuel is safeguarded by the IAEA, 55% will be directly disposed, 35% reprocessed, 6% exported, and 4% unknown. These numbers show that more than half of the spent fuel under safeguards will eventually be put in a deep geological repository and only a few nations will reprocess or export their fuel. Examining these spent fuel policies gives an overall picture of spent fuel management around the world.

Table 4: Summary of Spent Fuel Management Policies Around the World (International Atomic Energy Agency 2010), (World Nuclear Association 2011), (Högselius 2009)

Country	**Number of Reactors**	**Energy Generated (MWe)**	**Percent of Total Electricity Generation**	**Current Policy**
Argentina	2	935	7.0%	Undecided
Armenia	1	375	45.0%	Undecided
Belgium	7	5934	51.7%	Reprocessing
Brazil	2	1884	2.9%	Undecided
Bulgaria	1	1906	35.9%	Export to Russia
Canada	18	12569	14.8%	Direct disposal
China	13	10048	1.9%	Reprocessing
Czech Republic	6	3678	33.8%	Direct disposal
Finland	4	2716	32.9%	Direct disposal
France	58	63130	75.2%	Reprocessing
Germany	17	20490	26.1%	Direct disposal
Hungary	4	1889	43.0%	Direct disposal
India	19	4189	2.2%	Reprocessing
Italy	0	0	0.0%	Export to France with waste return
Iran	1	915	1.8%	Export to Russia
Japan	54	46823	29.2%	Reprocessing
Republic of Korea	21	18665	34.8%	Direct disposal
Lithuania	1	1185	76.2%	Direct disposal
Mexico	2	1300	4.8%	Undecided
Netherlands	1	487	3.7%	Export to France
Pakistan	2	425	2.7%	Undecided
Romania	2	1300	20.6%	Direct disposal
Russia	32	22693	17.8%	Reprocessing

Table 4: Summary of Spent Fuel Management Policies Around the World (International Atomic Energy Agency 2010), (World Nuclear Association 2011), (Högselius 2009), continued

Slovakia	4	1762	53.5%	Direct disposal
Slovenia	1	666	37.8%	Direct disposal
South Africa	2	1800	4.8%	Undecided
Spain	8	7514	17.5%	Direct disposal
Sweden	10	9303	37.4%	Direct disposal
Switzerland	5	3238	39.5%	Reprocessing
Taiwan	6	4980	20.7%	Direct disposal
Ukraine	15	13107	48.6%	Export some fuel to Russia with waste return, direct disposal
United Kingdom	19	10137	17.9%	Reprocessing
United States	104	100747	20.2%	Direct disposal but reconsidering

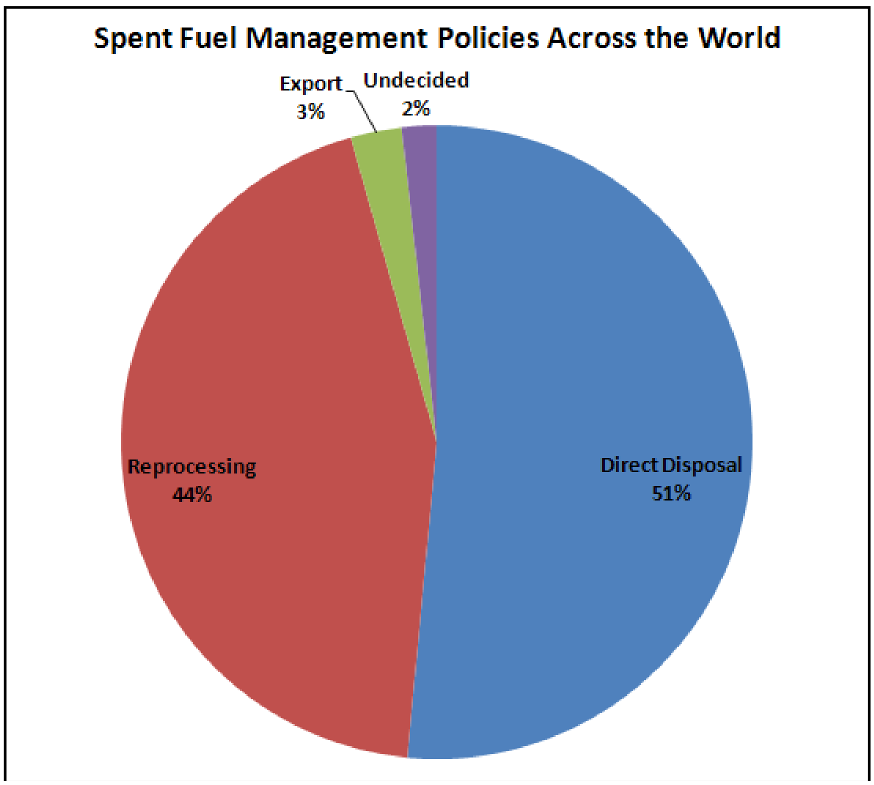


Figure 1: Fraction of spent fuel governed by each type of spent fuel management policy (data derived from Table 1).

IV. Safeguards Methods for Spent Fuel Storage

Several forms of spent fuel storage have been considered by various countries around the world. However, spent fuel storage can be lumped into three major categories: spent fuel pool storage, dry cask storage, and a geological repository. Centralized or international dry storage is essentially a form of dry cask storage but may require different safeguards measures. The IAEA, in conjunction with research laboratories around the world, has developed many safeguards methods for spent fuel storage. While many have been put into use to monitor and verify spent fuel, there is still a plethora of methods being developed. The following sections will outline the safeguards measures that are currently taken for various form of spent fuel storage, along with those that are still under research and development.

Spent fuel storage is considered an item facility in terms of safeguards because the nuclear material is counted as either an assembly or cask. The safeguards procedures for items include counting, identification, examination of integrity, and non-destructive measurements. The IAEA has experience in implementing these safeguards methods at spent fuel pools. However, in the case of dry storage, while casks may be counted themselves, it is difficult to verify the material in the casks because the integrity of casks should not be compromised (Pushkarjov 1986). In addition, research is still underway for safeguarding the transportation of spent fuel or a geological repository.

1. Current Safeguards Measures for the Spent Fuel Pool

Since the spent fuel pool is an item facility, spent fuel assemblies merely need to be counted to verify that they are all there. In reality, the operator can keep incorrect records or a dummy assembly may be put in the place of a diverted spent fuel assembly. In order to attempt to detect or prevent the diversion of spent fuel assemblies, containment and surveillance (C/S) measures are taken to ensure the continuity of knowledge (CoK) of the locations of the spent fuel assemblies. C/S measures include cameras and sealed access points to the spent fuel canal, for example. Surveillance measures can be improved by reducing the picture-taking interval, improving the resolution, using front-end scene-change detection, having better camera placements and using radiation sensors to trigger the cameras. Since C/S may fail to detect a diversion, spent fuel assemblies must also be verified (Boyer et al. 2007).

The IAEA currently uses several methods to verify the nuclear material in spent fuel pools by using non-destructive assay (NDA). The Improved Cerenkov Viewing Device (ICVD) allows the inspector to verify that spent fuel assemblies are indeed spent fuel by observing the glow of Cerenkov radiation. The ICVD amplifies the Cerenkov glow emitted from the fuel assembly, discriminating the glow from other sources of light. The IAEA inspector must judge the intensity of the glow relative to the assembly's declared burnup and cooling time. A dummy assembly may also be distinguished because it does not glow. Since verification with the ICVD depends on the inspector's experience and judgement, it is a qualitative measurement (Boyer et al. 2007).

Quantitative measurement devices, such as detectors, are also used by the IAEA to verify spent fuel. The Spent Fuel Attribute Tester (SFAT) verifies spent fuel by gamma ray spectroscopy usually using a CdZnTe or NaI detector connected to a Mini Multi Channel Analyzer (MMCA). The inspector observes the 661 keV Cs-137 peak to verify that the assemblies are indeed spent fuel. However, assemblies with low burnup or a very long cooling time may not show a peak distinguishable from background. Since the device must be inserted into the spent fuel pool to take measurements, some reactor operators may not allow it in fear of damaging the assemblies (Boyer et al. 2007).

The best NDA verification method used by the IAEA is the Fork Detector (FDET), which verifies spent fuel by measuring both gammas and neutrons (Boyer et al. 2007). The FDET consists of two detector arms that wrap around the fuel assembly. The detectors consist of an ionization chamber for gross gamma neutron measurements, a U-235 fission chamber for measuring thermal neutrons and a cadmium-shielded chamber for measuring fast neutrons (LaFleur et al. 2009).

While the ICVD, SFAT, and FDET devices should be able to detect a whole dummy assembly, they may not be able to detect a partial dummy assembly, where only some of the rods may have been removed. The FDET can only detect over 50% removal of the spent fuel rods. For this reason, spent fuel verification methods need to be further developed (Boyer et al. 2007).

Performing in situ verification measurements of irradiated fuel bundles in order to confirm specific spent fuel attributes without isolating the bundles has proven to be quite a challenge. However, this challenge has been partly solved with the development of enhanced cadmium-zinc-telluride (CdZnTe) detectors than can measure a gamma ray spectrum to verify the fission product content of the spent fuel. For the CANDU reactor at the Karachi Nuclear Power Plant in Pakistan, the irradiated fuel bundles are stored on horizontal trays in groups of 11. Eighteen filled trays are then stacked in the spent fuel pool. The detection probe must be lowered in between the stacks to adequately take measurements of single fuel bundles. The apparatus must be taken out of storage and assembled and mounted on the storage bay bridge in order to use it (Ahmed 2002).

2. Spent Fuel Pool Safeguards Measures under Development

More advanced detectors need to be developed because the detectors currently used by the IAEA cannot detect if less than half of a fuel assembly has been diverted and replaced by a dummy material. Therefore, there are many new or improved methods under research and development. The Digital Cerenkov Viewing Device (DCVD) has the enhanced capability of the previously described ICVD. The DCVD observes the glow of fuel assemblies but also records a picture for further scrutiny (Boyer et al. 2007). Studies are currently underway to characterize the UV photon intensity and derive information about the burnup and cooling history (Pratt, Bourva, and Carchon 2006). This device may be able to detect a 50% rod diversion. Gamma-ray tomography (TOMO) constructs an image of the spent fuel assembly by measurement of gamma spectrum from different angles, but the technology is still under development (Boyer et al. 2007).

In order to institute safeguards of special nuclear materials of the Advanced spent fuel Conditioning Process (ACP) by the Korea Atomic Energy Research Institute (KAERI), the ACP Safeguards Neutron Counter (ASNC) has been developed. This detector can verify spent fuel rod samples by non-destructive assay. The neutron counter counts the coincidences of emitted neutrons from the even numbered Pu and Cm isotopes, classifying them as single neutron counts (S), doubles (D), and triples (T). The masses of plutonium and uranium are then obtained from calibration curves and the curium ratio. The study was successfully completed and the ASNC was concluded to be one the of most efficient neutron counters for spent fuel (Lee 2008).

Another detection system under research and development is the Self-Interrogation Neutron Resonance Densitometry (SINRD) detector. This system utilizes the unique resonance structure of the fission cross sections of U-235 and Pu-239. The resonance absorption lines are measured using a set of U-235 and Pu-239 fission chambers. A bare U-235 fission chamber measures thermal neutrons and a B_4C covered U-235 fission chamber measures fast neutrons. Finally, Gd and Cd coated Pu-239 fission chambers are used to measure the resonance absorption from Pu-239. Simulations have estimated that this system can detect if more than 10-20% of the fuel pins have been removed, which is much better than the FDET system described previously. It has also been proposed to combine the FDET and SINRD detectors into one system for prime accuracy in detection (LaFleur et al. 2009).

3. Current Safeguards Measures for Spent Fuel Transfer to Dry Storage

As spent fuel pools at power plants fill up, many plants chose to invest in on-site dry storage to temporarily store spent fuel assemblies until they can be transferred to permanent storage or a reprocessing facility, for example. Spent fuel assemblies may be vulnerable to diversion in the transfer from pool to dry cask storage. The assemblies may be diverted off-site and possibly replaced by dummy assemblies. Therefore, safeguarding the actual transfer of spent fuel from the pool to dry storage is very important.

The IAEA safeguards this transfer by maintaining CoK and installing C/S equipment. Before the transfer campaign, the IAEA approves the procedures and installs reliable C/S equipment. Traditionally, an IAEA inspector is always on-site during the campaign, monitoring each move of the spent fuel assemblies and filled casks. However, since this involves many inspector days in the field, the IAEA has begun to move towards unattended, remotely monitored spent fuel transfer campaigns (Hanks and Tolba 2006). Two case studies, one for an attended spent fuel transfer campaign and one for an unattended transfer campaign, are discussed next.

The Zaporozhe power plant in Ukraine implemented dry spent fuel storage in 2001 and has the capacity for 110 ventilated storage containers, which is expected to increase to a capacity of 350 containers in the future. In order to maintain the CoK for the transfer of spent fuel, the following safeguards measures have been taken at the Zaporozhe power plant (Herrera et al. 2006).

i. Remote optical surveillance
ii. Attachment of VACOSS electronic seals
iii. Serial number identification of assemblies before loading
iv. Verification of gamma and neutron spectrum from spent fuel using the fork detector (FDET)
v. Verification of assemblies in container by item counting and serial number identification
vi. Sealing storage container with IAEA seals before transport

Using these safeguards measures along with verifying the declared reactor burnup, the IAEA can effectively ensure that no spent fuel is being diverted during transfer to dry storage (Herrera et al. 2006).

As mentioned previously, in order to minimize the inspector days required in the field, the IAEA has implemented unattended safeguards measures for the transfer of spent fuel from the pool to dry storage. This has been accomplished by installing unattended monitoring systems that maintain the CoK of the spent fuel during the transfer. Under a State-level approach to safeguards, these transfer campaigns are also subject to random inspections by the IAEA. This unattended approach has over a 50% probability to detect a post-transfer diversion (Hanks and Tolba 2006).

The Cernavoda Nuclear Site in Romania, a CANDU reactor, was the first of its kind to undergo an unattended safeguarded spent fuel transfer campaign. In order to ensure that all possible diversion methods could be counteracted, each step of the transfer was evaluated and safeguards implemented to ensure the CoK over the entire route. The safeguards measures taken through the spent fuel transfer are outlined below (Hanks and Tolba 2006).

i. Spent fuel, sealed by underwater ultrasonic seals, is unsealed and item counted.
ii. The plant notifies the IAEA of the full program of the intended transfer.
iii. IAEA inspectors use underwater surveillance to remotely monitor the re-batching of spent fuel bundles as they are loaded into dry storage baskets.
iv. The baskets are placed on a transport vehicle equipped with IAEA surveillance, including a neutron monitoring device.
v. The basket passes through the silo entry gamma monitor (SEGM) to verify the flow of spent fuel.
vi. Continuous surveillance remotely monitors the dry storage modules.

vii. Cabinets and wires for all surveillance and monitoring equipment are protected.

This unattended monitoring system has not only drastically reduced IAEA inspector days in the field, but has also allowed the plant to control its own spent fuel transfer schedule. In addition, the radiation dose has been reduced for the inspectors that would have been present on-site during the transfer (Hanks and Tolba 2006).

4. Current and Developing Safeguards Measures for Dry Cask Storage

It is important to maintain CoK of the spent fuel assemblies even after they have been placed in a cask for interim storage. Traditionally, the IAEA has implemented remote C/S measures to detect any movement of casks on the dry storage installation site. However, the IAEA is also moving towards attended and unattended monitoring of casks by gamma ray or neutron fingerprinting. Effective gamma and neutron detection could detect whether one or more assemblies were removed from a dry storage cask. The issue with this is that measurements must be done by nondestructive assay (NDA). Besides structural integrity and protection from weather or missiles, the main goal of a concrete and steel cask is to block radiation. Therefore, it is difficult to measure ample radiation outside of the cask in order to detect a partial removal of spent fuel assemblies. For example, a detector may not be able to distinguish between a full cask and a cask with a missing centerline assembly. Many cask fingerprinting detection systems are still being developed or experimentally used by the IAEA.

While the BN-350 nuclear facility in Kazakhstan is shut down, there remains much spent fuel in interim storage that is all under IAEA safeguards. As part of this project, the spent fuel assemblies were first quantitatively measured and documented using the Spent Fuel Coincidence Counter (SFCC). This gave the IAEA a baseline measurement for the material in the facility. After the assemblies are transferred into dry casks in groups of eight, the duel-slab verification detector (DSVD) can detect the removal of one or more of the assemblies by providing a “fingerprint” at the initial loading and then verifying the fingerprint periodically (Santi and Browne 2006).

In order to implement unattended safeguards at the BN-350 site, the UNARM (Unattended And Remote Monitoring) system was developed that collects, stores, and presents radiation data. To use this system on the dry cask storage, each cask is equipped with the MicroGRAND device. This device uses an ^{3}He tube to collect and store radioactive data for up to 90 days, to maintain CoK of the casks between IAEA inspections (Browne et al. 2006), (Browne et al. 2006).

Ultrasonic sealing has been used on multielement bottles (MEB), or spent fuel containers, at the British BNFL Sellafield plant as a safeguards measure. This technique installs a sealing bolt (SB) in place of one of the standard bolts on the container. The SB features a unique random signature and an internal breaking device, which prevents an unauthorized person from opening and resealing the container without being detected. Equipment involving a computer reads the signature from the SB and compares it to that of the stored reference signature, thereby indicating any tampering attempts. In addition, the SB can be read at any location, underwater or above ground (d'Agraives 1993).

The current reading and hardware technology used for the SB is being updated to make it a more portable, handheld device. This ultrasonic sealing technique can also be used for other applications, such as, safeguarding the underwater counterweights used in the British MAGNOX reactor or in the tagging of transport casks for nuclear materials. The ability to seal fresh PWR MOX (mixed oxide) fuel assemblies is also being considered as a further application for SBs (d'Agraives 1993).

Gamma-ray fingerprinting is currently being used to verify spent fuel for CANDU spent fuel storage canisters in case of a containment or surveillance failure at the site. However, this method may not be as effective as measuring the neutron spectrum emitted from the spent fuel in the canister. Therefore, prototype neutron detection systems, fabricated from He-3 and BF_3 detectors, are being studied to verify CANDU storage canisters. Source tests, and later field tests, found the optimum operating voltages for the different detectors. Further studies will include more tests using these detectors and eventually determine the validity of this neutron counting methodology for use in spent fuel verification (Park et al. 2006). It is expected that this system will meet the needs for initial verification while loading the cask along with reverification. In addition, this system may eventually be installed as a permanent remote monitoring system for the canisters (Lee et al. 2006)

Some studies have shown that cask fingerprinting is actually quite ineffective due to the inherent shielding of the radiation. Measurements of gamma-rays and thermal neutrons were taken on several different types of loaded casks at the Idaho National Laboratory. The results found that the radiation scattered by the shielding overwhelmed the unscattered radiation. This provided for an unclear cask “fingerprint.” However, it may still be possible to fingerprint casks by measuring high-energy radiation (Ziock, Caffrey, et al. 2005), (Ziock, Vanier, et al. 2005).

There are still many detection methods similar to those previously described that are being developed for dry cask storage verification. While the monitors used now can verify that there is radioactive material in the cask, a clear fingerprinting method has yet to be fully developed. In the case of verifying CANDU casks, the reverification tube built into the cask has allowed for measurement with minimal shielding. This may be the most effective way to verify material in casks. Still, containment and surveillance of the casks by seals and cameras is an important and effective measure for safeguards.

5. Geological Repository Safeguards Approach

As discussed in previous sections, many countries are adopting the policy of disposing of high level radioactive waste and spent fuel in a deep geological repository. Even though underground storage has many inherent security benefits, such facilities must be safeguarded by the IAEA because there is still a risk of diverting nuclear material. Safeguarding a geological repository poses many challenges, especially in maintaining CoK of the nuclear materials after the repository is closed. In particular, how can the IAEA and host country ensure that the nuclear material is not diverted in 100, 1000, or even 1 million years? This section will describe the progress in addressing this issue and then provide evaluation of some present safeguards methods for geological repositories.

The Waste Management Safeguards Project within the U.S. Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) has worked together with the IAEA to evaluate the safeguards measures that would need to be taken in order to effectively safeguard the transport, handling, and final storage of spent fuel in the case of an open fuel cycle. This project outlined the following tasks that need to be accomplished in order to meet safeguards objectives (Moran 1994):

i. Evaluate generic diversion paths at all possible back-end open fuel cycle facilities (including transportation).
ii. Identify the specific diversion paths.
iii. Evaluate the generic safeguards approaches against the specific diversion paths.
iv. Form specific safeguards approaches applicable to facility design requirements.
v. Identify research and development needs for the safeguards system.
vi. Develop proper design information documents for the IAEA.
vii. Determine the design information verification system.
viii. Determine appropriate technologies for design information verification system.
ix. Develop design requirements for verification techniques.
x. Evaluate the new safeguards approaches.
xi. Implement the new safeguards approaches.

Meanwhile, the general concepts that the IAEA has coined with regards to safeguarding a geological repository are as follows (Moran 2001):

i. Spent fuel in a geological repository is subject to safeguards
ii. Pre-operational design verification
iii. Safeguards by design
iv. Verify the receipt and flow of material

v. Verify nuclear material content

vi. Maintain continuity of knowledge of nuclear material content

vii. Detect undeclared activities

viii. Remote surveillance during post-operational phase

The above criteria have been described in more detail as IAEA guidelines and are thought to be sufficient for the safeguards of a geological repository.

The IAEA started the Program for the Development of Safeguards for the Final Disposal of Spent Fuel in Geological Repositories (SAGOR) in 1994 to develop a generic safeguards approach to protect against the diversion of spent fuel from geological repositories. They tried to address the primary point of when, if ever, safeguards should terminate in a geological repository, identified three phases of the lifetime of the repository, and proposed the appropriate safeguards measures (Fattah 1990), (Mosquera 2005).

The three phases of a repository lifetime are the pre-operational phase, the active or operating phase, and the closed phase. The pre-operational phase of a geological repository is as the repository is being licensed and built (Mosquera 2005). This phase may also involve a conditioning facility where fuel assemblies are prepared and packed in casks in hot cells (Tarvainen 1999). A conditioning facility is unnecessary if the spent fuel arrives in transport casks that are ready for disposal. The diversion paths for the conditioning facility include the removal of full casks or any whole or partial spent fuel assemblies in the storage areas and the conditioning process areas. The safeguards measures to be taken in this phase include (Tarvainen 1999):

- Design information verification of the repository design

- Shipper/receiver verification
- Containment and surveillance in storage areas
- Containment and surveillance in hot cell or conditioning areas
- Verification of spent fuel composition
- Verification of spent fuel in casks after conditioning

The active or operating repository stage involves identification and tracking of canisters that enter the repository and are sealed in their permanent locations (Mosquera 2005). The general concept is to ensure CoK of the casks throughout the transfer and permanent storage process through the following safeguards measures (Tarvainen 1999):

- Item counting
- Monitoring of cask flow using motion and radiation detectors
- Monitoring of cask flow using remote optical surveillance
- Sealing casks

The closed repository involves confirming that the containment provided by the geologic repository is not compromised to allow for diversion of casks. Safeguards may be performed via periodic visits or visual inspections of the area. The safeguards approach includes the following measures (Tarvainen 1999):

- Unannounced random visual inspections
- Satellite monitoring
- Active or passive seismic monitoring
- Environmental sampling

It is important to note that the safeguards for the closed repository does not verify that the actual spent fuel or nuclear materials are there but instead focuses on the integrity of the site. This is due to the fact that the tunnels into the repository are backfilled and the casks would need to be mined out (Tarvainen 1999).

