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Total system performance assessment of the proposed high level radioactive waste repository site at Genting Island, Karimunjawa, Indonesia

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Total system performance assessment of the proposed high level radioactive waste repository site at Genting Island, Karimunjawa, Indonesia

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1. GENERAL ASPECTS OF RADIOACTIVE WASTE MANAGEMENT

1.1. Introduction

Nuclear Power Plants (NPPs) utilize nuclear reactions to produce heat. The heat production is used to operate turbines and produce electricity using generators.

The safe operation of NPPs has been underway for more than three decades worldwide. However, two major accidents have occurred in nuclear technology history: TMI and Chernobyl. These accidents have significantly impacted public acceptance of nuclear power and nuclear waste disposal sites.

There are two basic criteria that must be followed in utilizing nuclear technology: safety and economy. These two criteria are applied sequentially.

In addition to producing electricity, NPPs also produce wastes, namely radioactive waste. These wastes are categorized into three levels: Low Level Radioactive Waste (LLRW), Transuranic Waste (TRUW) and High Level Radioactive Waste (HLRW).

LLRW is the radioactive waste that generates low decay heat, requires little or no shielding and contains very low levels of transuranic elements [1]. The sources of LLRW include contaminated clothing, plastics, laboratory glassware, etc.

TRUW is the radioactive waste that contains isotopes above Uranium in the periodic table. These wastes exhibit low radioactivity but long half-lives and little decay heat. TRUW requires shielding but no cooling [2]. Both wastes are generated during reactor operation and maintenance of NPPs.

HLRWs usually come from the nuclear spent fuel and liquid reprocessing wastes that are vitrified into solid wastes. Their initial activities exceed one thousand curies per liter and contain long-lived nuclides.

Figure 1.1 shows general flow of radioactive waste generated from a NPP station.

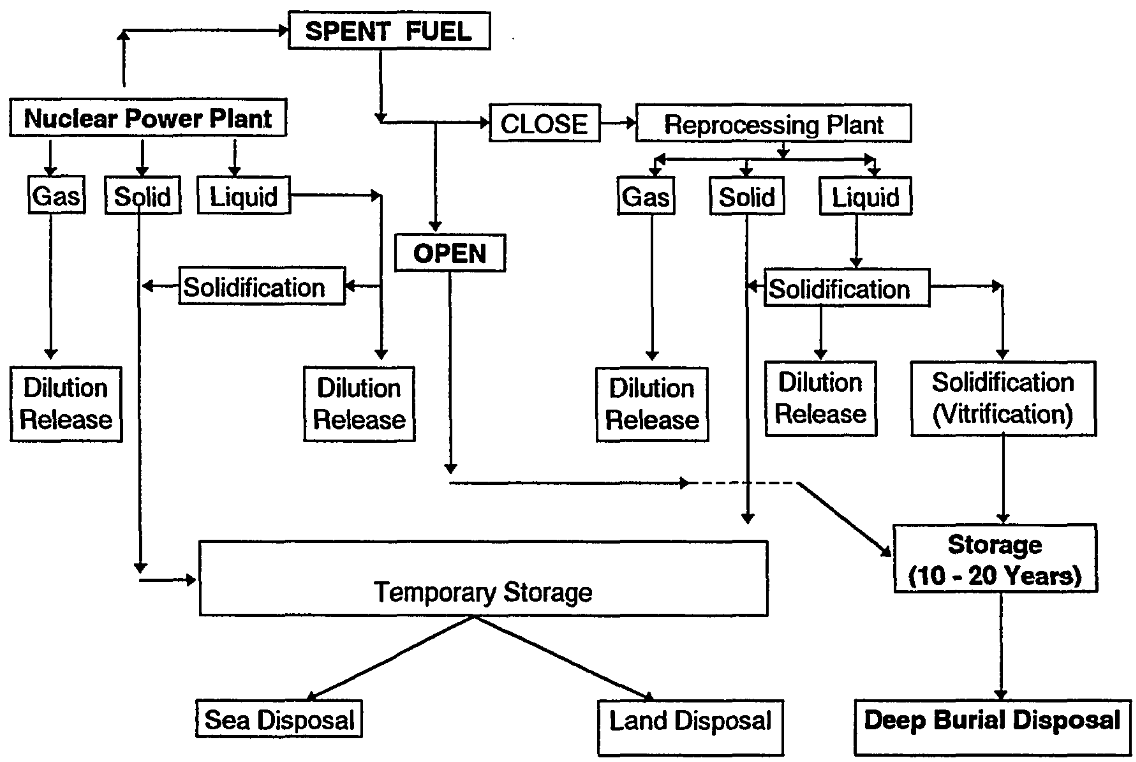

Figure 1.1. General Flow of Radioactive Wastes Generated from a NPP

In this figure, the fuel can either experience a closed or open fuel cycle. In a closed fuel cycle, the spent fuel is reprocessed and the liquid waste from the reprocessing plant is solidified prior to disposal as HLRW. In the open cycle, however, the spent fuel is not reprocessed. This fuel is usually stored in an interim storage facility for approximately 10 to 20 years to permit cooling before ultimate disposal. Note, that the open cycle is shown in boldface-type.

1.2. Management of High Level Radioactive Waste

As previously mentioned, HLRW is the spent fuel itself and vitrified liquids from reprocessing plants. In this study, reprocessing wastes will not be described further. Therefore, the discussion will be limited to an open fuel cycle or once-through cycle. Also, defense high-level waste is not considered here.

HLRW management includes the handling of spent fuel after discharge from the nuclear reactor and thus includes storage and disposal. One approach to HLRW management is to cool the spent fuel for approximately 10 - 20 years by placing it in an interim storage facility. This cooling time permits significant radioactive decay and reduces total activity and heat output. Thereafter, placing spent fuel in stable, deep geological formations with a number of containment barriers can safely isolate it from the environment.

Interim storage technologies have been developed for more than 40 years. Two techniques, which have received widespread interest, are water-filled pools and dry storage [2]. Spent fuel storage facilities are usually located at the reactor site. Some countries use a centralized storage site away from the reactor. Figure 1.2 shows the interim storage methods that have been widely used around the world.

In wet storage technologies, the spent fuel is immersed in pools of water, where radioactive decay heat is dissipated. Consequently, it is possible that corrosion of the spent fuel could occur. On the other hand, dry storage is essentially maintenance-free since natural convection heat transfer takes place and little corrosion will occur.

The repository is the place where the encapsulated spent fuel is placed for final disposal in a selected geological medium. The disposal site can be either above (unsaturated zone) or below the groundwater table (saturated zone).

The encapsulation of the fuel canister employs either single or multiple barriers. Fuel elements are designed to retain fission products and to have very low corrosion rates.

The repository facility is designed to ensure that if toxic radionuclides are released by the spent fuel, they remain isolated from the accessible environment (AE). This isolation should continue until at least the radionuclides have decayed to levels that will present no unacceptable risks to future generations.

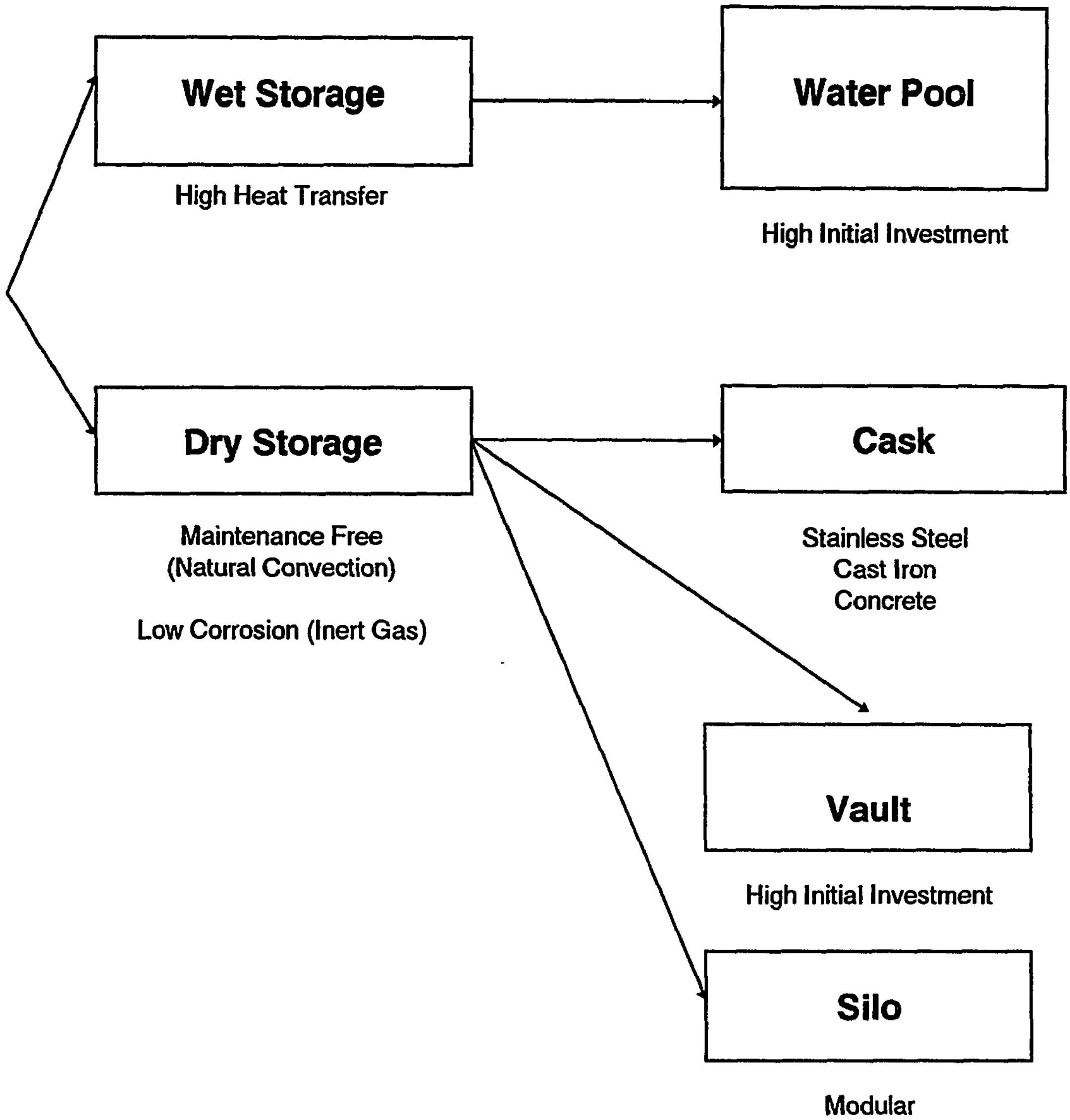

Figure 1.2. Interim Storage Method Elements

The main components of the repository barriers are the near-field, the geosphere and the biosphere. The stable waste form and corrosion-resistant package combined with engineered barriers are the near-field components. The geosphere is the geological media itself, where the ability to restrict groundwater flow, hence low permeability, is considered an ideal geological attribute. The biosphere can serve to dilute radioactivity and, although in a strict sense, may not constitute an isolation barrier.

1.3. Concepts and Criteria for Ultimate Repository Selection

Prior to ultimate disposal of HLRW, it should be stored for some time. This is due to its potential radiological hazard, which declines with time. The fission products retained in the spent fuel undergo natural radioactive decay. To measure the relative hazard of this waste when compared to other wastes, one method is to compare its hazard potential to the hazard presented by an equal volume of uranium ore [3]. The ratio of these two hazards can be considered as a relative hazard index. Figure 1.3 shows the relative hazard index versus storage years. This figure shows that in the first year of storage, the HLRW is about 1000 times as hazardous as the natural uranium ore, but after 10,000 years in storage the hazard is significantly less, decreasing by about two orders of magnitude.

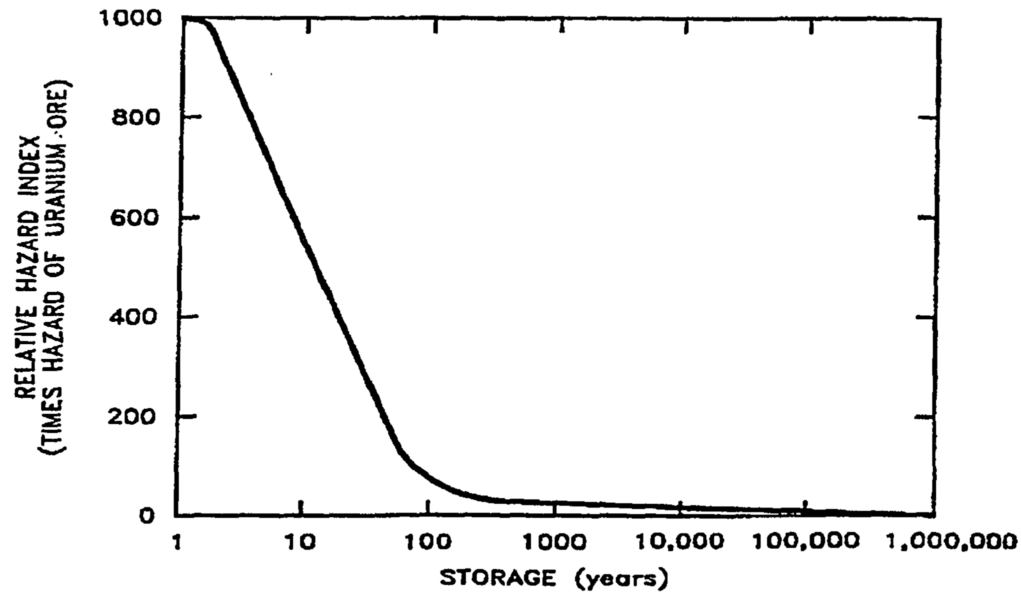

Figure 1.3. Hazard of HLRW versus Storage Time [1]

Several parameters must be addressed to evaluate the isolation of HLRW in a deep geological medium. These parameters include groundwater flow rate, sorption of radionuclides in the geologic medium, dispersion action, and dilution of radionuclides by surface water. Therefore, minimal permeability, maximal flow dispersion, minimal chance of forming apertures, and minimal thermal disturbance will make up ideal characteristics of a geologic medium [1]. Furthermore, an understanding of the exposure pathways is also very important for prediction of the eventual fate of any radionuclides.

Several countries such as France, Canada, Germany, Belgium, and Japan primarily consider salt, granite, and clay as the geologic media for the HLRW disposal facilities [3]. In the United States, however, the Yucca Mountain site (located in the state of Nevada), which has tuff geological formations, has been selected for site characterization [4].

1.4. Universal Container System (UC System)

In addition to identifying the ideal criteria for a geologic medium, along with understanding the pathways and release mechanisms of radionuclides, the containment system that isolates the spent fuel itself must be carefully selected. The containment materials should be well-chosen such that the disposal container can be safely emplaced in the geologic medium. The containment design must also be economically sound.

The conventional approach for containment management is usually divided into several steps. The first step is loading spent fuel in an interim storage container. Second, unloading the spent fuel from that container, and placing the fuel into transportation casks. The casks are transported from a reactor site to a centralized interim storage facility. Finally, another container must be prepared for ultimate disposal. These handling steps and procedures are cumbersome and should be reduced to avoid unnecessary radiation exposure.

One of the most promising containment schemes is known as the Universal Container System (UC System). The UC system is an integrated system in which spent fuel assemblies would be loaded and sealed in multi-assembly containers at a receiving facility [5]. The spent fuel would be stored, transported, and finally placed in the ultimate repository without ever reopening. The advantages of this system is that the number of required handling steps and procedures are significantly reduced. Two classes of UC system includes Multi-Purpose Containment (MPC) and Multi-Element Sealed Canisters (MESC).

MPC is a sealed multi-assembly container. It is a thick-walled, fully shielded container and is capable of handling 6 to 10 MTU of spent fuel. This container is the prime candidate for achieving direct disposal of the spent fuel without ever going into reprocessing in the future. On the other hand, MESC is a sealed, multi-assembly container with thin-walled storage baskets containing 6 to 10 metric tons of spent fuel [5]. MESC may be used as low cost interim storage and shielding just before final disposal. If spent fuel reprocessing is considered as a future option, MESC can be considered as the choice for container selection due to its greater flexibility.

There is another container design concept in the UC system that may provide a more robust waste package. This extra robustness increases the certainty of meeting containment requirements, provides tolerance to a wide range of repository conditions, employes multi-barriers, and uses a defense-in-depth approach and lend itself to drift emplacements [6]. This design is also referred to as a hybrid design of the UC system. The robust waste package increases the breaching time beyond that of the non-robust waste package.

The materials selected for a container design must be tolerant of the geologic conditions. The container will be placed in the deep geologic repository for a long time, over 10,000 years. A wet geologic medium requires the use of container materials that are different from a dry environment. In the Yucca Mountain site, for example, the inner containment material will be one of the following candidate alloys: Alloy 825, Alloy C-4 or Titanium Grade 12 [6]. Outer containment materials can be either corrosion-allowance or corrosion-resistant materials, based on data identified in degradation mode surveys for the specific environment of interest.

1.5. Objective of the Research

This thesis will discuss the research required to complete a total system performance assessment (TSPA) analyses of the proposed disposal site for HLRW in Indonesia. Up to now, the final site has not yet been officially determined by the government of Indonesia. However, since Indonesia has constantly been pursuing an interest in building NPP stations to produce electricity, radioactive waste will become an important issue in the future. This research effort will serve as a scientific reference for future investigations.

The research will include an estimation of the amount of HLRW that will be generated when the first NPP begins in operation in the year 2004. The HLRW inventory will be estimated by extrapolation for 25 years thereafter. A recommendation of the number of containers required can then be presented.

Since the Indonesian archipelago lies along the equator, the climates are very different from those in subtropical areas. The rate of rainfall is significantly higher than in subtropical areas. The temperature ranges from 20°C to 35°C throughout the year. Therefore, the type of materials for waste package containment must be carefully examined.

The ease of transportation and other accommodations in developing countries is usually less efficient compared to developed countries. Especially in Indonesia, which consists of 13,677 islands, management of the transportation system becomes even more difficult. Therefore, the site proposed should be carefully chosen such that it will not further complicate the transportation system.

One of the most important considerations is modeling some scenarios of disruptive events that might occur at the repository site. This is very important to understand the possibility of radionuclide releases from the repository facility to the AE. Therefore, caution may be taken to anticipate such possibilities. In this research, the potential repository facility is assumed located in the saturated zone (below the groundwater table). Integrating repository scenarios will be completed by modeling the proposed site with a computer code called Repository Integration Program (RIP), developed by Golder Associates, Inc. [7]. The results of the RIP model will be analyzed. Integrating the analyses with the other relevant information will serve as the basis for the TSPA of the proposed facility. The analyses of the results from the RIP computer code will be mainly on the impacts of the releases of the radionuclides within the exclusion xone limit used in this study.

In terms of dose/risk evaluation, the radiation doses of several radionuclides that have long half-lives and a range of retardation factors will also be calculated. The dose calculations use the assumptions that a individual lives on the vicinity of the AE and ingests these radionuclides through drinking water. The drinking water source is the groundwater. The seawater will not be used as the source of the drinking water. This will entail the impact to human when he or she lives within the area of repository facility.

In addition to the above, critical data necessary to reduce the uncertainty in site performance predictions will also be identified and may be useful for future investigations.

1.6. Overview of Thesis

This chapter has been a survey of the general aspects of radioactive wastes. Some definitions that have immediate relevance have also been given. More detailed discussions will be presented in the following chapters.

Chapter 2 discusses the results of a feasibility study that was conducted in Indonesia to determine the likelihood of having NPP programs in the country. It discusses the regulatory body dealing with nuclear and radiation activities in the country.

Chapter 3 serves as the literature review. This chapter presents the relevant research that has been conducted elsewhere.

Chapter 4 discusses the methodology used to conduct for the total system performance assessment of the Genting Island repository site facility.

Chapter 5 is the core of discussion of the repository integration modeling. The analysis of the results from the RIP code are presented. Furthermore, the radiation doses that might be received by a person lives in the area are also presented.

Finally, Chapter 6 summarizes the results and provides suggestions for some activities that can be pursued in the future to make the research in this area even more applicable to the regulatory requirements.

2. PROJECTION AND PLANNING FOR BUILDING THE NUCLEAR POWER PLANT PROGRAM IN INDONESIA

2.1. Forecasting Electricity Demands

Indonesia is currently one of the member nations of the Organization of Petroleum Exporting Countries (OPEC). It exports crude oil and condensate products to countries such as Japan, USA, South Korea and others. However, due to rising domestic oil consumption, the export of crude oil has declined. To maximize domestic oil refinery output in meeting specific product needs, Indonesia has become a crude oil and refined product importer over the past several years. In 1990, for example, Indonesia imported 45.7 million barrels of crude oil and 24.2 million barrels of refined products [8].

According to the Ministry of Mining and Energy of the Indonesian Government, Indonesia's energy generating capacity primarily includes Liquefied Petroleum Gas (LPG), natural gas, geothermal, hydropower, coal, and uranium resources. Oil reserves have been estimated to be of 5.3 billion barrels. The maximum production capacity is 467 million barrels per year. The LPG and natural gas reserves are estimated at around 216.8 trillion standard cubic feet (tscf). The total proven and probable coal reserves are about 4.8 and 18.8 billion tons, respectively. The geothermal potential reserve is approximately 10,825 MW. The largest shares of geothermal reserves are within the Java-Bali Islands. The potential of hydropower is of approximately 75 GW in which only 3.2 GW has been used for electricity generation. It is unfortunate that the hydropower resources are located in thinly populated areas such as the Kalimantan and Irian Jaya Islands. Uranium resources exploration is still at an early stage. However, 10,380 tons of Uranium have been found in West Kalimantan [9].

The State Electricity Generation Company (Perusahaan Listrik Negara - PLN) of Indonesia is the government owned that produces electricity. Its primary function is to generate and distribute electricity in the country. According to PLN, the growth rate of energy used to generate electricity has been of about 18.1% annually from 1970 to 1990 [10].

In the feasibility study report of the first NPP in Indonesia, the energy and electricity demands have been forecast [11]. The electricity demand was estimated based on macroeconomic growth rates. The electricity demand includes generation of electricity from PLN and private companies (non-PLN), as shown in Table 2.1. This table shows that by the beginning of the next century the country will need an additional electricity generation of 88,926.94 GWh. This is more than a 66% increase of the current demand. Therefore, the government needs to determine well-planned electricity supplies to meet the projected demand. Until present time, PLN regulates the electricity tariff in the country.

Table 2.1. Projected Electricity Demand from 1990 - 2019 (in Gwh) [11]

Demand Sector	Actual		Projection		
	1980	1990	2000	2010	2019
Households	2,910	9,004	27,880	83,066	195,340
Manufacture	14,706	39,285	94,955	214,618	408,555
Services	1,800	5,093	19,473	59,552	126,264
Total	19,415	53,382	142,309	357,236	730,158

Combining all potentially energy sources available, the study concluded that Indonesia will need to use nuclear energy to meet the projected demand. It is estimated that nuclear energy will play a role by the year 2004. Figure 2.1 shows the estimated role of nuclear energy in meeting the demand for electricity [11]. This figure provides the expected energy utilization by fuel type in barrels of oil equivalent (boe) for Indonesia for the time period from 1990 to 2018. The utilization of NPPs in Indonesia will be based upon the ability of these units to produce electricity at reasonable rates. The cost of electricity production from NPPs should compete fairly with other energy sources.

While building costs of conventional nuclear power plants such as light water and CANDU reactors can be calculated using well-established models, the costs of building of the advanced type of reactors, though less predictable, are projected to be lower. However, the advanced reactor types are still in the licensing stages. Therefore, caution must be taken due to the greater uncertainties involved. Capital cost and fuel cycle cost are the most significant part of generation cost. The advanced reactor types promise the possibility of building one with modular approach. They should be examined carefully.