6. Planned Geological Repository Safeguards

While many countries have adopted the policy for the final disposal of spent fuel in a geological repository, one has yet to be opened for commercial spent fuel. However, some U.S. government facilities have underground disposal facilities and other countries are in the process of implementing pre-operational safeguards on their commercial repositories. The safeguards approaches and measures of each disposal site are evaluated in this section to help provide a model for a geological repository with optimal safeguards.

An underground storage facility for nuclear materials has many inherent security features. As an example, the security features of the Radioactive Scrap & Waste Facility (RSWF) located at Argonne National Laboratory-West, now Idaho National Laboratory, are examined. RSWF was constructed in 1965 as an interim storage facility and continues to be the primary storage facility for the EBR-II spent fuel. This is a silo storage facility with carbon steel liners set and buried vertically in the ground. In addition to the spent fuel, the stored nuclear material also includes thorium, depleted uranium, natural uranium, enriched uranium, uranium-233, and plutonium (Moore and Zahn 1997).

The security features of the underground dry storage facility consist of detection and response systems along with physical barriers. The features include:

- High-quality fenced perimeter
- Bi-static microwave intrusion detection system
- Floodlights and infrared lighting system
- Intrusion reporting to a Central Alarm Station (CAS)
- CAS monitored by security response force
- Adequate response force to intrusion

The adversary would need to defeat the detection systems in order to obtain access to the storage casks. However, the adversary would need heavy equipment to be able to remove and haul the storage containers. It would be very difficult to reach the casks undetected with heavy equipment and succeed in removal while in an open area under fire. Therefore, underground storage has many inherent security features that make it difficult to successfully remove nuclear material (Moore and Zahn 1997).

Sweden and Finland have both adopted the policy of a once-through fuel cycle and plan to dispose of spent fuel in a geological repository after interim wet and/or dry storage. The two countries are working together and have the same strategy for the final disposal of spent fuel. Each country will encapsulate the spent fuel in cast iron enforced copper canisters. Finland has already decided on a repository site while Sweden is still undergoing site selection between two sites (Fritzell et al. 2008).

The general safeguards approach to the Swedish and Finnish geological repositories will be to verify the spent fuel placed in casks by NDA and then maintain CoK throughout the final disposal process. If the C/S measures for CoK fail after NDA, re-verification must be established again by NDA. The safeguards system should also not interfere with the operator's regular activities. Therefore, NDA measurements after the loss of CoK should be done without the need for a backflow of material. Dual C/S in both the tunnels and during the encapsulation process should be instituted in order to minimize the risk of complete loss of C/S. In addition, fingerprinting casks would enhance and make the C/S system more robust. Table 5 outlines the various safeguards methods that will be used in different processes leading to the final disposal of spent fuel in the Finnish geological repository (Fritzell et al. 2008).

While the general safeguards approach and goals for a geological repository seem to be well-defined, there are many challenges still facing the proposed system. Some of the technology described in Table 5 is still under research and development. Specifically, an NDA verification system to detect partial diversion and a fingerprinting method for assemblies and casks have yet to be fully developed. In addition, the performance of portal monitors in the final disposal tunnels needs investigation. Finally, a secure way to transfer information to future generations needs to be addressed (Fritzell et al. 2008).

Table 5: Safeguards methods for the final disposal of spent fuel in a geological repository in Finland (Fritzell et al. 2008)

Process	**Technical Safeguards Methods**
NDA verification	Tomography Other partial defect method
Assembly fingerprinting	Surface fingerprinting Radiation fingerprinting Measuring assembly weight
Buffer storing of verified assemblies	Camera surveillance
Transport cask loading	Authenticated fingerprints maintain CoK Camera surveillance
Cask transport/transfer	Authenticated fingerprints maintain CoK Seals
Cask storing	Authenticated fingerprints maintain CoK Seals
Cask unloading	Authenticated fingerprints maintain CoK Camera surveillance
Encapsulation process	Verification of fingerprints Remote monitoring cameras Inspector presence Portal monitors (All diversion paths should be covered)
Canister fingerprinting	Surface fingerprinting Other novel fingerprinting method Radiation fingerprinting Measuring canister weight
Canister transfer to the emplacement hole	Authenticated fingerprints maintain CoK Portal monitors (All diversion paths should be covered)
Canister emplacement	Verification of fingerprints Remote monitoring cameras Inspector presence Portal monitors (All diversion paths should be covered)
Backfilling	Remote monitoring cameras Inspector presence Portal monitors Other novel methods
After backfilling	Sealing Geophysical methods Satellite monitoring Other novel methods

7. Summary of Spent Fuel Safeguards

The past sections described many current and developing safeguards measures for spent fuel in various types of storage. The safeguards methods used in spent fuel pools, dry cask storage, and geological repositories have been summarized in Table 6, Table 7, and Table 8, respectively. For each safeguards method, the instrument or name, type of detection, description, level of verification, limitations, and current use are tabulated. As can be seen, there are many different spent fuel safeguards methods currently in use or under development. The newest 2011 version of the IAEA publication on "Safeguards Techniques and Equipment" can also be consulted for the most recent developments in safeguards technology (IAEA 2011).

Table 6: Summary of spent fuel pool safeguards

Instrument	**Type of Detection**	**Description**	**Level of Verification**	**Limitations**	**Current Use**
E-cup Seals	Tamper-indicating	Seals object, doorway, or other access point	Physical integrity	Inspector must visit site, remove seal, and send to headquarters for authentication	Most common seal used in IAEA
Electronic Seals[2]	Tamper-indicating	Seals object, doorway, or other access point	Physical integrity	Need electricity supply or battery	Minimal use by IAEA, under development
Visual Inspection[3]	Visual confirmation by inspector or remote monitoring	Count and inspect assemblies Monitor unauthorized movement	Physical integrity	Must have visual access Cannot identify missing fuel rods or substitution of dummy rods	Accountancy by plants and IAEA
Improved Cerenkov Viewing Device (ICVD)[, 4, 5](Doyle 2008)	Cerenkov	Amplifies light to identify physical characteristics of SNF in storage pool	Irradiation exposure and physical integrity	Cannot identify missing fuel rods or substitution of dummy rods	Most common type of verification used for SNF by IAEA
Digital Cerenkov Viewing Device (DCVD)[4, 6]	Cerenkov	Observes Cerenkov glow and records image	Irradiation exposure and physical integrity	Cannot identify less than half of missing fuel rods Cannot identify substitution of dummy rods	Used by IAEA

[2] (Tzolov, Goldfarb, and Penot 2007)
[3] (Doyle 2008)
[4] (Boyer et al. 2007)
[5] (Phillips 1991)
[6] (Pratt, Bourva, and Carchon 2006)

Table 6: Summary of spent fuel pool safeguards, continued

Instrument	**Type of Detection**	**Description**	**Level of Verification**	**Limitations**	**Current Use**
FORK detector (FDET)[7, 8]	Passive Gamma-Ray and Neutron Total Counting	Measures gamma rays using ionization chambers and neutrons using fission chambers	Irradiation exposure level found from gamma and neutron activity and verified to operator-declared information Can verify operating history of LWR	Assembly must be partially lifted in pool Cannot identify less than half of missing fuel rods or substitution of dummy rods Must have trustworthy operator declarations Cooling time must be greater than one year	Common use by IAEA
Irradiated Fuel Attribute Tester (IRAT)'	Gamma-ray Spectroscopy	CdZnTe detector measures gamma spectrum from fission products	Fission product and actinide content determined and verified to operator-declared information	Assembly must be partially lifted in pool Cannot identify less than half of missing fuel rods or substitution of dummy rods Must correct for shielding effects of assembly Must have trustworthy operator declarations	Used by IAEA
Spent Fuel Attribute Tester (SFAT)'	Gamma-ray Spectroscopy	CdZnTe or NaI(Tl) detector measures gamma spectrum from fission products	Fission product and actinide content determined and verified to operator-declared information	Cannot identify less than half of missing fuel rods or substitution of dummy rods Must correct for shielding effects of assembly Must have trustworthy operator declarations	Used by IAEA

[7] (Doyle 2008)
[8] (Phillips 1991)

Table 6: Summary of spent fuel pool safeguards, continued

Instrument	Type of Detection	Description	Level of Verification	Limitations	Current Use
Gamma-ray Tomography[9, 10]	High-Energy Gamma Tomography	Measures gamma-rays to reveal rod-to-rod gamma distribution and construct total image of assembly	Verifies irradiation exposure level in each rod	Assembly must be partially lifted in pool Must have trustworthy operator declarations	Research and development
Silo Entry Gamma Monitor (SEGM)[11]	Passive gamma-ray counting	Detector mounted at mouth of where spent fuel transfer occurs	Verifies presence of spent fuel during transfer	Cannot identify missing fuel rods or substitution of dummy rods Cannot distinguish between spent fuel assembly and other gamma-ray source	Used by IAEA at Romania's CANDU reactor
Mobile Unit Neutron Detector (MUND)	Passive neutron counting	Detector mounted on top of transport flask	Verifies presence of spent fuel during transfer	Cannot identify missing fuel rods or substitution of dummy rods Cannot distinguish between spent fuel assembly and other neutron source	Used by IAEA at Romania's CANDU reactor
Gamma-Ray Fingerprinting[12]	Active gamma-ray counting	Gamma-ray detector mounted on spent fuel storage canister	Continuous gamma-ray measurement verifies that spent fuel is in canister	Cannot fully distinguish between spent fuel assembly and other gamma-ray source	Used by IAEA
Neutron Fingerprinting	Active neutron counting	Neutron detector (He-3 or BF_3)	Continuous neutron measurement verifies spent fuel	Unknown	Research and development

[9] (Doyle 2008)
[10] (Boyer et al. 2007)
[11] (Hanks and Tolba 2006)
[12] (Park et al. 2006)

Table 6: Summary of spent fuel pool safeguards, continued

Instrument	**Type of Detection**	**Description**	**Level of Verification**	**Limitations**	**Current Use**
Miniature CdZnTe Gamma-Ray Detector[13]	Gamma-ray Spectroscopy	Miniature CdZnTe gamma-ray detector measures isotopic ratio of Cs-134 to Cs-137	Can verify declared burnup within 10%, cooling time within 3%, and U-235 enrichment within 10%	Long count times needed for enrichment verification	Research and development
In Situ CdZnTe Gamma-Ray Detector[14]	Gamma-ray Spectroscopy	CdZnTe gamma-ray detector lowered into spent fuel pool to measure Cs-137 spectrum	Gamma ray spectrum used to verify fission product content of spent fuel	Apparatus must be lower into pool and possibly decontaminated when removed	Used by IAEA at Karachi NPP (CANDU)
Advanced Spent Fuel Conditioning Process (ACP) Safeguards Neutron Counter (ASNC)[15]	Neutron Coincidence Counting	ASNC measures neutron coincidences of Cm-244 from spent fuel rods	Can measure the contained mass of U and Pu in spent fuel	Unknown	Research and development
Self-Interrogation Neutron Resonance Densitometry (SINRD)[16]	Neutron Detection	U-235 and Pu-239 fission chambers	10 – 20% pin removal	Uknown	Research and development

[13] (Abbas 1998)
[14] (Ahmed 2002)
[15] (Lee 2008)
[16] (LaFleur et al. 2009)

Table 7: Summary of dry cask storage safeguards

Instrument	**Type of Detection**	**Description**	**Level of Verification**	**Limitations**	**Current Use**
Visual Inspection[17]	Visual confirmation by inspector or remote monitoring	Count and inspect dry storage containers Monitor unauthorized movement	Physical integrity, no movement or opening of casks	Must have visual access Cannot identify missing fuel rods or substitution of dummy rods C/S failure is possible	Accountancy by plants and IAEA
Sealing Bolt (SB)[18]	Ultrasonic sealing	SB installed in place of standard bolt on spent fuel container	SB features unique random signature and internal breaking device to indicate tampering	Does not detect tampering real time Must be verified by inspector	Used on spent fuel containers in Britain
Gamma detectors[19]	Gamma ray fingerprinting	Verify gamma signature of cask from outside	Verify that radioactive elements are in cask	Cannot identify missing fuel rods or substitution of dummy rods	Used by IAEA
Neutron Fingerprinting[19]	Active neutron counting	Neutron detector (He-3 or BF_3) mounted on spent fuel storage canister	Continuous neutron measurement re-verifies that spent fuel is in canister	Unknown	Research and development
Cask Fingerprinting[20]	Measurement of gamma rays and thermal neutrons	Measurement of whole cask from slight distance	Provide a cask fingerprint to distinguish and identify any cask	Radiation scatters due to the cask shielding and thus the fingerprint is unclear	Research and development

[17] (Hanks and Tolba 2006)
[18] (d'Agraives 1993)
[19] (Park et al. 2006), (Lee et al. 2006)
[20] (Ziock, Caffrey, et al. 2005), (Ziock, Vanier, et al. 2005)

Table 8: Summary of geological repository safeguards

Instrument	Type of Detection	Description	Level of Verification	Limitations	Current Use
Visual Inspection[21]	Visual confirmation by inspector or remote monitoring	Count and inspect dry storage containers Monitor unauthorized movement	Physical integrity, no unauthorized movement of spent fuel or opening of casks	Must have visual access Cannot identify missing fuel rods or substitution of dummy rods C/S failure is possible	Used by IAEA for other storage methods
Assembly Fingerprinting	Gamma and/or neutron detection	Provide fingerprint for each assembly to allow for CoK throughout transfer process	Unique assembly identification	May not detect partial removal of rods	Research and Development
Canister Fingerprinting	Gamma and/or neutron detection	Provide fingerprint for each canister to allow for CoK throughout transfer process	Unique canister identification	May not detect partial removal of rods or assemblies	Research and Development
Portal Monitor	Gamma and/or neutron detection	Track and verify movement of canisters underground	Identification of canister presence	May not detect partial removal of rods or assemblies	Research and Development
Seals	N/A	Seal repository once filled	Verify that nothing has entered or left repository after closed	Must be verified by inspectors May be tampered with	Used by IAEA for other storage methods
Seismic Monitoring	Geophysical movement detection	Detect attempted clandestine underground access to repository	Verify that nothing has entered or left repository after closed	Unknown	Used by IAEA for nuclear test monitoring

[21] (Fritzell et al. 2008)

Table 8: Summary of geological repository safeguards, continued

Instrument	**Type of Detection**	**Description**	**Level of Verification**	**Limitations**	**Current Use**
Satellite Monitoring[22]	Detection of movement	Detect attempted clandestine surface access to repository	Verify that nothing has entered or left repository after closed	Unknown	Used by IAEA for nuclear facility monitoring

[22] (Fritzell et al. 2008)

V. Assessment of Proliferation Resistance

1. Various Proliferation Resistance Projects

Defining and assessing the proliferation resistance of various aspects of the nuclear fuel cycle has been an ongoing world-wide project. Many countries and organizations have participated in research to find an accurate and quantitative way to measure the proliferation resistance of parts of or whole nuclear fuel cycles, as described in the following paragraphs.

The U.S. Non-proliferation Alternative System Assessment Program (NASAP), started in 1976, defined that the proliferation resistance attributes are resources required, time required and risks of detection. The project concluded that the LWR fuel cycle with spent fuel storage is more proliferation resistant than other fuel cycles that involve HEU or Pu (Kang 2005).

The International Fuel Cycle Evaluation (INFCE) involved 66 countries and 5 international organizations and started in 1977. This evaluation defined the proliferation resistance attributes as resources required, time required, detectability, and safeguardability. Safeguardability was said to be the most important and no real difference was found in the proliferation resistance between the once-through and closed fuel cycles (Kang 2005).

There has also been extensive research performed in the area of plutonium disposition. The “spent fuel standard” is a condition in which weapons-grade plutonium has become intrinsically as proliferation resistant as plutonium in spent fuel. Intrinsic barriers include the concentration of plutonium, difficulty of separation, and ease of detection (Kang 2005).

The U.S. DOE Nuclear Energy Research Advisory Committee (NERAC) Task Force on Technical Opportunities for Increasing the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS) identified areas to advance technically using an integrated safeguards evaluation methodology (ISEM) in 2000. New research and development programs were suggested in the areas of assessing proliferation resistance, material control and accountability, and enhancing intrinsic barriers (Kang 2005).

Innovative Nuclear Reactors and Fuel Cycles (INPRO) project was initiated by the IAEA in 2003. The study worked to reduce proliferation risk to allow fulfilling of energy needs with nuclear power. It defined basic principles and user requirements for proliferation resistance but did not propose a specific method for evaluating proliferation resistance (Kang 2005).

The INPRO study identified intrinsic proliferation resistant features as technical features that reduce the attractiveness for nuclear weapons, inhibit the diversion of nuclear material, inhibit undeclared production of direct-use material, and facilitate verification. Extrinsic features includes states’ commitments to nonproliferation and disarmament, export/import agreements, control to nuclear material access, application of IAEA safeguards, and legal action with violations of the above agreements. Finally, the Generation IV Nuclear Energy System (Gen IV) project from the U.S. DOE in 2002 defined intrinsic barriers by material quality and defined extrinsic barriers by institutional controls (Kang 2005).

In an effort to develop methods for nonproliferation analysis, the NNSA has created the Nonproliferation Assessment Methodology (NPAM) Working Group consisting of representatives from DOE laboratories and academia. Proliferation resistance models can be used to evaluate the proliferation resistance of different fuel cycles in order to guide policy makers (Bari et al. 2003).

There are multiple methods of analysis that may be used to assess nonproliferation characteristics. Attribute analysis identifies specific attributes of a fuel cycle that affect the proliferation potential. In scenario analysis, hypothetical scenarios of proliferation pathways are analyzed. Finally, the two-sided method examines the interplay between two opponents in a proliferation scenario (Bari et al. 2003).

To aid further in the analysis, barriers to proliferation must be identified and evaluated in terms of effectiveness. Barriers are characterized as either intrinsic (inherent to the system or material) or extrinsic (extra safeguards approaches taken by the facility or state). The specific proliferation threats to the fuel cycle must be characterized. Also, the fuel cycle may be divided into different facilities, or subdivided even further, to aid in the analysis. Finally, metrics for the total proliferation analysis must be defined (Bari et al. 2003).

2. Working Definition of Proliferation Resistance

The concepts of proliferation resistance can be divided into three major categories: (1) reducing total fissile inventory, (2) making access to or the separation of fissile material more difficult, and (3) minimizing the weapons value of plutonium by unfavorable isotopic combinations (Stanbro and Olinger 2002). A common definition of proliferation resistance is "a measure of the relative increase in barriers [both intrinsic to the material or process and extrinsic (or engineered)] to impede the proliferation of nuclear weapons either by diversion of material by a state in possession of a system or theft of material by a terrorist or sub-national group" (Charlton, LeBouf, and Aghara 2003).

Therefore, proliferation resistance is defined in terms of intrinsic properties that are built into a fuel cycle system and extrinsic properties that include the decisions of the state. However, intrinsic resistance does not necessarily mean making nuclear material as radioactive as possible. This approach makes the fuel cycle very expensive in terms of handling and fuel fabrication and thus is an unattractive option for states looking to pursue or expand their nuclear power infrastructure. Additionally, a state may still develop enrichment and separation technology to handle "hotter" materials and convert them to weapons-grade materials (Pasamehmetoglu 2006).

It is important to outline the exact proliferation resistant characteristics of a system involving special nuclear material in order to evaluate the total proliferation resistance. Determining a quantitative way to describe these characteristics will also give a quantitative measure of total proliferation resistance. Various methodologies for analyzing proliferation resistance will be examined.

3. Existing Methods to Asses Proliferation Resistance or Risk

As described in the previous section, the proliferation resistant characteristics of nuclear materials are described by the intrinsic and extrinsic barriers. These include, but are not limited to, the form and quantity of the material, accessibility, and added safeguards and institutional controls. It is beneficial to attempt to describe these characteristics quantitatively in order to determine the actual attractiveness, proliferation risk, or proliferation resistance of nuclear materials, nuclear facilities, or even entire fuel cycles. This section will describe different methods that have and are being used to assess the proliferation risk or resistance of nuclear materials, facilities, or cycles.

Some parameters that may be used to assess proliferation risk are described by the IAEA. Significant amounts of nuclear materials needed to create a weapon are termed significant quantities (SQs). In particular, 1 SQ of Pu is 8 kg, and 1 SQ of LEU is 75 kg (in terms of only the isotope U-235). In addition, the IAEA estimates that it would take a minimum of 1 to 3 weeks to divert and process 1 SQ of Pu in spent fuel, but a minimum of about a year to divert and process 1 SQ of U-235 in spent fuel (Kiriyama and Pickett 2000). Therefore, it seems that the plutonium in spent fuel is a greater proliferation risk than the uranium. The increased time for diverting and processing U-235 is derived from the fact that more of it needs to be diverted and it must undergo enrichment instead of chemical separation.

The first proliferation-resistance methodology examined tracks the proliferation resistance of a unit mass of nuclear material through its entire cycle. The proliferation resistance is viewed as the probability that proliferation would be avoided per unit mass input per unit time. This methodology is based on the Multi-Attribute Utility Analysis (MAUA) method and examines 1-MT of fuel over 100 years in the fuel cycle (Charlton, LeBouf, and Aghara 2003).

The static proliferation resistance value PR_{ij} for a given sub-objective j in process i is determined by

$$PR_{ij} = \sum_{k=1}^{K_j} w_{jk} u_{jk} (x_{ijk})$$

where w_{jk} is the weight, u_{jk} is the utility function, and x_{ijk} is the input value for the utility function for each attribute k of each sub-objective j. The sub-objectives and attribute measures taken into account include (Charlton, LeBouf, and Aghara 2003):

1. Host nation diversion
    i. Attractiveness level
    ii. Concentration
    iii. Handling requirements
    iv. Type of accounting system
2. Theft by insider
    i. Attractiveness level
    ii. Concentration
    iii. Handling requirements
    iv. Type of accounting system
    v. Accessibility
3. Theft by outsider
    i. Attractiveness level
    ii. Concentration
    iii. Handling requirements
    iv. Type of accounting system
    v. Accessibility
4. International acceptance

i. Treaty compliance

ii. IAEA cost to safeguard

iii. Similarity to weapons production

5. Meets NRC requirements

i. Meet NRC requirements

6. Technology misuse

i. Technology transfer

ii. Export control

Each utility function requires a numerical input from the user. The static proliferation resistance value PR_i for the total process i is determined by

$$PR_i = \sum_{j=1}^{J} w_j PR_{ij}$$

where w_j is the weight for the sub-objective j. Finally, the total nuclear security measure NS for the system is determined by

$$NS = \frac{\sum_{i=1}^{I} m_i \cdot \Delta t_i \cdot PR_i}{M \cdot T}$$

where m_i is the amount of material that is in process i for time Δt_i, M is the total amount of material evaluation (1 megaton) and T is the total time in the cycle (100 years). A higher PR value gives a higher NS value which means that the material in the process is less likely to undergo proliferation (Charlton, LeBouf, and Aghara 2003).

The above analysis was used to find the proliferation resistance of various fuel cycles and reprocessing techniques. The results yielded NS values of 0.772 for PWR and 0.637 for CANDU fuel in the reactor for a certain period of time. The NS values for reprocessing were 0.395 for PUREX and 0.519 for UREX. Wet storage was also analyzed for 1-MT of spent fuel with a burnup of 50,000 MWd/MT stored for 100 years. The PR value decreased over time and the total NS value was found to be 0.716 (Charlton, LeBouf, and Aghara 2003).

Nonproliferation Assessment Tool (NAT) has been developed as a continuation of the above described proliferation resistant analysis. This software uses “Design for Nonproliferation” parameters which include all of the aspects of a nuclear fuel cycle facility that affect the proliferation resistance or diversion of nuclear materials. The Multi-Attribute Utility Analysis (MAUA) tool is used again in NAT to make complex decisions by assigning numerical values to all options. The measures used to assess the proliferation resistance (PR) are attractiveness level, concentration, handling requirements, type of accounting system, and accessibility (Pratt, Biegalski, and Landsberger 2007).

The NAT software has a simple graphical user interface that includes a section for facilities and a section for chains. Furthermore, ORIGEN 2.2 software is used within NAT to calculate the nuclide composition and characteristics of materials after irradiation and decay. ORIGEN 2.2 derives the quantities of americium, plutonium, and uranium, and further breaks it down into the quantities of various Np, Pu, Th, and U isotopes. The output information also includes the heating rate from plutonium and the radioactivity from actinides and fission products. All of these output values along with the measures stated above determine the PR value of a facility. An entire fuel cycle can also be described in the NAT software to determine a PR value (Pratt, Biegalski, and Landsberger 2007).

The purpose of the protected plutonium production (PPP) project is to research various aspects of plutonium protection against proliferation due to the potential to use plutonium for peaceful energy production. As a part of the PPP-project, studies have been done on the proliferation resistance of nuclear heavy metals. A particular study focuses on the concept of "attractiveness" of a material in terms of proliferation potential (Artisyuk, Saito, and Ezoubtchenko 2008).

In this study, the IAEA SQ concept and guideline that Pu with greater than 80% Pu-238 is "proliferation resistant," as well as the NRC formula quantity (FQ), have all come under reconsideration. The FQ was termed by the NRC as the quantity of nuclear material theoretically required for the fabrication of one nuclear weapon and is equal to 5 kg of U-235 in HEU, 2 kg of U-233, and 2 kg of Pu. To review, the IAEA SQ is equal to 25 kg of U-235 in HEU, 8 kg of U-233, and 8 kg of Pu. HEU is considered uranium enriched to at least 20% U-235 for both the FQ and SQ. The cause for concern in these classifications is that, for both the FQ and SQ, there is no difference in the mass between the quantities needed of Pu and U-233. However, the critical mass for U-233 is slightly smaller than for Pu. Also, the specific isotopic composition of Pu is not considered other than for the SQ, in which greater than 80% Pu-238 is proliferation resistant. For this reason, a new attractiveness level assessment for plutonium, dependant on isotopic composition, has been developed (Artisyuk, Saito, and Ezoubtchenko 2008).

The coefficient α-Rossi is defined as a "ratio of supercriticality (k-eff – 1) to prompt neutron life time," which describes the energy yield of a certain configuration of fissionable material. The attractiveness (ATTR) is based on this coefficient according to the following equation (Artisyuk, Saito, and Ezoubtchenko 2008):

$$ATTR = \frac{\alpha^n}{\left(\frac{\rho}{\rho_0}\right)M} \ ,$$

where *n* is adjustable to reflect energy release in a particular configuration, ρ/ρ_0 is the ratio of the material density reached in a particular configuration to density at normal conditions, and *M* is the mass of the material. Results show that the attractiveness versus density ratio of certain masses of Pu, U-234, Np-237, and U-235 (with varying enrichments) all increase with increasing density ratio. The attractiveness of Pu also depends on the technological barriers to handling the material, such as neutron emission (SF) and decay heat (DH). Another equation for attractiveness may be used to account for these barriers (Artisyuk, Saito, and Ezoubtchenko 2008):

$$ATTR = \frac{\alpha^3}{SF \times DH} \ .$$

The attractiveness of Pu decreases with increasing amounts of even isotope doping (Pu-238 and Pu-240), (Artisyuk, Saito, and Ezoubtchenko 2008).

A method to evaluate the nuclear nonproliferation credibility of a country has also been proposed. First a tree is made that shows the different criteria used to evaluate the state’s nonproliferation credibility, as shown in

Table 9. Then, weight coefficients are found for each criterion using the Analytic Hierarchy Process (AHP). This process involves a questionnaire comparing pairs of criteria that is filled out by experts. Table 10 shows the final weight of each Level 3 criterion in descending order. As can be seen, the NPT system and compliance category has the heaviest weight (Kwon and Ko 2009).

Table 9: Tree for the criteria of nonproliferation credibility (Kwon and Ko 2009)

Level 1	Level 2	Level 3
NPT system & compliance	NPT system	Safeguards
		Exports control
		Physical protection
		Duration of participation
	Compliance	Domestication
		Transparency
		Diplomatic activities
International politics	Military threats	Nuclear threats
		Conventional threats
		Potential threats
		Perceived threats
		Security assurance
	National prestige	Global power status
		Regional power status
		Pariah status
Domestic politics	Parochial groups & beliefs	Decision makers
		Military
		Scientists
		General public
		Political power of parochial groups
	Additional aspects of domestic politics	Technology
		Economy
		Society

Table 10: Weight coefficients of different nonproliferation credibility criteria, in order of importance (Kwon and Ko 2009)

Parameter	Total Weight Coefficient
Duration of participation	0.36
Safeguards	0.149
Domestication	0.092
Exports control	0.06
Physical protection	0.049
Nuclear threats	0.044
Transparency	0.041
Diplomatic activities	0.036
Political power of parochial groups	0.029
Conventional threats	0.028
Perceived threats	0.02
Decision makers	0.02
Security assurance	0.014
Military	0.011
Pariah status	0.009
General public	0.008
Economy	0.008
Potential threats	0.007
Regional power status	0.004
Scientists	0.004
Society	0.003
Global power status	0.002
Technology	0.001

Using these coefficients, another questionnaire is made to rate four states based on each criterion. The final results give a value to each country in terms of proliferation resistance. The countries analyzed in this study were Switzerland (9.006), Japan (8.404), South Korea (7.116), and North Korea (1.405). The study recommends that the criteria should be periodically updated, statistical indicators should be developed, and a relationship with a group of experts should be established in reference to this evaluation method (Kwon and Ko 2009).

In another study, different reactor types proposed by Russia to build in developing countries are reviewed according to proliferation resistance. The study proposes that the country is provided a floating reactor that sits on the shore, but is shipped back to the supplying country (e.g. Russia) for maintenance and refueling. The customer would not be handling any of the fuel or fuel cycle facilities, thus largely decreasing the proliferation risk by eliminating the risk of diversion during enrichment, reprocessing, and final disposal. There are various types of reactors considered for this model, including: PWR, KLT-40S, and ABV-6, all which have similar properties; and the GT-MGR with low-enriched uranium (LEU) fuel, weapons-quality plutonium fuel, or mixed U-Pu fuel (Petrunin et al. 2008).

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) has determined a methodology for assessment of the proliferation resistance of new reactors and fuel cycles. Using the INPRO methodology and its modification, proposed by KAERI for the DUPIC (Direct Use of PWR Spent Fuel in CANDU) fuel cycle analysis, along with the parameters of each reactor and supposed attractiveness of each fuel, the proliferation risk of each reactor is calculated. The results show that the proposed, supposedly more proliferation-resistant technologies (KLT and ABV) that are to take the place of the PWR, actually have the same relative risk. The GT-MGR with LEU is the most proliferation-resistant while the GT-MGR designs with weapons-quality Pu and mixed U-Pu fuels are the least proliferation-resistant (Petrunin et al. 2008).

Another quantitative approach to the assessment of relative proliferation risk of nuclear fuel cycles actually examines the proliferation risk of spent fuel in various forms of storage and material flow cases. In this assessment, the diversion risk u_i from material i

is a compound function of projections u_i^j on a set of criteria $[x_j]$ each of which possesses a weight w_j, defined by the relation (Silvennoinen 1981):

$$u_i = F\left(u_i^j \{w_j\}\right).$$

The assessment criteria are defined as (Silvennoinen 1981):

x_1: Minimum cost to produce a weapon from given material

x_2: Marginal cost of using civil nuclear fuel cycle to make weapons

x_3: Minimum time required to construct a weapon

x_4: Detectability of weapons construction

x_5: Ease of diversion (accessibility)

x_6: Quality of separated fissile material

The diversion risk is calculated for various source materials throughout the fuel cycle. However, only the spent fuel sources are outlined in Table 11 since only spent fuel storage will be considered in this project. Here, the diversion risk of the spent fuel in each storage type is shown as a function of the annual material flow, or total material as in the case of the closed repository. The utility for each factor described above is shown, along with the final total diversion risk utility. The final utility is graphed in Figure 2. As can be seen, spent fuel in interim storage after a long cooling time is subject to the highest diversion risk, while spent fuel in a closed repository has the least risk of diversion (Silvennoinen 1981).

Table 11: Diversion risk of the source materials as a function of the annual or total material flow (Silvennoinen 1981)

Source Material	Annual Material Flow (tHM)	Minimum Cost	Marginal Cost	Minimum time	Detect-ability	Divert-ability	Quality	Total Diversion Risk
Spent Fuel (short cooling time)	30	0.2	0	0.19	0.13	0.01	0.15	0.1
	500	0.8	0	0.19	0.13	0.03	0.15	0.18
	1000	1	1	0.19	0.13	0.03	0.15	0.31
Spent fuel in an interim storage (long cooling time)	30	0.22	0	0.23	0.28	0.06	0.15	0.15
	500	0.9	0	0.23	0.28	0.15	0.15	0.26
	1000	1	1	0.23	0.28	0.18	0.15	0.31
Spent fuel in a final repository (operating)	30	0.2	N/A	0.19	0.22	0.02	0.15	0.12
	1000	1	N/A	0.19	0.22	0.06	0.15	0.22
Spent fuel in a final repository (closed)	900	0.07	N/A	0.06	0.16	0.01	0.15	0.07
	30000	0.4	N/A	0.06	0.16	0.01	0.15	0.11

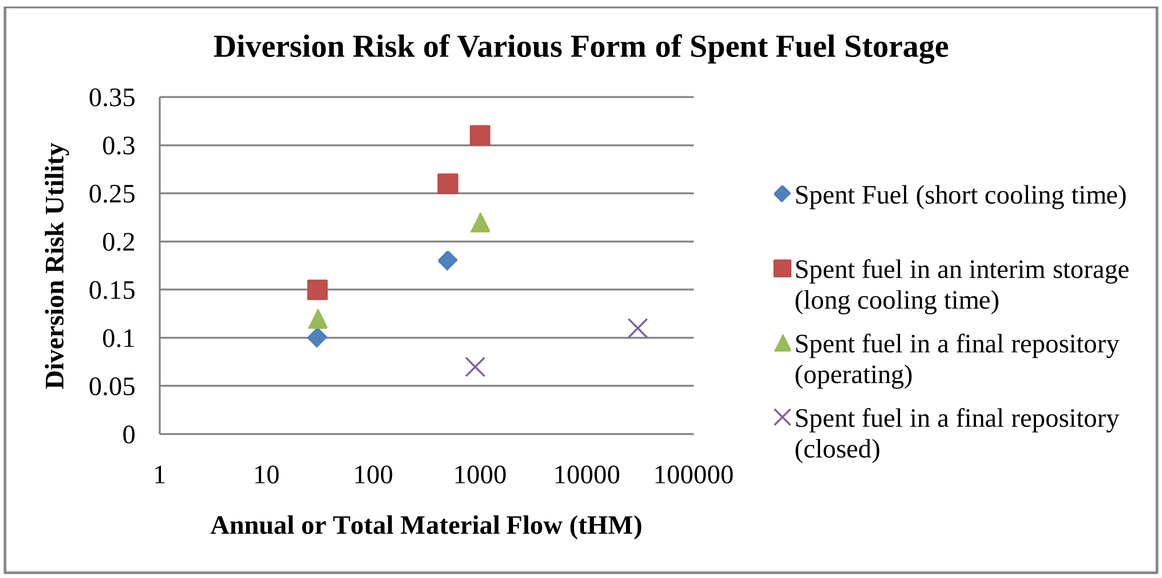


Figure 2: Diversion risk of the source materials as a function of the annual or total material flow, data taken from Table 11 (Silvennoinen 1981)

In an effort to re-describe proliferation resistant characteristics of fuel cycles, a study assessed the existing methodologies of proliferation resistance assessment. The past methods were analyzed and compared according to defined desired characteristics

(Giannangeli 2007). In reference to the proliferation resistant analysis of spent fuel in storage, the most desired characteristics derived from this study are: produces a time-dependent analysis, considers safeguards system implementation, considers physical protection measures, ability to assess multiple facility types with consistent set of metrics, and considers geological storage of material. An evaluation similar to this should be done to evaluate different proliferation resistance methods in terms of spent fuel analysis.

The aforementioned study categorized the proliferation resistant characteristics in four different phases of nuclear material becoming a nuclear weapon: diversion, transportation, transformation, and weapon fabrication. As related to spent fuel analysis, the material is all in the same form (except for dependence on cooling time), therefore only the diversion stage is different for the various storage options of spent fuel. The final inputs for proliferation resistance in the diversion stage as determined by this study are (Giannangeli 2007):

1) Material handling difficulty during diversion
    a) Mass/SQ of nuclear material
    b) Volume/SQ of nuclear material
    c) Number of items/SQ
    d) Material form (solid, liquid, powder, gas)
    e) Radiation level in terms of dose
    f) Chemical reactivity with common substances
    g) Temperature of source system
    h) Heat load of material
2) Difficulty of evading detection by the accounting system

a) Uncertainty in accountancy measurements

b) Expected vs. actual material unaccounted for

c) Frequency of measurement

3) Difficulty of evading detection by the material control system

a) Probability of detection based on vulnerability analysis of material control system in place

4. Analysis of Selected Proliferation Resistance Methodologies

Many methodologies for assessing the proliferation resistance of processes, facilities, and/or fuel cycles have been presented in the previous section. Selected methodologies are now analyzed in terms of relevance or applicability to spent fuel storage. The methods that are analyzed are derived from the research of (Silvennoinen 1981), (Poplavskii 2001), (Charlton et al. 2007), (Petrunin et al. 2008), (Kwon and Ko 2009), and shown in Table 12, Table 13, Table 14, Table 15, and Table 16, respectively. For each attribute, the applicability to spent fuel storage is determined and listed as “Yes” or “No.”

An important assumption being made in this assessment is that the material type (spent fuel) is constant for each storage type. This is an important assumption because it eliminates many of the proliferation resistant attributes. For the initial analysis of the spent fuel storage methods, the material in storage should not be considered since the actual storage *method* is going to be analyzed. The material in storage is assumed to be PWR or BWR assemblies with a normal burnup and at least five years out of the core. Therefore, the material in storage is relatively the same for any type of storage.

The applicability to spent fuel storage of the PR attributes in the following tables is determined by whether or not the attribute can be applied to the spent fuel storage facility rather than the material itself. Attributes pertaining to the nuclear material facility, such as capacity, or considering safeguards, such as measurement uncertainty, are marked with a “Yes.” Attributes that can be modified to be applicable to spent fuel storage are also marked with a “Yes.” An example of this is the radiation dose rate, which can be changed to be counted from outside of the storage method rather than of the material itself. Attributes marked with a “No” are mainly material-specific, such as isotopic composition or material heating rate. Inapplicable attributes also include those that have to do with the actual weapon fabrication, since only the diversion phase is being considered. Country-specific attributes are also not applicable since the storage methods are going to be analyzed, and not the countries in which they are stored.

Table 12: Proliferation resistant attributes and weights for methodology to assess the proliferation resistance of military and civil plants (Silvennoinen 1981). The applicability to spent fuel storage is listed for each attribute.

PR Attribute	**Weight**		**Applicable to Spent Fuel Storage?**
	Military Processing Plant	**Civil Processing Plant**	
Minimum cost to produce a weapon from given material	0.11	N/A	**No**
Marginal cost of using civil nuclear fuel cycle to make weapons	N/A	0.20	**No**
Minimum time required to construct a weapon	0.15	0.10	**No**
Detectability of weapons construction	0.17	N/A	**No**
Ease of diversion (unseparated material)	0.30	N/A	**Yes**
Ease of diversion (weapons-grade material)	N/A	0.35	**No**
Quality of separated fissile material	0.17	0.19	**No**

Table 13: Proliferation resistant attributes and weights for methodology to assess the proliferation resistance of facilities (Poplavskii 2001). The applicability to spent fuel storage is listed for each attribute.

PR Attribute (Level 1)	**Weight**	**PR Attribute (Level 2)**	**Weight**	**PR Attribute (Level 3)**	**Total Weight**	**Applicable to Spent Fuel Storage?**
Resistance to theft	0.0613	Material characteristics	0.2647	Attractiveness to thieves as estimated by US DOE	0.0120	**No**
				Fissioning material in a different form	0.0042	**No**
		Environment	0.3840	Overall capacity	0.0067	**Yes**
				Production stages	0.0090	**No**
				Maximum plutonium content	0.0013	**No**
				Number of runs	0.0034	**No**
				Safe shipment distance	0.0031	**Yes**
		Guarantees and safety	0.3513	Measurement error	0.0078	**Yes**
				Type of accounting system for nuclear material	0.0063	**Yes**
				Accessibility	0.0062	**Yes**
				American classification	0.0011	**No**

Table 13 Continued.

Resistance to switching	0.1631	Materials characteristics	0.2984	Attractiveness to switching according to IAEA assessment	0.0487	**No**
		Circumstances	0.3598	Overall capacity	0.0207	**Yes**
				Production stages	0.0185	**No**
				Maximum plutonium content	0.0194	**No**
		Guarantees and safety	0.3418	Measurement error	0.0240	**Yes**
				Type of accounting system for nuclear material	0.0192	**Yes**
				Accessibility	0.0075	**Yes**
				International classification	0.0051	**No**
Irreversibility	0.2218	Final form of material			0.0887	**No**
		Final placement of material			0.0887	**No**
		Plutonium residue			0.0444	**No**
International collaboration	0.3316	Collaboration with US and Europe			0.2884	**No**
		Civilian utilization of plutonium			0.0432	**No**
Timeliness	0.2222	Storage time			0.1802	**Yes**
		Completion time			0.0420	**No**

Table 14: Proliferation resistant attributes and weights for methodology to assess the proliferation resistance of facilities (Charlton et al. 2007). The applicability to spent fuel storage is listed for each attribute.

PR Measure	**PR Attribute**	**Weight**	**Applicable to Spent Fuel Storage?**
Attractiveness level	DOE attractivess level (IB - IVE)	0.10	**No**
	Heating rate from Pu in material (W)	0.05	**No**
	Weight fraction of even Pu isotopes	0.06	**No**
Concentration	Concentration (SQs/tonne)	0.10	**No**
Handling requirements	Radiation dose rate (rem/hr at 1 meter)	0.08	**Yes**
	Size/weight	0.06	**No**
Type of accounting system	Frequency of measurement	0.09	**Yes**
	Measurement uncertainty (SQs/year)	0.10	**Yes**
	Separability	0.03	**No**
	Percentage of processing steps that use item counting	0.05	**No**
Accessibility	Probability of unidentified movement	0.07	**Yes**
	Physical barriers	0.10	**Yes**
	Inventory (SQs)	0.05	**Yes**
	Fuel load type (batch or continuous)	0.06	**No**

Table 15: Proliferation resistant attributes and weights for methodology to assess the proliferation resistance of facilities (Petrunin et al. 2008). The applicability to spent fuel storage is listed for each attribute.

PR Attribute	**Barrier**					**Applicable to Spent Fuel Storage?**
	Very Weak	**Weak**	**Average**	**Strong**	**Very Strong**	
Pu-239/Pu (mass %)	>93	80–93	70–80	60–70	<60	**No**
U-235/U (mass %)	>90	50–90	20–50	5–20	<5	**No**
U-238/U-233 contamination (ppm)	<1	1–100	100–4000	4000–7000	>7000	**No**
Material type	Depleted Uranium	Natural Uranium	LEU	Direct use of unirradiated materials	Direct use of irradiated material	**No**
Equivalent dose rate (mSv/h)	<10	10–150	150–1000	1000–10000	>10000	**Yes**
Pu-238/Pu heat release (mass %)	<0.1	0.1–1	1–10	10–80	>80	**No**
Spontaneous neutron radiation: (Pu-240 + Pu-242)/Pu (mass %)	<1	1–10	10–20	20–50	>50	**No**
Fuel assembly mass (kg)	10	10–100	100–500	500–1000	>1000	**No**
Number of fuel assemblies per 1 SQ	1	1–10	10–50	50–100	>100	**No**
Number of SQ in the total fuel flow	>100	50–100	10–50	10–1	<1	**Yes**
Material form: uranium	Metal	Oxide or solution	Compound	Off-loaded fuel	Wastes	**No**
Material form: plutonium	Metal	Oxide or solution	Compound	Off-loaded fuel	Wastes	**No**
Material form: thorium	Metal	Oxide or solution	Compound	Off-loaded fuel	Wastes	**No**

Table 16: Proliferation resistant attributes and weights for methodology to assess the proliferation resistance of facilities (Kwon and Ko 2009). The applicability to spent fuel storage is listed for each attribute.