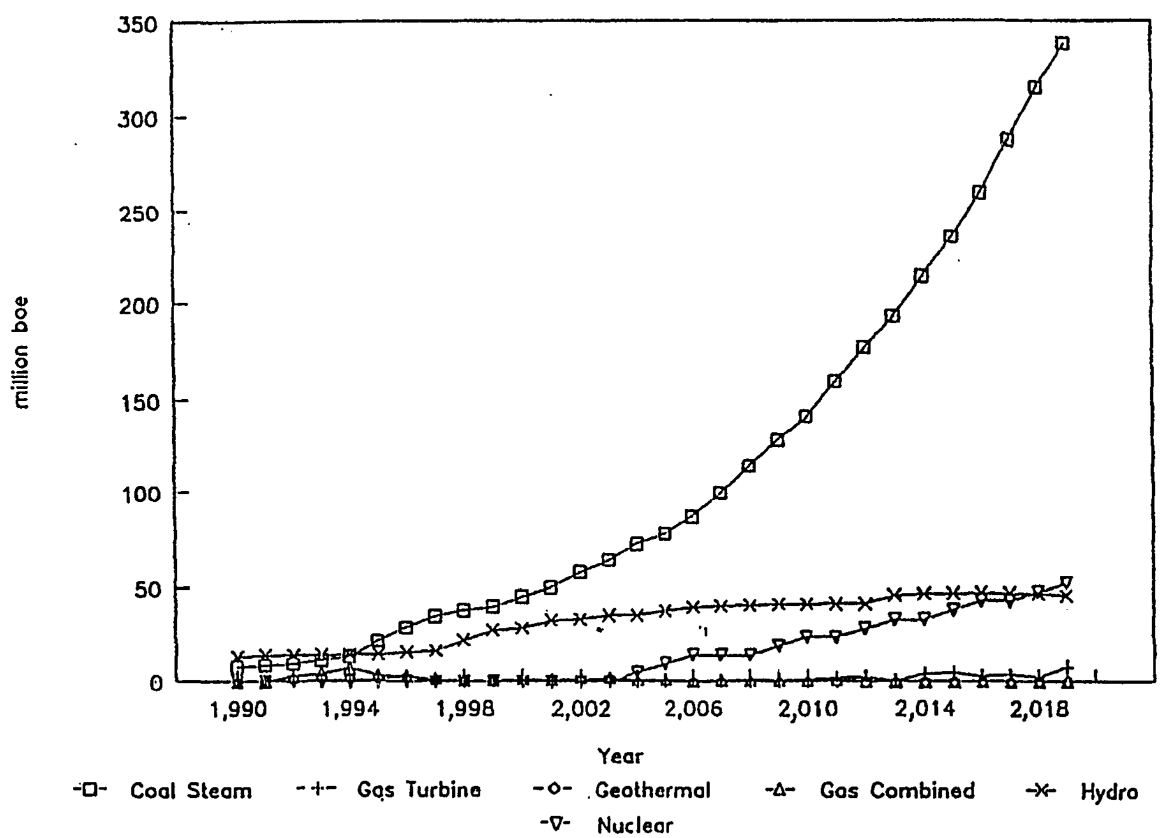

Figure 2.1. PLN Electricity Supply Plan [11]

2.2. The First Nuclear Power Plant Programs

The need for large electricity generation by the beginning of the next century encourages Indonesia to consider building NPP stations. Nuclear energy has several advantages compared to other generating power plants.

First, it uses a relatively small area of land. This fact is primarily attractive to Indonesia, since the plant will be built in Java Island. Sixty percent of the Indonesian population lives on Java. This is because Java is the most developed Island.

Second, from an economic point of view, the feasibility study shows that the generation cost of a 600 MW(e) NPP is lower than that of similar sizes of coal-fired power plant [11]. It also shows that the generation cost of the NPP is still competitive with the combined-cycle plant with similar power.

Third, the uranium fuel market price is currently very low. Therefore, Indonesia can take advantage of purchasing uranium fuels from overseas in the near future. Although the technology to explore uranium resources in the country must continue to be developed as well. The ability to self-sustain fuel stocks for domestic needs must be considered to compensate for the possibility of rising uranium fuel price overseas in the future.

The development of the Indonesian NPP program dates back to October 1, 1973, when a seminar discussing the possibility for using NPP for electricity generation was first held [12]. From that point there have been a number of seminars, symposiums and workshops held to promote the use of nuclear energy.

The feasibility study also discusses the safety aspects of each reactor candidate and the economics for building each of the light-water type reactors. The study offers a comparison of a number of safety related instruments from a variety reactor types. It is very likely that the first NPP will be a pressurized water reactor (PWR) type using a 600 MW(e) design. PWR is a proven technology and has been in operation safely for decades around the world. Besides PWRs, the other types of reactor being considered are the advanced types, namely AP600 and SBWR.

According to the plan, 10 NPP stations are planned to be built in Muria Peninsula on Java Island over a 25 year period starting in 2004 [13]. The peninsula is located at the northern side of Central-Java province.

2.2.1. The Fuel Description of the Nuclear Power Plant

The detailed description of the first NPP in Indonesia has not yet been made available for public review. However, there is a strong indication that the 600 MW(e) PWR will be selected as the first NPP [13]. Therefore, throughout this report the standard PWR parameters will be used when necessary.

The description of the PWR that is relevant to this report encompasses the overall fuel assembly characteristics. The generic data is taken from the characteristic database of Light Water Reactor (LWR) supplied by the US Department of Energy (DOE) [14]. Regarding the HLRW, the pertinent information relates to the nuclear fuel assembly. The overall assembly characteristics are provided in Table 2.2.

Table 2.2. The Overall Assembly Characteristics [14]

Assembly Class	Westinghouse 15 X 15
Assembly width (inches)	8.43
Assembly length (inches)	159.71
Rod pitch (inches)	0.56
Average weight of Uranium (Discharge fuel)	453.90
Enrichment range (% 235)	1.86 - 3.99
Average discharge burnup (MWd/MTIHM)	29,456
Maximum discharge burnup(MWd/MTIHM)	44,720

This description is for a Westinghouse standard fuel assembly (15 X 15). There are 225 fuel rod positions in the assembly. Each of the fuel assemblies consists of 204 fuel rods. The remaining positions are used for burnable poisons and control rods. The fuel rod description can be seen in Table 2.3.

Table 2.3. Fuel Rod Description [14]

Fuel rod positions per assembly	225
Number of fuel rods per assembly	204
Rod diameter (inches)	0.422
Rod length (inches)	148.59 - 151.88
Active fuel length (inches)	142.00 - 144.00
Weight per rod (lbs)	6.77 - 6.85
Clad material	Zircolay-4
Clad thickness (inches)	0.0242
Fuel-clad gap (inches)	0.0038
Fill gas used	He
Initial gas pressure (psig)	0 - 475
Nitrogen content of fill gas (percentage)	4-78
Fuel pellet material	Uranium Oxide
Fuel pellet shape	dished, chamfered
Fuel pellet diameter (inches)	0.365- 0.366
Fuel pellet length (inches)	5.52
Fuel pellet weight per rod (lbs)	0-3
Grain size (microns)	8-20
Fuel density (% theoretical)	95
Smear density (gr/cm^3)	10.07
Plenum spring material	St. Steel 302
Plenum spring weight per assembly (lbs)	0.038 - 0.044
Plenum length (inches)	8.20
Plenum volume (cubic inches)	1.25

2.2.2. Physical Description of the First Nuclear Power Plant Site

The design and construction of a NPP must ensure that the occurrence of natural phenomena will not cause the reactor containment structure to collapse and a loss of safety function to occur. The natural phenomena include effects of earthquakes, tornadoes, hurricanes, and floods. Therefore, seismology, meteorology, hydrology and geology of the plant site must be investigated in great detail.

Geologists believe that the surface of the earth is composed of large structures called tectonic plates [15]. An earthquake takes place as the result of the movement of these tectonic plates relatives to one another. Because of high stress energy along the edges of the plates or faults, these plates can undergo sudden movement resulting in an earthquake.

Meteorology is important in a sense of the dispersion of effluents from a power plant. A NPP must be designed and constructed to withstand large storms. The design must not only withstand direct wind force, but also impact of the objects that have been picked up by the winds.

Investigation of the site hydrology is necessary to prevent large quantities of water from entering the site. If the site is located on the seashore, as in the case of NPP in Muria peninsula, an investigation to estimate the largest tidal wave possible must be conducted. Watertight structures must be designed to withstand the maximum expected water intrusion.

Geological structure of the site must be investigated to determine whether the area can adequately support the reactor building. In the Muria peninsula, there are two volcanoes composed of alkali-potassic and ultra-potassic rocks that were active in the late Tertiary to Quartenary ages [11]. The volcanoes are located at a distance of about 100 km north the volcanic axis of Sunda Arc.

The proposed site of the first NPP is in Ujung Watu. It is located on seashore of the Java sea on Muria Peninsula. Around the Ujung Watu site, lapilli-tuff of Ujung

Watu formation is widely distributed. It is composed of calcareous sandstone, claystone, and limestone rich in fossils.

Other available data about Ujung Watu include the moisture content of the soil (54.5 - 58.4%), the rate of rainfall (2,992 mm/yr.), and the groundwater flowrate (0.15 cm/day) [16].

2.3. Projection on High Level Radioactive Waste Management

In parallel to the plans for building NPPs in Indonesia, the projection of HLRW that will be generated should also be considered. This consideration is very important since HLRW will pose significant challenges when it accumulates as the reactors are operated to produce electricity. HLRW mainly consists of the spent fuel itself.

Eventhough the generation of HLRW will not be in significant quantities for at least 10 to 20 years after the beginning of plant operations, the technology and management of these wastes should be evaluated and implemented as soon as construction of the NPP begins. Development of HLRW management techniques will also show the Indonesian people that the scientists, engineers and utility managers responsible for safe operation of these facilities are aware of the ramifications of nuclear power. This, hopefully, will increase public acceptance by the people of Indonesia and the world.

The technological solutions for HLRW generally accepted around the world include a moderate cooling period (approximately 10 to 20 years) to decrease radioactivity. Afterwards, the management effort is focused on development of ways to safely isolate the HLRW from the AE. This is done by placing HLRW in stable, deep geological formations with a number of containment barriers.

Table 2.4 shows the estimated amount of spent fuel generated from 10^3 GWh output of LWR and PHWR, which is based on the calculation with capacity factor of 80% [17]. From the table, we can see that for a 10,000 GWh output from a PWR, 3.1 MTU of spent fuel will be generated. Using a CANDU 3 as the reactor, however, the spent fuel generated will be 23.7 MTU.

The principal activities in the final disposal stages consist of encapsulation of the spent fuel and underground disposal. The spent fuel is required to be packed in a specially designed container and then emplaced in either the unsaturated zone (above the groundwater table) or the saturated zone (below the groundwater table).

The investigation of the proposed HLRW disposal site facility from a geological stand point has been conducted in Indonesia. One of the promising option of this investigation is to dispose the HLRW in an Island in close proximity to the Ujung Watu site (northern side of the Central Java island), and this means it is convenience in transportation. The proposed repository site facility is on the southern side of Genting Island, located in Karimunjawa archipelago [18]. The archipelago is administered by Central Java province. The Genting Island is the most eastern part of the Karimunjawa archipelago. It takes approximately 2 hours by motorboat to reach Genting Island from the main archipelago [18].

Table 2.4. Spent Fuel Generated from Various Types of Nuclear Power Plant [17]

Type/Vendor	600 MWe			900-1000 MWe					Advanced Type		
	PWR		PHWR	PWR			BWR	PHWR	AP600	SBWR	CANDU3
	WH	NPI	AECL	WH	NPI	NPI	GE	AECL	WH	GE	AECL
Elect. Output (1) MWe	615	645	638	866	1060	994	952	881	631	635	432
Fuel Cycle Length (2) EFPH	10.6	11.6	12	11	11.6	11	17.3	12	15.3	22.2	12
Ave. Dis. Burnup (3) GWD/HTU	41.1	45.3	7.3	42.2	45.2	47.7	38.4	6.5	40.7	38.2	6.5
Generated SF for 1 Fuel Cycle (4)	14.8	14.8	112	20.3	24.0	20.3	39.7	174	22.1	35.4	89.5
Generated SF for Ave. 1 Year Operation	13.4	12.2	89.6	7.7	19.9	17.7	22.0	139.2	13.9	15.3	71.6
Generated SF for 1000 Gwh Generation	3.1	2.7	20.0	2.9	2.7	2.5	3.3	22.6	3.1	3.4	23.7

(1), (2), (3), (4) = Source "Fuel Cycle Evaluation" (INPB-D-002)
(5) = (4)/{(2)/12 X 1/0.8} (80% capacity factor)
(6) = (4)/{(1) X (2)/12 X 8760 / 1000}
EFPM = Effective Full Power Month

The main reasons for considering the southern side of Genting Island as the proposed location of HLRW disposal site facility are:

1. The area is remote
2. Very low in population density
3. High strength basaltic rock formation
4. The economic growth potential of the island is very small
5. Close proximity to the proposed NPP site

The Genting Island is about 2.6 km long and 800 m wide. The total area is approximately 135 Ha (333.6 acres). The southern tip of the Genting Island covers an area of approximately 30 Ha (74.1 acres). The elevation of this area is between 0 m to 40.5 m above sea level. The average temperature is about $20^{o}C$ - $35^{o}C$ with a relatively high humidity due to the influence of sea winds. Like any other place in Indonesia, Genting Island has two seasons: dry season and rainy season. Dry season runs from June until August. Rainy season runs from November through March. Transitional season runs from April through May and September through October.

2.4. Constraints Facing the HLRW Management

At this time Indonesia does not have practical experience handling HLRW. Therefore, continuous monitoring of the progress of the technology taking place in other countries should be maintained. Accordingly, first-hand experience by training of Indonesian's personnel to conduct research and to learn the technical skills would be very beneficial.

The infrastructure that supports transportation of the hazardous materials should be built according to the international standards. Regarding radioactive materials transportation, Indonesia uses the guidelines of the International Atomic Energy Agency (IAEA) with some modifications to suit the conditions in Indonesia. However, no specific guidelines have been established regarding HLRW management.

The development of rules and regulations for HLRW management should be done as conservatively as possible. The main purpose of rules and regulations is to ensure health and safety for radiation workers and public in large. However, the approach must also be practical. The regulations applied in other countries must be carefully examined and adopted, whenever necessary.

2.5. Regulatory Body of Atomic Energy Activities

Basic stipulations of atomic energy and its implementing regulations in the Indonesia have been based on the Act No. 31, Year 1964. The National Atomic Energy Agency (Badan Tenaga Atom Nasional - BATAN) was established in accordance with Government Regulation No. 33, Year 1965 [12].

The principle function of BATAN is to develop, regulate, monitor, and research the application of atomic energy in Indonesia for the safety, health and prosperity of Indonesian society [19]. Therefore, BATAN should also initiate the promotion of developing NPP programs in the country.

Regulations by BATAN on principle rules for transporting radioactive elements can be found on the regulation No. 07/DJ/5/II/74 [20]. However, transportation of HLRW is not specifically discussed in the regulation.

Besides BATAN, the Department of Demographics and Living Environment (Departemen Kependudukan dan Lingkungan Hidup) of the Indonesian Government has the jurisdiction regarding analyses of the impacts on the environment of a NPP [21].

3. LITERATURE REVIEW

3.1. The Integrated Repository Performance Assessment

Many aspects regarding HLRW repository parameters must be addressed in order to analyze performances of the repository site. They include near-field environment, geosphere, and biosphere conditions. Several papers have discussed and investigated the PA of repository sites.

A paper by Andersson and Norbby discusses the PA program in relation to final disposal of spent nuclear fuel and other HLRW. The paper describes five main deterministic models to address conceptual uncertainties and coupled effects [22] :

1. A near-field model. This model describes the transport of radionuclides through the buffer zone into a fractured rock matrix.

2. The groundwater flow and transport code using a finite element method. In the model, the rock is considered as porous medium and the flow is governed by Darcy's law.

3. Estimates the variability of flow conditions within a repository region.

4. A computer code used for flow and stress analysis in deformable, saturated, fractured rock including thermo-hydro-mechanical effects.

5. A three-dimensional model to analyze mechanical and thermomechanical behavior of a repository with surrounding rock due to glaciation and thermal loading.

The purpose of the investigation is to develop and apply the methodology that can be used as the basis for developing regulations of HLRW management.

O'Connel emphasizes the integrated PA of the waste packages and engineered barrier system (EBS) at the Yucca Mountain site [23]. The model development for single waste packages indicates that the radionuclide release rate performance is sensitive to water flux.

Evaluation of near-field thermal environmental conditions for HLRW in tuff geologic medium was done by Altenhofen and Eslinger [24]. A three-dimensional heat conduction model for the underground repository facility was used to evaluate near-field host rock temperatures throughout the 10,000-year isolation period. The result can be seen in Table 3.1.

Shaw proposed a methodology of HLRW repository PA [25]. The methodology is divided into eleven nodes.

Table 3.1. Waste Emplacement Design Parameters Used in Repository-Scale Thermal Model [24]

Panel	Panel Area $m^2(x10^{-5})$	Average Age Year	Average Heat KW/MTU	Thermal Load W/m^2	Mass Load Kg/m^2
1	0.43	31.3	0.36	13.58	37.22
2	1.29	29.8	0.52	13.82	26.83
3	3.00	27.4	0.65	14.23	21.92
4	3.13	24.8	0.72	14.70	20.53
5	3.00	23.3	0.84	14.97	17.85
6	2.35	22.5	0.93	15.11	16.31
7	1.70	21.9	1.00	15.23	15.25
8	1.57	21.5	1.03	15.31	14.89
9	0.85	21.3	1.04	15.35	14.76
10	0.20	21.3	1.04	15.35	14.76
11	1.31	20.6	1.07	15.49	14.48
12	2.22	20.2	1.10	15.57	14.21
13	2.61	19.6	1.12	15.70	14.05
14	3.26	18.8	1.14	15.84	13.88
15	3.66	17.9	1.15	16.05	13.95
16	3.66	17.0	1.16	16.25	14.01
17	3.66	16.3	1.14	16.41	14.37
18	1.83	-	-	-	-

Node 1 represents the average long hydrologic flux into the repository. Node 2 represents earthquake-induced canister failures. Node 3 represents water table changes caused by earthquakes. Node 4 represents potential occurrences of volcanoes in the vicinity of the site. Node 5 represents the water table changes due to the volcano scenario. Borehole stability is represented by node 6. The distribution of canister lifetime is represented by node 7. Geochemical effects are represented by node 8 with uncertainty in the solubility of the Uranium waste. Rock fracture modeling is represented by node 9. The representation includes how groundwater flow is partitioned between fracture flow and matrix flow. Node 10 represents the effective porosity of fractured rock, both under fracture flow and under matrix flow. The uncertainty in the retardation of nuclides during hydrologic transport through the tuff is represented by node 11. The methodology was then developed and applied to the Yucca Mountain site. The code is called IMARC [26], which is an acronym for Integrated Multiple Assumptions and Release Calculations.

Figure 3.1 shows the master logic tree for demonstration calculations. The methodology employs a logic tree approach to model uncertainties that are used in the probabilistic assessment of repository performance.

The linkage between source term and near-source for high level repository PA was conducted by researchers from Risk Engineering, Inc. [27]. The assessment was mainly to examine and define the interactions among the source term, the near-source region, the effects of temperature variations caused by emplacement of waste material, the behavior of the waste containers, and the effects on local groundwater hydrology. The study was conducted primarily for the Yucca Mountain proposed repository site. Phase II of this study was to further explore and elaborate in four areas: waste containment, source term, thermal loading, and hydrologic flow [28].

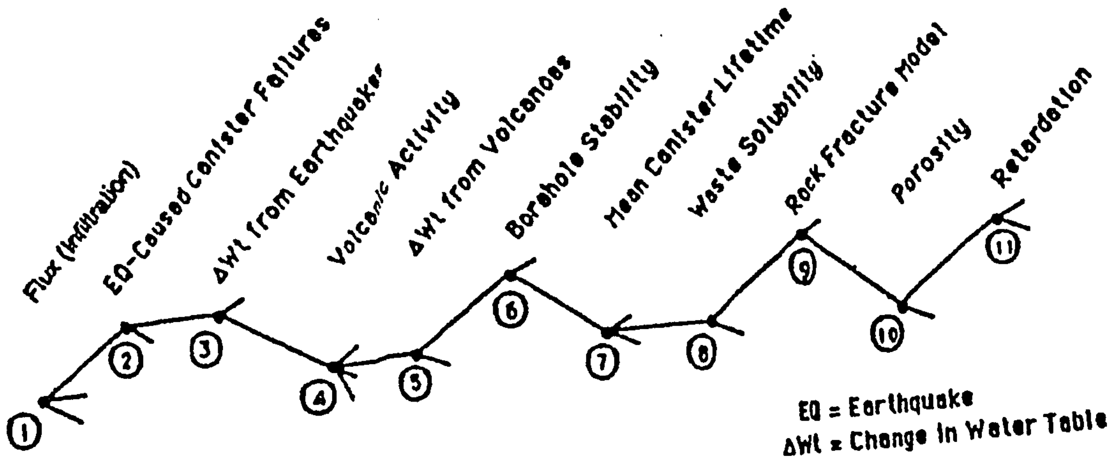

Figure 3.1. Master Logic Tree for Demonstration Calculations [25]

A more general model for integrated PA has been developed by Golder Associates, Inc. [7]. The computer code is called RIP, which stands for Repository Integration Program. It concentrates on the integration of the entire system, and utilizes relatively high-level descriptive models and parameters. The integration consists of four coupled components that address waste package behavior and radionuclide release, fluid flow and radionuclide transport through the geosphere, disruptive events that can affect system parameters, and radionuclide fate and effect in the biosphere. The RIP computer code is employed to conduct TSPA of the proposed repository site at southern tip of Genting island, Karimunjawa archipelago, Central Java, Indonesia. The complete analysis is described in great detail on Chapter 5.

The identification of fault displacement hazards and seismic hazards of a geologic repository was pointed out by McConnell [29]. The paper outlined appropriate investigations that can be used to identify fault displacement hazards and seismic hazards at a geologic repository. The identification leads to three types of faults. Type III faults are the faults located outside the controlled area and require no further investigation. Type II faults are the faults located in the controlled area or outside the controlled area but may affect isolation within the controlled area. These may be subject to further investigation. The faulting that occurred during the Quarternary Period is considered characteristic of the controlled area. This is considered Type I and subject to further investigation.

3.2. The Near-field Environmental Conditions

Andersson suggests that the near-field region of a geological repository is a spatially complex region composed of both engineered and natural barrier materials [22]. The functional requirements for geological disposal of nuclear waste are:

1. The waste must not be released to the biosphere at a rate or in concentrations deemed to present an unacceptable hazard.

2. The waste must be removed and isolated from the effects of human activity or catastrophic natural events.

3. The technology to implement disposal must be readily available, and achievable at a reasonable cost.

4. The potential future retrieval of some types of disposed nuclear wastes if so required by national policy.

5. The process that control safe performance of nuclear waste disposal must be sufficiently well characterized and understood for modeling, and adequate, relevant data can be obtained and used in such models to reliably demonstrate predicted performance.

To meet the regulatory requirements, the predominant role of the engineered barriers system (EBS) has gained a growing consensus worldwide. The uncertainties of the performance of natural phenomena lead to the design of the EBS using a multiple-barrier approach. In this concept, a series of engineered and natural barriers is nested one inside the other. A schematic of the EBS can be seen in Figure 3.2. The innermost barrier is the waste form. The outer layers are a container, a backfill or buffer, and the geologic formation. The primary function of the waste form matrix is to immobilize the radioactive materials. The container is used to isolate the nuclear waste from groundwater for a designated period of time. The backfill or buffer is used to reduce possible tectonic shearing forces on containers, prevent the container from settling within the emplacement tunnel, conduct heat from the engineered barriers, filter fine particles and colloids that may form during waste package reactions, retard the diffusional-transport rate of dissolved radionuclides, and chemically buffer the composition of intruding groundwater [30].

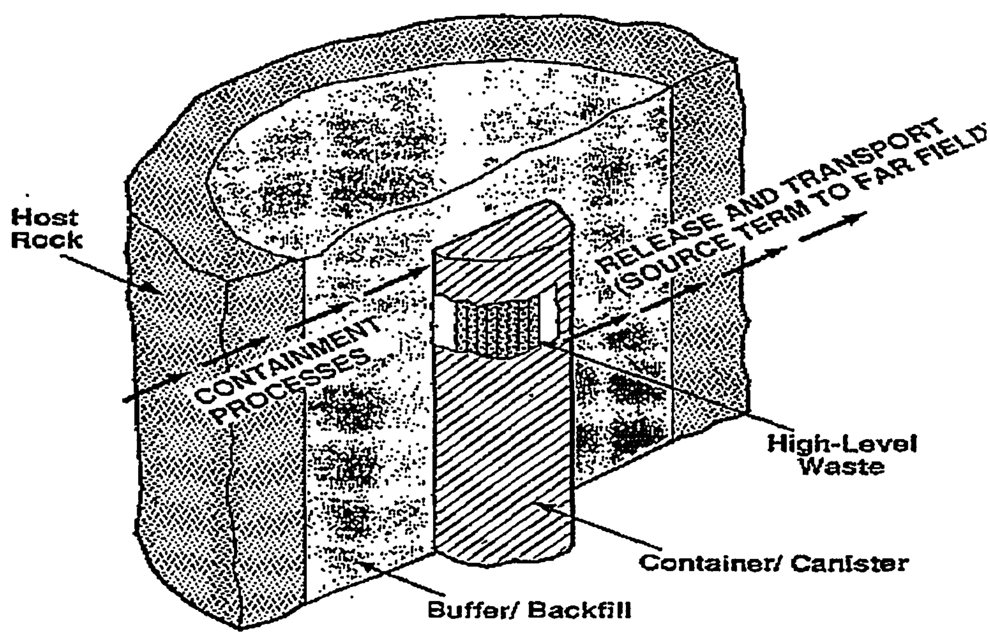

Figure 3.2. Schematic of Engineered Barriers System [31]

3.2.1. Source-term of the Waste Package

The Characteristic Database System (LWR assemblies database by United States DOE) can be used to determine the abundance of radionuclides in a spent fuel assembly as a function of burnup [14]. However, the information is not adequate. The uncertainty of location of radionuclides in the spent fuel is high because the location of these radionuclides is not uniform. The researchers have estimated the proportion of the key radionuclides within the separate region of spent fuel [31]. This information is important to understand the release pathway of a particular radionuclide to the environment. Figure 3.3 shows the compositions of the key radionuclides within fuel assembly before and after fissioning take place.

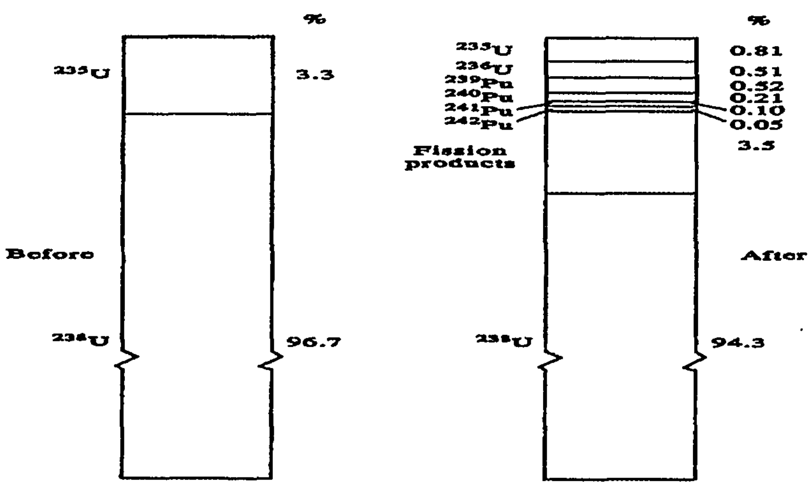

Figure 3.3. Schematic Diagram of Spent Fuel Showing Different Source Region with Characteristic Radionuclides [31]

Einziger discusses the key factors that affect the spent fuel source term. They are fuel and cladding oxidation, cladding splitting kinetics, cladding rupture (breach) kinetics, and the physical parameters of the fuel [32]. The model was then developed to calculate the amount of fuel oxidized from UO_2 and U_3O_8 as a function of time and temperature when canister and rod breaches. The model is applicable for temperature ranges from 100°C to 300°C. The study shows that below 200°C, the preponderance of total splitting time is due to incubation. Below 150°C, the cladding oxidizes completely before the cladding crack can propagate down the complete rod. Below 100°C, the spent fuel is not expected to form U_3O_8 during the lifetime of a potential repository. Above 300°C, all fuel cladding can be considered as oxidized to ZrO_2 and the fuel oxidized to U_3O_8 in a repository time frame.

Manaktala pointed out several characteristics of the spent fuel, cladding, and the waste package that are likely to influence the long-term performance of spent fuel in a geologic repository but may not be adequately addressed in current PA models [33]. For instance, the increase of surface area of fuel as a result of pellet cracking may not be adequately addressed. This information is important to determine the release of radionuclides from the fuel upon contact with water. Other information related to the distribution of fission products and actinides in the discharged fuel is valuable for the development of release models. Most studies related to the failure of the waste package have concentrated on the corrosion failure of metallic container as a result of contact of groundwater with the container.

These studies do not address electrochemical effects between the fuel, the cladding, and other waste package components. The possible detrimental effects of waste package materials on the spent-fuel degradation kinetics need to be considered in development of the source-term models.

3.2.2. Temperature Conditions at Waste Package

The analyses of thermo-hydrological behavioral of some areal mass loadings were done by Buscheck for the Yucca Mountain site [34]. The examination was completed to estimate the temporal and spatial extent of the temperature and saturation changes during the first 100,000 years. Three primary strategies for thermal loading were introduced:

1. The possibility to limit thermal load and distribute it such that it has no impact on hydrological performance

2. The impact of thermo-hydrological process for intermediate thermal loads to the geological disposal system

3. The impact of thermo-hydrological process for high thermal loads to the geological disposal system

The calculations were done using computer code called V-TOUGH (Vectorized Transport of Unsaturated Groundwater and Heat). The other issue raised in the report was the challenge to adequately understand repository heat-driven vapor and condensate flow utilizing data from long-term in situ heater tests. This is required to determine the potential for the major repository-heat-driven sources of fracture flow to impact waste package performance and radionuclide transport.

The thermal conditions in the vicinity of the waste package are important factors in predicting container corrosion and radionuclide dissolution processes as reported by Lingineni [35]. The study determines waste package temperatures from the time history of the average repository temperature. The repository is assumed to be a rectangular panel that has dimension 427 m X 937 m.

3.2.3. The UC System

The robust waste package design has been introduced to fulfill the multiple barrier concept. One of the designs was presented by Doering [36]. Each barrier in this design contributes to the overall performance of the package. It is expected that the robust waste package design will provide complete containment and isolation for more than 1000 years. Figure 3.4 shows the conceptual design.

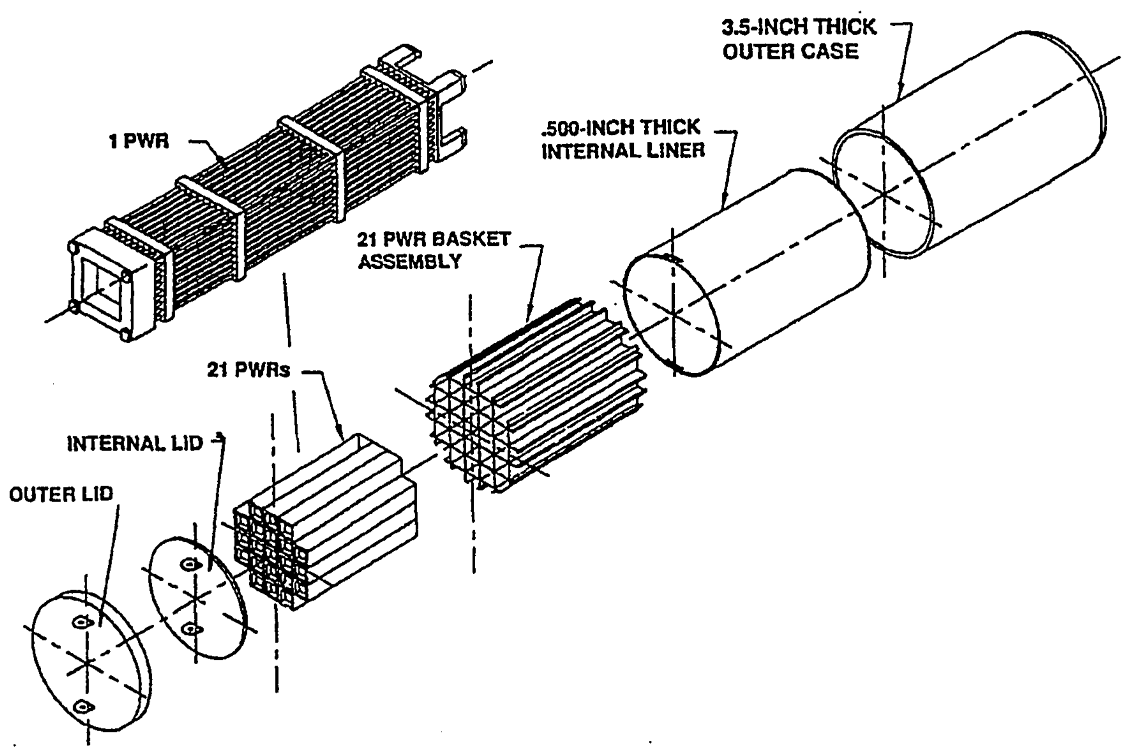

Figure 3.4. Multi-Barrier Robust Waste Package Design [36]

One of the most important factors regarding the material selection is the ability to minimize the possibility of failure of the waste package over some period of time. The material degradation can be caused by oxidation or corrosion. The two basic metallic containment barriers are corrosion-allowance materials and corrosion-resistant materials. In the MPC concepts, the outer barrier uses a corrosion allowance material and the inner barrier uses a corrosion resistant material [37].

Stuart described the MPC designed by Nuclear Assurance Corporation (NAC) that has been reviewed by the US Nuclear Regulatory Commission (USNRC) [38]. Some advantages of the NAC design include high capacity of spent fuel in a single transportable package, criticality control, and sufficient heat transfer capability to keep cladding temperatures within the regulatory limits. The fuel parameters can be as high as 4.2% U235 initial enrichment, residual heat content as high as 45,000 MWd/MTU, and a decay heat as short as 6.5 years. The weight of the fully loaded package and its accessories is maintained to be less than 125 tons for all handling conditions. The container design can be seen in Figure 3.5.

The design was tested using several different computer codes. The ANSYS code was used to analyze the structural model using finite element. In the shielding analysis the use of XSDRNPM and MORSE (Monte Carlo methodology) were used. In this study, approximately 12 million neutrons and gammas were tracked through the shielding model. The tests incorporated two types of scenarios. First, the test for storage of spent fuel at reactor site followed by transporting the container to another location where it may be stored for extended period of time. Second, testing of the possibility for immediate transport of the cask. In addition to that, demonstration of containment integrity was also conducted.

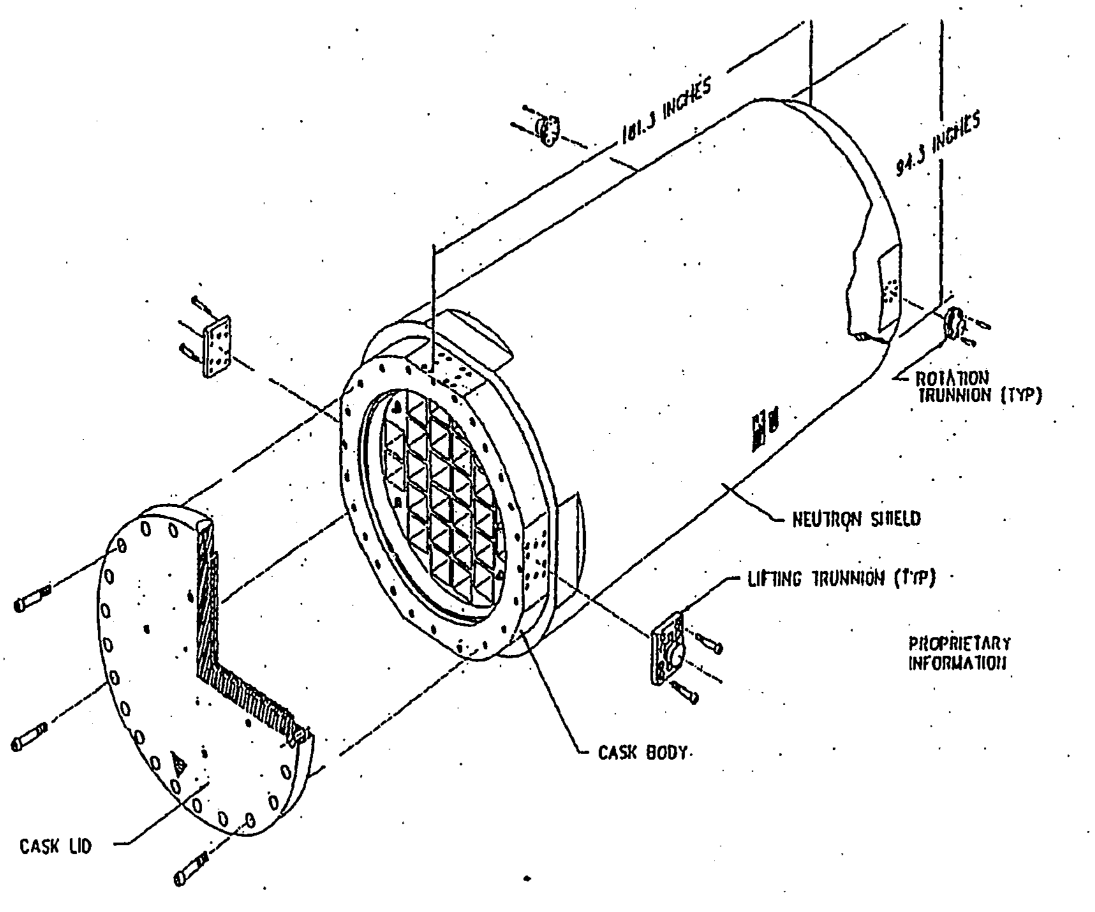

Figure 3.5. NAC-STC Dual Purpose Cask [38]

3.2.4. The EBS Failure Modeling

The EBS is the waste package and the engineered barrier in which the waste package is emplaced. The performance of the EBS must also be evaluated to determine its reliability. Since it is expected to perform adequately for approximately 100000 years, many EBS failure models have been introduced.

Bullen has completed several studies to determine the impact of container failure mechanisms and container failure rates on radionuclide release rates from the EBS [39,40,41,42,43]. These efforts use a mathematical model to predict the cumulative failure distribution for the Containment Barrier System (CBS) employed in a deep geological facility. The model can incorporate several designs from single, thin-walled metal barriers to multiple, redundant barriers and also thick-walled containment. The cumulative failure rate is considered as a function of the mean container lifetime, the threshold container failure time, and the failure rate at the mean container lifetime. These parameters are the variables in Weibull and exponential distributions. In addition, the model also includes factors to describe containers failed at emplacement, early failure rate, and the effects of multiple, concurrent repository environments on the performance of the entire container population [40]. Previous evaluations completed on the CBS with this mathematical model include single metal alloy 825 container, a multiple barrier container consisting of titanium clad, Alloy C-4, and carbon steel containers. The results show that the use of a multiple barrier container will delay the failure time as much as 50% longer than single barrier container [40]. The multiple barrier system using alloy 825 and steel container appears superior in design reliability. This is due to the robustness and the nature of redundancy that are employed. Figure 3.6 shows the cumulative failure distribution for multiple barrier, single barrier Alloy 825, and single barrier steel containers. The dominant heat transfer mechanism in this study is conduction and the areal power density is 114 KW/acre.

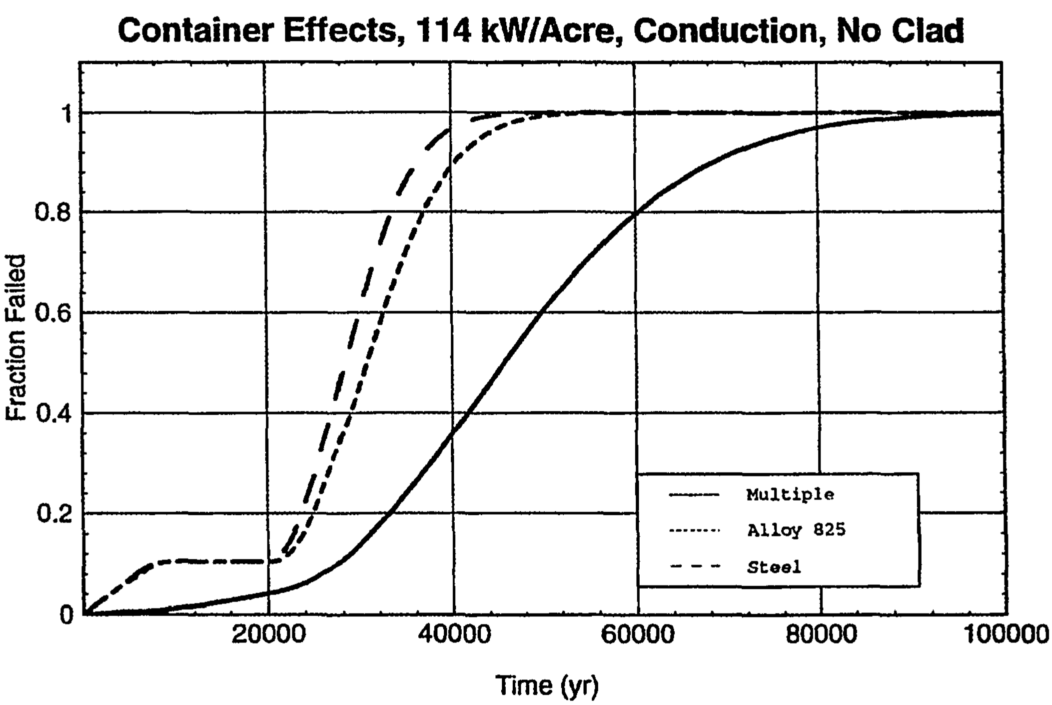

Figure 3.6. Cumulative Container Failure Distribution for Multiple Barrier, Alloy 825, and Steel Containers with Conduction Heat Transfer and Areal Power Density of 114 KW/acre [40]

Three different thermal regions have been developed by Bullen for three thermal loading scenario, namely 36 KW/acre, 57 KW/acre, and 114 KW/acre. These thermal regions are applied for conduction, convection, and heat-pipe thermal redistribution mechanisms.

3.3. Materials Selection for HLRW Container

Regarding the materials for a potential HLRW container McCright has done some testing for conceptual metal barrier materials for a tuff repository [44]. Initially, 17 alloys were selected as the candidate materials using criteria such as mechanical properties, weldability, corrosion resistance, and cost. After the alloys were ranked, the team decided to consider four materials for further investigation. They are AISI 304L, 321, 316L, and Alloy 825. Besides the stainless-steel based alloys, the copper-base materials were also considered, namely CDA 102 (oxygen-free copper), CDA 613 (aluminum bronze), and CDA 715 (70-30 copper-nickel). The nominal compositions of these materials can be seen in Table 3.2 and Table 3.3. 304L and other austenistic alloys are expected to show excellent general corrosion resistance in aerated dry steam environments, in wet steam, and in vadose water [44]. The limiting factor for using 304L is the possibility of much more rapid penetration via localized or stress-assisted forms of corrosion.

Table 3.2. Materials Composition for Austenitic Alloys [44,46]

Materials	C	Mn	Cr	Ni	Other elements
			Chemical Compositions (wt %)		
304L	0.03 (max)	2.0 (max)	18-20 (max)	8-12 (max)	
316L	0.03 (max)	2.0 (max)	16-18 (max)	10-14 (max)	2 - 3 % Mo
321	0.08 (max)	2.0 (max)	17-19 (max)	9-12 (max)	9-12 (5X%C) Ti min
825	0.05 (max)	1.0 (max)	19.5-23.5 (max)	38-46 (max)	Mo: 2.5 - 3.5 Ti: 0.6 - 1.2 Cu: 1.5-3.0 Al: 0.2 max

Table 3.3. Materials Composition for Copper and Copper-Based Alloys [44]

Materials	Cu	Fe	Pb	Sn	Al	Mn	Ni	Zn
				Chemical Compositions (wt %)				
CDA 102	99.95 (min)	-	-	-	-	-	-	-
CDA 613	92.7 (nom)	3.5	-	0.2-0.5	6.0-8.0	0.5 (max)	0.5	-
CDA 715	69.5 (nom)	0.4-0.7	0.5 (max)	-	-	1.0 (max)	29.0-33.0	1.0 (max)

However, Alloy 825 contains 2-3% Mo and is Ti-stabilized to combat localized and intergranular forms of corrosion. The copper and copper-base alloys show several advantages, co-exist thermodynamically with water (under some conditions), the driving force for corrosion and oxidation is smaller, and localized corrosion is less severe [44,45]. However, in high gamma radiation field the corrosion and oxidation rates of these materials are higher [45]. High gamma radiation field occur during early part of the containment period.

General corrosion rates for austenitic stainless steel materials have been determined for 304L, 316L, and Alloy 825 [44]. These data were drawn from an exposure of the materials in water at different temperatures under radiation environment for two month period. The exposure data for 304L in the irradiated environments can be seen in Table 3.4. The corrosion rates can be seen in Table 3.5.

Table 3.4. Oxidation Test Results for 304L in Irradiated Environments [44]

		Corrosion penetration rate (μm/yr)	
Materials =	Environment =	105°C 3 x105 rads/hr	105°C 6 x 105 rads/hr
304L (Solution annealed)	water	0.31 ; 0.31	0.36
	water and tuff	0.29 ; 0.32	0.31
304L (Solution annealed and heat-treated condition)	water	0.23 ; 0.37	0.51
	water and tuff	0.25 ; 0.30	0.55

Table 3.5. General Corrosion Rates for Austenitic Stainless Steels in Water at Different Temperatures [44]

Alloy	Temp (°C)	Time (hr)	Medium	Corrosion rate (μm/yr)	
				Average	Std. Dev.
304L	50	11,512	water	0.13	0.02
316L	50	11,512	water	0.15	0.01
825	50	11,512	water	0.21	0.01
304L	80	11,056	water	0.09	0.01
316L	80	11,056	water	0.11	0.01
825	80	11,056	water	0.11	0.01
304L	100	10,360	water	0.07	0.02
316L	100	10,360	water	0.04	0.01
825	100	10,360	water	0.05	0.02

McCright reported that copper is expected to resist attack by pure steam [45]. However, if facilitated by oxygen, the corrosion may occur. Figure 3.7 shows the effect of nickel on the corrosion of copper-nickel alloys in oxygen-containing water and steam at saturated pressure.