PR Attribute (Level 1)	**Weight**	**PR Attribute (Level 2)**	**Weight**	**PR Attribute (Level 3)**	**Weight**	**Total Weight**	**Applicable to Spent Fuel Storage?**
NPT system & compliance	0.787	NPT system	0.786	Safeguards	0.241	0.149	**Yes**
				Exports control	0.097	0.060	**Yes**
				Physical protection	0.080	0.049	**Yes**
				Duration of participation	0.582	0.360	**No**
		Compliance	0.214	Domestication	0.548	0.092	**No**
				Transparency	0.241	0.041	**No**
				Diplomatic activities	0.211	0.036	**No**
International politics	0.128	Military threats	0.883	Nuclear threats	0.385	0.044	**No**
				Conventional threats	0.251	0.028	**No**
				Potential threats	0.064	0.007	**No**
				Perceived threats	0.173	0.020	**No**
				Security assurance	0.127	0.014	**No**
		National prestige	0.117	Global power status	0.106	0.002	**No**
				Regional power status	0.261	0.004	**No**
				Pariah status	0.633	0.009	**No**
Domestic politics	0.085	Parochial groups & beliefs	0.856	Decision makers	0.271	0.020	**No**
				Military	0.158	0.011	**No**
				Scientists	0.06	0.004	**No**
				General public	0.111	0.008	**No**
				Political power of parochial groups	0.400	0.029	**No**
		Additional aspects of domestic politics	0.144	Technology	0.106	0.001	**No**
				Economy	0.633	0.008	**No**
				Society	0.261	0.003	**No**

The five methodologies presented all differ in how proliferation resistance is evaluated. The PR attributes in the previous tables can be categorized into four types: diversion (including ease of detection and safeguards), material characteristics, effort to create the weapon, and state level factors. The presented methodologies evaluate a combination of one or more of these types of attributes. The methodologies by (Silvennoinen 1981) and (Poplavskii 2001) focus on the diversion of the material as well as the effort needed to construct the weapon, with some emphasis on material characteristics. On the other hand, the evaluation method by (Petrunin et al. 2008) only takes into account the material characteristics and how they serve as a barrier to making a weapon. The PR assessment methodology by (Charlton et al. 2007) takes into account the material characteristics, as well as attributes related to diversion, such as accessibility and safeguards. The PR attributes in the methodology by (Kwon and Ko 2009) differ from the others in that they focus on state level factors, meaning nonproliferation system and international and domestic politics, in order to evaluate proliferation resistance. (Poplavskii 2001) does also include international collaboration as a PR attribute.

The analyses of the five methodologies in the previous tables also show many PR attributes that are, are not, and could be applicable to spent fuel storage. The types of attributes that are applicable include the accessibility to and safeguards of the spent fuel. Whereas, the material characteristics, state level factors, and effort to make a weapon are not applicable since the material is assumed constant from storage site to storage site. In addition, may attributes could be altered to be applicable for spent fuel. This includes the state-specific attributes, such as in (Kwon and Ko 2009). If the state that the storage site is in were being considered, then these attributes would become important. However, the methodology with the most applicable PR attributes to spent fuel is (Charlton et al. 2007). This methodology seems to place the most value on safeguards. The material characteristics can also be adapted to be applicable to spent fuel storage methods. This methodology will be used in the next section to create a PR assessment methodology specific to spent fuel storage.

METHOD OF ANALYSIS

In the last section of the Literature Review, various proliferation risk/resistance methodologies were described and analyzed. Each methodology was evaluated in terms of how applicable it is to spent fuel storage. There were many proliferation-resistant (PR) attributes from the different methodologies that were not applicable to spent fuel and some that were. From this evaluation, it was determined that the methodology with the most applicable PR attributes was the "Proliferation resistance assessment methodology for nuclear fuel cycles" (Charlton et al. 2007). The following sections will describe proliferation resistant attributes and barriers that will be used the methodology proposed in this thesis.

I. Analysis of Proliferation Resistant Attributes

The method of analysis for determining the proliferation-resistant (PR) value of various spent fuel storage methods is based on multiattribute utility analysis (MAUA), (Clemen 1996), (Charlton et al. 2007). The method used by Charlton et al. is described in the Literature Review. The PR measures, attributes, and associated weights are outlined in Table 17. The applicability to spent fuel storage is determined for each attribute. The primary assumption used to compare the PR values of spent fuel storage is that the material is constant for each storage type. Therefore, the attractiveness level, amount of plutonium, significant quantity concentration, etc., are not applicable. The radiation dose rate, frequency of measurement, measurement uncertainty, probability of unidentified movement, physical barriers, and inventory are the attributes derived from this study that are applicable to spent fuel storage.

An important assumption being made in this assessment is that the material type (spent fuel) is constant for each storage type. This is an important assumption because it eliminates many of the proliferation resistant attributes. For the initial analysis of the spent fuel storage methods, the material in storage should not be considered since the actual storage *method* is going to be analyzed. The material in storage is assumed to be PWR or BWR assemblies with a normal enrichment and burnup of 4% and 45 GWd/tHM, respectively. The fuel is also assumed to be at least five years out of the core since the largest amount of radioactive material decays in this time. Attributes such as material attractiveness, weight, and plutonium concentration will not be considered. The only material-specific attribute that should be accounted for is dose rate from the shielded material. This will be considered because the storage method may be more proliferation-resistant if it does not shield the fuel, thus making it harder to handle.

Also in Table 17, the weights from the applicable attributes are normalized to one to calculate new weights. As can be seen, the most important attributes are measurement uncertainty and physical barriers, each having a weight of 0.21 and. This is an interesting observation because, in terms of spent fuel storage, as measurement uncertainty increases, the physical barriers also increase. For example, spent fuel in the pool is easier to measure with a greater confidence than in dry cask storage. However, dry cask storage provides a greater physical barrier (the cask) than the pool. Overall, the accessibility attributes contribute most to the PR value in terms of spent fuel storage. This is correct because it seems that as spent fuel is transferred to dry cask storage, and then a geological repository, the radiation barriers increase and the accessibility decreases.

Table 17: Measures, attributes, and weights for PR assessment (Charlton et al. 2007) and associated applicability to spent fuel storage with a new, normalized weight.

PR Measure	**PR Attribute**	**Weight**	**Applicable to Spent Fuel Storage?**	**New Weight**
Attractiveness level	DOE attractiveness level (IB - IVE)	0.10	No	N/A
	Heating rate from Pu in material (W)	0.05	No	N/A
	Weight fraction of even Pu isotopes	0.06	No	N/A
Concentration	Concentration (SQs/tonne)	0.10	No	N/A
Handling requirements	Radiation dose rate (rem/hr at 1 meter)	0.08	Yes	0.16
	Size/weight	0.06	No	N/A
Type of accounting system	Frequency of measurement	0.09	Yes	0.18
	Measurement uncertainty (SQs/year)	0.10	Yes	0.21
	Separability	0.03	No	N/A
	Percentage of processing steps that use item counting	0.05	No	N/A
Accessibility	Probability of unidentified movement	0.07	Yes	0.14
	Physical barriers	0.10	Yes	0.21
	Inventory (SQs)	0.05	Yes	0.10
	Fuel load type (batch or continuous)	0.06	No	N/A

1. Intrinsic Proliferation Resistant Attributes

The radiation dose rate, physical barriers, and inventory are considered intrinsic PR attributes because they are specific to the spent fuel storage methods themselves. The utility function for each attribute is described in the following section.

The radiation dose rate attribute is determined as the dose rate concentration, *x*, in rem/h-SQ for the unshielded material. This attribute is actually not directly applicable to spent fuel storage because it would be constant for the unshielded material. Therefore, for this analysis, the definition of this attribute should be for the shielded material in order to distinguish between the different spent fuel storage types. The units for *x* therefore also have to be changed to mrem/hr because the radiation dose rate is significantly lower and specific to the facility and not the amount of material. The utility function for dose rate is given by (Charlton et al. 2007) as

$$u(x) = \begin{cases} 0\,, \; if\ x \leq 0.2, \\ 0.0520833x - 0.010416\,, \; if\ 0.2 \leq x \leq 5, \\ 0.0035714x + 0.232143\,, \; if\ 5 < x \leq 75, \\ 0.00095238x + 0.428571, \; if\ 75 < x \leq 600, \\ 1\,, \; if\ x > 600, \end{cases}$$

The physical barriers to the spent fuel also play a major role in determining the accessibility to the nuclear material. Less accessible material will most likely be less attractive to divert. The utility function for the physical barriers is given in Table 18. In the case of spent fuel storage, the spent fuel pool is considered a vault, dry cask storage is secure, and a geological repository is remote (Charlton et al. 2007). However, it seems that spent fuel in the geological repository would have a much bigger physical barrier than in the pool.

Table 18: Physical barriers utility function (Charlton et al. 2007).

Physical Barrier	Utility Function Value
Inaccessible	1.00
Canyon	0.90
Vault	0.75
Secure	0.50
Remote	0.25
Hands-on	0.00

The final utility function relevant to spent fuel storage is the total inventory of the facility. This utility function is describe as

$$u(x) = \begin{cases} 1\,, \;\; if\ x < 1, \\ \left[\dfrac{(30-x)^{\frac{1}{3}}}{7.18}\right] + 0.574\,, \;\; if\ 1 \le x \le x_{max} \\ 0\,, if\ x > x_{max} \end{cases}$$

where x is the total inventory and x_{max} is the maximum possible inventory, originally set at 100 SQs total of plutonium and uranium (Charlton et al. 2007), where 1 SQ equals either 8 kg of plutonium (all isotopes) or 75 kg of Uranium-235 in LEU. This utility function is used mainly to discriminate between very large and very small facilities. However, in the case of spent fuel storage, all facilities contain greater than 100 SQs so the x_{max} is changed to an arbitrary value of 5000 SQs. This value is allows for the distinction of very large and very small facilities for the inventory of spent fuel storage sites. In order for the equation to be valid, this new maximum must be accounted for by multiplying x by the original x_{max} (100 SQ) divided by the new $x_{max,}$ (5000 SQ). Therefore, the x in the main part of the above equation becomes $x(100/x_{\mathrm{max}})$.

2. Extrinsic Proliferation Resistant Attributes

The frequency of measurement, measurement uncertainty, and probability of unidentified movement are considered extrinsic PR attributes because they have been added to the spent fuel storage method by institutional controls. The utility function for each attribute is described in the following section.

The frequency of measurement utility function defines how frequently the nuclear material is checked. The frequency of measurement can vary from continuous to never. Table 19 shows the utility function value for each frequency of measurement. It is important to note that the frequency of measurement may vary between similar facilities from country to country. Also, does continuous monitoring necessarily include factors like the physical security around a site? Physical security may be breached without notice or by the host country. To distinguish between spent fuel storage sites, the frequency of measurement will only include the actual measurements taken to verify that the spent fuel is actually present. The “probability of unidentified movement” utility function will be presented to take continuous monitoring and other such safeguards measures into account.

Table 19: Frequency of measurement utility function (Charlton et al. 2007).

Frequency of Measurement	Utility Function Value
Continuous	1.00
Hourly	0.95
Daily	0.85
Weekly	0.75
Monthly	0.50
Quarterly	0.25
Annually	0.10
Never	0.00

To coincide with the frequency of measurement, the measurement uncertainty is also important. The utility function for the uncertainty in measurement in the original assessment methodology is found in SQs/year (Charlton et al. 2007). However, as explained later in this chapter, spent fuel does not have an uncertainty in the measurement during verification but rather in the number of assemblies that are actually verified. Therefore, this attribute is represented as the fraction of material that is left unverified after an inspection. The utility function is therefore given by

$$u(x) = 1 - x$$

where *x* is the percent of the total inventory of nuclear material verified per inspection.

As mentioned before, surveillance is an important measure in safeguards and proliferation resistance. Even though containment and surveillance may fail, the presence of it still increases the PR value. Therefore, the probability of unidentified movement is a utility functions which includes the presence of video cameras, automatic bar code readers, global positioning system devices, metal detectors, radiation portable monitors, and other radiation detection equipment. The utility function is given by

$$u(x) = \frac{1}{2} - \frac{1}{2}\tanh(4x - 2)$$

where *x* is the probability that an SQ is moved without the detection of a surveillance system (Charlton et al. 2007). This probability is difficult to define because it depends on many factors. For spent fuel storage, these factors include how many cameras there are and their placement, for example. Also, the probability will be considered in a general case, and not per SQ, because the size of the nuclear material (spent fuel) is considered constant for each storage type.

3. Summary of Proliferation Resistant Attributes

The utility functions described above provide a good foundation for analyzing the PR of spent fuel storage facilities. The main factors in determining PR are the ability to measure and verify the fuel and the disability to access the fuel. The radiation dose rate, physical barriers, and inventory are considered intrinsic attributes of the spent fuel storage facility. Frequency of measurement, measurement uncertainty, and probability of unidentified movement are extrinsic attributes, meaning that they have been added to spent fuel storage method by institutional controls.

Table 20 summarizes each PR attribute and the associated utility function and adjusted weight. The result of each utility function, $u_i(x_i)$, is multiplied by its weight, w_i, and then the products are added to obtain the total proliferation resistance value, *PR*, as shown in the equation below:

$$PR = \sum_{i=1}^{i} w_i u_i(x_i)$$

The total PR value varies from 0 to 1, where 0 means that the spent fuel storage method does not have any proliferation resistant characteristics and 1 denotes that it is very proliferation resistant.

The presence of physical barriers and low measurement uncertainty increase the proliferation resistance of a storage method by the greatest factor, as shown in Table 20. The total inventory, an intrinsic attribute to the storage method, seems to have the least effect on the PR value. When adding the weight values, it can be seen that intrinsic and extrinsic properties have about the same weight, 0.47 and 0.53 respectively.

The next step in the method of analysis is to collect data for each storage method to input into each utility function. The data to be collected are:

- Radiation dose rate (rem/h-SQ)
- Physical barriers
- Inventory in total SQs
- Frequency of measurement
- Measurement uncertainty in percent per inspection
- Probability of unidentified movement

Table 20: Summary of Intrinsic and Extrinsic Proliferation Resistant Attribute Equations

	PR Attribute	Weight	Value of *x*	Utility Function
Intrinsic	**Radiation Dose Rate**	0.16	mrem/hr	$u(x) = \begin{cases} 0, & if\ x \le 0.2, \\ 0.0520833x - 0.010416, & if\ 0.2 \le x \le 5, \\ 0.0035714x + 0.232143, & if\ 5 < x \le 75, \\ 0.00095238x + 0.428571, & if\ 75 < x \le 600, \\ 1, & if\ x > 600, \end{cases}$
	Physical Barriers	0.21	N/A	Inaccessible: 1.00 Canyon: 0.90 Vault: 0.75 Secure: 0.50 Remote: 0.25 Hands-on: 0.00
	Inventory	0.10	Total SQs	$u(x) = \begin{cases} 1, & if\ x < 1, \\ \left[\frac{\left(30 - x\frac{100}{x_{max}}\right)^{\frac{1}{3}}}{7.18}\right] + 0.574, & if\ 1 \le x \le x_{max} \\ 0, & if\ x > x_{max} \end{cases}$
Extrinsic	**Frequency of Measurement**	0.18	N/A	Continuous: 1.00 Monthly: 0.50 Hourly: 0.95 Quarterly: 0.25 Daily: 0.85 Annually: 0.10 Weekly 0.75 Never: 0.00
	Measurement Uncertainty	0.21	Measurement uncertainty	$u(x) = 1 - x$
	Probability of Unidentified Movement	0.14	Probability of unidentified movement	$u(x) = \frac{1}{2} - \frac{1}{2}\tanh(4x - 2)$

II. Proliferation Resistant Attribute Values for Spent Fuel Storage

The following sections will present and discuss the data needed to calculate the proliferation resistant values from the equations in Table 20 on page 84. The data to be collected are radiation dose rate (rem/h-SQ), physical barriers, inventory in total SQs, frequency of measurement, measurement uncertainty in SQs/year and probability of unidentified movement of 1 SQ. The information collected will be used to determine the proliferation resistance value of each spent fuel storage method.

1. Radiation Dose Rate

The radiation dose rate for spent fuel in storage varies widely and is dependent on the original fuel composition and enrichment, burnup, cooling time, and the amount of fuel. Since the type of fuel is not being considered in this analysis, the fuel is assumed to be a normal enrichment and burnup around 4% and 45,000 MWd/tHM, respectively. The fuel has also been out of the core for at least 5 years. Still, it is difficult to obtain an exact value for the dose rate outside of the spent fuel pool or cask.

The fuel assemblies in the spent fuel pool are required to be under at least 20 feet of water. This amount of shielding causes the dose rate immediately above to spent fuel pool to be under 2 mrem/hr in normal conditions (Nuclear Energy Institute 2011). The dose rate at the surface of the pool should always be held under the continuous occupational exposure limit of 2.5 mrem/hr (Saling and Fentiman 2002). Additionally, a spent fuel pool that has more than 25 feet of water above the fuel assemblies has almost a negligible dose rate of less than 1 mrem/hr (Cummings 2010). The NRC does not have a set limit for the radiation dose rate on the surface of a spent fuel pool.

The NRC requires that the annual dose equivalent to an individual outside of the controlled area of the dry cask storage area does not exceed 25 mrem. The NRC does not, however, have dose rate limits inside of the control area because the dose varies on many different factors, such as the geometry of the storage array and the time that employees spend in the controlled cask area. The NRC does regulate the total dose that workers can receive. However, the NRC has accepted specific cask dose rates of 20 to 400 mrem/hour (U.S. NRC 1997). The NRC does specify the dose rate limit on the surface of a transportation cask to be 200 mrem/hour (U.S. NRC 2004). As a real-life example, the Calvert Cliffs Nuclear Power Plant in Maryland uses the NUHOMS 32P dry cask storage system with 32 PWR assemblies in each cask. The dose rate at the surface of the cask is 35 mrem/hour and 12 mrem/hour at 30 cm (Serra 2011).

In the case of a geological repository, the NRC requires that, for a Category 2 design basis event, an individual located outside of the controlled area should not receive more than a total effective dose equivalent of 5 rem (U.S. NRC 1981). For normal operation, the radiation exposure limits are that set by 10CFR20 in which the yearly limit is 5 rem for individual occupational exposure and 0.1 rem for individual public exposure (U.S. NRC 1991). However, the goal for a closed geological repository is to have negligible radiation exposure on the surface of the repository.

2. Physical Barriers

Proliferation resistance decreases as the difficulty in accessing the material decreases. Therefore, a spent fuel storage method with large physical barriers is more proliferation resistant. Table 20 on page 84 shows that the level of physical barrier can be categorized to be, from largest to smallest: inaccessible, canyon, vault, secure, remote, and hands-on.

The spent fuel pool is considered a "vault" facility because a large structure, the reactor containment or auxiliary building, impedes access to the facility. Dry cask storage is a "secure" facility because the material is stored in sealed containers. A geological repository can be considered both a "remote" and a "canyon" facility. A remote facility is inaccessible to the proliferator solely because of its location. This may be true for a geological repository, but the geologic feature would impede access to the material more than the location. Therefore, a geological repository should be considered a "canyon" facility because it is a completely enclosed, underground structure. Table 21 presents the physical barrier type for each spent fuel storage type, along with the corresponding utility function value, as taken from Table 20 on page 84.

Table 21: Physical barrier utility function for spent fuel storage

Storage Type	Physical Barrier	Utility Function Value
Spent Fuel Pool	Vault	0.75
Dry Cask Storage	Secure	0.50
Geological Repository	Canyon	0.90

3. Inventory

The total inventory of a facility is important to consider in PR calculations because a facility with a large inventory is more attractive to proliferators since there is more material and missing material is less likely to be detected. Inventories for spent fuel storage facilities are usually given in number of assemblies. However, the equation for the inventory utility function in Table 20 on page 84 calls for the inventory in the total amount of SQs. As mentioned before, the IAEA measures the amount of special nuclear material in terms of significant quantities (SQs), or the estimated amount of that material theoretically needed to create a nuclear weapon. An SQ of plutonium is 8 kg, and an SQ of low-enriched uranium is 75 kg, measured in terms of the contained U-235 (Doyle 2008).

Table 22 shows the characteristics of typical PWR and BWR fuel assemblies. It is important to know that a typical PWR fuel assembly contains 461 kg of uranium and a BWR assembly has 189 kg (Saling and Fentiman 2002). In order to convert this original uranium mass to SQs in spent fuel, it is assumed that a typical fuel assembly with a burnup of 45,000 MWd/t and original U-235 enrichment of 4% yields 1% U-235 and 1% Pu-239 of the heavy metal in the spent fuel (World Nuclear Association 2009). This gives 4.61 kg each of U-235 and Pu in a PWR spent fuel assembly and 1.89 kg each of U-235 and Pu in a BWR spent fuel assembly. Converting these masses to SQs, there are 0.061 SQs of U-235 and 0.576 SQs of Pu in a PWR spent fuel assembly and 0.025 SQs of U-235 and 0.236 SQs of Pu in a BWR spent fuel assembly. Finally, this calculation shows that there are at total of 0.638 SQs in a typical PWR spent fuel assembly and 0.261 SQs in a typical BWR spent fuel assembly.

Table 22: Typical fuel assembly parameters for PWR and BWR with calculation of total SQs per assembly (Saling and Fentiman 2002), (World Nuclear Association 2009), (Doyle 2008)

	PWR	**BWR**
Fuel Assembly Characteristics		
Total Fuel Pins	264	63
Fuel Pin Array	17x17	8x8
Total Mass (kg)	658	320
Uranium Mass (kg)	461	189
Spent Fuel Parameters		
Burnup (MWd/t)	45000	45000
Original Enrichment of U-235	4%	4%
Fuel Assembly Characteristics After Burnup		
Percent U-235	1%	1%
Percent Pu	1%	1%
U-235 Mass (kg)	4.61	1.89
Pu Mass (kg)	4.61	1.89
SQs of U-235	0.061	0.025
SQs of Pu	0.576	0.236
Total SQs per Assembly	**0.638**	**0.261**

Another important calculation is the conversion from tons of heavy metal (tHM) to significant quantities. As mentioned previously, spent fuel yields about 1% Pu and 1% U-235. In 1 tHM of spent fuel, this equates to 10 kg each of Pu and U-235. Since 1 SQ of Pu is 8 kg, there are about 1.25 SQ of Pu per tHM. Similarly, since 1 SQ of U-235 is 75 kg, there are about 0.133 SQ of U-235 per tHM. Adding these two values yields approximately 1.38 SQ/tHM of spent fuel.

Spent fuel pool inventory largely varies from plant to plant in the United States and in the world. For this analysis, the inventory of the spent fuel storage type will be evaluated as the storage type's capacity. The most recent data available from the IAEA for spent fuel pool capacities around the world is presented in Table 23. This table only includes countries that have mainly Western-design PWR and BWR power plant designs that are operating. The data shows that average capacity in tHM per spent fuel pool varies from country to country, with the United States having the highest average capacity of 555 tHM/pool. The average capacity per pool for all of the selected countries is 388 tHM (International Atomic Energy Agency 1999). Using the conversion of 1.38 SQ/tHM in spent fuel, the average spent fuel pool capacity for a typical PWR or BWR is 536 SQ. In the United States, this is significantly higher, at 767 SQ per pool.

Table 23: Spent fuel pool capacities in countries with typical Western-type PWR and BWR power plants (International Atomic Energy Agency 1999)

Country	Facility Type	Number of Spent Fuel Pools	Total Capacity (tHM)	Average Capacity (tHM)	Average Capactiy (SQ)
France	900 MW PWR	34	5870	173	239
	1300 MW PWR	20	5420	271	375
Germany	PWR	13	3176	244	338
	BWR	6	1385	231	319
Japan	PWR	20	6460	323	447
	BWR	23	8410	366	506
Spain	PWR/BWR	9	3820	424	587
Sweden	PWR/BWR	12	1500	125	173
Switzerland	PWR/BWR	5	705	141	195
United States	PWR/BWR	110	61000	555	767
Total		**252**	**95746**	**388**	**536**

Another set of data for spent fuel pool capacities in the United States is shown in Table 24. The capacity data in assemblies per spent fuel pool is derived from (Bunn et al. 2001). These capacities are calculated to SQ per pool using the reactor type (PWR or BWR) and the typical SQ/assembly values calculated in Table 22 on page 89. The average value of the capacity per spent fuel pool is approximately 943 SQ. This number is significantly higher than that presented in Table 23, or 767 SQ in the United States.

The data in Table 24 originally comes from the NRC in 1998 while the data in Table 23 is from the IAEA in 1997. The IAEA states that, at the end of 1997, the United States had 210,000 assemblies, or 61,000 tHM, of spent fuel pool storage capacity across the 110 reactor sites. In addition, 27 reactors were to run out of spent fuel pool space by 1998 (International Atomic Energy Agency 1999). In 1998, NRC data shows that the spent fuel pool capacity increased to 220,919 assemblies (Bunn et al. 2001). This nearly 11,000 assembly increase may have occurred because many reactors were about to run out of pool space and were required to re-rack the pools in order to accommodate more assemblies until some could be transferred to dry storage.

Table 24: Spent fuel pool capacity for operating nuclear power plants in the United States (Bunn et al. 2001), (U.S. NRC 2011-2012)

Reactor	Type	Capacity (Assemblies)	Capacity (SQ)
Arkansas 1	PWR	968	618
Arkansas 2	PWR	988	630
Beaver Valley 1	PWR	1627	1038
Beaver Valley 2	PWR	1088	694
Braidwood 1	PWR	2870	1831
Braidwood 2	PWR	2027	1293
Browns Ferry 1	BWR	3471	906
Browns Ferry 2	BWR	3133	818
Browns Ferry 3	BWR	2353	614
Brunswick 1	BWR	1767	461
Brunswick 2	BWR	1767	461
Byron 1	PWR	2781	1774
Byron 2	PWR	2026	1293
Callaway	PWR	1340	855
Calvert Cliffs 1	PWR	1830	1168
Calvert Cliffs 2	PWR	2016	1286
Catawba 1	PWR	1418	905
Catawba 2	PWR	1418	905
Clinton	BWR	2515	656
Columbia 2	BWR	2654	693
Comanche Peak 1	PWR	556	355
Comanche Peak 2	PWR	735	469
Cooper	BWR	2366	618
Crystal River 3	PWR	1357	866
Davis-Besse	PWR	718	458
Diablo Canyon 1	PWR	1324	845
Diablo Canyon 2	PWR	1317	840
D.C. Cook 1	PWR	3613	2305
D.C. Cook 2	PWR	2017	1287
Dresden 2	BWR	3537	923
Dresden 3	BWR	3536	923
Duane Arnold	BWR	2411	629
Hatch 1	BWR	5946	1552
Hatch 2	BWR	2018	527
Fermi 2	BWR	2383	622

Table 24: Spent fuel pool capacity for operating nuclear power plants in the United States (Bunn et al. 2001), (U.S. NRC 2011-2012), continued

Reactor	Type	Capacity (Assemblies)	Capacity (SQ)
Fort Calhoun	PWR	1083	691
Grand Gulf 1	BWR	4348	1135
Robinson	PWR	544	347
Hope Creek	BWR	4006	1046
Indian Point 2	PWR	1374	877
Indian Point 3	PWR	1345	858
FitzPatrick	BWR	2797	730
Farley 1	PWR	1407	898
Farley 2	PWR	1407	898
Kewaunee	PWR	990	632
LaSalle 1	BWR	7932	2070
LaSalle 2	BWR	2023	528
Limerick 1	BWR	2832	739
Limerick 2	BWR	3921	1023
McGuire 1	PWR	1351	862
McGuire 2	PWR	1425	909
Millstone 2	PWR	1263	806
Millstone 3	PWR	756	482
Monticello	BWR	2209	577
Nine Mile Point 1	BWR	2776	725
Nine Mile Point 2	BWR	4049	1057
North Anna 1	PWR	1737	1108
North Anna 2	PWR	2020	1289
Oconee 1	PWR	1312	837
Oconee 2	PWR	1312	837
Oconee 3	PWR	825	526
Oyster Creek	BWR	2645	690
Palisades	PWR	771	492
Palo Verde 1	PWR	1205	769
Palo Verde 2	PWR	1205	769
Palo Verde 3	PWR	1205	769
Peach Bottom 2	BWR	3819	997
Peach Bottom 3	BWR	3819	997
Perry 1	BWR	4020	1049
Pilgrim	BWR	3859	1007

Table 24: Spent fuel pool capacity for operating nuclear power plants in the United States (Bunn et al. 2001), (U.S. NRC 2011-2012), continued

Reactor	Type	Capacity (Assemblies)	Capacity (SQ)
Point Beach 1	PWR	1502	958
Point Beach 2	PWR	2013	1284
Prairie Island 1	PWR	1386	884
Prairie Island 2	PWR	2014	1285
Quad Cities 1	BWR	3657	954
Quad Cities 2	BWR	3897	1017
River Bend	BWR	2680	699
Ginna	PWR	1879	1199
St. Lucie 1	PWR	1706	1088
St. Lucie 2	PWR	1076	686
Salem 1	PWR	1632	1041
Salem 2	PWR	1632	1041
San Onofre 2	PWR	1542	984
San Onofre 3	PWR	1542	984
Seabrook	PWR	1236	789
Sequoyah 1	PWR	2091	1334
Sequoyah 2	PWR	2021	1289
Shearon Harris 1	PWR	4184	2669
South Texas 1	PWR	1969	1256
South Texas 2	PWR	1969	1256
Surry 1	PWR	1044	666
Surry 2	PWR	2013	1284
Susquehanna 1	BWR	2840	741
Susquehanna 2	BWR	2840	741
Three Mile Island	PWR	1338	854
Turkey Point 3	PWR	1395	890
Turkey Point 4	PWR	1389	886
Vermont Yankee	BWR	2863	747
Summer	PWR	1276	814
Vogtle 1	PWR	1475	941
Vogtle 2	PWR	1998	1275
Waterford 3	PWR	2398	1530
Watts Bar 1	PWR	1612	1028
Wolf Creek	PWR	1327	847
Average Capacity		**2124**	**943**

The World Nuclear Association estimates that there are 270,000 metric tons of spent fuel in storage around the world, 90% of which is in spent fuel pools and the remaining 10% in dry storage (World Nuclear Association 2011). Since this number is in metric tons, and not tHM, it can be assumed that roughly 65% of the mass of the spent fuel bundle is heavy metal. This assumption is made from the typical parameters of fuel assemblies given in Table 22 on page 89. Therefore, there are approximately 158,000 tHM (218,000 SQ) in pool storage and 17,500 tHM (24,200 SQ) in dry storage around the world. Since there are 432 commercial reactors operating in the world (World Nuclear Association 2011), it can be assumed that there are also approximately 432 spent fuel pools storing spent fuel. This gives a rough estimate of a world average of spent fuel pool inventory of 500 SQ, which is close to the 536 SQ calculated in Table 23 on page 90.

The capacity for dry cask storage facilities also varies from plant to plant and across the United States and the world. In the United States, dry casks are placed on licensed sites termed independent spent fuel storage installations (ISFSIs). As of 2011, there are 63 licensed ISFSIs storing 1,220 loaded dry casks. Each cask holds up to 40 PWR assemblies or up to 68 BWR assemblies (U.S. NRC 2011-2012).

Since many dry cask storage facilities are still under construction or gaining capacity, current data on dry cask storage facilities is difficult to aggregate. As of 1999, the United States had the most dry cask storage facilities across different sites, as seen in Table 25. The design capacity for each site varies, with the average being 454 tHM or 626 SQ (International Atomic Energy Agency 1999).

Table 25: Spent fuel capacity for dry cask storage facilities in the United States (International Atomic Energy Agency 1999)

Dry Cask Storage Facility	**Type**	**Design Capacity (tHM)**
Arkansas Nuclear	PWR	150
Dresden 1	BWR	70
North Anna	PWR	840
Palisades	PWR	233
Point Beach	PWR	447
Prairie Island	PWR	724
Surry	PWR	808
Trojan	PWR	358.9
Average Capacity (tHM)		**454**
Average Capacity (SQ)		**626**

Another estimate of spent fuel pool and dry cask inventory in the United States can be gained from data indicating there were approximately 65,000 tHM of spent fuel stored in the United States at the end of 2010. 75% of this fuel was stored across 100 spent fuel pools and 25% across 63 dry cask storage sites. Using these totals and the conversion of tHM to SQ, it can be estimated that spent fuel pools in the United States contain an average of about 670 SQ and dry cask storage sites contain an average of about 360 SQ (Alvarez 2011). These values are lower than those previously obtained because they represent the actual inventory of the storage types rather than the capacity.

In the case of spent fuel storage or disposal in a geological repository, the proposed Yucca Mountain Nuclear Waste Repository in the United States will be used as an example. Although the NRC license application for this geological repository has been withdrawn, the design information is still available to be used as an example. Yucca Mountain was proposed to hold both civilian and defense radioactive waste. The repository should hold 63,000 tHM of civilian radioactive waste in the form of spent fuel.

This amounts to approximately 292,000 spent fuel assemblies: 167,000 BWR assemblies and 125,000 PWR assemblies (U.S. DOE 2008). Using the previously calculated conversion of 1.383 SQ/tHM in spent fuel, the proposed inventory of approximately 87,000 SQ will be used for evaluating the geological repository.

Since the retraction of the Yucca Mountain licensing application, Finland is the furthest along in siting and building the first commercial geological repository for spent fuel. The facility in Finland will be near the Olkiluoto nuclear power plant and contain 9000 tonnes of spent fuel. Sweden is also siting a geological repository that will have a capacity of 12,000 tonnes (World Nuclear Association 2011). Since these numbers are in metric tons, and not tHM or SQ, it can be assumed that roughly 65% of the mass of the spent fuel bundle is heavy metal and that 1.383 SQ/tHM are in typical spent fuel. Therefore, the geological repositories in Finland and Sweden will contain approximately 8000 SQ and 11,000 SQ, respectively. Besides the United States, Finland, and Sweden, no other countries that are planning to directly dispose of their spent fuel have been ambitious in planning and designing their repositories. Thus no other numbers for capacity are available. Therefore, a European geological repository capacity average of 9500 SQ for Finland and Sweden will be used. This is likely to be the size of the repository for other countries in Europe since many of them typically have only a few reactors, far fewer than the United States. However, a larger repository in Europe may need to be under safeguards if several countries agree to have a single, shared repository.

Due to the slow pace of many countries making concrete policies, the use of large interim storage facilities has become more popular. For example, The Rokkasho complex in Japan is designed to hold 20,400 tonnes of fuel before it is reprocessed. South Korea is also planning to have a 20,000 tonne centralized interim storage facility before a more permanent policy for spent fuel is made (World Nuclear Association 2011). Taking the average of these two capacities and using the conversions detailed previously, interim storage facilities in preparation for reprocessing have an approximate capacity of 18,000 SQ. This number will be considered in the dry cask storage scenarios since a centralized interim storage facility has a similar design to a dry cask storage facility at a reactor site.

The data collected above for the estimated average inventory of spent fuel storage types can now be used in the intrinsic PR analysis. Due to the varying data, five scenarios for spent fuel pool storage capacity will be considered: the low 2011 world average of 500 SQ, the medium 1998 world average 536 SQ, the medium 1997 U.S. average of 767 SQ, the high 1998 U.S. average of 943 SQ, and finally the low 2010 U.S. estimation of 670 SQ. The average 1998 U.S. dry cask storage site capacity of 626 SQ will be used along with the 2010 U.S. estimation of 360 SQ. Also, the planned capacity of centralized interim storage in Asia (18,000 SQ) will be modeled. Finally, the 87,000 SQ capacity of the proposed U.S. geological storage facility will be considered along with the European estimate of 9,500 SQ. These values are summarized in Table 26. As can be seen, the geological repositories and Asian centralized facilities have a significantly higher inventory than the spent fuel pool or dry cask storage, which are all under 1000 SQ.

Table 26: Summary of estimated average inventory for spent fuel storage types

Spent Fuel Storage Type	Region and Year of Data	Average Estimated Inventory
Spent Fuel Pool	World 1998	536 SQ
	World 2011	500 SQ
	U.S. 1997	767 SQ
	U.S. 1998	943 SQ
	U.S. 2010	670 SQ
Dry Cask Storage	U.S. 1998	626 SQ
	U.S. 2010	360 SQ
	Asia 2011	18,000 SQ
Geological Repository	U.S. 2011	87,000 SQ
	Europe 2011	9,500 SQ

4. Frequency of Measurement

The frequency of measurement of the special nuclear material in a storage site is an extrinsic PR attribute. For this analysis, the frequency of measurement can be continuous, hourly, daily, weekly, monthly, quarterly, annually, and never. Continuous measurement has the highest value and therefore is the most proliferation resistant, while no measurement is obviously the least proliferation resistant.

The measurement, in the form of an inspection, is performed by IAEA inspectors. During an inspection, IAEA inspectors count the material, verify the integrity of seals, and verify random samples from the material. Since spent fuel storage is an item facility, accounting for the nuclear material can simply be done by counting assemblies or casks. However, it is also important to verify, by radiation measurement, a random selection of assemblies in order to ensure that the assemblies are indeed the nuclear material that they are claimed to be (Doyle 2008).

IAEA safeguards inspectors lay out an inspection schedule for a material balance period of a facility. This period is the time between Physical Inventory Verification (PIV) inspections, usually one calendar year and no more than 14 months long. The IAEA has established conversion times from estimates of the time to convert different forms of nuclear materials into nuclear materials usable for weapons. These conversion times give the IAEA timeliness goals for inspection. Because plutonium is contained in spent fuel, a few months is considered the conversion time for usable material, giving the IAEA a timeliness detection goal of 3 months (Doyle 2008).

The IAEA works with the facility to set up an effective safeguards system. The facility utilizes a material accounting system on the state level, called the States' Systems of Accounting and Control (SSAC). An effective SSAC assists in the implementation of IAEA safeguards. The IAEA considers three basic diversion scenarios for spent fuel at an LWR:

1. Spent fuel assembly diversion by substitution of a dummy element for actual element
2. Spent fuel pin diversion by substitution of a dummy element for an actual element
3. Unreported plutonium production by the insertion of fertile targets for irradiation in core fuel (PWR guide tubes or burnable poison rod)

These diversions can occur in the reactor pool, spent fuel pool, or spent fuel transfer cask. The IAEA therefore sets up a material balance area (MBA) with key measurement points (KMP) around areas at the LWR facility containing nuclear material. The spent fuel pool itself is usually considered as an MBA (Doyle 2008).

Due to the 3-month timeliness goal of spent fuel, an LWR is inspected on a quarterly basis. This includes a yearly PIV and three interim inspections under the traditional INFCIRC/153 safeguards agreement. IAEA inspectors perform three basic activities to verify the operator's declarations:

1. Check reactor's nuclear material accounting and operating records
2. Verify material itself by visual and non-destructive assay (NDA) techniques
3. Use C/S to check that the nuclear material is not being diverted

In the case of the spent fuel pool, the inspector is given a map of the pool charting where each assembly is. A random sample of assemblies is also verified using NDA techniques. The random sampling plan is designed to give the appropriate confidence level for the desired probability of detection (Doyle 2008).

Since the dry cask storage sites considered in this analysis are at the reactor site, it can be assumed that these sites are also inspected at the same time as the spent fuel pool, on a quarterly basis. While casks can be counted, the spent fuel assemblies inside cannot be. Therefore, dry cask storage safeguards depends heavily on containment and surveillance. This will be taken into account in the "probability of unidentified movement" PR attribute.

Since an operational commercial geological repository does not exist yet, the IAEA has not determined how often the site will be inspected. Since spent fuel will be stored in the repository, the same timeliness goal of 3 months used for the spent fuel pool will likely be used for the repository. Therefore, the operational repository should be inspected quarterly. However, the closed repository will not be inspected and thus the frequency of measurement will be considered as "never."

5. Measurement Uncertainty

The measurement uncertainty for this analysis should be found in a fraction of the total inventory instead of SQs/year as originally used by this method. The original proliferation assessment methodology also assumes that there is no measurement uncertainty for material that can be accounted for using item counting (Charlton et al. 2007). However, even though item counting has no measurement uncertainty, verification of random samples of the nuclear material does. As noted previously, spent fuel, especially in spent fuels pools, is verified to indeed be spent fuel by various measures. This actual measurement has no uncertainty because it yields "yes" or "no" that the item is spent fuel. However, the IAEA only verifies a certain number of fuel assemblies in a facility, which is where measurement uncertainty is presented. Those assemblies that are not verified could actually be dummy assemblies instead of spent fuel.

At an item facility, the IAEA chooses a certain number of items to be verified, *n*, as a sample out of the total amount of items, *N*, using the following equation:

$$n = N\,(1 - \beta^{x/M})$$

where β is the nondetection probability, *M* is the goal amount, and *x* is the average nuclear material weight of an item (Doyle 2008). In the case of a spent fuel storage facility, such as the spent fuel pool, N is the total inventory of the facility. In this case, both the sample amount, *n*, and inventory, *N*, will be in SQs since the measurement uncertainty is quantified in SQs/year. The nondetection probability, β, is the confidence level that the IAEA has in the facility that it is not diverting nuclear material. For these calculations, a medium confidence level of 50%, or β equals 0.5, will be used. In the case of spent fuel assemblies, the goal quantity, M, is 1 SQ or 8 kg of plutonium. (Pu contributes significantly more to the SQ amount in a spent fuel assembly than U-235.) Finally, *x* is the average nuclear material, or plutonium, weight of a spent fuel assembly.

The average masses of plutonium in PWR and BWR assemblies are calculated in Table 22 on page 89 as 4.61 and 1.89 kg, respectively. Since the spent fuel facilities considered in this evaluation can contain either PWR or BWR assemblies, the two values yield an average of 3.25 kg of Pu per assembly. Using this value in the equation above gives that *n/N*, or the fraction of material to be verified, is equal to 0.25. This means that 25% of the assemblies must be verified in order to have a confidence level of 50% in the facility. For a low facility confidence level of 10% (β = 0.1), 61% of the material must be verified, and for a high facility confidence level of 90% (β = 0.9), 4% of the material must be verified.

If this fraction *n/N* of material to be verified is multiplied by the total inventory of the facility to obtain the measurement uncertainty in SQ/year, then the amounts are much higher than 1 SQ. For this reason, the utility function is changed to $u(x) = 1 - x$, where *x* is the measurement uncertainty, or the fraction of material that is not verified in the facility per inspection (1 – *n/N*). In order words, $u(x) = n/N$. Therefore, in the case of the spent fuel pool, the utility function is equal to 0.25 for a medium confidence level where *n/N* = 0.25. As discussed in the Literature Review, the IAEA currently does not have a robust method to verify PWR and BWR assemblies that are inside of a cask whether at a dry storage facility or in a geological repository. For this reason, the measurement uncertainty for these facilities is 100%, making the utility function equal to zero.

6. Probability of Unidentified Movement

The IAEA relies heavily on the use of unattended monitoring systems (UMS) to complement other forms of safeguards measures at nuclear material facilities. The goal of these systems is to maintain the continuity of knowledge (CofK) of the nuclear material flow and storage. The system should never lose safeguards significant data under even the most challenging infrastructure and operational environments. The UMS automatically monitors nuclear material using sensors 24 hours a day and 365 days per year. In the case of spent fuel storage, these sensors are mostly optical, i.e. cameras. Radiation sensors can also be used in spent fuel transfers or to trigger cameras (Doyle 2008).

In addition to not losing safeguards significant data, the UMS must also assure the authenticity of the data. This is accomplished by sealing the equipment and wires and also by encryption. The system must also not interfere with the normal facility operations or safety features. Independent redundant components are used to monitor the same event or area to prevent the loss of data in case of a failure. In the case of spent fuel storage, this can be done by having two or more cameras monitoring the storage area (Doyle 2008).

In order to determine the probability of unidentified movement, a detailed vulnerability study of a specific facility must be completed. For a general case, this is not possible to do. In the proliferation resistance assessment methodology developed by

(Charlton et al. 2007), this utility function is set to unity for a hypothetical case in order to not affect the relative comparisons. However, the methods used for C/S vary for each spent fuel type. Surveillance by dual cameras can be used for the spent fuel pool, dry cask storage site, and opening of the geological repository during operation. However, the level of confidence is increased for the spent fuel pool because the cameras are actually viewing the spent fuel assemblies themselves. The surveillance in a dry cask storage facility is monitoring the actual casks. In the case of a geological repository, the confidence in surveillance is quite low for both the open and closed stages. In the open stage, the cameras monitor the opening for unauthorized movement. Constant surveillance is absent in the case of the closed repository and may only be done by satellite or seismic monitoring for unauthorized movement. However, satellite monitoring would only be periodic and seismic monitoring would only detect large explosions that would open another entrance to the repository. Finally, cameras and seals may be deliberately altered by the state or adversary and not detected by the IAEA

All of the factors discussed above contribute to the probability of unidentified movement for each storage type. The IAEA does not provide risk assessments of or probabilities for the state or adversary being able to evade IAEA detection for the unauthorized movement of nuclear material. Therefore, a ranking system, similar to that used in the physical barriers and frequency of measurement attributes, is used for this attribute. In this case, there seem to be five stages in the confidence of the ability to catch unauthorized movement of spent fuel in storage, as described in Table 27. The confidence level varies according to the level of surveillance of the nuclear material and the ability to alter the C/S system.

Table 27: Probability of unidentified movement values for varying levels of confidence of C/S system

Confidence Level	Description	Probability of Unidentified Movement
100%	Complete surveillance of spent fuel Alteration of C/S system impossible	0.00
75%	Complete surveillance of nuclear material Alteration of C/S system possible	0.25
50%	Complete surveillance of facility storing nuclear material Alteration of C/S system possible	0.50
25%	Partial surveillance of facility storing nuclear material	0.75
0%	No surveillance of facility storing nuclear material	1.00

A spent fuel pool allows for complete surveillance of the nuclear material but the alteration of the C/S system is possible. Alteration of C/S is also possible for a dry cask facility and the surveillance can only monitor the casks and not the actual nuclear material. Finally, only partial surveillance of the geological repository is possible, which makes the ability to alter the C/S irrelevant. Therefore, the probability of unidentified movement for the spent fuel pool, dry cask storage, and geological repository facilities are 0.25, 0.50, and 0.75, respectively.

7. Summary of Intrinsic and Extrinsic PR Attribute Values

The above sections describe in detail the values that will be used for each spent fuel storage type to input into the utility function for the proliferation resistance assessment. These values are summarized in Table 29. These values are the *x* values that will be entered into the utility functions in Table 20 on page 84. The proliferation resistance methodology uses the equations from Table 20 and the values from Table 29 in an Excel spreadsheet that gives the output for the intrinsic PR value, extrinsic PR value, and total PR Value. As described previously, the *x* value is entered into the utility function and then multiplied by the weight of the attribute, these values are then simply added, as viewed in Table 28. The total PR value will vary from 0 to 1, with a larger number being more proliferation resistant. These values will be determined for various cases and assessed in the next section.