The report concludes that copper-nickel alloys typically corrode in dry and good-quality wet steam at rates less than 2.54 μm/yr. When temperature is in the range between 300°C and 350°C, a sharp increase in corrosion rate is expected to take place.

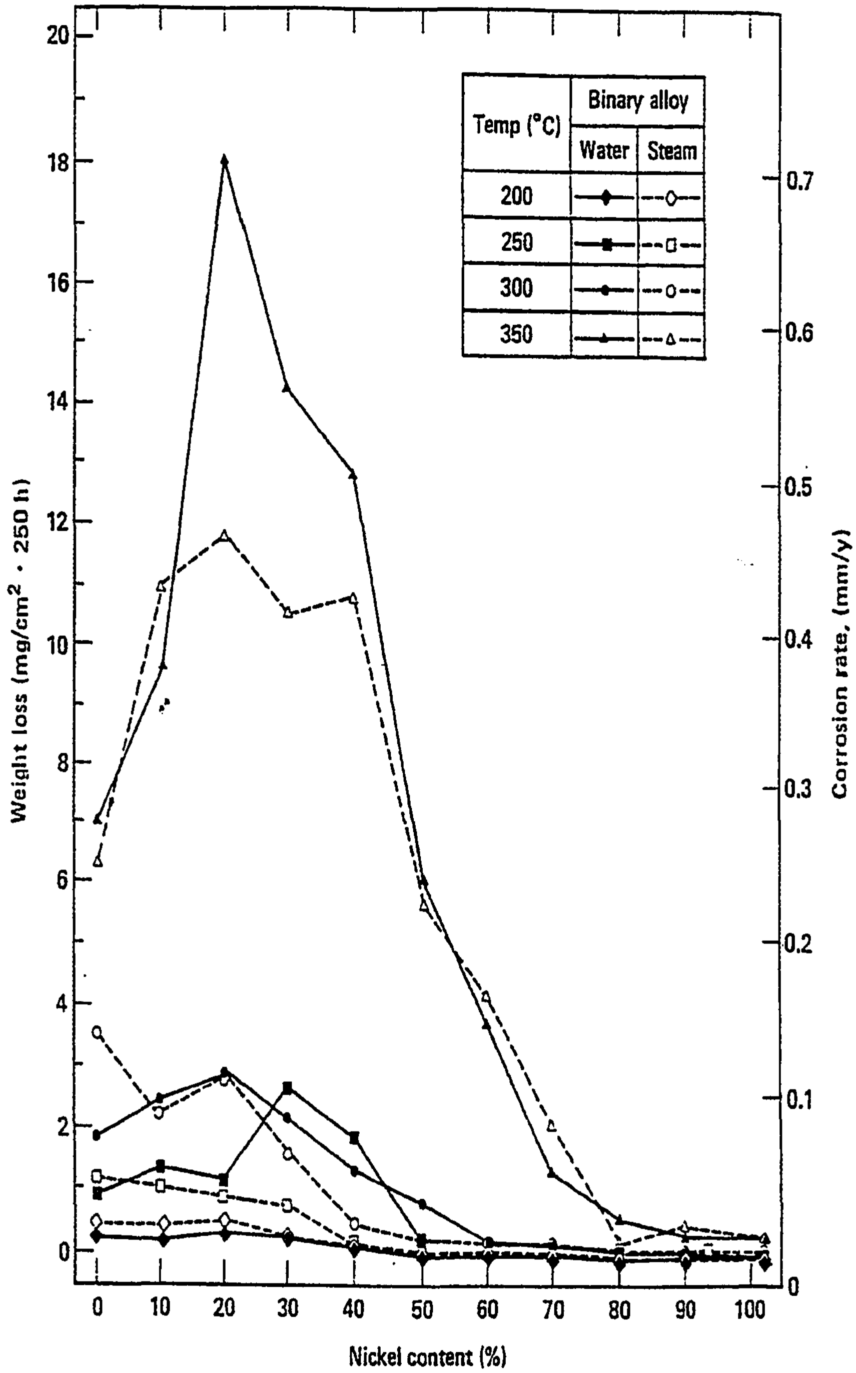

Figure 3.7. Effect of Nickel on the Corrosion of Copper-Nickel Alloys in Oxygen-Containing Water and Steam at Saturated Pressure [45]

3.4. Assessment of Disruptive Events

As equally important as understanding the aspects in the near-field conditions is an assessment of the possibilities of occurrences of natural catastrophic phenomena within disposal facility. The assessment of volcanic and tectonic hazards to the waste repositories for Yucca Mountain site was conducted by Wallmann [47]. The analyses were conducted using the computer code RIP [7]. The events are specified as a disruptive rate (events/year), event characteristics and magnitude, and consequences associated with an event. The results show that the disruptive events have no visible effect on the overall release, which is dominated by C-14.

Davidson discussed the dominant pathways for the large circulation of groundwater through the granite batholith. Low-dipping fracture zones are the most dominant pathway [48]. In the fracture zones, the permeabilities can vary by as much as 6 orders of magnitude over lateral distances of only a few meters. The regions with high permeability are associated with low stress magnitudes, whereas regions of low permeability correspond to the locations with high stress magnitude.

Ahokas shows that fracture zones control hydraulic head and groundwater flow [49]. Simulations and modeling of fracture zones were also done on fractured crystalline rock using discrete-fracture model [50,51].

4. TSPA METHODOLOGY FOR THE GENTING ISLAND REPOSITORY SITE

4.1. Identification of Source Terms/Inventory of HLRW

The identification of the estimated radionuclide inventory, or source term, is a very important parameter in PA due to the potential impact of the release of these radionuclides to the AE. These radionuclides will be encapsulated into waste-forms which have high mechanical integrity, low solubility, low surface area exposed to water, and low solid-phase diffusion rates. However, the radionuclides are expected to diffuse to the waste-form surface if brought into contact with groundwater. One potential mechanism for radionuclide release is long-term dissolution of the UO_2 fuel matrix. Another release mechanism is diffusion of radionuclides along the surface of grains and cracks in the fuel.

The waste-form is assumed to be spent fuel assemblies from PWR reactors. The waste package design includes a modified MPC design with 10 cm of carbon steel surrounded by 1 cm of Alloy 825.

It is projected that 10 NPPs will be built in Muria Peninsula, Central Java, over the next 25 years beginning in the year 2004. After the first NPP reaches the end of its design life (2036), the number of spent fuel assemblies (SFA) which will have already accumulated will be 7,489 assemblies. Each small MPC design can hold 9 PWR fuel assemblies, then the required number of containers will be approximately 850 containers.

Table 4.1 shows 13 long-lived radionuclides that will remain in the waste-form for very long times (over 10000 years) which have 3.27% initial enrichment and 33,000 MWd/MTHIM burnup. Table 4.2 shows the total number SFA of PWR calculated.

Table 4.1. The Significant Inventory of Radionuclides in the SFA [14]

Decay Time (Years)	Curies/MTIHM												
	C14	Se79	Tc99	I129	Cs135	Ra226	U234	U235	U238	Np237	Pu239	Pu240	Pu242
1	1.39	4.9e-1	1.6e1	3.8e-2	5.6e-1	5.8e-8	1.2	1.8e-2	3.1e-1	4.8e-1	4.0e2	5.8e2	2.2
10	1.39	4.9e-1	1.6e1	3.8e-2	5.6e-1	4.3e-7	1.3	1.8e-2	3.1e-1	4.9e-1	4.0e2	5.8e2	2.2
20	1.39	4.9e-1	1.6e1	3.8e-2	5.6e-1	1.3e-6	1.4	1.8e-2	3.1e-1	5.0e-1	4.0e2	5.8e2	2.2
30	1.39	4.9e-1	1.6e1	3.8e-2	5.6e-1	2.7e-6	1.5	1.8e-2	3.1e-1	5.1e-1	4.0e2	5.9e2	2.2
50	1.39	4.9e-1	1.6e1	3.8e-2	5.6e-1	7.3e-6	1.7	1.8e-2	3.1e-1	5.4e-1	4.0e2	5.9e2	2.2
100	1.37	4.9e-1	1.6e1	3.8e-2	5.6e-1	3.0e-5	2.0	1.8e-2	3.1e-1	6.1e-1	4.0e2	5.8e2	2.2
200	1.36	4.9e-1	1.6e1	3.8e-2	5.6e-1	1.4e-4	2.4	1.8e-2	3.1e-1	7.5e-1	4.0e2	5.8e2	2.2
300	1.34	4.9e-1	1.6e1	3.8e-2	5.6e-1	3.3e-4	2.6	1.8e-2	3.1e-1	8.8e-1	4.0e2	5.7e2	2.2
500	1.31	4.9e-1	1.6e1	3.8e-2	5.6e-1	9.9e-4	2.7	1.8e-2	3.1e-1	1.1	3.9e2	5.6e2	2.2
1000	1.23	4.9e-1	1.6e1	3.8e-2	5.6e-1	4.1e-3	2.8	1.9e-2	3.1e-1	1.3	3.9e2	5.3e2	2.2
2000	1.09	4.8e-1	1.5e1	3.8e-2	5.6e-1	1.5e-2	2.8	1.9e-2	3.1e-1	1.5	3.8e2	4.8e2	2.2
5000	7.6e-1	4.7e-1	1.5e1	3.8e-2	5.6e-1	7.1e-2	2.7	2.0e-2	3.1e-1	1.6	3.5e2	3.5e2	2.1
10000	4.2e-1	4.4e-1	1.5e1	3.8e-2	5.6e-1	1.8e-1	2.6	2.2e-2	3.1e-1	1.6	3.0e2	2.1e2	2.1
20000	1.2e-1	4.0e-1	1.5e1	3.8e-2	5.6e-1	4.0e-1	2.5	2.4e-2	3.1e-1	1.6	2.3e2	7.1e1	2.0
50000	3.3e-3	2.9e-1	1.3e1	3.8e-2	5.5e-1	9.5e-1	2.2	2.9e-2	3.1e-1	1.5	9.7e1	2.9	2.0
100000	7.7e-6	1.7e-1	1.1e1	3.8e-2	5.4e-1	1.5	1.7	3.1e-2	3.1e-1	1.5	2.3e1	1.5e-2	1.8

Table 4.2. Number of Spent Fuel Generated

Year of Operation		Plant 1	Plant 2	Plant 3	Plant 4	Plant 5	Plant 6	Plant 7	Plant 8	Plant 9	Plant 10	Total
2004-2006	spent	64	0	0	0	0	0	0	0	0	0	64
2006-2008	spent	128	64	0	0	0	0	0	0	0	0	192
2008-2010	spent	192	128	64	0	0	0	0	0	0	0	384
2010-2012	spent	256	192	128	64	0	0	0	0	0	0	640
2012-2014	spent	320	256	192	128	64	0	0	0	0	0	960
2014-2016	spent	384	320	256	192	128	64	0	0	0	0	1344
2016-2018	spent	448	384	320	256	192	128	64	0	0	0	1792
2018-2020	spent	512	448	384	320	256	192	128	64	0	0	2304
2020-2022	spent	576	512	448	384	320	256	192	128	64	0	2880
2022-2024	spent	640	576	512	448	384	320	256	192	128	64	3520
2024-2026	spent	704	640	576	512	448	384	320	256	192	128	4160
2026-2028	spent	768	704	640	576	512	448	384	320	256	192	4800
2028-2030	spent	832	768	704	640	576	512	448	384	320	256	5440
2030-2032	spent	896	832	768	704	640	576	512	448	384	320	6080
2032-2034	spent	960	896	832	768	704	640	576	512	448	384	6720
2034-2036	spent	1153 (final core)	960	896	832	768	704	640	576	512	448	7489
2036-2038	spent	0	1153 (final core)	960	896	832	768	704	640	576	512	7041
2038-2040	spent	0	0	1153 (final core)	960	896	832	768	704	640	576	6529
2040-2042	spent	0	0	0	1153 (final core)	960	896	832	768	704	640	5953
2042-2044	spent	0	0	0	0	1153 (final core)	960	896	832	768	704	5313
2044-2046	spent	0	0	0	0	0	1153 (final core)	960	896	832	768	4609
2046-2048	spent	0	0	0	0	0	0	1153 (final core)	960	896	832	3841
2048-2050	spent	0	0	0	0	0	0	0	1153 (final core)	960	896	3009
2050-2052	spent	0	0	0	0	0	0	0	0	1153 (final core)	960	2113
2052-2054	spent	0	0	0	0	0	0	0	0	0	1153 (final core)	1153

Each reactor contributes = 1153 spent fuel assemblies during the reactor lifetime
Total number of fuel assemblies = 1153 X 10 = 11,530 spent fuel assemblies

4.2. Site Selection Criteria

There are two options for siting a deep geologic repository site. These include placement of the facility in the unsaturated zone (above the groundwater table) and placement in the saturated zone (below the groundwater table). The proposed repository at Genting Island, however, can be best designed such that the wastes are emplaced in the saturated zone since the site has a relatively shallow groundwater table.

The approach for site selection is also impacted by the thermal loading of the waste packages. There are two thermal loading strategies being considered: high-temperature loading where the surface temperature of the container exceeds the boiling point of water , and ambient-temperature loading where the container surface temperature is maintained below the boiling point of water.

4.2.1. Design of the Repository Site

The proposed design for the HLRW repository site at the southern tip of Genting Island, can be described by dividing the site into two areas or rings. The inner ring will contain a group of waste packages with an areal power density of approximately 100 KW/acre. This region is called the high temperature ring. The outer ring contains a group of waste packages with an areal power density of approximately 30 KW/acre, which represents the ambient temperature ring. The inner ring includes 75% of the waste packages and the outer ring includes the remaining waste-packages. Figure 4.1 shows schematically the proposed design of the repository site.

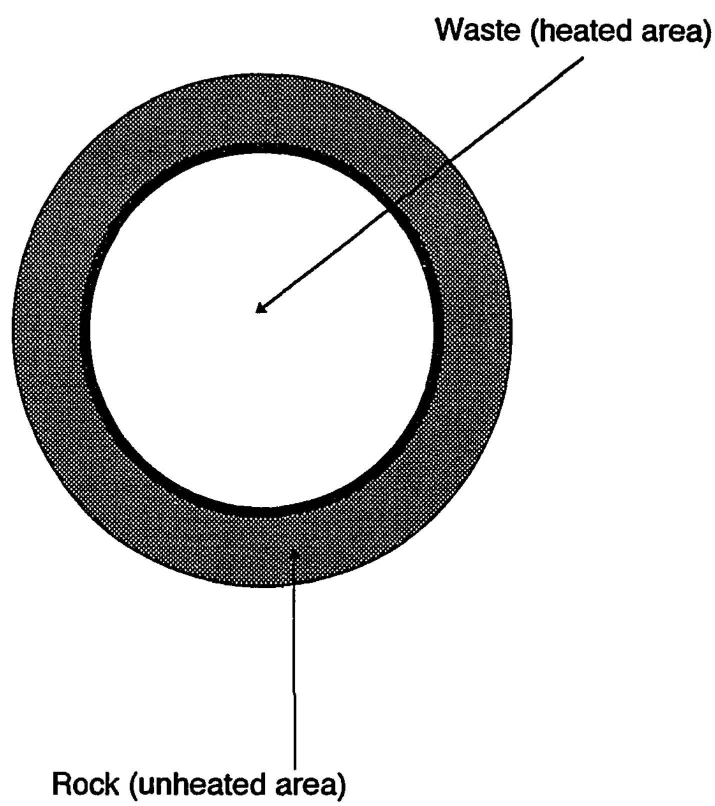

Figure 4.1. Schematic Diagram of the Cross Section of Area

The geological map of Karimunjawa archipelago can be seen in Figure 4.2 [18]. The groundwater level at Genting Island is considered shallow. It is influenced by the sea level and the rainfall rate. Figure 4.3 shows the groundwater level map of the southern tip of Genting Island [18].

The lithology of the area is such that the top layer of soil is mainly alluvium consisting of pebble, gravel, clay, coral limestone and coarse grained rocks. The thickness of this layer is approximately between 1.5 - 3.5 m. Below the layer is basalt, which consists of basaltic lava or alkaline basalts. The thickness of the layer is approximately between 24 - 35 m. The depth to groundwater is approximately 103 m.

The basaltic rock at Genting Island is classified as a strong rock. In the area, it is found that the approximate strength is 1550.36 kg/m^2 in compression. Basaltic rock is favorable because of its strength and the interlocking of fracture blocks, which can limit displacement along fractures. Therefore, the diffusion time of radionuclides along the rock fractures can be delayed, that eventually it will take longer for the radionuclides to reach the AE.

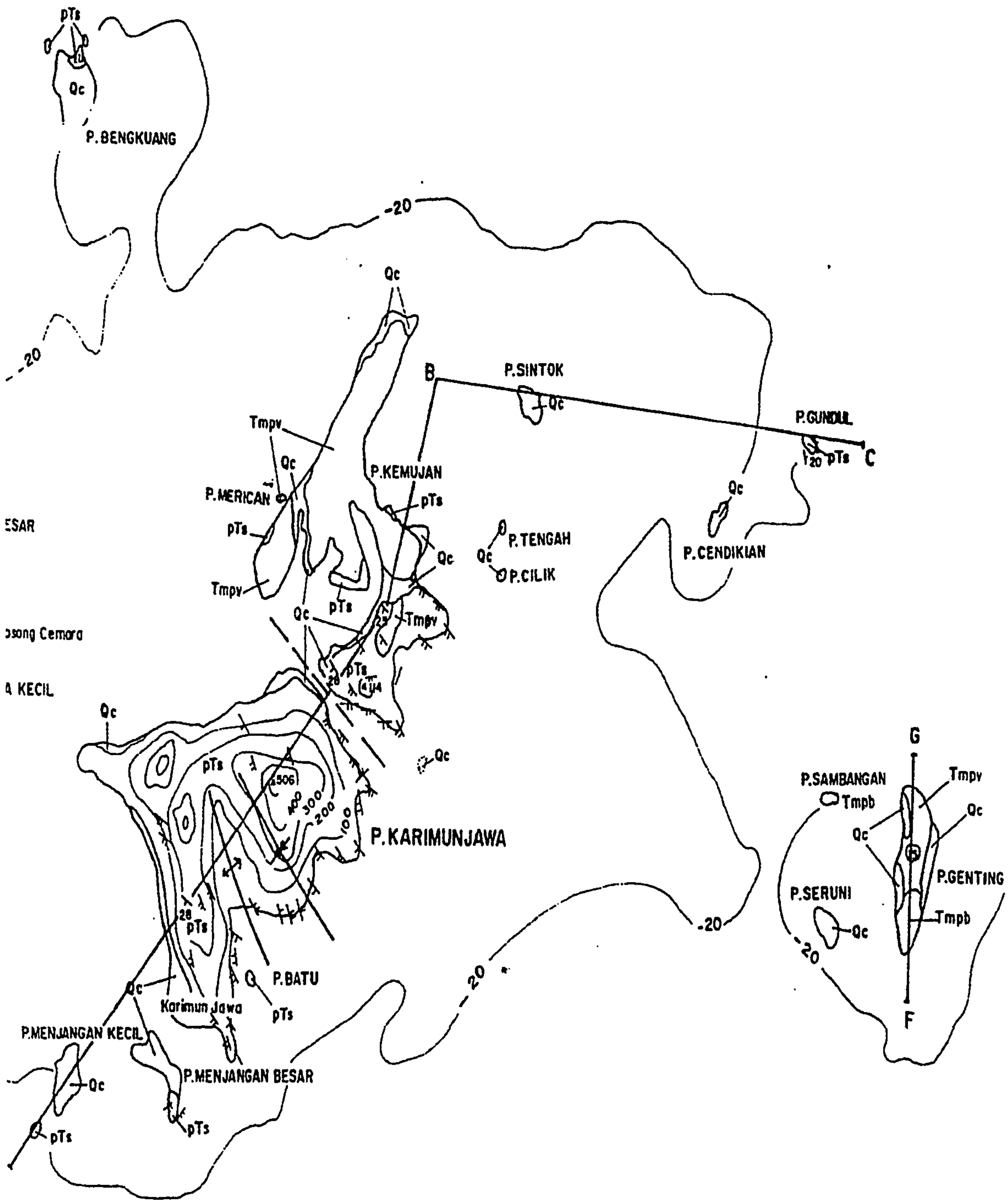

Figure 4.2. The Geological Map of Karimunjawa Archipelago [18]

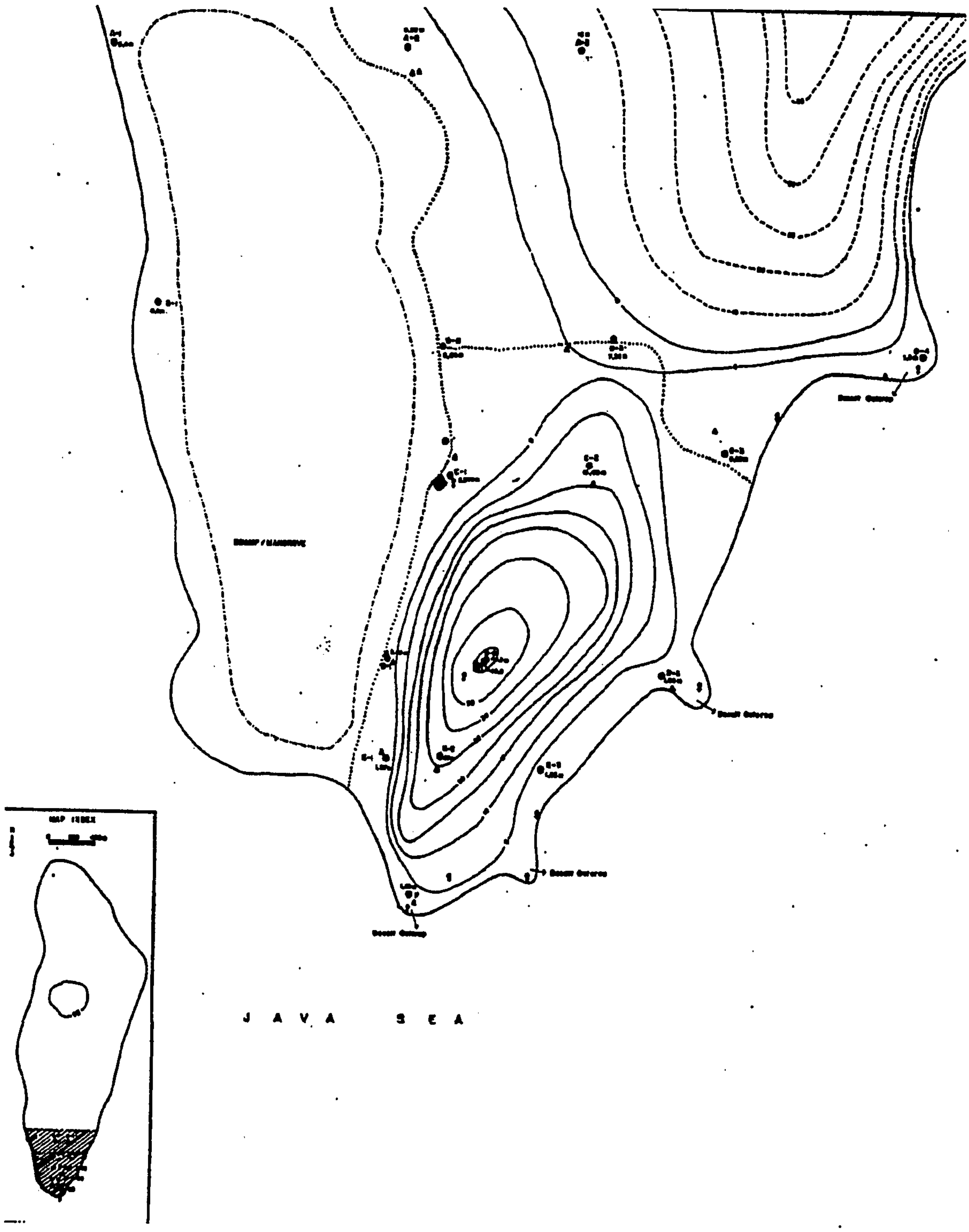

Figure 4.3. The Groundwater Level Map of the Genting Island [18]

Having minimal permeability in the geologic medium is very important. The coefficient of permeability determines the ability of a material to let a liquid pass through its cross-sectional area. Permeability is defined as the discharge that occurs through a unit cross-section of aquifer under a hydraulic gradient of 1.00 and has units of speed (m/s) [52]. The permeability measurements from Genting Island are summarized in Table 4.3.