Table 28: Representation of Excel spreadsheet for the calculation of PR values

PR Attribute	**Weight**	**Value of *x***	**Utility Function**	**Weighted Attribute**	**PR Value**
Radiation Dose Rate	0.16	*Input*	*Calculate*	*Calculate*	*Sum of Intrinsic Values*
Physical Barriers	0.21	*Input*	*Calculate*	*Calculate*	
Inventory	0.10	*Input*	*Calculate*	*Calculate*	
Frequency of Measurement	0.18	*Input*	*Calculate*	*Calculate*	*Sum of Extrinsic Values*
Measurement Uncertainty	0.21	*Input*	*Calculate*	*Calculate*	
Probability of Unidentified Movement	0.14	*Input*	*Calculate*	*Calculate*	
					Total PR Value

Table 29: Summary of Intrinsic and Extrinsic PR Attribute Values (*x* values for utility functions in Table 20)

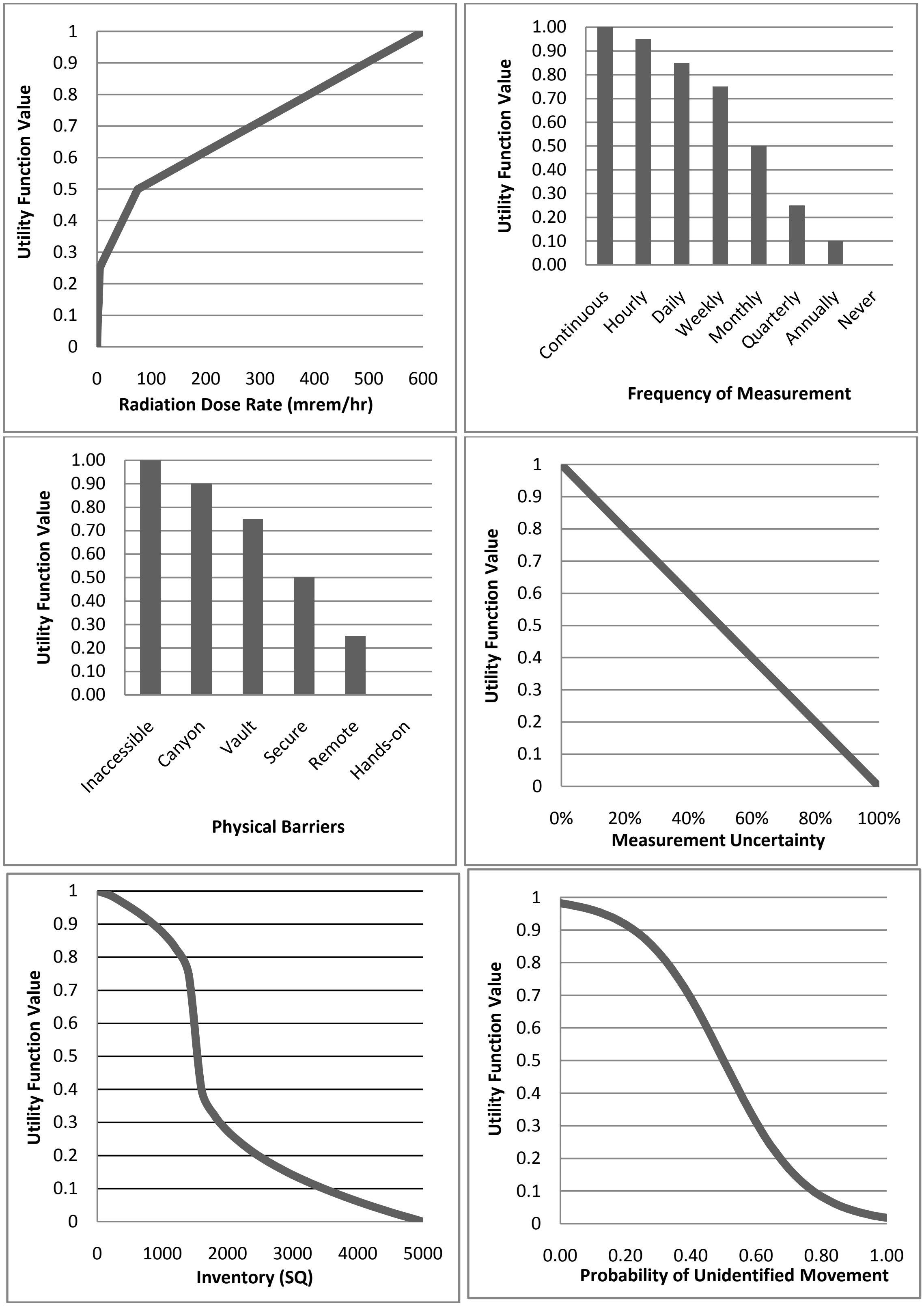
	PR Attribute	Spent Fuel Pool	Dry Cask Storage	Geological Repository
Intrinsic	**Radiation Dose Rate**	<1 – 2.5 mrem/hr	20 – 400 mrem/hr	Open: 200 mrem/hr Closed: 0 mrem/hr
	Physical Barriers	Vault (0.75)	Secure (0.50)	Canyon (0.90)
	Inventory	World 1998 – 536 SQ World 2011 – 500 SQ U.S. 1997 – 767 SQ U.S. 1998 – 943 SQ U.S. 2010 – 670 SQ	U.S. 1998 – 626 SQ U.S. 2010 – 360 SQ Asia 2011 – 18,000 SQ	U.S. 2011 – 87,000 SQ Europe 2011 – 9,500 SQ
Extrinsic	**Frequency of Measurement**	Quarterly (0.25)	Quarterly (0.25)	Open: Quarterly (0.25) Closed: Never (0.00)
	Measurement Uncertainty	High Confidence: 96% Medium Confidence: 75% Low Confidence: 39%	100%	100%
	Probability of Unidentified Movement	0.25	0.50	0.75

RESULTS & ANALYSIS

Using the proliferation resistance assessment method detailed in the Method of Analysis, the different methods of storing spent fuel will be examined based on proliferation resistance (PR) values. While different inputs for some of the PR attributes were found, the first assessment will be a normal case involving the most credible inputs. Afterwards, the effects of various inputs into certain PR attributes will be examined. Finally, an assessment of each PR attribute's affect on the overall PR value will be presented.

As a review, Figure 3 on the next page gives a visual representation of each PR attribute in terms of the input versus the utility function value. Each utility function value is between 0 and 1 and then multiplied by the weight of the factor to contribute to the total PR value, a value also between 0 and 1, where a higher number means more proliferation resistant. The exact equations for each factor can be found in Table 4 of the Method of Analysis.

Figure 3: Visual representation of each utility function value versus input.



I. PR Assessment of Normal Case

In the Method of Analysis, various input values were found for each PR attribute for each storage method. For some factors, more than one input value was found. In this "normal case" analysis, the input value with the most credibility (i.e. most supporting data available) will be used. For the radiation dose rate, 2 mrem/hr is considered average for the spent fuel pool, as stated by the Nuclear Energy Institute. Since the Nuclear Regulatory Commission does not regulate the dose rate limit directly outside of a dry storage cask, the 200 mrem/hr limit for the transportation cask is used. Since transportation casks are also used for final disposal, the same value is used for the geological repository. In addition, the geological repository is evaluated for the operating, or "open," phase for this normal case analysis.

There were also many different values found for the inventory of all three storage types. The United States has the most data available for estimating inventories in SQs, and those numbers will therefore be used. The U.S. data in 1998 detailed the inventory of each pool and dry cask storage site, with averages of 943 SQ and 626 SQ respectively. The United States was also the furthest along in designing a repository at a particular site with a proposed inventory of approximately 87,000 SQ. Finally, different values for the confidence level for the spent fuel pool were found. The medium confidence level (measurement uncertainty of 75%) will be used for this normal case.

The values from Table 29 in the Method of Analysis, with the clarifications above, serve as the inputs to utility functions. The utility functions are multiplied by their respective weights to add to the total PR value, as shown in Table 30. The input, or value of x, is shown along with the calculated PR values. The three different spent fuel storage types all have relatively similar total intrinsic PR values. However, they differ greatly in the total extrinsic PR values. The main difference is that the spent fuel pool has a lower measurement uncertainty, and thus a higher PR value, because spent fuel assemblies are actually verified to be there but not in the dry cask storage facility or the geological repository. In addition, the spent fuel pool and dry cask storage have a higher value in the probability of unidentified movement. From the total PR value, it can be seen that the spent fuel pool ranks first in proliferation resistance while the geological repository ranks last. This first observation will now lead to the examination of what factors influence the PR value the most in terms of spent fuel storage. This examination is important in order to be able to make recommendations for improving the proliferation resistance of spent fuel storage.

Table 30: PR values for normal case of spent fuel storage methods.

PR Attribute	Weight	Value of *x*			PR Value		
		Spent Fuel Pool	**Dry Cask Storage**	**Open Geological Repository**	**Spent Fuel Pool**	**Dry Cask Storage**	**Open Geological Repository**
Radiation Dose Rate	0.16	2	200	200	0.02	0.10	0.10
Physical Barriers	0.21	Vault	Secure	Canyon	0.16	0.11	0.19
Inventory	0.10	943	626	87000	0.09	0.09	0.00
Frequency of Measurement	0.18	Quarterly	Quarterly	Quarterly	0.05	0.05	0.05
Measurement Uncertainty	0.21	75%	100%	100%	0.05	0.00	0.00
Probability of Unidentified Movement	0.14	0.25	0.50	0.75	0.12	0.07	0.02
Intrinsic PR Value (max 0.47)					**0.27**	**0.30**	**0.29**
Extrinsic PR Value (max 0.53)					**0.22**	**0.12**	**0.07**
Total PR Value (max 1.00)					**0.49**	**0.42**	**0.36**

II. Assessment of Intrinsic PR Attributes

In the previous section, the normal case for the proliferation resistance of spent fuel storage was presented. While the total intrinsic PR values for the various types of storage only differed slightly, the PR values for the individual intrinsic attributes did vary significantly in some cases. The following sections include an analysis of how each intrinsic attribute affects the total PR value.

1. Radiation Dose Rate

In the Method of Analysis, a range of values was found for the dose rates of all three of the spent fuel storage types. Figure 4 shows the total PR value as a function of dose rate, with all other values in Table 30 held constant. Even though the spent fuel pool's low dose rate makes it less proliferation resistant, it still has a much higher PR value than dry cask storage or the geological repository. The NRC has accepted dose rates of 20 to 400 mrem/hr for dry cask storage licenses, as shown in Figure 4. In the normal case, the NRC transport cask dose rate limit of 200 mrem/hr was used for both dry cask storage and geological repository. Increasing this dose rate to 400 mrem/hr would not significantly increase the total PR value. Therefore, the dose rate limit for transport casks should not be increased because there would be little effect on the proliferation resistance but likely a larger effect on worker and public safety.

It is also important to note, as shown in Figure 4, that the dose rate limit for a closed repository is 0 mrem/hr, as mandated by the NRC. According to the radiation dose rate attribute, a closed repository would have a PR value of only 0.25, a 30% decrease from the normal case. The PR value of the closed repository will be further examined in the next section.

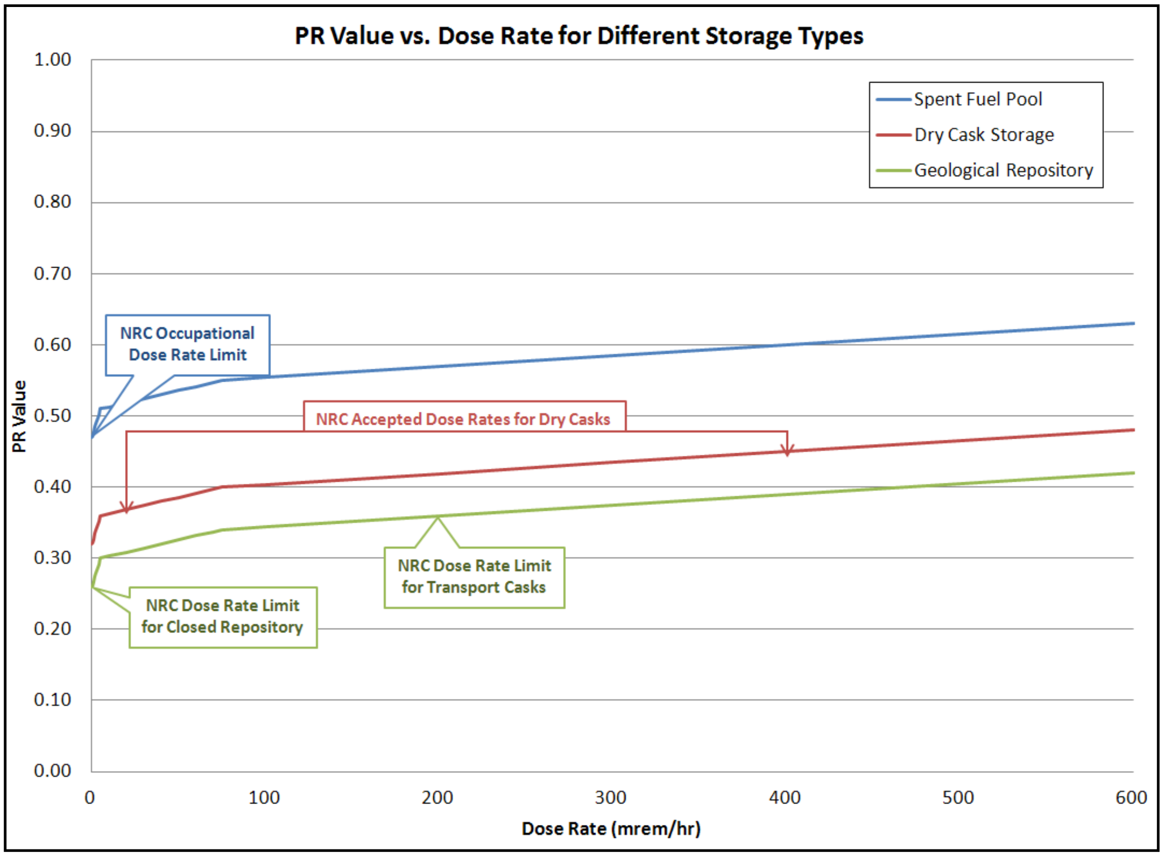


Figure 4: Total PR value as a function of dose rate.

2. Physical Barriers

Unlike the radiation dose rate attribute, the physical barriers attribute is inherent to the storage type and therefore cannot vary. To review, the physical barrier to the nuclear material is determined by the type of facility. The spent fuel pool is a vault, the open geological repository is a canyon, and dry cask storage is a secure facility. The geological repository is generally placed in a landform, such as a mountain, and therefore has a larger physical barrier than the other two types of storage, thus increasing its proliferation resistance. Because the spent fuel pool is inside of a building, it has a higher

PR value for the physical barriers attribute than dry cask storage. If the dry cask storage facility were to be placed in a secure building and thus also be considered a vault facility, the PR value would increase from 0.42 to 0.47. This is about a 12% increase in the proliferation resistance and makes the PR value quite close to that of spent fuel storage (0.49), so placing the dry cask storage facility in a large secure structure could be beneficial.

It is also important to note that a closed geological repository would technically be an “inaccessible” facility. While this would increase the proliferation resistance, as discussed in the previous section, there would also be no radiation dose outside of the facility, thus decreasing the proliferation resistance. With these factors in mind, the PR value of a closed repository would be 0.28, which is a 22% decrease from the open repository PR value of 0.36. The PR value of the closed repository will be further discussed while analyzing extrinsic attributes.

3. Inventory

In the Method of Analysis, the values found for the inventory of the different spent fuel storage types varied greatly. Since the most accurate data was found from the United States, those values were used in the normal case. However, the inventories for U.S. spent fuel storage facilities are also generally larger than the rest of the world because the United States has the largest nuclear energy industry. The United States has many power plants that have been online for a long time; thus more spent fuel storage capacity is needed. Because the inventories would be smaller in other countries, the PR values for the storage types would be larger since the facilities would be less attractive for nuclear material diversion.

Figure 5 shows the total PR value as a function of inventory, with all other values in Table 30 held constant. The data points found for average inventories in the United States and for the world are noted in the figure. Because spent fuel pool inventories are relatively small, the total PR value does not vary significantly for the noted data points. The U.S. 1998 average of 943 SQ was used for the normal case, yielding a PR value of 0.49. Since the inventory of many spent fuel pools has decreased in the past decade due to the utilization of dry cask storage, the U.S. average of spent fuel pool inventory for 2010 decreased to 670 SQ. Calculating with this value would not change the PR value. The average estimate for spent fuel pool inventory for the world in 2011 was found to be even lower at 500 SQ. Calculating with this new inventory value would actually be quite insignificant and increase the total PR value by only 0.01. The spent fuel inventory would have to be 1150 SQ or higher to decrease the PR value by 0.01. Therefore, variation in the spent fuel pool inventory does not make a significant difference in the total PR value calculation.

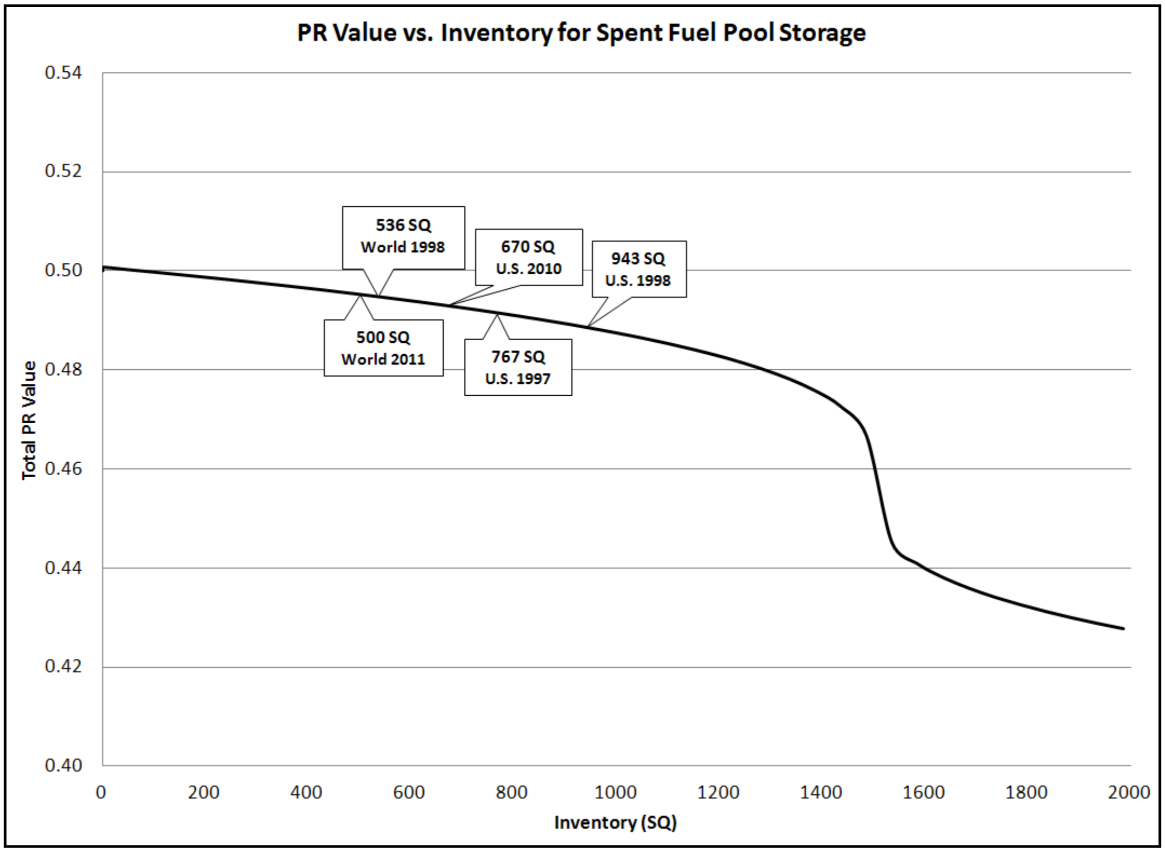


Figure 5: Total PR value as a function of inventory for spent fuel pool storage.

Figure 6 shows the total PR value as a function of inventory for dry cask storage, with all other values in Table 30 held constant. The data points found for average inventories in the United States and for Asia are also noted in Figure 6. The 1998 U.S. average of 626 SQ was used for the normal case and yielded a PR value of 0.42. U.S. data from 2010 gave an average dry cask storage inventory of only 360 SQ. This number is lower than the 1998 average because it accounts for the actual inventory, and not capacity, of the facilities. Many of the dry cask storage facilities are actively being filled, thus increasing the inventory. Calculating the PR value with 360 SQ would not change the PR value. Therefore, similar to spent fuel pools, the variation in the inventory of dry cask storage does not have a significant effect on the total PR value unless the storage facility is very large, as in the case for centralized interim storage.

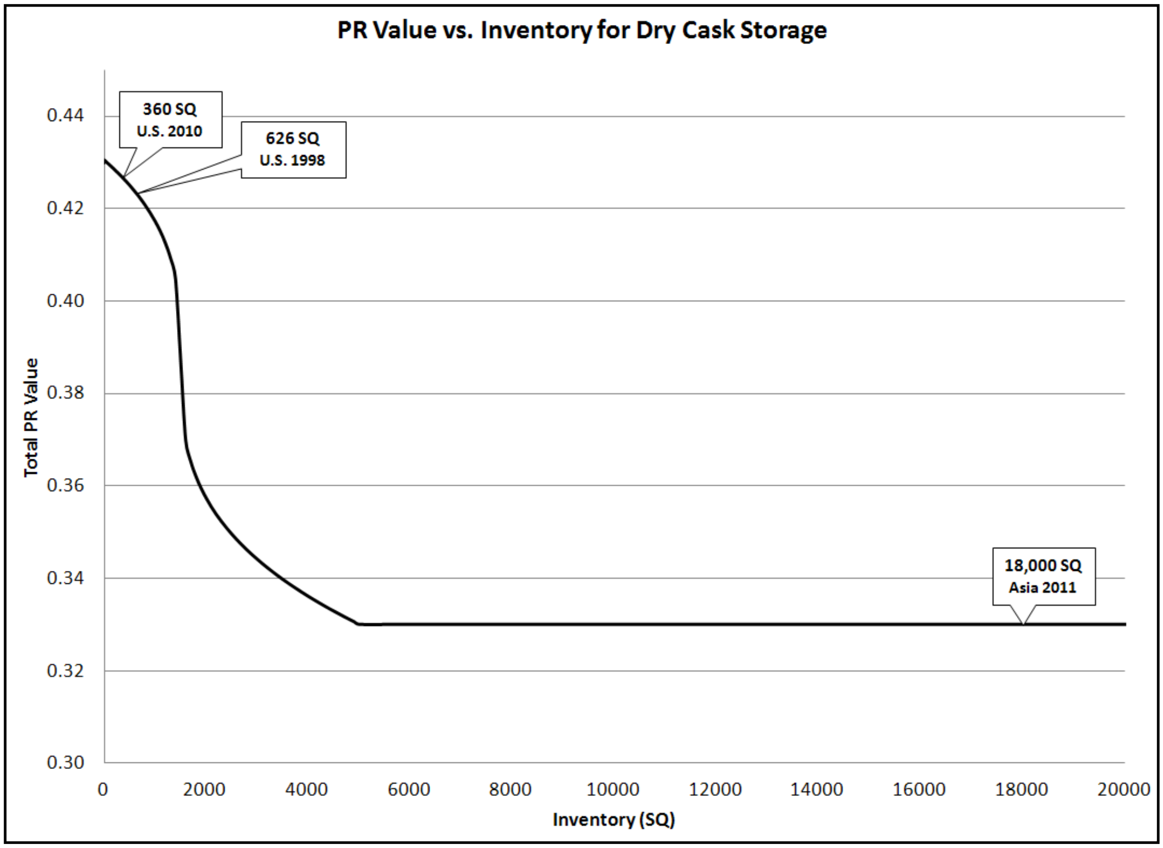


Figure 6: Total PR value as a function of inventory for dry cask storage.