Table 4.3. The Coefficient of Permeability [18]

Coefficient of Permeability (m/s)	Type of Rocks
1.82×10^{-6}	Soil and Basalt
1.75×10^{-6}	Basalt
1.26×10^{-7}	Basalt
5.55×10^{-6}	Basalt
4.79×10^{-6}	Basalt

The water chemistry of Genting Island shows that the concentrations of Mg^{2+}, Na^{+}, and Cl^{-} are rather high at the coastal area. The content of HCO_3^{-} tends to increase on the southern side of the Island. The existence of HCO_3^{-} in the groundwater is due to the influence of decomposed plants and swamp materials.

The geotechnical investigation shows that the intact rocks can be classified into several strengths. The classifications of strength are very soft, soft, medium, strong, and very strong. The unconfined compression strength value is 360.86 kg/cm^2. The mean value of Poissons ratio is 0.31. The mean value of the natural density is 2.797 g/cm^3. The mean value of cohesiveness is 57.87 kg/cm^2. Table 4.4 shows the results of the investigation.

The southern side of the Genting Island is primarily a volcanic cone region. The highest point of this area is 40.5 m above mean sea level, while the lowest point is about 5.0 m above mean sea level.

Table 4.4. Technical Classification of Intact Rocks [18]

Classification of Strength	Uniaxial Compressive Strength (Kg/cm^2)	Type of Rock
Very soft	10 - 250	Limestone, Salt rock
Soft	250 - 500	Coal, Silstone, Schist
Medium	500 - 1000	Sandstone, Slate, Shale
Strong	1000 - 2000	Marble, Granite
Very strong	> 2000	Quartz, Basalt

4.2.2. Near-Field Conditions

Near-field conditions refer to environmental factors, such as pH, water contact mode, temperature, etc. Table 4.5 gives the summary of the best estimates of the sorption properties and solubilities of the principal radionuclides as affected by the geochemical environments of the near-field mineralogy.

In the model, pH, water contact mode, and temperature are included. Regarding pH, along the 50 m depth of an experimental borehole, the pH varies from 7.7 to 6.8. The average pH is 7.25, which can be considered neutral. For the saturated zone, the water contact mode can be divided into two categories: zero velocity-leading to diffusive transport, and high velocity-leading to advective transport. Since Genting Island is an ocean island, it is most likely that the water contact mode will lead to primarily diffusive transport.

Pathway analyses of Genting Island show that radionuclides can only be transported from the site in groundwater. Therefore, this study employs the groundwater pathway as the most dominant pathway. The exclusion zone, as required by the law, is 5 km. However, in this study, the exclusion zone is conservatively assumed to be only 1 km. No biotic transport mechanisms are available in this design except for transport of radionuclide by bioaccumulation in marine life.

Table 4.5. Relative Solubilities and Retardation Factors of Selected Elements in Various Geologic Media [53]

	Solubility (log ppm)					Retardation Factor ($1 + 10K_d$)				
		Reducing: Eh=-0.2		Oxidizing: Eh=-0.2						
Element	Most Prob.	pH=9	pH=6	pH=9	pH=6	Granite	Basalt	Tuff	Clay, Soil	Salt
Se	-3 (?)	-	-	-	-	5	5	5	5	20
						50	50	50	50	200
						200	200	200	200	1000
Sr	high	-0.2	high	-0.2	high	10	50	20	50	1
						200	200	200	200	10
						2000	2000	10000	5000	100
Zr	-4	-	-6	-4	-6	500	500	500	500	300
						5000	5000	5000	5000	1000
						30000	10000	10000	50000	5000
Tc	-3	-10	high	high	high	1		1	1	1
						5	5	5	5	5
						40	40	100	20	20
Sn	-3	-4	-4	-4	-4	100	100	200	200	100
						1000	1000	1000	1000	100
						5000	5000	5000	5000	1000
Sb	-3 (?)	-	-	-	-	10	10	10	10	5
						100	100	100	100	50
						1000	1000	1000	1000	500
I	high	high	high	high	high	1	1	1	1	1
						1	1	1	1	1
						1	50	1	1	1
Cs	high	high	high	high	high	100	100	60	200	1
						1000	1000	500	1000	10
						10000	10000	10000	20000	2000
Pb	-1	-1	0	-1	0	10	20	20	20	5
						50	50	50	50	20
						200	500	500	500	100
Ra	-2	-3	-1	3	-1	50	50	50	50	5
						500	500	500	500	50
						5000	5000	5000	5000	500
Th	-3	-4	-4	-4	-4	500	500	500	500	300
						5000	5000	5000	5000	1000
						10000	10000	10000	50000	5000
U	-3	-3	-5	high	high	10	20	5	50	10
						50	50	40	200	20
						500	1000	200	5000	60
Np	-3	-4	-4	-2	-2	10	10	10	10	10
						100	100	100	100	50
						500	500	500	400	300
Pu	-3	-5	-4	-5	-3	10	100	50	500	10
						200	500	200	1000	200
						5000	5000	5000	20000	10000
Am	-4 (?)	-8	-5	-8	-5	500	60	300	200	300
						3000	500	1000	800	1000
						50000	50000	50000	50000	5000

4.3. Waste Package Selection

The waste package technology currently proposed for the Genting Island site employs the UC System. It is an integrated system in which spent fuel assemblies can be loaded and sealed in multi-assembly containers at the reactor site or at a receiving facility [5]. The spent fuel is then stored, transported, and finally emplaced in the ultimate repository without ever reopening the container. The proposed UC system can employ two types of waste package designs: the Multi-Purpose Container (MPC) design and the Multi-Element Sealed Canister (MESC) design. Both designs have advantages such as:

1. High capacity for spent fuel in a single transportable package.

2. Criticality control with burnup credit.

3. Sufficient heat transfer capability to keep cladding temperatures within regulatory limits.

4. The number of required handling steps and procedures are significantly reduced.

The conceptual EBS and its associated isolation processes can be seen in Figure 4.4. The container shown in this figure is called the small MPC design. This figure shows schematically the innermost set of barriers that include the waste form, corrosion-allowance barrier and corrosion-resistant barrier.

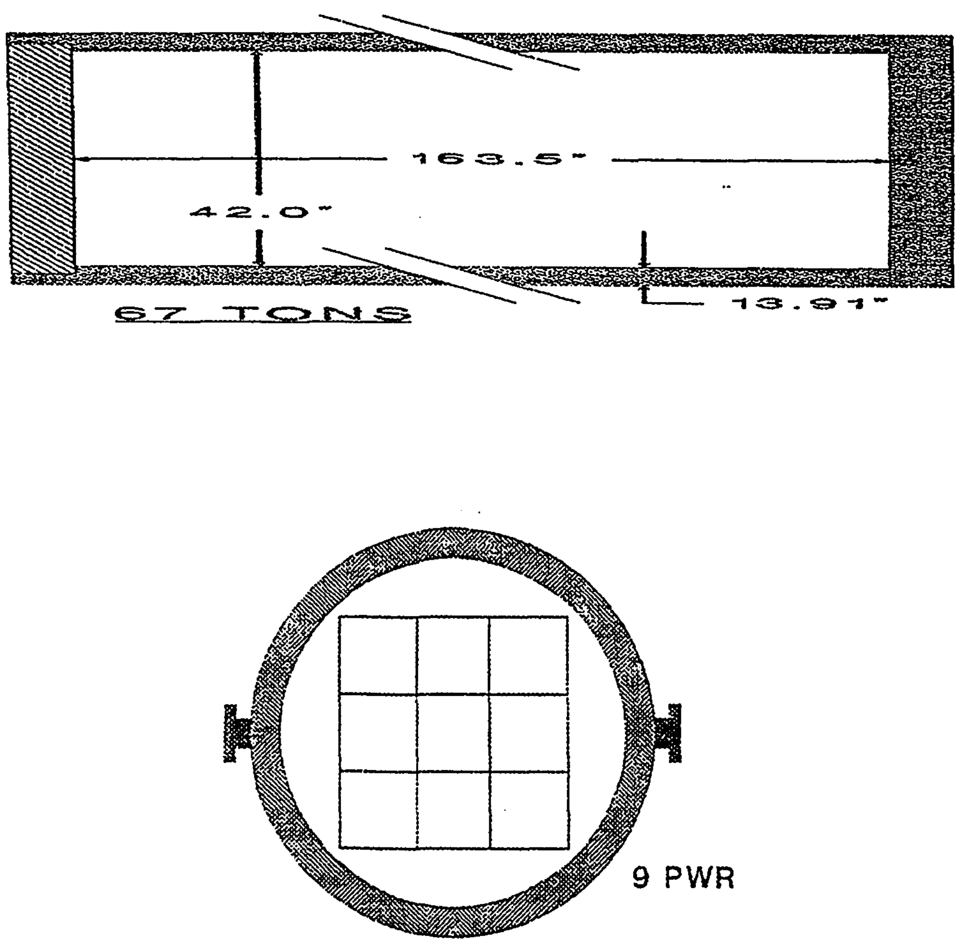

Figure 4.4. Schematic Diagram of Small MPC Design [5]

4.3.1. Possible Failure Mechanisms of the Containment Barrier Systems

The EBS includes the waste package and the near-field region in which the waste package will be emplaced. The performance of the EBS must be evaluated to determine the failure rate of containers as a function of time. This container failure rate is an important parameter in the PA analyses of the long-term performance of the site. The container failure mode is assumed to be general corrosion for the carbon steel and pitting corrosion for the corrosion resistant materials. The cladding failure mode is assumed to be creep rupture.

A schematic representation of an emplaced container and possible corrosion and degradation mechanisms can be seen in Figure 4.5.

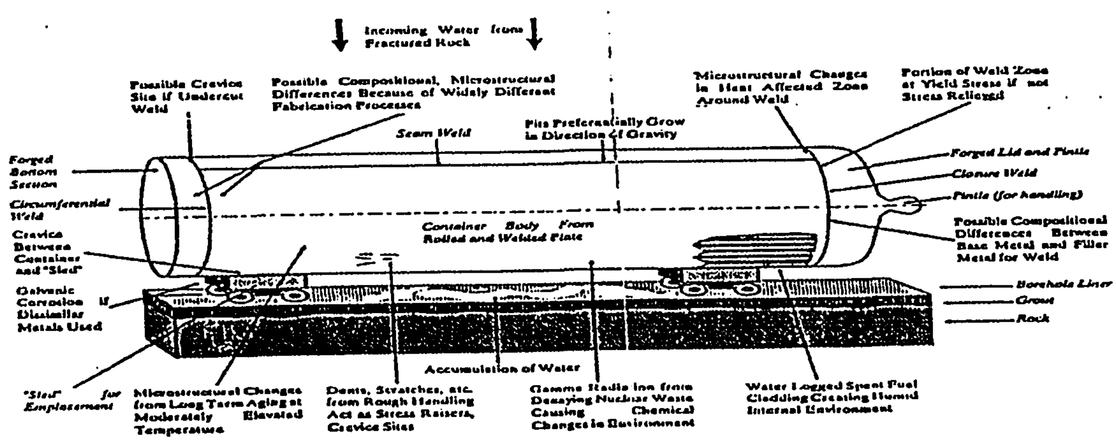

Figure 4.5. The Emplaced Waste Package Container, Possible Corrosion and Degradation Problems [44]

4.4. Dose/Risk Evaluation

As the first step of modeling radiation dose, Dose / risk evaluation is the calculation of drinking-water doses. The calculation is relatively straightforward, that the release rate from the RIP calculation is multiplied by the dose-conversion factors provided. The conversion factors are extracted from the table of the study conducted by INTERA Inc., SANDIA Report for the Yucca Mountain project, and the use of GENII computer code [54,55,56]. These dose-conversion factors are based on the ICRP 30 (International Council on Radiation Protection) standards and are employed to RIP computer code. The listing of theses dose-conversion factors from SANDIA REPORT can be seen in Appendix A.

The average annual dose to individual group can be calculated using the following equation:

$$\text{Dose} = \sum_j \sum_r (DF).(C) \tag{4.1}$$

DF = annual effective dose equivalent of receptor group g from all radionuclides present in all pathways (rems/(μCi/L)).

C = concentration of each radionuclide presents in the pathways (μCi/Liters).

The concentration of the radionuclides are based on the mean release rates of these isotopes to the AE (Ci/yr.). The average volumetric flow rate of the groundwater (liters/yr.) are given. Therefore, the concentration of each isotope in the AE can be determined.

Two calculations will be done with regard to dose evaluation. Firstly, the calculation without incorporating seawater dilution factor, and the other is when the seawater dilution factor is incorporated.

4.5. Disruptive Event Scenario

The disruptive events scenarios for the island are assumed to be flooding of the repository and disruption of the repository by earthquakes. There is no volcanic activity in the area. Since the area is remote, if either disruptive event occurs, the resulting radionuclide release from the site will be limited by the significant dilution effects of the ocean. This dilution will prevent significant radiological impacts. The ocean will mitigate the impact of any radionuclide leakage from repository due to geohydrological factors [57]. The primary benefit that is derived from an ocean island repository is the protective mechanism resulting from the great dilution of any release of radionuclides from the repository into the surrounding seawater.

5. TSPA ANALYSES OF THE GENTING ISLAND REPOSITORY FACILITY

In this study, the RIP computer code is used to conduct the PA analyses of the Genting Island repository facility. The code employs "top-down" approach. This approach relies on the expert interpretation of the available data about the repository facility. It integrates the entire system and utilizes relatively high-level descriptive models and parameters. In this study, the RIP code employs 1,000 Monte Carlo realizations to evaluate possible events based upon statistical distributions to describe each scenario. The time histories recorded for this study are up to 100000 years following initial emplacement of the waste containers.

5.1. Assumptions

In order to conduct the analyses for the proposed facility, several assumptions must be made. The assumptions are:

Near-field environment:

1. The pathway length is conservatively assumed to be only 1 km.

2. The waste packages are emplaced in the saturated zone (below the groundwater table).

3. The most dominant pathway is groundwater flow.

4. The facility is 2.6 km long and 800 m wide. The total area is approximately 334 acres.

5. Water contact mode is diffusive transport.

6. Two thermal loadings are employed: high-temperature loading and ambient-temperature loading.

7. Mean pH of the groundwater is 7.25.

8. The rock formation is basaltic rock.

Inventory of HLRW:

1. The eleven most important radionuclides with long half-lives are considered. They are: C-14, Se-79, Tc-99, I-129, Cs-135, Ra-226, U-234, Np-237, Pu-239, Pu-240, and Pu-242.

2. The waste-form is spent fuel assemblies from PWR reactors.

3. The waste package design is a modified MPC design with 1 cm of Alloy 825 and surrounded by 10 cm of carbon steel. Each container can hold maximum of 9 PWR fuel assemblies.

4. Ten PWRs are expected to be built on Java over the next 25 years beginning in the year 2004. The estimated number of spent fuel assemblies produced by these reactors is 7,489 by the year 2036 (see Table 4.2). The required number of containers will be approximately 850.

5. The dominant degradation mechanism is general corrosion for the outer barrier and pitting corrosion for the inner barrier of the MPC.

Dose conversion:

1. The dose conversion table employed in this study is similar to one used to by INTERA, Inc. for TSPA of Yucca Mountain project [54].

2. The conversion of the impact of the ingestion of contaminated drinking water in terms of rem uses a table derived from the GENII computer code which is based on International Council on Radiation Protection (ICRP 30) standards [55,56].

Disruptive event scenario:

1. Two disruptive events are considered: earthquakes and flooding of the repository facility. The data for the annual rate of occurrence for each event are taken from the seismic analyses conducted for the nearby NPPs selected site [11]. The annual rate of occurrence for earthquake is $1x10^{-7}$ and for flooding of the repository is $1x10^{-5}$.

The complete listing of the input to the RIP code can be found in Appendix B.

5.2. The Analyses of the Results from the RIP Computer Code

Several radionuclides must receive special consideration in PA analyses due to their long half-lives and their high solubilities in groundwater. These radionuclides include C-14, Se-79, Tc-99, I-129, Cs-135, Ra-226, Np-237. The early releases of these radionuclides to be considered are C-14, Se-79, Tc-99, and Cs-135. The radionuclides that are significant for a long-term releases are Np-237 and Ra-226. In addition, the total releases of all radionuclides considered in this study is also presented.

Transport of radionuclides in the saturated zone yields retardation of radionuclide migration due to sorption and dispersion. During this period of retardation, radioactive decay reduces the inventory of most radionuclides. Table 5.1 shows the calculated mean annual release rate of these radionuclides to the AE.

From the table, it can be seen that Ra-226 and U-234 have high annual release rates compared to other radionuclides. These isotopes do not retard significantly in the saturated zone. They have low retardation factors (see Table 4.5). Ra-226 is the second ingrowth product in the U-234 decay chain. That is U-234 decays to Th-230 and then Th-230 decays to Ra-226.

The domination of releases by these radionuclides does not appear at the same time. For instance, the isotope C-14 dominates the releases up to approximately 30000 years.

Table 5.1. The Mean Annual Release Rate (Ci/year) for Various Radionuclides

Radionuclide	AE
C-14	7.35×10^{-5}
Se-79	5.67×10^{-5}
Tc-99	2.70×10^{-3}
I-129	8.02×10^{-6}
Cs-135	1.20×10^{-4}
Ra-226	4.61×10^{-1}
U-234	1.07
Np-237	1.03×10^{-4}
Pu-239	1.86×10^{-2}
Pu-240	3.47×10^{-3}
Pu-242	4.24×10^{-4}

5.2.1. Results of Early Releases to the AE

To demonstrate the variation of releases to the AE up to 35000 years, several radionuclides must be considered. They are C-14 (5715 years), Se-79 (6.5×10^4 years), Tc-99 (2.13×10^5 years), I-129 (1.7×10^7 years), and Cs-135 (2.3×10^6 years).

Figure 5.1 shows the release of C-14 (Ci/yr.) as a function of time. This figure suggests that C-14 dominates the early releases up to approximately 20000 years. Each peak corresponds to the failure of a group of containers. The maximum total release rate of C-14 to the AE is 1.16×10^{-2} Ci/yr.

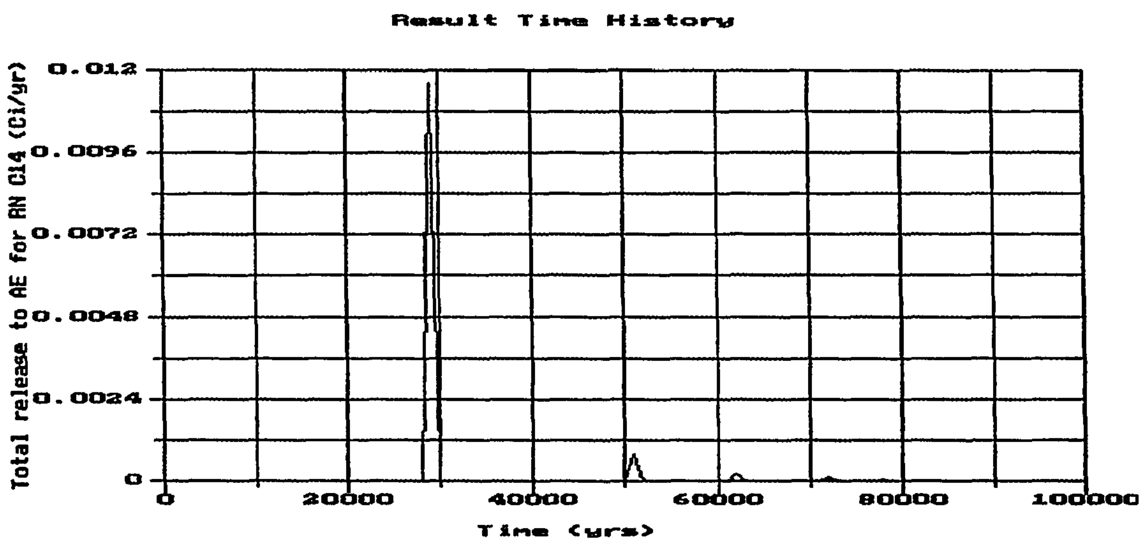

Figure 5.1. Release of C-14 (Ci/yr.) to the AE as a Function of Time

A distribution of the release for C-14 in terms of a Complimentary Cumulative Distribution Function (CCDF) can be seen in Figure 5.2, which is plotted on a linear-log scale. The results presented in this figure indicate that the most probable cumulative release to the AE is approximately 7.35 Ci.

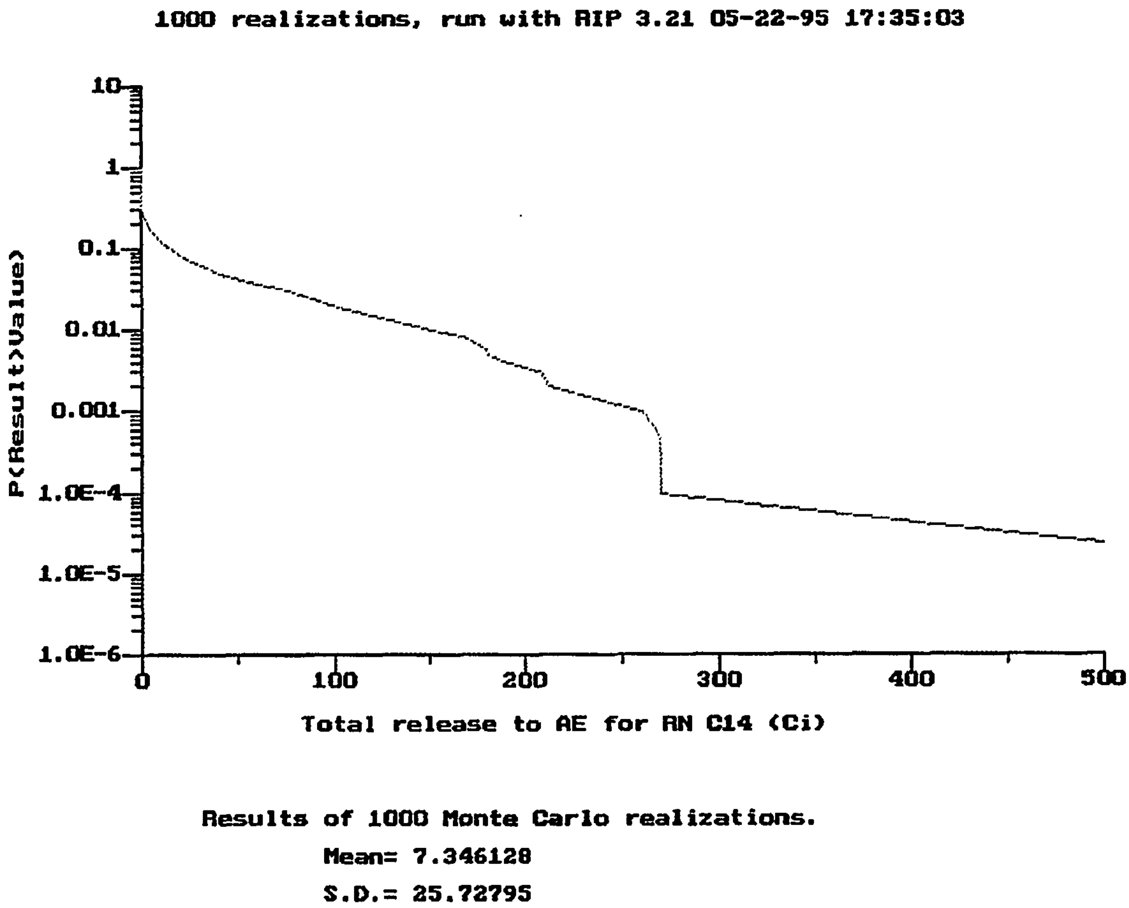

Figure 5.2. CCDF Plot of C-14 for Release to the AE

Figure 5.3 shows the release of Se-79 (Ci/yr.) as a function of time. This figure suggests that Se-79 is a dominant radionuclide over the time period up to approximately 35000 years. The maximum total release rate to the AE is 6.74×10^{-3} Ci/yr. The most probable release inventory for Se-79 to the AE is approximately 5.67 Ci. Figure 5.4 shows the CCDF plot for Se-79.

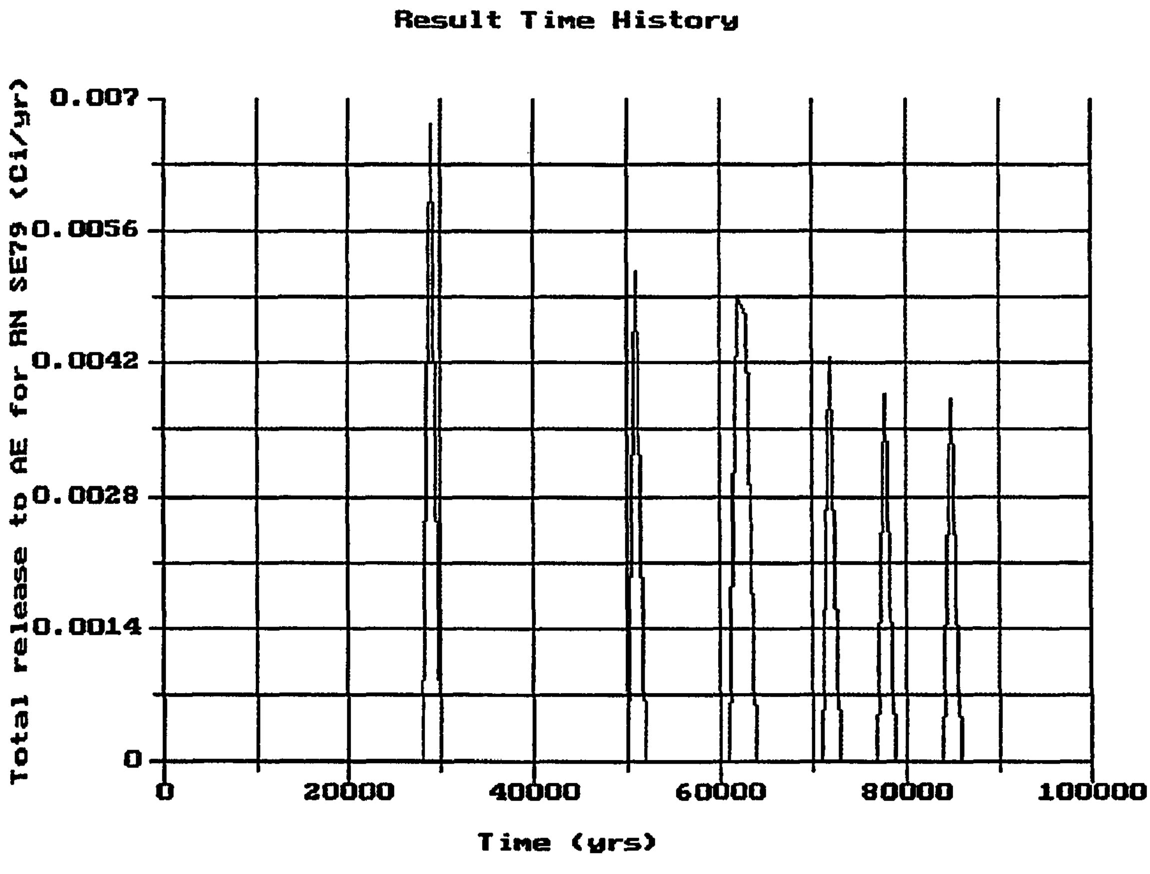

Figure 5.3. Release of Se-79 (Ci/yr.) to the AE as a Function of Time

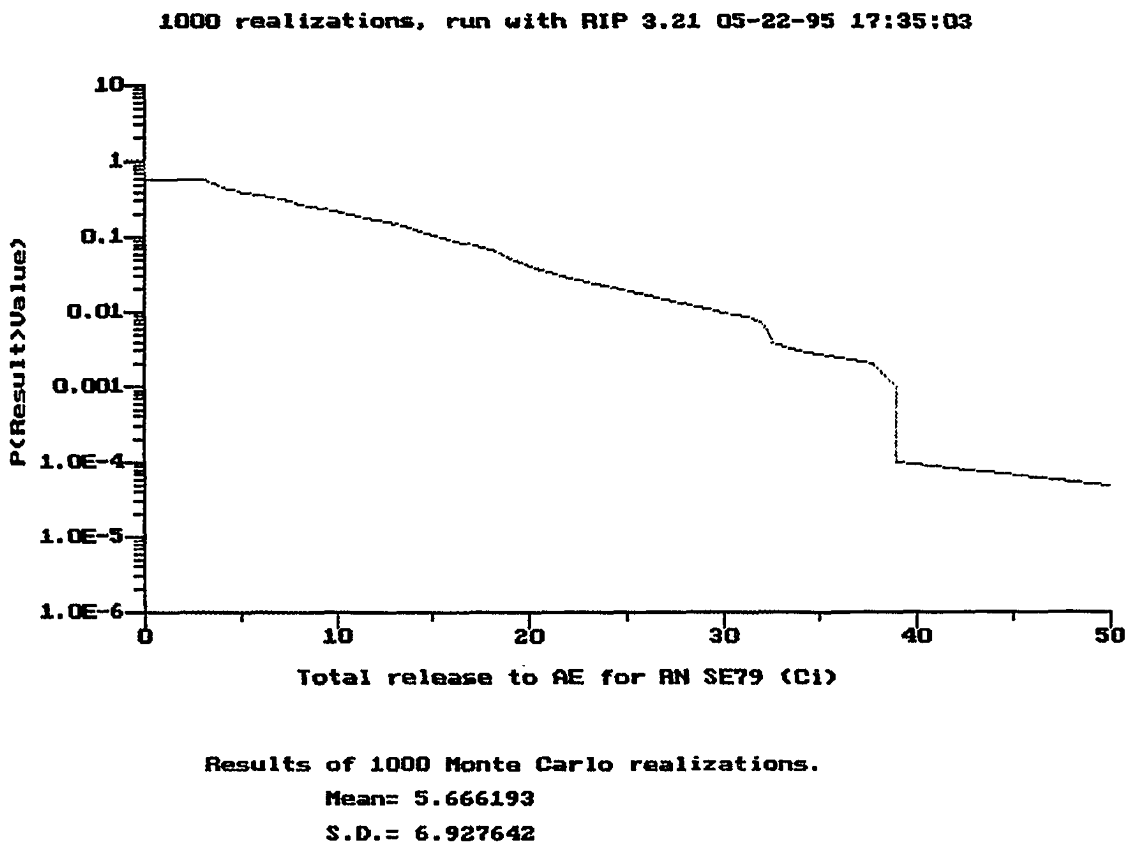

Figure 5.4. CCDF Plot of Se-79 for Releases to the AE

Figure 5.5 shows the release of Tc-99 (Ci/yr.) as a function of time. This figure suggests that Tc-99 is a dominant radionuclide over the time period up to approximately 80000 years. The maximum release rate of Tc-99 to the AE is 2.61×10^{-1} Ci/yr. The most probable release inventory for Tc-99 to the AE approximately 269.27 Ci. Figure 5.6 shows the CCDF plot for Tc-99.

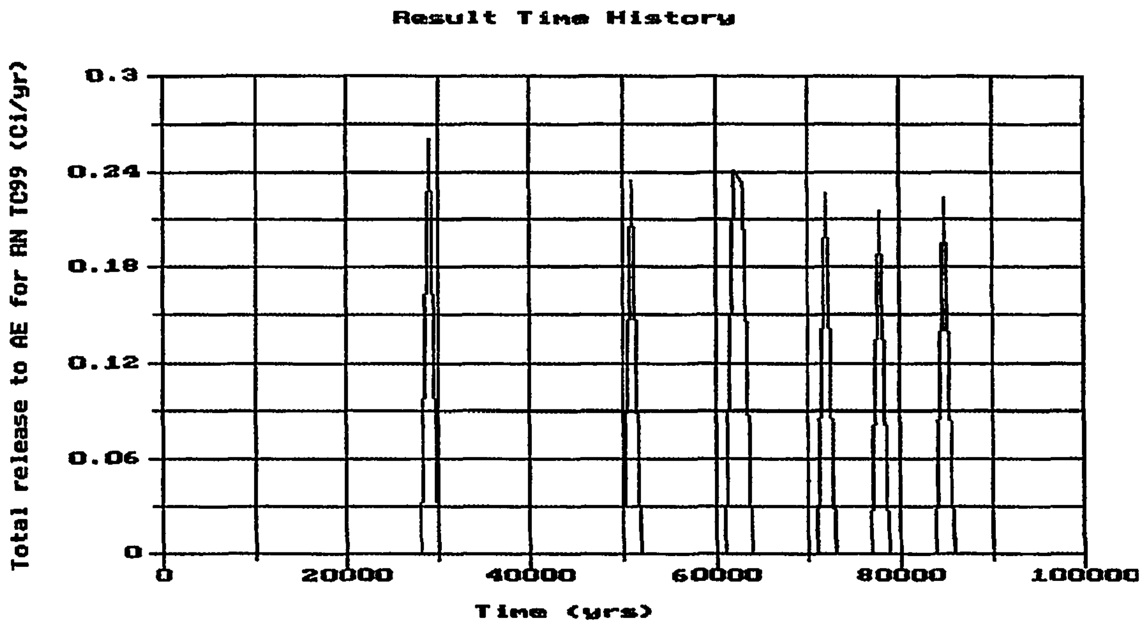

Figure 5.5. Release of Tc-99 (Ci/yr.) to the AE as a Function of Time

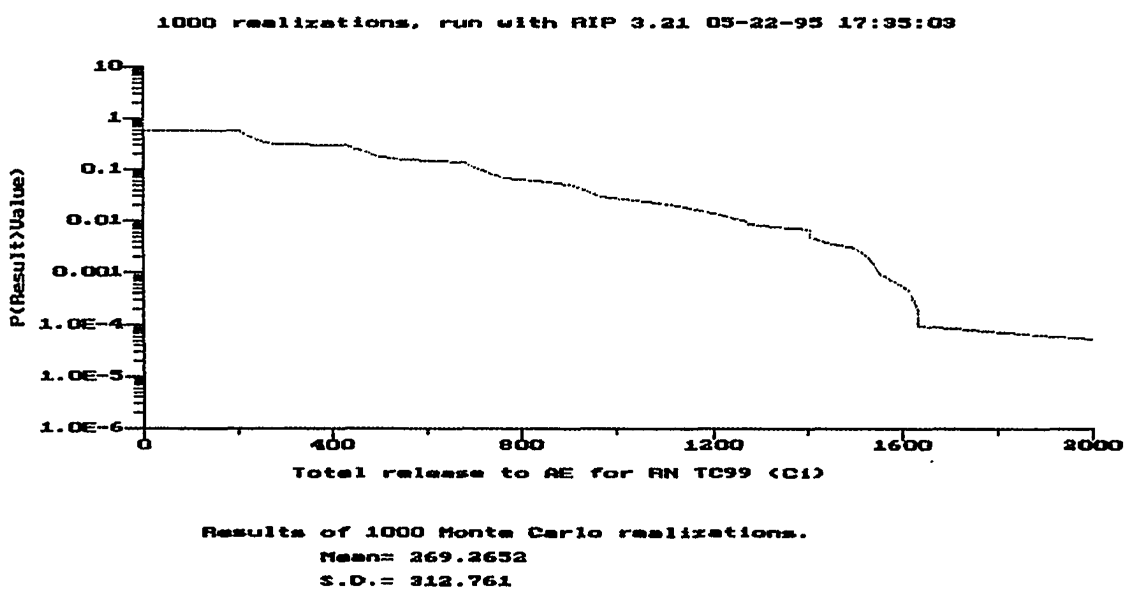

Figure 5.6. CCDF Plot of Tc-99 for Releases to the AE

Figure 5.7 shows the release of I-129 (Ci/yr.) as a function of time. This figure suggests that I-129 is a dominant radionuclide over the time period up to approximately 35000 years. The maximum total release rate of I-129 to the AE is 7.23×10^{-4} Ci/yr. The most probable release inventory for I-129 to the AE is approximately 0.80 Ci. Figure 5.8 shows the CCDF plot for I-129.

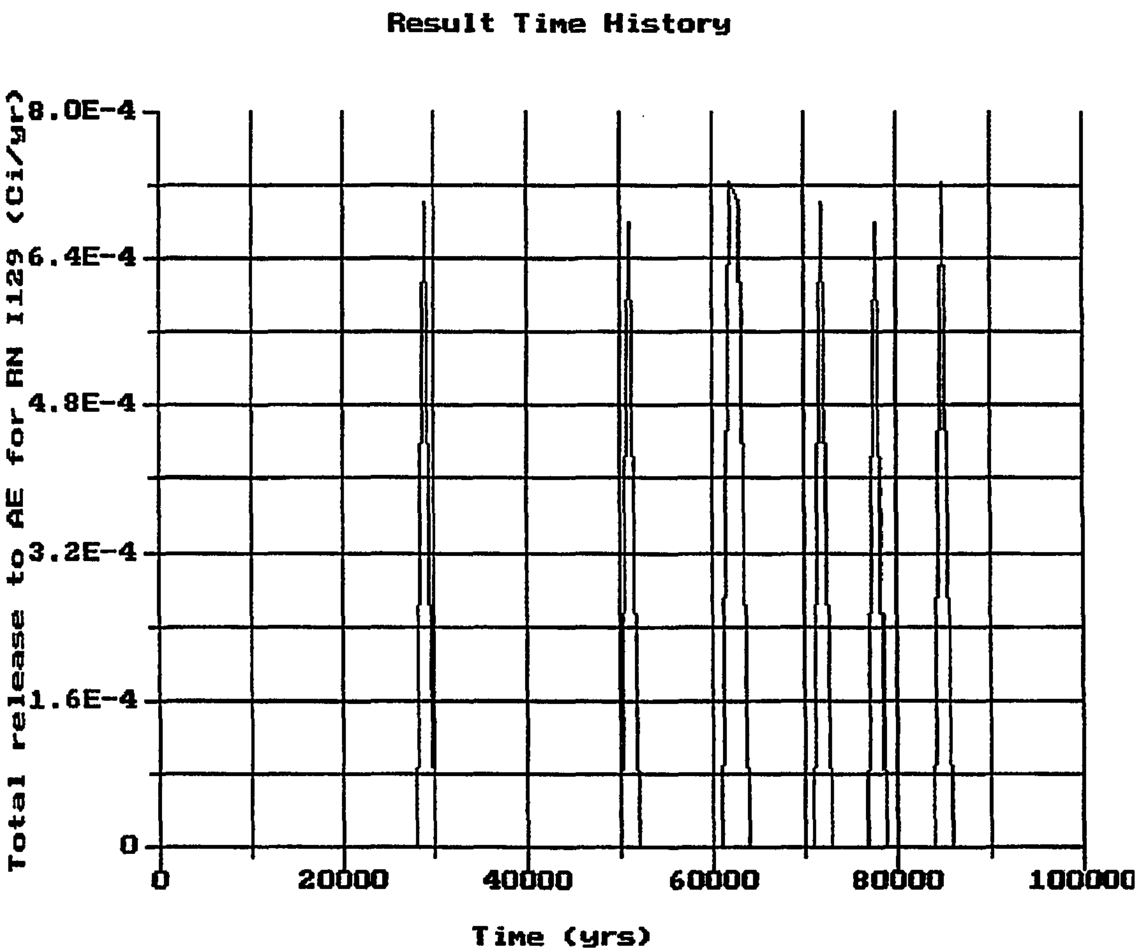

Figure 5.7. Release of I-129 (Ci/yr.) to the AE as a Function of Time

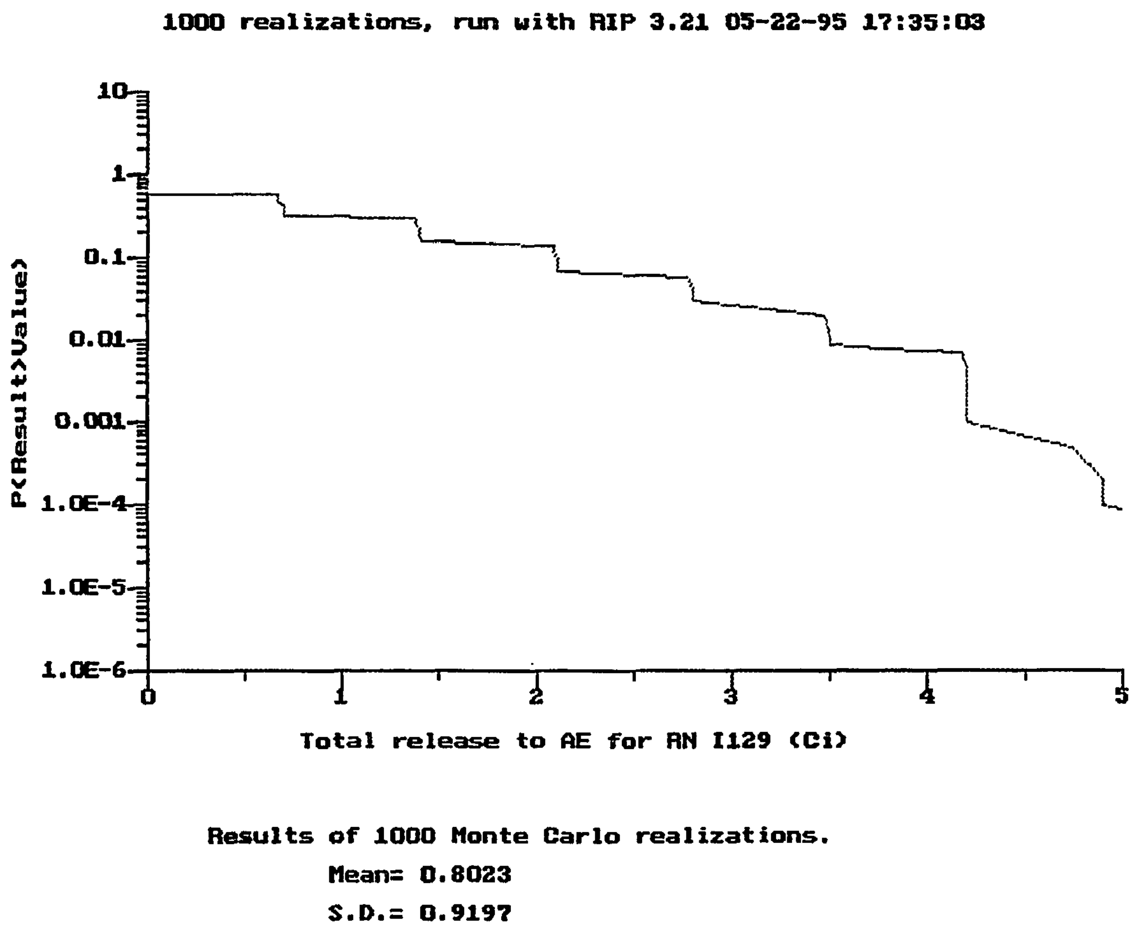

Figure 5.8. CCDF Plot of I-129 for Releases to the AE

Figure 5.9 shows the release of Cs-135 (Ci/yr.) as a function of time. This figure suggests that Cs-135 is a dominant radionuclide over the time period up to approximately 35000 years. The maximum total release rate of Cs-135 to the AE is 1.08×10^{-2} Ci/yr. The most probable release inventory for Cs-135 to the AE is approximately 12.04 Ci. Figure 5.10 shows the CCDF plot for Cs-135.

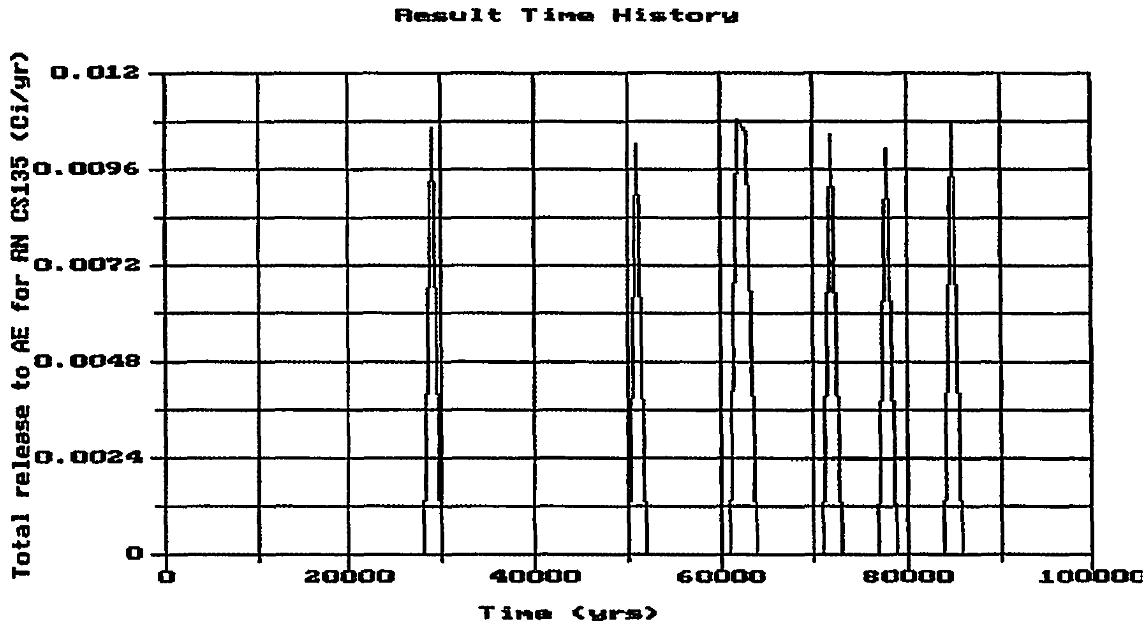

Figure 5.9. Release of Cs-135 (Ci/yr.) to the AE as a Function of Time

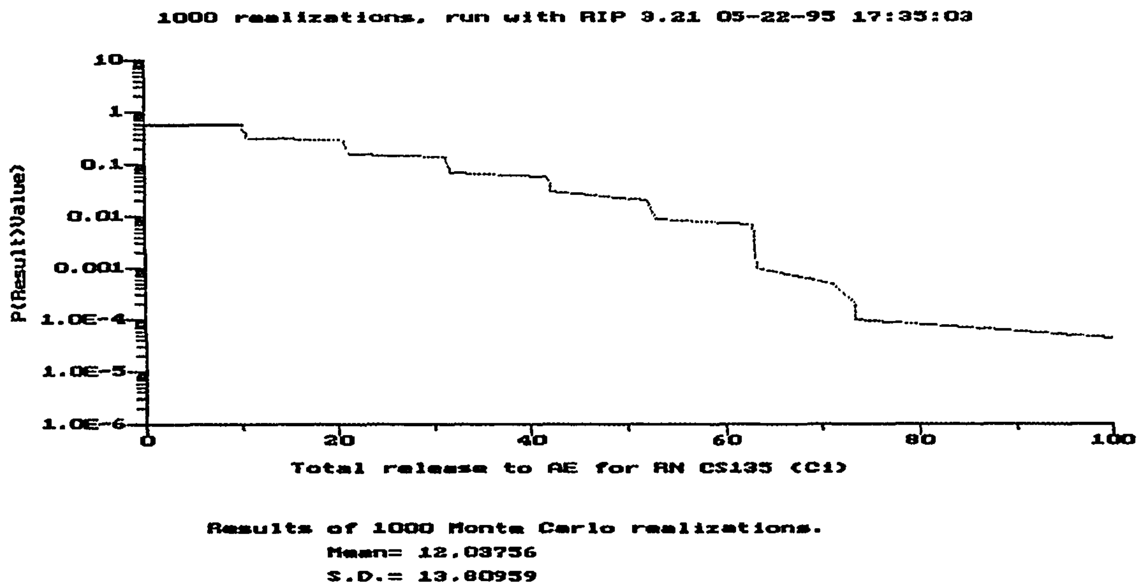

Figure 5.10. CCDF Plot of Cs-135 for Releases to the AE

5.2.2. Results of Releases to the AE up to 100000 Years

Ra-226 (1599 years) and Np-237 (2.14×10^6 years) dominate the long-term releases from 1000 years to 100000 years. Np-237 domination occurs since the nuclide has a very long half-life and a low retardation factor.