The total PR value is significantly affected if the dry cask storage facility is a centralized facility with a much larger inventory. For example, the centralized interim dry storage facility in operation in Japan and the one planned in South Korea will each hold about 18,000 SQ. Using this inventory value, the total PR value would be 0.33, which is a 21% decrease from the normal case of 0.42. Thus a centralized facility may be less proliferation resistant than smaller on-site dry storage facilities, unless extrinsic factors, such as safeguards, are improved.

The geological repository has a very large inventory and a total PR value of 0.36 for the normal case. Since geological repositories are meant to dispose of all or most of the fuel of a country or region, the inventory cannot be made smaller in order to increase the PR value.

III. Assessment of Extrinsic PR Attributes

The total proliferation resistance values as a function of extrinsic PR attributes are analyzed in the following sections. For each extrinsic attribute, the normal case from Table 30 is used as a baseline. The frequency of measurement, measurement uncertainty, and probability of identified movement all have a significant effect on the total PR value, as shown in the following sections.

1. Frequency of Measurement

The frequency of measurement is characterized by how often, per year, an inspection is conducted at a certain facility. Figure 7 shows the total PR value for each storage type as a function of the frequency of measurement, with all other values in Table 30 held constant. The normal case baseline is highlighted in the figure at the “Quarterly” frequency of measurement. It is clear from the graph that an increase in the frequency of measurement would significantly increase the total PR value for all storage types. However, an increase in the frequency of measurement would also increase the number of inspections, thus putting an extra burden on both the IAEA and the facility. In the case of the spent fuel pool, increasing the measurement frequency to monthly would triple the number of inspections in a year but only increase the PR value by 0.04, or 8%. Similarly, increasing the measurement frequency from monthly to weekly would more than quadruple the number of inspections in a year but only increase the PR value by 0.05, or 9%. If the measurement frequency were daily or continuous, an inspector would need to work onsite, thus increasing the cost significantly.

At this time, remote measurement, or remote verification, of spent fuel is unavailable. Measurements are often done by item counting, visual inspection, and by using detectors, all which need to be done by an inspector. Unless remote inspections can be developed, the frequency of measurement will likely remain quarterly in the future.

It is important to note here that the frequency of measurement for a closed geological repository would be “never” instead of quarterly. Including the changes in the intrinsic factors described in the previous section, the final total PR value for a closed repository would be 0.23. This value is 36% lower than the PR value for an open repository, 0.36. The main cause of this loss in PR value is caused by the fact that the closed repository should have a negligible dose rate on the surface, as mandated by the NRC. Technically, the adversary would be exposed to a dose rate when handling the storage casks. However, even this dose rate could be very low because the spent fuel would have had a long time to decay in the repository. Therefore, a closed repository is significantly less proliferation resistant than an open repository due to the decrease in radiation dose rate and lack of inspections, even though the closed repository’s inaccessibility slightly increases its PR value.

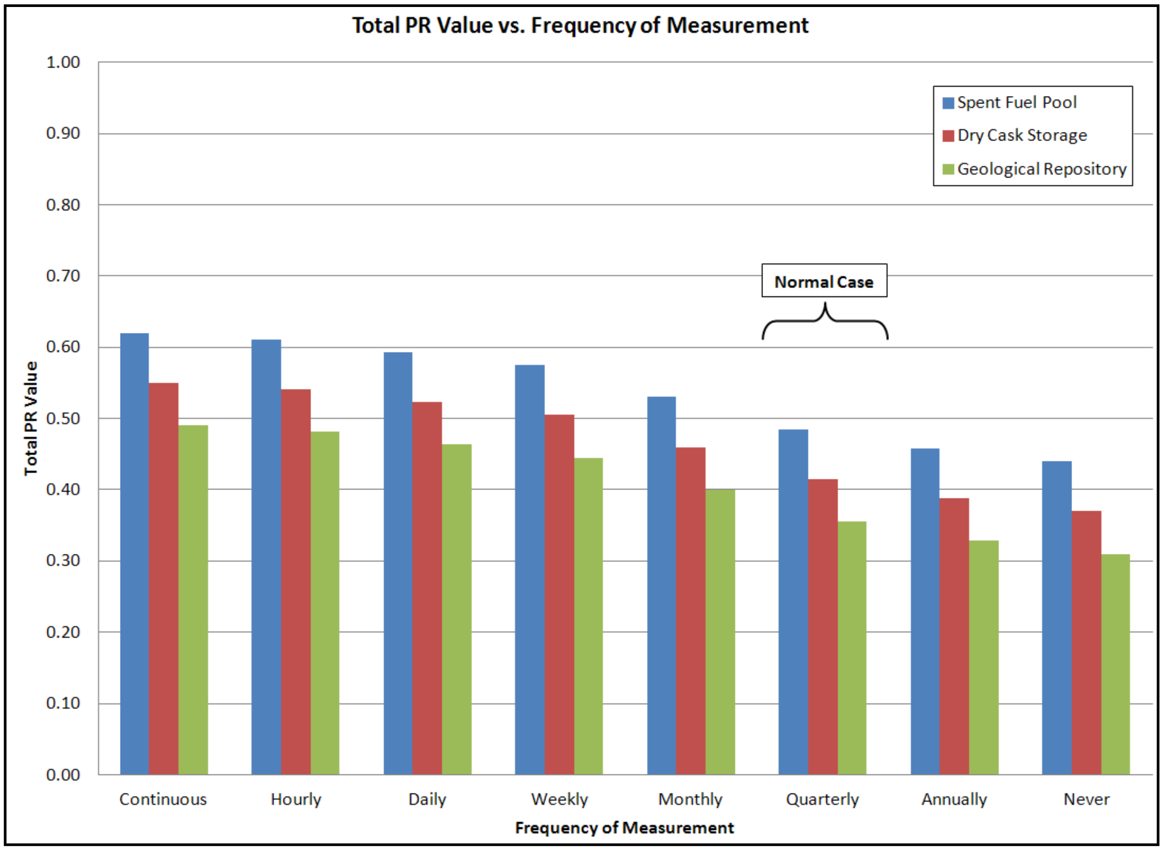


Figure 7: Total PR value as a function of frequency of measurement.

2. Measurement Uncertainty

Measurement uncertainty is defined as the fraction of the total inventory of nuclear material that is not verified during an inspection. In order to be verified, the spent fuel assembly in the storage facility must be measured by a Cerenkov viewing device or a detector to determine that it is actually spent fuel and not a dummy assembly. Because not all of the spent fuel assemblies can be verified during each inspection, the inspector chooses a number of random assemblies to verify. This number is determined by the confidence in the facility that it will not divert nuclear material.

For the normal case in Table 30, a medium confidence level was used to calculate the PR value for the spent fuel pool. A medium confidence level means that 25% of the nuclear material is verified per inspection, giving a measurement uncertainty of 75%. Since spent fuel assemblies in dry cask storage and the geological repository are not normally verified by the IAEA, the measurement uncertainties are 100%. These values, along with the total PR value as a function of measurement uncertainty with all other values in Table 30 held constant, are shown in Figure 8.

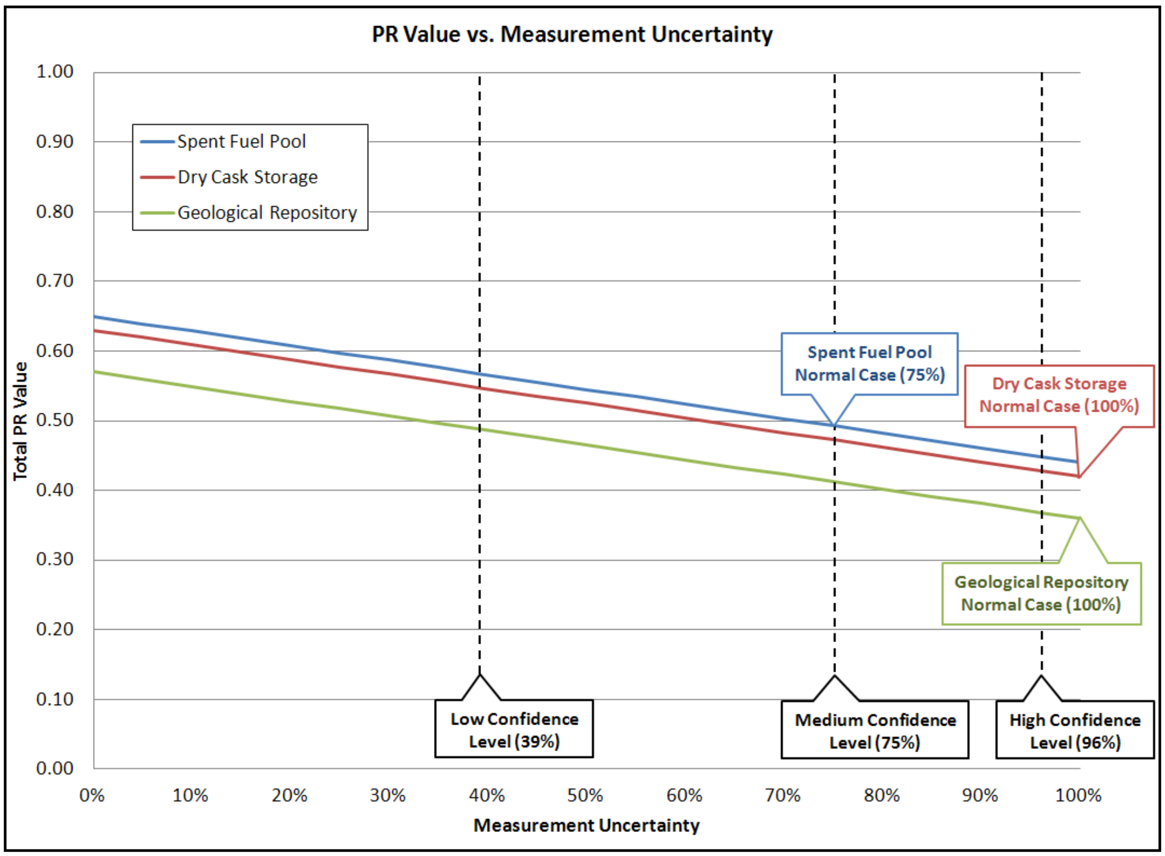


Figure 8: Total PR value as a function of measurement uncertainty.

It is apparent in Figure 8 that a decrease in the measurement uncertainty would significantly increase the total PR value. If dry cask storage and geological repository were verified to the same level as a spent fuel pool (75% measurement uncertainty), then the PR values would increase to 0.47 for dry cask storage and 0.41 for geological repository. In addition, the PR values would significantly increase if the storage facilities were treated with a low confidence level and thus had a measurement uncertainty of only 39%, as described in the Method of Analysis. However, that means that 61% of the nuclear material would have to be verified each inspection which would put a significant burden on both the inspector and operator.

3. Probability of Unidentified Movement

The containment and surveillance (C/S) aspect of safeguards is taken into account through the probability of unidentified movement attribute. The probability of unidentified movement is evaluated according to the range of surveillance and the possibility of deliberate alteration of the C/S system, as describe in Table 27 of the Method of Analysis. Since any installed C/S system can be altered, the probability of unidentified movement can never reach 0% for spent fuel storage. The normal case for each storage type, along with the total PR value as a function of the probability of unidentified movement with all other values in Table 30 held constant, is shown in Figure 9.

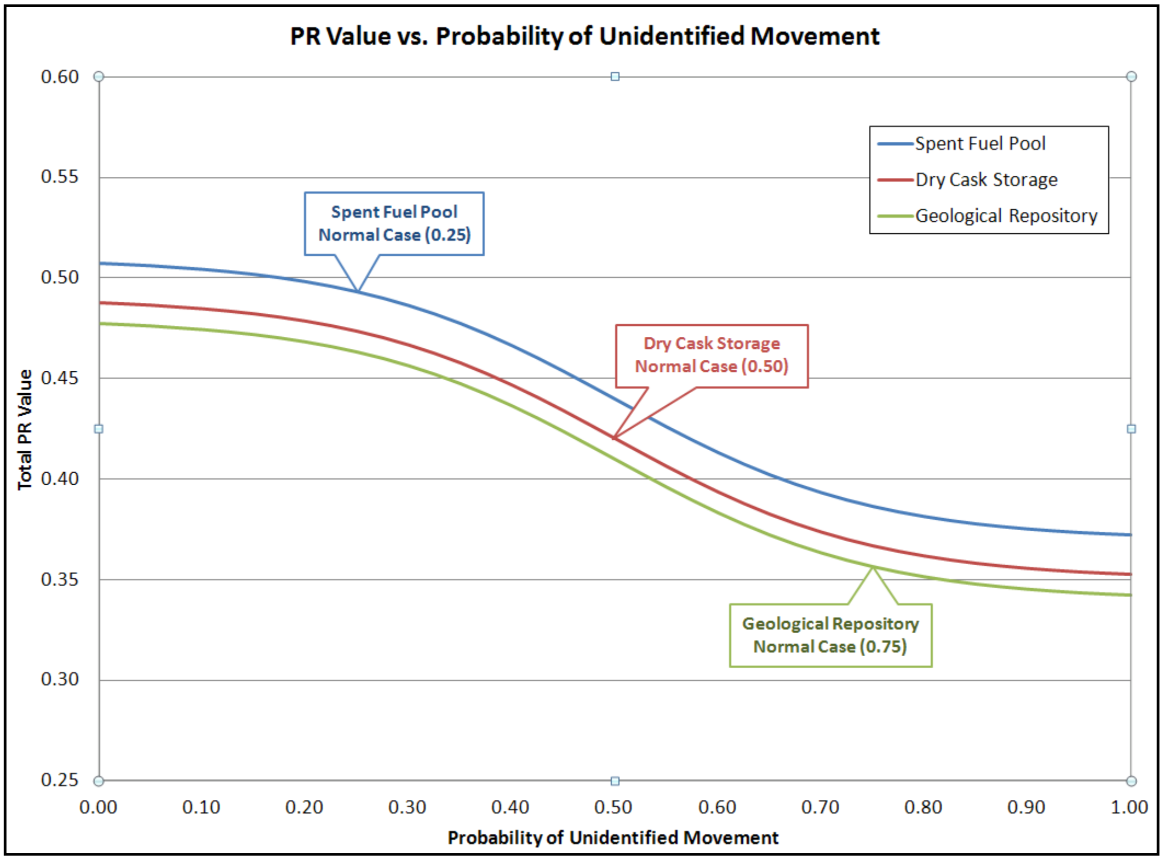


Figure 9: Total PR value as a function of the probability of unidentified movement.

It is apparent in Figure 9 that the total PR value would increase significantly with a decrease in the probability of unidentified movement. The spent fuel pool has the highest PR value in this regard because the actual spent fuel assemblies are under surveillance. Since surveillance of the inside of a cask is unlikely, ultrasonic sealing bolts that send a real-time signal to the IAEA can be used to decrease the probability of unidentified movement to 0.25 for dry cask storage. Likewise, surveillance of the entire repository and ultrasonic sealing bolts would decrease the probability of unidentified movement to 0.25 for the geological repository. This would increase the cost of safeguards for the initial installation but not significantly increase the burden of safeguards for the IAEA or operator. With a 0.25 probability of unidentified movement, dry cask storage and geological repository would have PR values of 0.47 and 0.46, respectively. These values are very close to the normal case PR value of 0.49 for the spent fuel pool.

IV. Recommendations to Increase Proliferation Resistance

In the previous sections, the proliferation resistance of each type of spent fuel storage was analyzed. The effect of each intrinsic and extrinsic attribute on the total PR value was also presented. Some of the attributes have a more significant effect on the total PR value than others. In addition, some attributes can be changed in order to increase the PR value of the spent fuel storage type. Therefore, several recommendations can be made in order to increase the PR value for all three spent fuel storage types. Utilizing these recommendations, an "optimal case" for the PR values of spent fuel storage is presented.

First of all, changes in the intrinsic PR attributes had less of an effect on the total PR value than the extrinsic attributes. In addition, the intrinsic attributes are more difficult to change because they are inherent to the actual storage type. For example, the radiation dose rate of the storage facility should not be changed in order enhance the proliferation resistance. Even though an increase in the dose rate increases the PR value, higher dose rates could be dangerous to the workers and inspectors in the facility.

The physical barriers PR attribute is inherent to the spent fuel storage facility, as well, and cannot be altered to increase the proliferation resistance. The exception is with dry cask storage because it can be made into a "vault" facility by simply placing it in a secure building. Dry cask storage facilities are typically placed outside and surrounded by only a fence. A secure building placed around the facility would increase the PR value but would not put a significant burden on the IAEA or the operator, after the initial construction of the facility.

While the values found for average inventories of spent fuel storage facilities in the United States and across the world seemed to vary greatly, only extremely large changes in the inventory have an effect on the total PR value. The large inventories (over 5000 SQ) of centralized storage facilities and geological repositories significantly decrease the total PR value. Geological repositories will always have large inventories because they are meant to dispose of all or most of the fuel in a country or region. However, centralized interim dry storage facilities can be avoided by employing on-site dry storage and minimizing the build-up of spent fuel storage at reprocessing facilities, for example. If a centralized facility is necessary, then more safeguards should be put into place in order to increase the proliferation resistance.

As mentioned previously, all three of the extrinsic PR attributes had a significant effect on the total PR value. The reason for this is that any increase in safeguards measures would increase the proliferation resistance of a facility. However, increases in safeguards measures can also put a significant burden on the IAEA and on the operator. This is certainly the case for the frequency of measurement. While an increase in the frequency of measurement would increase the PR value, it would also require more inspections and thus more resources from the IAEA. It is recommended that the frequency of inspection remain quarterly for efficient and effective safeguards. However, if remote verification of spent fuel were to be developed, then the frequency of measurement could increase without putting a heavy burden on the IAEA. Increasing the frequency of measurement would also significantly increase the PR value for all spent fuel storage types. Since remote verification is not currently being developed for spent fuel storage, it will not be considered in the optimal case presented later in this section.

The IAEA currently does not verify spent fuel inside of casks at every facility, thus making the measurement uncertainty for dry cask storage and geological repository at 100%. In order to increase the proliferation resistance of these facilities, the measurement uncertainty should be decreased to the level of the spent fuel pool (75%). This would mean that a quarter of the spent fuel assemblies in the facility would need to be verified during each inspection. With the right technology and cask design, this could be done in an efficient manner, without putting a significant burden on the inspector or operator. Developing the technology and optimal cask designs to facilitate verifying spent fuel assemblies inside of the casks would significantly increase the proliferation resistance of dry cask storage and geological repository.

Research institutions around the world are working on developing verification technologies for dry cask storage, as described in Table 7 in the Literature Review. The IAEA has already tested is using some of these technologies at CANDU reactors These technologies involve passive gamma ray counting where the spent fuel transfer into the cask occurs and neutron counting on the top of the cask once it is loaded. Gamma ray and neutron fingerprinting methods are also being developed. Verification methods for dry casks could be simply altered to work for transportation and storage casks in the geological repository.

The probability of unidentified movement can also be decreased in dry cask storage and geological repository facilities in order to increase the proliferation resistance. The C/S system at a spent fuel pool has total knowledge of the spent fuel assemblies, unless the system is compromised. When spent fuel assemblies are inside of casks, surveillance does not have a direct view of the assemblies themselves. For this reason, ultrasonic sealing bolts that can send a tampering alert real time should be used on casks. In addition, surveillance of the entire geological repository should be in place, especially while it is in the operation phase. These measures would decrease the probability of unidentified movement to 0.75 and thus increase the PR value for both dry cask storage and geological repository. Sealing bolts and increased surveillance would also not put a significant burden on the IAEA and operator after the initial development and installation.

In addition, it is important to discuss the closed geological repository because it has a different PR value from the open repository. While the closed repository's inaccessibility increases the PR value, the lack of radiation dose rate outside the repository and lack of inspections decreases the PR value. Since the closed repository is sealed, actual inspections of the casks and fuel cannot occur. However, real-time surveillance can be employed to alert the IAEA of any tampering with the spent fuel. There should be no activity in a closed repository, so the surveillance would be quite simple because any activity should set off an alarm. Therefore, it is recommended that a closed repository have IAEA surveillance in order to increase the proliferation resistance.

Several recommendations have been made in order to enhance the proliferation resistance of spent fuel storage and disposal, as listed below:

1. Do not increase radiation dose rate of facility to increase the proliferation resistance due to the safety risk.
2. Place dry cask storage facility inside of a secure building.
3. Minimize the use of centralized interim dry storage facilities, or increase safeguards measures if the facility is necessary.
4. Inspect all facilities on a quarterly basis until remote verification for spent fuel storage is developed to take the place of inspections.
5. Develop the technology and optimal cask designs to facilitate verifying spent fuel assemblies inside of casks.
6. Use ultrasonic sealing bolts that can send a tampering alert real time on spent fuel casks.
7. Employ surveillance of an entire geological repository, especially while it is in the operation phase.
8. Employ full surveillance of a closed geological repository.

The normal case from Table 30 can now be altered to reflect the above recommendations, presenting the optimal case in Table 31. As can be seen, the total PR values for dry cask storage and geological repository were significantly increased and are higher than that for the spent fuel pool. The input values of *x* that were altered are highlighted in bold in the table. The total PR value for spent fuel pool storage remains unchanged because no recommendations were able to be made without putting a significant burden on the inspectors and operators. However, any spent fuel storage facility could be made more proliferation resistant by simply decreasing the measurement uncertainty by verifying more spent fuel assemblies during inspections or by developing remote verification to increase the frequency of measurement. With the recommendations employed, dry cask storage would be the most proliferation resistant method of spent fuel storage. A closed geological repository is significantly less proliferation resistant than one in operation because of the lack of the ability to inspect it. Therefore, surveillance should be increased for a closed geological repository in order to increase the proliferation resistance. In conclusion, the extrinsic PR attributes have the greatest affect on the total PR value and safeguards measures should be enhanced in order to increase the proliferation resistance of a spent fuel storage facility.

Table 31: PR values for optimal case of spent fuel storage methods.