Figure 5.11 shows the release of Np-237 (Ci/yr.) as a function of time. This figure suggests that Np-237 is a potentially significant contributor to the total radiological impact over the time period from 1000 year to 100000 years. The maximum total release rate of Np-237 to the AE is 1.03×10^{-4} Ci/yr. The most probable release inventory for Np-237 to the AE is approximately 10.30 Ci. Figure 5.12 shows the CCDF plot for Np-237.

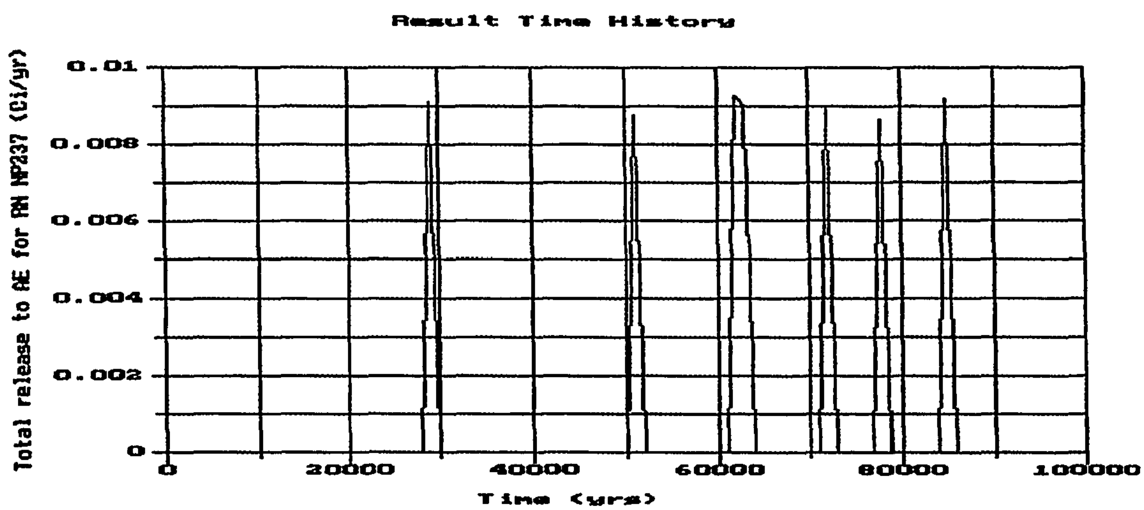

Figure 5.11. Release of Np-237 (Ci/yr.) to the AE as a Function of Time

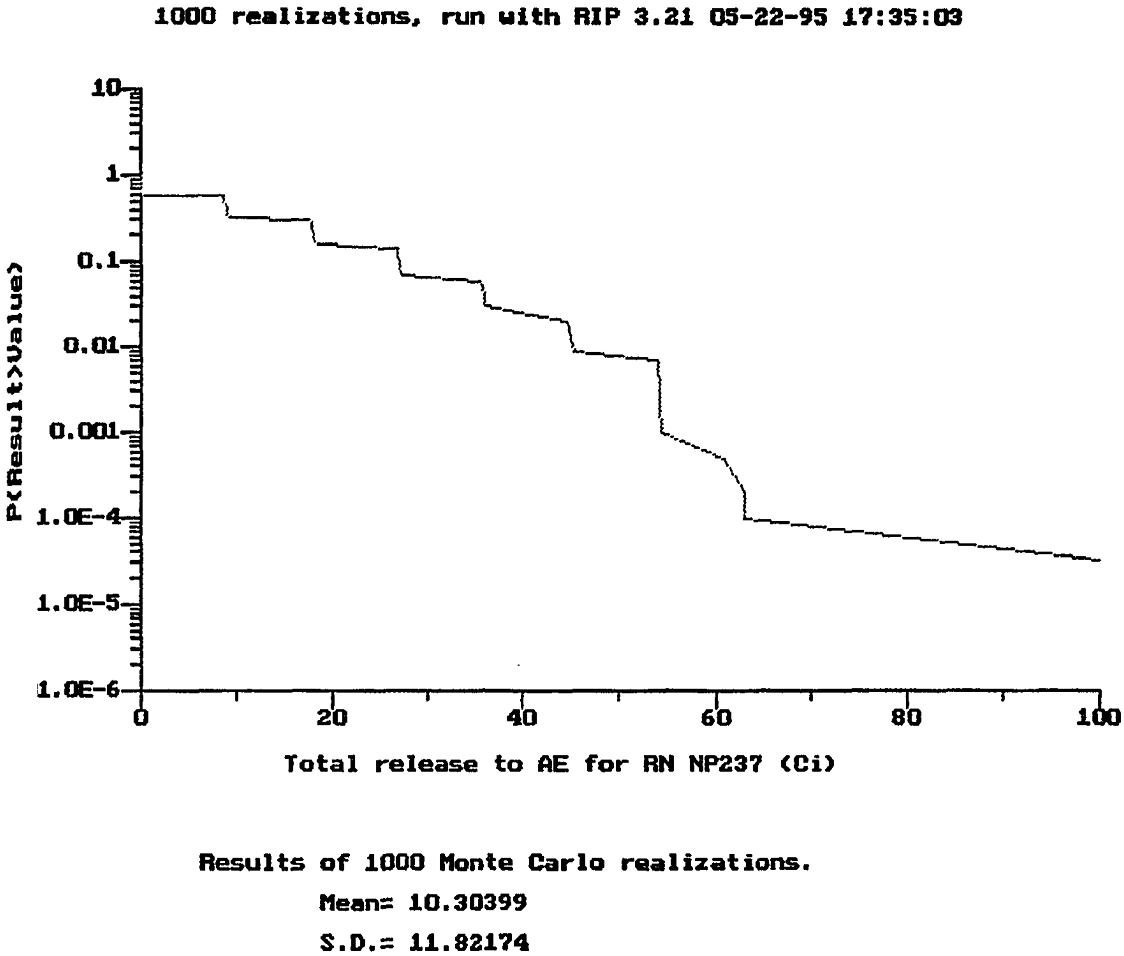

Figure 5.12. CCDF Plot of Np-237 for Releases to the AE

Figure 5.13 shows the release of Ra-226 (Ci/yr.) as a function of time. The maximum total release rate of Ra-226 to the AE is 4.61×10^{-1} Ci/yr. The most probable release inventory for Ra-226 to the AE is approximately $4.67 \mathrm{X} 10^{+4}$ Ci. Figure 5.14 shows the CCDF plot for Ra-226.

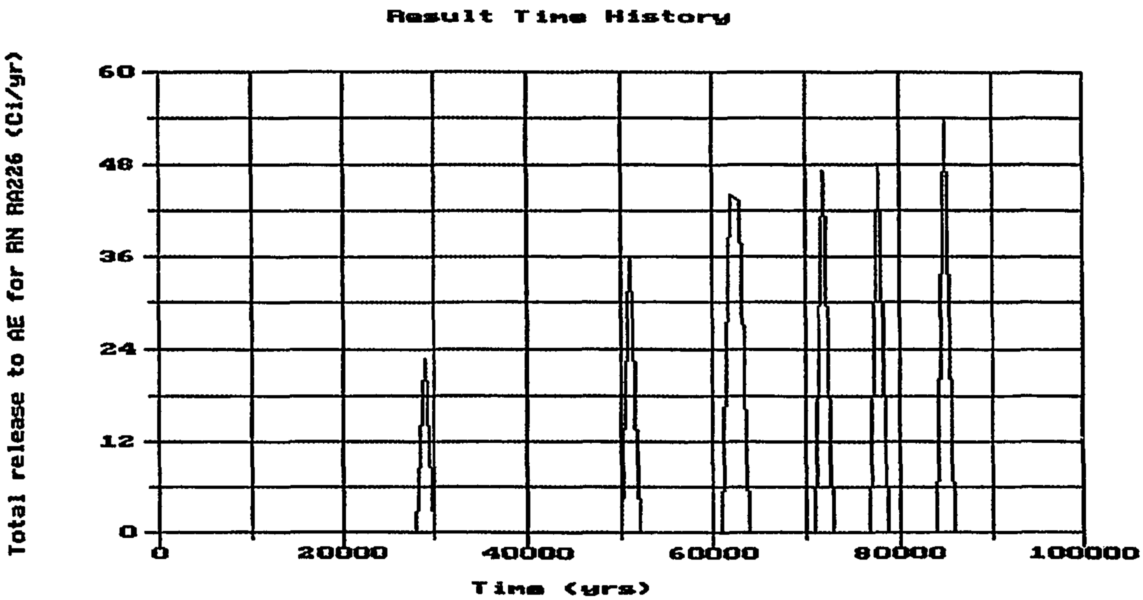

Figure 5.13. Release of Ra-226 (Ci/yr.) to the AE as a Function of Time

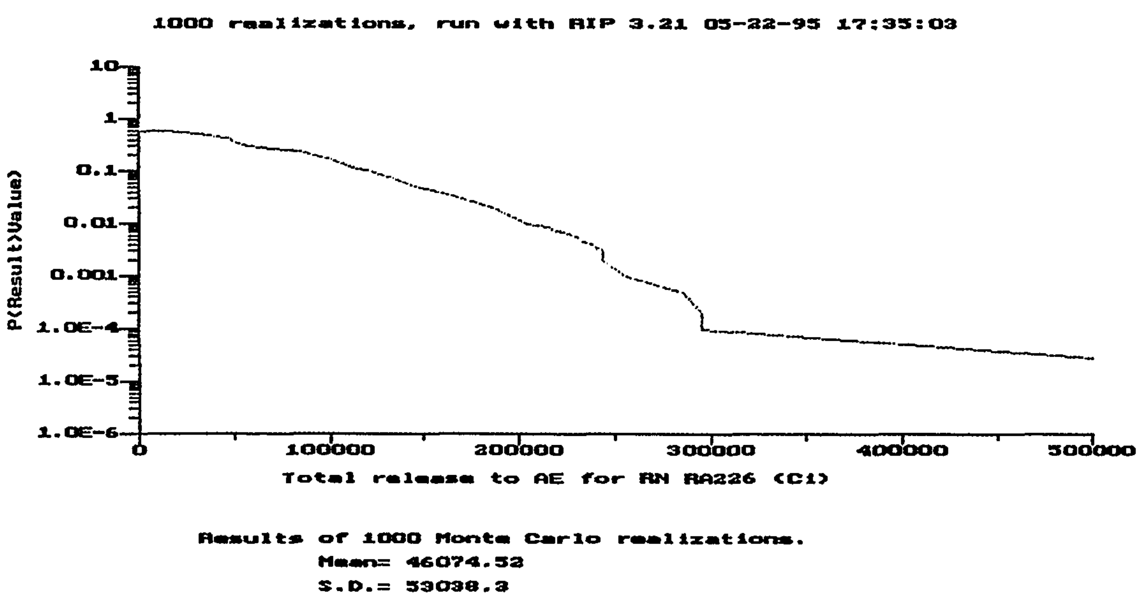

Figure 5.14. CCDF Plot of Ra-226 for Releases to the AE

Figure 5.15 shows the total release to the AE of all radionuclides in CCDF plot. The figure suggests that the most probable release from the inventories of radionuclides to the AE is approximately 2.61X10^{+5} Ci during the time period of initial emplacement up to 100000 years later.

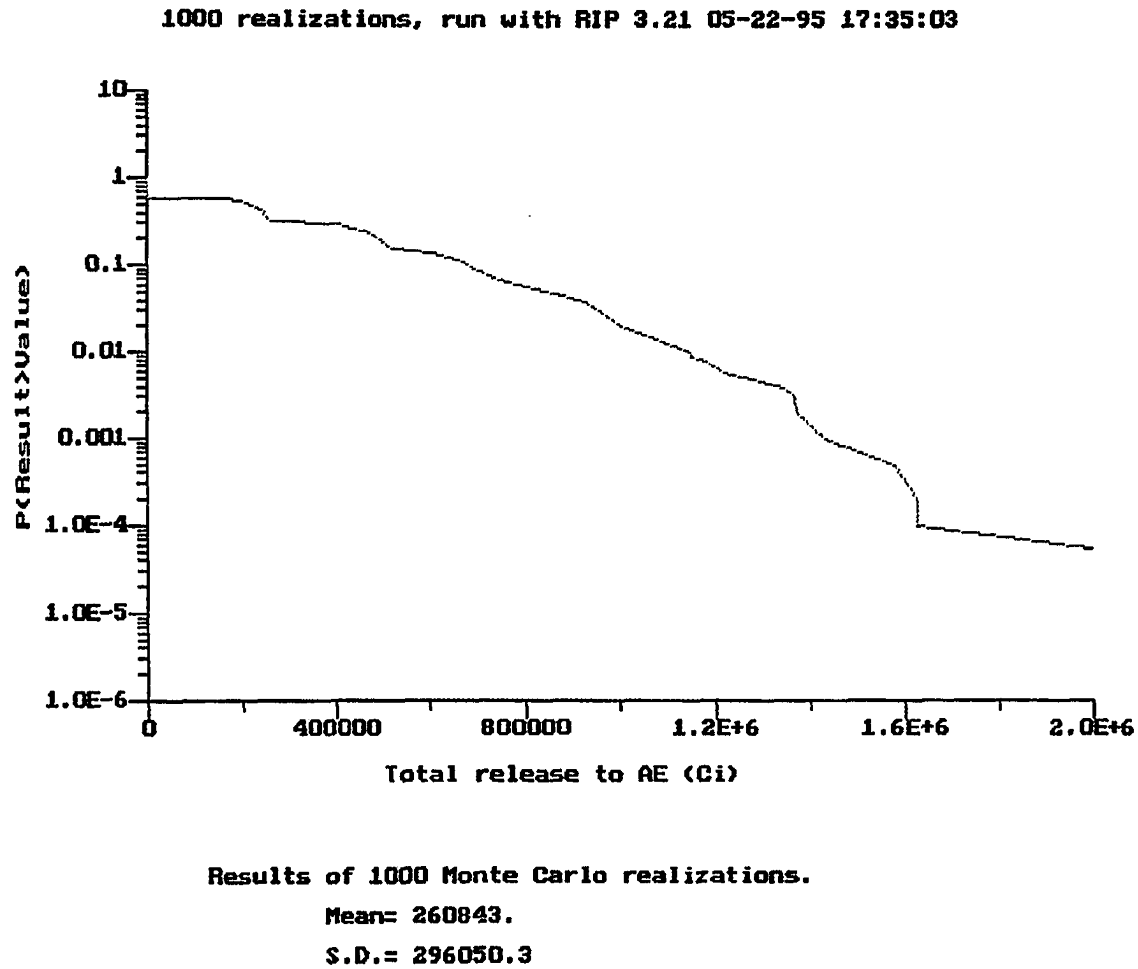

Figure 5.15. CCDF Plot of Total Release to the AE

5.3. Dose/Risk Evaluations

The PA analyses in terms of dose/risk evaluations is very important. The analyses of radiation hazard is very involved for several reasons: interaction of radiation and matter, and the relationships of dose and observable effects to humans. The basic philosophy for radiation release and effect to humans is expressed in the term ALARA (As low As Reasonably Achievable).

The evaluation is based on the assumptions that the person lives near the repository facility, and uses the contaminated groundwater as the source for drinking water. The annual water consumption is assumed to be 730 liters/yr.

The Dose conversion factors relate dose in millirems (mrem) to the activity of each radionuclide in the drinking water in picocuries (pCi). A relatively new concept called the Committed Effective Dose Equivalent (CEDE) is used in this study. The time spans used in this study are taken to be 1 year exposure and 50 year exposure. The resulting 1 year dose from a 1 year exposure is called the annual effective dose equivalent (AEDE). The resulting 50 year dose from a 1 year exposure is called the committed effective dose equivalent (CEDE). The resulting 50 year dose from a 50 year exposure is called the cumulative committed effective dose equivalent (CCEDE). The unit of whole body equivalent doses is rem (radiation equivalent men). Table 5.2 shows the dose conversion factor based on the calculation using the GENII computer code [54,55,56].

Table 5.2. Dose Conversion Factors (Ingestion - Drinking Water, 1 μCi/liters of Groundwater, Dose in rems) [54,55,56]

Isotope	1 Year Exposure		50 Year Exposure
	1 Year Dose (AEDE)	50 Year Dose (CEDE)	50 Year Dose (CCEDE)
C-14	1.5	1.5	$7.5\times10^{+1}$
Se-79	5.2	6.1	$3.0\times10^{+2}$
Tc-99	1.6	1.6	$8.1\times10^{+1}$
I-129	$1.6\times10^{+2}$	$1.8\times10^{+2}$	$9.1\times10^{+3}$
Cs-135	4.5	5.0	$2.5\times10^{+2}$
Ra-226	$8.0\times10^{+1}$	$7.0\times10^{+2}$	$2.6\times10^{+4}$
U-234	$1.4\times10^{+1}$	$1.9\times10^{+1}$	$8.8\times10^{+2}$
Np-237	$1.0\times10^{+2}$	$3.8\times10^{+3}$	$1.0\times10^{+5}$
Pu-239	$1.3\times10^{+1}$	$3.6\times10^{+1}$	$1.3\times10^{+3}$
Pu-240	$1.3\times10^{+1}$	$3.6\times10^{+1}$	$1.3\times10^{+3}$
PU-242	$1.2\times10^{+1}$	$3.4\times10^{+1}$	$1.2\times10^{+3}$

As an illustration, let us calculate the dose resulting from a release of C-14 in radioactive groundwater in the repository facility. As shown in Table 5.1, the mean release of C-14 to the AE is 7.35×10^{-5} Ci/yr. The average flow rate of the groundwater pathway of Genting Island is approximately 315.58 m^3/yr (see Appendix B). Therefore, the concentration of C-14 is

$$(7.35\times10^{1}\ \mu\text{Ci/yr.}) \,/\, (315.58\times10^{3}\ \text{liters/yr.}) = 2.33\times10^{-4}\ \mu\text{Ci/liters}$$

To determine the total body dose using AEDE and make use of Table 5.2, the result of the product factor is

$$(2.33\times10^{-4}\ \mu Ci/liters) \times (1.5\ rems/(\mu Ci/liters)) = 3.5\times10^{-4}\ rems\ (0.35\ mrems).$$

The same result is obtained for the CEDE case since the Dose conversion factor is the same. For the CCEDE case, the total body dose is 1.75×10^{-2} rems (17.5 mrems).

The results of the total body dose calculation for the isotopes in Table 5.2 can be seen in Table 5.3.

Table 5.3. Total Body Dose Calculations (No Dilution Factor)

Isotope	AEDE (rems)	CEDE (rems)	CCEDE (rems)
C-14	3.5×10^{-4}	3.5×10^{-4}	1.8×10^{-2}
Se-79	9.3×10^{-4}	1.1×10^{-3}	5.4×10^{-2}
Tc-99	1.4×10^{-2}	1.4×10^{-2}	6.9×10^{-1}
I-129	4.1×10^{-3}	4.6×10^{-3}	2.3×10^{-1}
Cs-135	1.7×10^{-3}	1.9×10^{-3}	9.5×10^{-2}
Ra-226	$1.2\times10^{+2}$	$1.0\times10^{+3}$	$3.8\times10^{+4}$
U-234	$4.7\times10^{+1}$	$6.4\times10^{+1}$	$3.0\times10^{+3}$
Np-237	3.3×10^{-2}	1.2	$3.3\times10^{+1}$
Pu-239	7.6×10^{-1}	2.1	$7.6\times10^{+1}$
Pu-240	1.4×10^{-1}	4.0×10^{-1}	$1.4\times10^{+1}$
Pu-242	1.6×10^{-2}	4.6×10^{-2}	1.6

From the table, it can be seen that the actinides have a significant radiological impact to humans when a person lives within the exclusion zone. These results are as anticipated. The seawater dilution factor, however, is not incorporated in the calculations.

The assumption employed in this model for determining the dilution factor is that the exclusion zone is 1 km. The depth of sea surrounding the island is approximately 20 m, as shown in Figure 5.16.

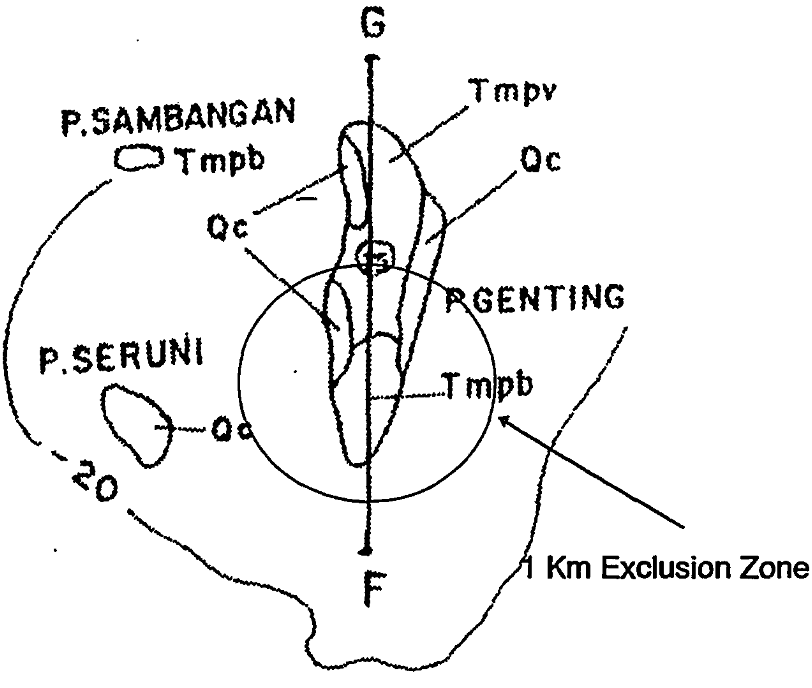

Figure 5.16. Exclusion Zone

Using a cylindrical model to determine the seawater volume and assume that only one third of the volume occupied by water, the calculation suggests

Volume = π x $(1,000)^2$ x 20 x 0.3 = 2.1 x 10^7 m^3

Dilution factor = 315.58 m^3 / 2.1 x 10^7 m^3 = 1.5 x 10^{-5}

Table 5.4 shows the result of calculations when the seawater dilution factor is incorporated. The existence of ocean clearly has significant impact in reducing the concentration of each radionuclide in the perimeter of the exclusion zone used in this study. It should be noted that these dose rates are calculated assuming direct consumption of drinking water. Seawater will not be employed as a drinking water source. Additional pathway analyses (seafood and biota) should be employed in subsequent PA analyses using these diluted radionuclide concentrations.