PR Attribute	**Weight**	**Value of *x***				**PR Value**			
		Spent Fuel Pool	**Dry Cask Storage**	**Open Geological Repository**	**Closed Geological Repository**	**Spent Fuel Pool**	**Dry Cask Storage**	**Open Geological Repository**	**Closed Geological Repository**
Radiation Dose Rate	0.16	2	200	200	0	0.02	0.10	0.10	0.00
Physical Barriers	0.21	Vault	**Vault**	Canyon	Inaccessible	0.16	0.16	0.19	0.21
Inventory	0.10	943	626	87000	87000	0.09	0.09	0.00	0.00
Frequency of Measurement	0.18	Quarterly	Quarterly	Quarterly	Never	0.05	0.05	0.05	0.00
Measurement Uncertainty	0.21	75%	**75%**	**75%**	100%	0.05	0.05	0.05	0.00
Probability of Unidentified Movement	0.14	0.25	**0.25**	**0.25**	**0.50**	0.12	0.12	0.12	0.07
					Intrinsic PR Value (max 0.47)	**0.27**	**0.35**	**0.29**	**0.21**
					Extrinsic PR Value (max 0.53)	**0.22**	**0.22**	**0.22**	**0.07**
					Total PR Value (max 1.00)	**0.49**	**0.57**	**0.51**	**0.28**
					Percent Change from Normal Case	**0%**	**+36%**	**+42%**	**N/A**

V. Evaluation of Proliferation Resistance Assessment Methodology

The proliferation resistance methodology developed in this thesis is quite effective in evaluating the PR of spent fuel storage and disposal. Data was able to be collected to evaluate the PR value of typical spent fuel storage facilities, as shown in Table 30. The PR values obtained seem to accurately depict the proliferation resistance of each spent fuel storage type. The methodology also allowed for the evaluation of certain PR attributes as they affect the total PR value. Finally, the methodology can be used to evaluate the proliferation resistance of any spent fuel storage type.

As with any assessment methodology, there is always room for improvement. The radiation dose rate PR attribute should be re-evaluated because it does not necessarily take into account the dose rate while handling the spent fuel. For example, the closed geological repository technically has a negligible dose rate but there would be a dose rate once the adversary reached the actual spent fuel storage area. The probability of unidentified movement attribute should also be further defined and possibly include a complete vulnerability assessment of the C/S system in a spent fuel storage facility.

The research performed by Silvennoinen 1981 was highlighted in the Literature Review (Table 11 and Figure 2) and is the only proliferation risk assessment that specifically looks at spent fuel storage, along with the rest of the fuel cycle. It was found from this research that spent fuel in interim storage after a long cooling time is subject to the highest diversion risk, while spent fuel in a closed repository has the least risk of diversion. This does not align with the results from the previous section because Silvennoinen 1981 takes into account other factors than presented in this research.

The assessment criteria used by Silvennoinen 1981 are: minimum cost to produce a weapon from given material, marginal cost of using civil nuclear fuel cycle to make weapons, minimum time required to construct a weapon, detectability of weapons construction, ease of diversion (divertability), and quality of separated fissile material. These attributes have a much greater focus on the material type and the actual process to developing the nuclear weapon. The only similar attribute to this research is the divertability, which is defined as "the ease of material diversion in terms of accessibility to and unaccountability of the flow of source material." Isolating just this factor still yields the same results that spent fuel in interim storage after a long cooling time is subject to the highest diversion risk, while spent fuel in a closed repository has the least risk of diversion. To make this comparison side by side, the below lists show the ranking in terms of proliferation resistance of the normal case presented in this research versus Silvennoinen 1981:

Normal Case	Silvennoinen 1981
1. Spent Fuel Pool	1. Closed Geological Repository
2. Dry Cask Storage	2. Spent Fuel Pool
3. Open Geological Repository	3. Open Geological Repository
4. Closed Geological Repository	4. Dry Cask Storage

The ratings for the divertability in Silvennoinen 1981 were "deduced by judgemental technique" and thus not based on actual institutional controls and intrinsic factors to the spent fuel storage method, as in this research. The main deciding factors it seems were the cooling time of the spent fuel and the accessibility to the storage site. Therefore, it is difficult to compare these two methods because the input factors are much different.

CONCLUSIONS & FUTURE WORK

The IAEA safeguards nuclear facilities in order to prevent the diversion of special nuclear material into a nuclear weapons program. The plutonium and uranium in spent fuel presents the largest build-up of nuclear material in the open nuclear fuel cycle. In most countries, spent fuel is stored in spent fuel pools and dry cask storage onsite at the plant and will eventually be disposed of in a geological repository. However, each method of spent fuel storage presents different proliferation risks due to the nature of the storage method and the safeguards techniques that are utilized. Previous proliferation resistance/proliferation risk assessments have considered nuclear material through the whole fuel cycle and not specifically focused on spent fuel. This project evaluates different methods of spent fuel storage in terms of proliferation resistance, taking intrinsic and extrinsic features into account.

The first step in this project was to define the spent fuel storage types to be analyzed. Afterwards, existing proliferation-risk evaluation methods were reviewed and the methodology by Charlton et al. 2007 was chosen for this assessment and modified as necessary. The proliferation-resistant characteristics to be used in the assessment of spent fuel storage were identified and then used to evaluate the proliferation resistance. With the current data obtained, the PR values are 0.49 for the spent fuel pool, 0.42 for dry cask storage, 0.36 for the operating geological repository, and 0.28 for the closed geological repository. The maximum PR value is 1.00 and means that the facility is completely proliferation resistant. Therefore, the spent fuel pool is currently the most proliferation resistant method for storing spent fuel.

Furthermore, each PR attribute was evaluated in terms of its affect on the total PR value. From this assessment, the extrinsic PR attributes, which consist of safeguards measures added to the spent fuel storage facility, had the greatest effect on the proliferation resistance. Therefore, several recommendations to increase the proliferation resistance of spent fuel storage were presented, as listed below:

1. Do not increase radiation dose rate of facility to increase the proliferation resistance due to the safety risk.
2. Place dry cask storage facility inside of a secure building.
3. Minimize the use of centralized interim dry storage facilities, or increase safeguards measures if the facility is necessary.
4. Inspect all facilities on a quarterly basis until remote verification for spent fuel storage is developed to take the place of inspections.
5. Develop the technology and optimal cask designs to facilitate verifying spent fuel assemblies inside of casks.
6. Use ultrasonic sealing bolts that can send a tampering alert real time on spent fuel casks.
7. Employ surveillance of an entire geological repository, especially while it is in the operation phase.
8. Employ full surveillance of a closed geological repository.

Taking these recommendations into account, the PR values of dry cask storage and the closed geological repository were able to be significantly increased, to 0.57 and 0.51, respectively. Therefore, with increased safeguards to match the safeguards level of the spent fuel pool, dry cask storage would be the most proliferation resistant method to store spent fuel. However, the spent fuel pool could also be more proliferation resistant by developing remote verification to increase the frequency of measurement without putting a significant burden on the IAEA.

More work can be done to further evaluate the proliferation resistance of spent fuel storage and even the entire back end of the fuel cycle. First of all, specific spent fuel storage sites should be evaluated and compared to the normal case. Evaluating specific sites would also allow for a complete vulnerability assessment of the C/S system to provide an accurate value for the “probability of unidentified movement” value. More accurate values would also be able to be obtained for the other intrinsic and extrinsic attributes by evaluating existing spent fuel storage sites since many of the values used in this assessment were typical or average values.

As discussed previously, a cost-benefit analysis of safeguards should also be performed. This would allow more concrete recommendations to be made for the level of safeguards in a facility. An increase in safeguards does not necessarily provide an increase in proliferation resistance of the same magnitude. More safeguards usually results in an increased burden on the IAEA and facility operator.

In addition, other PR attributes to add to this methodology should be examined. Specifically, there may be a more appropriate attribute to replace the “radiation dose rate” since this methodology is not material-specific. If the methodology were to take into consideration the type of material in storage, it would be very important to take the cooling time into account. For example, spent fuel in a geological repository would have a longer cooling time and thus be easier to handle than spent fuel in a spent fuel pool.

Other PR attributes could also be introduced into this methodology to assist in evaluating the entire back end of the fuel cycle. A more advanced methodology could evaluate the PR of the transportation of spent fuel assemblies and also the reprocessing cycle.

In conclusion, the proliferation resistance assessment methodology developed in this research effectively calculates the PR values of spent fuel storage and disposal facilities. These PR values can be used to compare the relative proliferation resistance of each storage type in terms of several intrinsic and extrinsic PR attributes. The extrinsic attributes, mainly involving safeguards measures, affect the total PR value most. It was found that for current data the spent fuel pool is significantly more proliferation resistant than dry cask storage or geological repository. As a result, several recommendations were made to improve the proliferation resistance of spent fuel storage. With more safeguards in place, on-site dry cask storage would be the most proliferation resistant. Therefore, the IAEA should continue to develop remote monitoring and cask storage verification techniques in order to improve the proliferation resistance of spent fuel.

REFERENCES

REFERENCES

Abbas, K. 1998. Use of miniature CdZnTe X/γ detector in nuclear safeguards: Characterization of spent nuclear fuel and uranium enrichment determination. *Nuclear instruments & methods in physics research. Section A, Accelerators, spectrometers, detectors and associated equipment* 405 (1):153-158.

Ahmed, I. Iqbal. 2002. An in situ safeguards verifier for spent CANDU-type fuel bundles stored on stacked trays. *JNMM* 31 (1):45-52.

Alvarez, Robert. 2011. Spent Nuclear Fuel Pools in the U.S.: Reducing the Deadly Risks of Storage. Institute for Policy Studies.

Artisyuk, Vladimir, Masaki Saito, and Alexey Ezoubtchenko. 2008. Development of methodology to assess proliferation resistance of nuclear heavy metals. *Progress in Nuclear Energy* 50 (2-6):647-653.

Bari, R., J. Roglans, R. Denning, and S. Mladineo. 2003. Assesssing the Proliferation Resistance of Innovative Nuclear Fuel Cycles. In *International Conference on Innovative Technologies for Nuclear Fuel Cycles*. Vienna, Austria.

Boyer, Brian D., Eleanor T. Dixon, Michael C. Browne, and Robert F. Parker. 2007. Spent fuel safeguards and physical protection issues relevant for advanced fuel cycles. In *GLOBAL 2007 Advanced Nuclear Fuel Cycles and Systems*. Boise, Idaho.

Browne, Michael C., Robert F. Parker, Peter A. Santi, and Richard B. Williams. 2006. Overview of Safeguards Approach for the BN-350 Dry Storage Facility. *Transactions of the American Nuclear Society* 95:43.

Bunn, Matthew, John P. Holdren, Allison Macfarlane, Susan E. Pickett, Atsuyuki Suzuki, Tatsujiro Suzuki, and Jennifer Weeks. 2001. Interim Storage of Spent Nuclear Fuel. Harvard University and University of Tokyo.

Charlton, W. S., R.F. LeBouf, and S. Aghara. 2003. Proliferation Resistance Assessment Method for Analyzing ATW Fuel Cycles. Paper read at Sixth International Meeting on Nuclear Applications of Accelerator Technology, at San Diego, California.

Charlton, W. S. William, D. G. William Ford, S. Sheldon Landsberger, M. Michael Whitaker, R. F. Ryan Lebouf, C. Carl Beard, and C. Claudio Gariazzo. 2007. Proliferation resistance assessment methodology for nuclear fuel cycles. *Nuclear technology* 157 (2):143-156.

Clemen, Robert T. 1996. *Making hard decisions : an introduction to decision analysis*. 2nd ed. Belmont, Calif.: Duxbury Press.

Cummings, Kristopher W. 2010. *Nuclear Engineering Handbook*: CRC Press.

d'Agraives, B. C. Bertrand. 1993. Ultrasonic sealing techniques. A possible solution for safeguarding the containment or storage of spent fuel in an underwater or dry environment. *Journal of Nuclear Materials Management* 21 (3):31-36.
Repeated Author. 1993. Ultrasonic sealing techniques. A possible solution for safeguarding the containment or storage of spent fuel in an underwater or dry environment. *JNMM* 21 (3):31-36.
Doyle, James E. 2008. *Nuclear Safeguards, Security and Nonproliferation: Achieving Security with Technology and Policy*. Amsterdam ; Boston ; Oxford: Butterworth-Heinemann.
Fattah, A. 1990. International safeguards aspects of spent fuel in permanent geological repositories. *JNMM* 19 (1):41-45.
Fritzell, A., T. Honkamaa, P. Karhu, O. Okko, A. Håkansson, and G. Dahlin. 2008. C/S in Final Disposal Processes – Swedish and Finnish Perspectives. *ESARDA Bulletin* 38:10-16.
Fukuda, K., Y. Peryoga, H. Sagara, M. Saito, and M. Ceyhan. 2008. Perspectives of inventories of secondary nuclear materials generated in the nuclear energy production and mass balance in PPP cycle. *Progress in Nuclear Energy* 50 (2-6):654-659.
Giannangeli, Donald D. J. 2007. Development of the Fundamental Attributes and Inputs for Proliferation Resistance Assessments of Nuclear Fuel Cycles, Nuclear Engineering, Texas A&M University.
Hanks, D.H., and A. Tolba. 2006. Successful implementation of an unattended safeguards approach for the transfer of spent fuel to interim dry storage. Paper read at International Safeguards Symposium on Addressing Verification Challenges, at Vienna.
Herrera, I. Perez, K. Murakami, H.-J. Schreiber, C. Olivieri, Y.-G. Lee, A. Ignachenko, and V. Kulkakova. 2006. IAEA safeguards implementation at dry spent fuel storage at Zaporozhe Nuclear Power Plant. Paper read at International Safeguards Symposium on Addressing Verification Challenges, at Vienna, Austria.
Högselius, Per. 2009. Spent nuclear fuel policies in historical perspective: An international comparison. *Energy Policy* 37:254-263.
IAEA. 2011. Safeguards Techniques and Equipment. International Atomic Energy Agency.
International Atomic Energy Agency. 1999. Survey of wet and dry spent fuel storage. Vienna.
Repeated Author. 2010. IAEA Annual Report.
Kang, Jungmin. 2005. Analysis of Nuclear Proliferation Resistance. *Progress in Nuclear Energy* 47 (1-4):672-684.
Kiriyama, Eriko, and Susan Pickett. 2000. Non-Proliferation Criteria for Nuclear Fuel Cycle Options. *Progress in Nuclear Energy* 37 (1-4):71-76.
Kwon, Eun-ha, and Won Il Ko. 2009. Evaluation method of nuclear nonproliferation credibility. *Annals of Nuclear Energy* 36 (7):910-916.
LaFleur, Adrienne M., William S. Charlton, Howard O. Menlove, Martyn T. Swinhoe, and Steve Tobin. 2009. Comparison of Using Self-Interrogation Neutron Resonance Densitometry

(SINRD) versus the Fork Detector (FDET) to Measure Partial Defects in Spent Fuel Assemblies. Los Alamos, NM: Los Alamos National Laboratory.

Repeated Author. 2009. Comparison of Using Self-Interrogation Neutron Resonance Densitometry (SINRD) versus the Fork Detector (FDET) to Measure Partial Defects in Spent Fuel Assemblies. Los Alamos, NM: Los Alamos National Laboratory.

Lee, Sang-Yoon, Michael C. Miller, Il-Jin Park, and Gyung-Sik Min. 2006. Canister Loading Monitor for CANDU Spent Fuel Dry Storage. *Transactions of the American Nuclear Society* 94:82-83.

Lee, T. Tae-Hoon. 2008. Development of the ACP safeguards neutron counter for PWR spent fuel rods. *Nuclear instruments & methods in physics research. Section A, Accelerators, spectrometers, detectors and associated equipment* 589 (1):57-65.

Mathews, C., C. Kessler, and O. Elkhamri. 2006. Strategic safeguards considerations of the global nuclear energy renaissance. Paper read at International Safeguards Symposium on Addressing Verification Challenges, at Vienna, Austria.

Moore, R. David, and Tom Zahn. 1997. Inherent Security Benefits of Underground Dry Storage of Nuclear Materials. Argonne National Laboratory-West.

Moran, B. W. Bruce. 2001. International atomic energy agency safeguards impacts on geological repositories. *JNMM* 29 (3):18-21.

Moran, B.W. 1994. Waste management safeguards project: History of and recommendations for development activities in support of safeguards of final disposal of spent fuel. Oak Ridge National Lab., TN (United States).

Mosquera, A. D. Alfredo. 2005. Safeguarding geological repositories for spent nuclear fuel: Past, present, and future. *JNMM* 33 (3):34-38.

Nuclear Energy Institute. 2011. *Used Fuel Pools at Nuclear Power Plants* 2011 [cited September 28 2011]. Available from http://www.nei.org/resourcesandstats/documentlibrary/nuclearwastedisposal/factsheet/used-fuel-pools-at-nuclear-power-plants/?page=3.

Nuclear Regulatory Commission. 2010. *Spent Fuel Pools* 2007 [cited January 22 2010]. Available from http://www.nrc.gov/waste/spent-fuel-storage/pools.html.

Repeated Author. 2010. *Fact Sheet on Dry Cask Storage of Spent Nuclear Fuel* 2008 [cited January 2010]. Available from http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/dry-cask-storage.html.

Office of Civilian Radioactive Waste Management. 1994. Conceptual Design For a Waste-Management System that uses Multipurpose Canisters. U.S. Department of Energy.

Park, I.J., G.H. Ahn, Kim, J.S., G.S. Min, and S.Y. Leeb, Miller, M.C. 2006. Simulation and test of spent fuel verification system for CANDU dry storage canister. Paper read at International Safeguards Symposium on Addressing Verification Challenges, at Vienna, Austria.

Pasamehmetoglu, Kemal O. 2006. Proliferation Resistance of Advanced Nuclear Fuel Cycles. *Journal of Nuclear Materials Management* 34 (3):21.

Petroski, Robert. 2005. Centralized Interim Storage of Nuclear Waste and a National Interim StorageStrategy. *Journal of Engineering and Public Policy* 9.

Petrunin, V. V., V. I. Polunichev, Yu P. Sukharev, V. I. Alekseev, and A. N. Lepekhin. 2008. Analysis of questions concerning the nonproliferation of fissile materials for low-and medium-capacity nuclear power systems. *Atomic Energy* 105 (3):159-164.

Phillips, J. R. 1991. Passive Nondestructive Assay of Nuclear Materials. Los Alamos National Laboratory.

Poplavskii, V. M. 2001. Systems-Analysis Assessment of the Effectiveness of Measures for Decreasing the Proliferation Risk of Fissioning Materials. *Atomic energy* 91 (5):896.

Pratt, V., L. Bourva, and R. Carchon. 2006. Quantification of Cerenkov light for safeguard applications. Paper read at International Safeguards Symposium on Addressing Verification Challenges, at Vienna, Austria.

Pratt, Victoria, Kendar Foltz Biegalski, and Sheldon Landsberger. 2007. Development of the Nonproliferation Assessment Tool (NAT) Software Package for the Calculation of Proliferation Resistance Values of Nuclear Fuel Cycle Facilities. *Journal of Nuclear Materials Management* 35 (2).

Pushkarjov, V. 1986. INTERNATIONAL SAFEGUARDS ASPECTS OF SPENT-FUEL STORAGE. *IAEA bulletin* 28 (1):53-57.

Saling, James H., and Audeen W. Fentiman. 2002. *Radioactive waste management*. 2nd ed. New York: Taylor & Francis.

Santi, Peter A., and Michael C. Browne. 2006. Attended Safeguards for the BN-350 Dry Storage Facility. *Transactions of the American Nuclear Society* 95:44-45.

Serra, Paul A. 2011. *Dry Cask Storage Questions* 2011 [cited September 28 2011].

Silvennoinen, P. 1981. An approach to quantitative assessment of relative proliferation risks from nuclear fuel cycles. *The Journal of the Operational Research Society; Houndmills* 32 (6):457.

Stanbro, William D., and Chad T. Olinger. 2002. Proliferation Resistance: New Visibility and Myths. *Journal of Nuclear Materials Management* 30 (3):39-43.

Tarvainen, M. 1999. The SAGOR project and its outcome. In *International seminar on retrievability of high level waste and spent nuclear fuel*. Saltsjoebaden, Sweden: International Atomic Energy Agency.

Tzolov, Roumen, Michael Goldfarb, and Laurent Penot. 2007. Development and Evaluation of New Electronic Seals at the IAEA.

U.S. DOE. 2008. Yucca Mountain Repository License Application - General Information. edited by U. S. D. o. Energy.

U.S. NRC. 1981. Part 60 - Disposal of High-Level Radioactive Wastes in Geologic Repositories. In *10CFR60*, edited by U. S. N. R. Commission.

Repeated Author. 1991. Part 20 - Standards for Protection against Radiation. In *10CFR20*, edited by U. S. N. R. Commission.

Repeated Author. 1997. Standard Review Plan for Dry Cask Storage Systems (NUREG-1536, Initial Report). edited by U.S. Nuclear Regulatory Commission.

Repeated Author. 2004. Part 71 - Packaging and Transportation of Radioactive Material. edited by U.S. Nuclear Regulatory Commission.

Repeated Author. 2011-2012. Information Digest (NUREG-1350). edited by U. S. N. R. Commission.

World Information Service on Energy. 2010. *Nuclear Fuel Material Balance Calculator* 2009 [cited January 21 2010]. Available from http://www.wise-uranium.org/nfcm.html.

World Nuclear Association. 2011. *Plutonium* 2009 [cited September 29 2011]. Available from http://www.world-nuclear.org/info/default.aspx?id=456&terms=plutonium.

Repeated Author. *Plutonium* 2009. Available from http://www.world-nuclear.org/info/default.aspx?id=456&.

Repeated Author. 2010. *Radioactive Waste Management* 2009 [cited January 29 2010]. Available from http://www.world-nuclear.org/info/inf60.html.

Repeated Author. 2011. *Country Briefings* 2011 [cited September 7 2011]. Available from http://www.world-nuclear.org/.

Repeated Author. 2011. *Radioactive Waste Management* 2011 [cited October 13 2011]. Available from http://www.world-nuclear.org/info/inf04.html.

Repeated Author. 2007. *WNA Nuclear Century Outlook Data* 2011 [cited September 7 2007]. Available from http://www.world-nuclear.org/outlook/nuclear_century_outlook.html.

Repeated Author. 2011. *World Nuclear Power Reactors & Uranium Requirements* 2011 [cited October 13 2011]. Available from http://www.world-nuclear.org/info/reactors.html.

Ziock, K.P., G. Caffrey, A. Lebrun, L. Forman, P. Vanier, and J. Wharton. 2005. Radiation Imaging of Dry-Storage Casks for Spent Nuclear Fuel. In *IEEE Nuclear Science Symposium and Medical Imaging Conference*. Fajardo, Puerto Rico: Lawrence Livermore National Laboratory.

Ziock, K.P., P. Vanier, L. Forman, G. Caffrey, J. Wharton, and A. Lebrun. 2005. The Feasibility of Cask "Fingerprinting" as a Spent-Fuel, Dry-Storage Cask Safeguards Technique. Lawrence Livermore National Laboratory.

Repeated Author. 2005. The Feasibility of Cask "Fingerprinting" as a Spent-Fuel, Dry-Storage Cask Safeguards Technique. Lawrence Livermore National Laboratory.