It is shown from the table that the concentration of each isotope is significantly reduced. The chemistry of the seawater may also effect the solubility of the actinides. In addition, geological isolation is ensured by the remote nature of the site and the regional environment.

This study has been undertaken in a conservative manner. However, the study expects to see some potential impacts to the AE when the repository is flooded and/or earthquakes occur.

Table 5.4. Total Body Dose Calculations (With Dilution Factor)

Isotope	AEDE (rems)	CEDE (rems)	CCEDE (rems)
C-14	5.3×10^{-9}	5.3×10^{-9}	2.6×10^{-7}
Se-79	1.4×10^{-8}	1.7×10^{-8}	8.1×10^{-7}
Tc-99	2.1×10^{-7}	2.1×10^{-7}	1.0×10^{-5}
I-129	6.2×10^{-8}	6.9×10^{-8}	3.5×10^{-6}
Cs-135	2.6×10^{-8}	2.9×10^{-8}	1.4×10^{-6}
Ra-226	1.8×10^{-3}	1.5×10^{-2}	5.7×10^{-1}
U-234	7.1×10^{-4}	9.7×10^{-4}	4.5×10^{-2}
Np-237	4.9×10^{-7}	1.9×10^{-5}	4.9×10^{-4}
Pu-239	1.2×10^{-5}	3.2×10^{-5}	1.2×10^{-3}
Pu-240	2.2×10^{-6}	6.0×10^{-6}	2.2×10^{-4}
Pu-242	2.4×10^{-7}	6.9×10^{-7}	2.2×10^{-5}

As required by law, the exclusion zone is set at 5 km. However, in this study only 1 km exclusion zone is employed. This exclusion zone was deemed adequate to ensure that no biotic transport mechanism was operable except possible transport to marine life via releases to the ocean environment. Such release would entail significant dilution effect and are deemed insignificant. However, subsequent analyses which consider bioaccumulation effects should be completed.

5.4. Critical Data

The PA analyses at this stage has been completed to provide preliminary estimates of repository performance. These analyses are necessary to provide input to the site-characterization and repository-design programs. This current work, however, should be considered as a preliminary developmental process. The results from this study can be used to assess the critical parameters/data that will contribute to better and more sensitive assessment of the proposed repository site in the future. The critical data are described and discussed in the following section.

5.4.1. Waste Package Design

The waste package that should be employed for the HLRW in this repository facility is a modified MPC or MESC design. These designs could provide more advantages for the ocean-island type of repository facility. For instance, the handling procedures are significantly less when compared to other designs. The design will limit radiological exposure of the workers handling the wastes, and it will enhance the ease in transportation. The ocean-island repository requires two types of transportation: land and sea.

The number of waste packages will not be excessive, so a small MPC design that can hold 9 PWR assemblies should be further considered.

The materials to be used for the container are subject to further investigation. Tropical areas requires materials that have low material corrosion rates. In addition, microorganisms may have a significant impact on container degradation. In this area, microorganisms can survive and exist deep underground in large quantity. Therefore, investigating corrosion rates that may include contributions from microorganisms should be conducted thoroughly. As a starting point, relevant information can be gathered from the underground oil-pipeline data available for tropical areas.

5.4.2. Radionuclide Inventory

The radionuclide inventory employed in this study is a representation of the isotopes that have long-term impact to the environment. Based on the results presented in the previous sections, it is important to conduct more detailed analyses of the impacts of Np-237, and other actinides in the AE.

The decay processes of radionuclides are predictable, although the burnup of spent nuclear fuel is unknown. The inventory of each isotope depends upon the decay of the parent isotopes, which vary with the burnup. Hence, total burnup will affect all the isotopes in the decay chain. Relatively accurate data regarding burnup is necessary for more detailed analyses.

5.4.3. Geological Characteristics

The most important parameters regarding geological properties of the ocean-island repository facility are the matrix, bulk, and fracture properties. The matrix properties include porosity of the rock, rock bulk density, saturated hydraulic conductivity, water-retention parameters, and residual degree of saturation.

The bulk properties include bulk saturated hydraulic conductivity, gas permeability, and combination of bulk and gas saturated hydraulic conductivities.

The fracture properties for flow and transport models include frequency of fracture, orientation, spacing, hydraulic aperture, and porosity.

In addition that the earthquake plays an important role in the repository integrity. When the rate of occurrence was selected to be 1×10^{-7} no effects shown in the results. But, when the rate of occurrence was taken to be 1×10^{-2} some significant effects were seen. Therefore, further study regarding seismic analyses of the site must be thoroughly conducted.

5.4.4. *Pathway Parameters*

For an ocean-island repository and sited below the groundwater table, the most important pathway for radionuclide releases is through the groundwater since the groundwater is usually shallow.

The radionuclides releases through the rock will eventually reach the groundwater. In the groundwater, the radionuclides will travel in a diffusive manner to the AE. Therefore, the groundwater is expected to be the primary agents affecting the performance of the Genting Island repository. In addition to being the transport mechanism for the radionuclides, the groundwater will also corrode the waste containers. Hence, understanding groundwater flow characteristics is essential when attempting to predict repository performance.

The shallow groundwater is assumed to exist under in reducing conditions. Consequently, the next assumption is to consider solubility under reducing conditions only rather than oxidizing conditions.

In regard to gaseous flow and transport, this study suggests that C-14 will not be released in significant quantity. However, a more accurate analyses should be conducted. The gas flows are driven by heat and in turn affect waste package temperatures. A coupled transient model of heat transfer and gas flow employing a relatively fine grid will be required.

The ocean dilution factor plays a very important role in reducing the concentration of radionuclides releases to the AE. Their concentrations become essentially negligible when this factor is incorporated in the analyses. Therefore, employing a more accurate model to calculate the dilution factor is necessary.

Seafood and biota should be analyzed as the potential transport pathways in the future study.

6. SUMMARY AND CONCLUSIONS

6.1. Summary and Conclusion of the Study

The TSPA of the proposed HLRW repository facility at Genting Island, Karimunjawa archipelago, Indonesia has been completed. The emphasis of this study has been to determine the parameters required and developing the methodology for completing a preliminary PA evaluation. The emphasis of the analysis is the evaluation of releases of radionuclides to the AE.

The parameters pertinent to the repository site include information from geology, climatology, and water chemistry perspectives. These data are important to predict the long-term performance of the site. Two disruptive events scenarios were incorporated: earthquakes, and flooding the repository site.

The development of the methodology to conduct the analyses can be categorized into three main areas: the inventory of HLRW, the natural and engineered barrier systems, and the geology of the site. The inventory of radionuclides employed in this study includes the isotopes that have long half-lives with a range of retardation factors. The waste-form considered in this study is spent nuclear fuel assemblies of the PWR type.

The natural barrier systems include basaltic rock, clay and soil as backfill materials. Basaltic rock possesses high compression strength.

The engineered barrier systems includes a modified MPC design with 1 cm of Alloy 825 and surrounded by 10 cm of carbon steel. This container can hold maximum of 9 PWR fuel assemblies.

Gaseous releases from C-14 is important only during the first 30000 years, since the half-life is sufficiently short. The fast release of this nuclide is primarily due to faster $^{14}CO_2$ transport time. The release of $^{14}CO_2$ will most likely have a negligible radiological impact on the environment. The total inventory in the proposed repository is less than the allowed release of $^{14}CO_2$ from an operating NPP.

Other radionuclides that have long half-lives and/or low retardation factors resulted in rather higher releases, especially Tc-99, Np-237, and other actinides. Eventhough Ra-226 has a relatively short half-life, this radionuclide appeared as in-growth product of U-234. It appeared after 20000 years.

The rate of occurrence of earthquakes in this study was taken to be $1x10^{-7}$. Using this number, the effect of the earthquakes on the performance of the repository was not shown. However, some significant effects were shown when the rate of occurrence of earthquakes was set at $1x10^{-2}$. Further analyses of the impact of earthquakes on repository design and performance are required.

The Dose Conversion Factors provided by the GENII computer code, SANDIA Report, and INTERA Report of the Yucca Mountain Project show compatibilities (See Appendix A). The calculation shows that for a person living in the vicinity for 50 years and ingesting Ra-226 through contaminated drinking water for 50 years, the CCEDE is $2.6x10^4$ rems. However, when the person lives in the perimeter of the exclusion zone, the CCEDE from Ra-226 ingestion is only approximately $5.7x10^{-1}$ rems.

In general, all the actinides show rather high dose in this study. One of the most important factors to study further is the behavior of Np-237 and its impact on the AE. This radionuclide lasts in the inventory for a very long period of time.

The materials properties for the containers used in this study were taken from the research elsewhere. Therefore, it is necessary to gather data that are more closely represent Genting Island repository.

The behavior of groundwater requires further analyses. This is one of the most important pathways for the release of radionuclides from the repository facility.

In order to ensure public safety, an exclusion zone of up to 5 km can be established. This can be maintained over an extended period of time to ensure public health.

This study has shown that the ocean-island repository concept can be used as a repository facility for the disposal of spent nuclear fuel in Indonesia. The area is considered wet environment.

One topic of investigation that has not been dealt with in this study is the economic impact. The cost of containers and transportation, for example, could be quite high. Also, when dealing with the wastes, several government institutions will be involved in decision making process.

6.2. Suggestions for Future TSPA Work

Significant additional work remains to be done on TSPA of the proposed HLRW repository site at Genting Island. Many important features, processes, and events have to be included in subsequent models.

Additional gaseous flow and transport calculations needs to be conducted in greater detail. Especially, variation C-14 transport time with variations in permeability of the different layers.

The geostatistical modeling of stratigraphy should be refined. Additional geostatistical modeling for other properties should also be conducted. For instance, finding the uncertainty in saturated-zone velocity will be important. The effects of matrix/fracture coupling in the saturated zone must also be investigated.

The kinetics of dissolution of UO_2 and other materials in seawater, and groundwater from the Genting Island should be investigated.

Materials selections for containers that are suitable for tropical climates should be further investigated. In addition, the microbiological-influenced corrosion rates should be included in further studies.

Seismic analyses for the island should be conducted in order to determine more accurate rates of occurrence of earthquakes surrounding the repository area. Such analyses will provide more accurate input parameters for the evaluation of the potential effects of earthquakes on TSPA of the proposed repository at Genting Island, Karimunjawa, Indonesia.

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APPENDIX A. DOSE CONVERSION FACTORS

Total-body dose-conversion factors for ingestion
SANDIA (P. 14-4, V. 2)

Species	Dose-Conversion Factor (mrem/Ci)
Pu-239	4.3×10^9
Np-237	3.9×10^9
U-234	2.6×10^8
Pa-231	1.1×10^{10}
I-129	2.8×10^8
Tc-99	1.3×10^6
Se-79	8.3×10^6

Assumptions:

1. The water is consumed at a rate of 2 liters per day (730 liters/year)

2. No mixing or dilution for the well-withdrawal process

3. 5 km exclusion zone

APPENDIX B. LIST OF INPUTS TO THE RIP COMPUTER CODE

The title of the data file is
TSPA of Genting Island, Rev. 10.4a
The file was last written on 05-22-95 at 17:35:03
The name of the data file is GENT4A.RP

This file is organized as follows:
GENERAL INFORMATION
OUTPUT
SIMULATION DETAILS
RN TABLE
WASTE PACKAGE DESCRIPTION
NEAR FIELD CONDITIONS
PATHWAY DESCRIPTIONS
RECEPTOR DESCRIPTIONS
DOSE CONVERSION TABLES
DISRUPTIVE EVENTS
STRATEGY
PARAMETER DATABASE

GENERAL INFORMATION

The name of the RN species table is genrel1.dat

The number of defined paths is 1
The paths are : PATH1
The number of defined receptors is 2
The receptors are : KIDS,ADULTS
The number of defined dose conversion tables is 1
The dose conversion tables are : TABLE1
The number of defined waste packages is 1
The WPs are : SPENT
The number of disruptive events is 2
The Events are : FLOOD,QUAKES
The Radionuclide Groups are :
Group 1 C14,CS135,I129,NP237,PU239,PU240,PU242

RA226,SE79,TC99,U234,U235,U238,U233,U236
PB210,TH230,PA231,TH229,TH232,AC227,RA228,TH228

OUTPUT

Output flags
a A b B c C d D e E f F g G j J m
1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0

Description of output flags used

Release from WPs by WP group and RN
Release from WPs by WP group
Release from all WPs by RN
Release from all WPs grand total
Release to the AE by path and RN
Release to the AE by path
Total release to the AE by RN
Total release to the AE
Release totals by path and RN
Release totals by path
Max release time histories saved for RN output
Max release time histories saved for totals
Maximum annual normalized release
Normalized total release to AE

Results are not being normalized
The WP model is being run

SIMULATION DETAILS

The number of timesteps used is 100
The number of years per timestep is 1000
The total number of realizations is 1000
Random results are being generated
Latin-Hypercube sampling is turned on

There are 1 event classes defined

For this class, 1 out of 1 realizations is kept
The realizations with the following events are in this class :
All other realizations

RN TABLE

RN ID	Decay	Regulatory	Activity	Daughter	Rate	Limit
C14	1.209E-04	1.0E-01	1.48E+00	-		
CS135	3.013E-07	1.0E+00	5.57E-01	-		
I129	4.414E-08	1.0E-01	3.72E-02	-		
NP237	3.238E-07	1.0E-01	4.87E-01	U233		
PU239	2.879E-05	1.0E-01	3.75E+02	U235		
PU240	1.060E-04	1.0E-01	5.73E+02	U236		
PU242	1.791E-06	1.0E-01	2.18E+00	U238		
RA226	4.331E-04	1.0E-01	2.64E-06	PB210		
SE79	1.067E-05	1.0E+00	4.80E-01	-		
TC99	3.254E-06	1.0E+01	1.51E+01	-		
U234	2.834E-06	1.0E-01	1.43E+00	TH230		
U235	9.845E-01	1.0E-01	1.68E-02	PA231		
U238	1.551E-01	1.0E-01	3.14E-01	U234		
U233	4.372E-06	1.0E-01	7.82E-05	TH229		
U236	2.959E-08	1.0E-01	2.93E-01	TH232		
PB210	3.108E-02	1.0E+00	7.51E-07	-		
TH230	8.999E-06	1.0E-02	3.79E-04	RA226		
PA231	2.115E-05	1.0E-01	3.59E-05	AC227		
TH229	9.441E-05	1.0E-01	4.32E-07	-		
TH232	4.932E-11	1.0E-02	4.71E-10	RA228		
AC227	3.183E-02	2.0E-01	1.97E-05	-		
RA228	1.205E-01	2.0E-01	3.36E-10	TH228		
TH228	3.622E-01	2.0E-01	0.00E+00	-		

WASTE PACKAGE DESCRIPTION

Details of Waste Package 1
ID : SPENT
Description : 7489 SFA from PWR, 9 assemblies per package
Numpackages : 8.3E+0002
MTHM per pack : 1.34000E+01
MWD per MTHM : 3.30000E+04

Repository Infiltration Rate (m/yr) : FLOW
Air Alteration Rate (1/yr) : AAR
Matrix dissolution Rate (g/m*m/yr) : DISS
Surface area of Matrix (m*m/g): 1.00000E-01
Water volume contacting Matrix (m*m*m): 1.00000E-01

Mass of Sorbent (kg): 0.00000E+00
Equilibrium Partition Coefficient for RN group 1 : 1.00000E-01

Container failure modes for the Waste Package 1

Container failure mode 1
general corrosion
Start when rewet
Aging rate of Failure mode: 0.00000E+00
Probability Failure mode is active: FAIL
Weibull Failure Mode
Alpha : 2.00000E+00
Beta - Epsilon : 5.00000E+03
Effective Catchment Area (m*m): CATCH
Geometric factor for diffusion (m): 1.00000E-02
Fraction of fuel which is wetted is 1.00000E-01

Cladding failure modes for the Waste Package 1

Cladding failure mode 1
creep rupture of cladding
Start when container fails
Probability Failure mode is active: FAIL2
Weibull Failure Mode
Alpha : 1.50000E+00
Beta - Epsilon : 1.00000E+04

Behavior of RNs in Waste Package 1

RN C14
Inventory (Ci/container) : 3.11000E+00
Free Mass Balance : FREE1
Gap Mass Balance : GAPS

This RN is in chemical group 1
This RN is gaseous
Gaseous mass transfer rate away from WP (g/yr): C14MTR
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000
RN CS135
Inventory (Ci/container) : 1.19000E+00
Free Mass Balance : 1.0000
Gap Mass Balance : 0.0300
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+02
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN I129
Inventory (Ci/container) : 7.81000E-02
Free Mass Balance : 1.0000
Gap Mass Balance : 0.0300
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+02
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN NP237
Inventory (Ci/container) : 1.02000E+00
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0300
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+02
Elemental solubility (g/m*m*m): 3.60000E+01
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN PU239

Inventory (Ci/container) : 7.88000E+02
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): 2.10000E-03
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN PU240
Inventory (Ci/container) : 1.20000E+03
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): 2.10000E-03
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN PU242
Inventory (Ci/container) : 4.58000E+00
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): 2.10000E-03
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN RA226
Inventory (Ci/container) : 5.54000E-06
Free Mass Balance : 1.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+02
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1

Path 1 : PATH1 Fraction of Balance = 1.0000

RN SE79
Inventory (Ci/container) : 1.01000E+00
Free Mass Balance : 1.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+02
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN TC99
Inventory (Ci/container) : 3.17000E+01
Free Mass Balance : 1.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+02
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN U234
Inventory (Ci/container) : 3.00000E+00
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN U235
Inventory (Ci/container) : 3.53000E-02
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous

Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN U238
Inventory (Ci/container) : 6.59000E-01
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN U233
Inventory (Ci/container) : 1.64000E-04
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN U236
Inventory (Ci/container) : 6.15000E-01
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN PB210
Inventory (Ci/container) : 1.50000E-06
Free Mass Balance : 0.0000

Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000
RN TH230
Inventory (Ci/container) : 7.96000E-04
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN PA231
Inventory (Ci/container) : 7.54000E-05
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN TH229
Inventory (Ci/container) : 9.07000E-07
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN TH232

Inventory (Ci/container) : 9.89000E-10
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN AC227
Inventory (Ci/container) : 4.14000E-05
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN RA228
Inventory (Ci/container) : 7.06000E-10
Free Mass Balance : 1.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+02
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

RN TH228
Inventory (Ci/container) : 9.07000E-07
Free Mass Balance : 0.0000
Gap Mass Balance : 0.0000
This RN is in chemical group 1
This RN is not gaseous
Effective diffusion coefficient (m*m/yr): 1.00000E+01
Elemental solubility (g/m*m*m): SOL
The number of release paths is 1

Path 1 : PATH1 Fraction of Balance = 1.0000

NEAR-FIELD CONDITIONS
WP groups as defined by the environmental conditions

Group #	Expected # WPs	WP TYPE	CONTAC	TEMPV	PH_V
1	68	1	1	0.6650	0.6650
2	68	1	1	0.6650	0.9950
3	70	1	1	0.6650	1.3300
4	68	1	1	0.9950	0.6650
5	68	1	1	0.9950	0.9950
6	70	1	1	0.9950	1.3300
7	70	1	1	1.3300	0.6650
8	70	1	1	1.3300	0.9950
9	72	1	1	1.3300	1.3300
10	23	1	2	0.6650	0.6650
11	23	1	2	0.6650	0.9950
12	23	1	2	0.6650	1.3300
13	23	1	2	0.9950	0.6650
14	23	1	2	0.9950	0.9950
15	23	1	2	0.9950	1.3300
16	23	1	2	1.3300	0.6650
17	23	1	2	1.3300	0.9950
18	24	1	2	1.3300	1.3300

CONTAC	% Balance	Description
1	0.750	diffusive tr
2	1.000	advective tr

The repository rewetting temperature is 1.00000E+02
Its variability of 0.500 is Uniform
The discretization levels are 0.330 0.660 1.000

The temperature time history for the repository is :

Time (yrs)	Temperature
0.0	35.00

1.0	80.00
10.0	125.00
100.0	185.00
1000.0	160.00
10000.0	85.00

The variability in the temperature is TEMPA
The variability in the time is TIMEA

The incremental temperature time history for WP SPENT is :

Time (yrs)	Temperature
0.0	25.00
1.0	30.00
10.0	50.00
100.0	70.00
1000.0	60.00
10000.0	40.00

The variability in the temperature is TEDGE

There are 1 other environmental factors
Environmental factor 1 has a value of MEANPH
Its variability of 0.500 is Uniform
The discretization levels are 0.330 0.660 1.000

PATHWAY DESCRIPTION

The tolerance values for recalculating are :
Fraction of total flow : 1.00000E-01
Mode velocity : 1.00000E-01
Mode porosity : 1.00000E-01
Pathway length : 1.00000E-01
Transition rate : 1.00000E-01

Specified flow balance

Path ID	Total Flow	Discharge (fraction)
PATH1	3.15580E+02	PATH1(1.000)

Details of Path 1
ID : PATH1
Description : geosphere pathways
ID for dose/conc. conversion table :
Total length of Pathway (m) : 1.00000E+03
Total Area of Pathway (m*m) : 5.76000E+02
Total Flow through Pathway (m*m*m/yr) : 3.15580E+02
The number of exit paths is 1
Path 1 : PATH1 Fraction of Balance = 1.0000

The number of flow modes is 2

Flow mode #1
Description : matrix flow
Fraction of balance of remaining total flow : FLOW
Flow mode velocity : VEL
Transition rate (m): POISON

The number of retardation factors is 1

RN group 1 Sorption SORP Matrix diffusion : MAT

Flow mode #2
Description : fracture flow
Fraction of balance of remaining total flow : 1.0000
Flow mode porosity : 1.00000E-01
Transition rate (m): N/A

The number of retardation factors is 1

RN group 1 Sorption 10.0000 Matrix diffusion : 20.0000

RECEPTOR DESCRIPTIONS

Details of receptor 1
Receptor ID : KIDS
Description : children under 10
Risk/Dose ratio : 1.00000E+00

Details of receptor 2

Receptor ID : ADULTS
Description : man and woman
Risk/Dose ratio : 5.00000E-01
DOSE CONVERSION TABLES

Details of dose table 1
Table ID : TABLE1
Description : dose/concentration table 1
Risk dose values for receptor KIDS

RN ID	Value
C14	1.72000E+06
CS135	1.23000E+03
I129	2.05000E+02
NP237	1.31000E+05
PU239	1.24000E+04
PU240	4.56000E+04
PU242	7.25000E+02
RA226	8.79000E+06
SE79	1.10000E+05
TC99	7.72000E+02
U234	4.56000E+02
U235	1.48000E-01
U238	3.63000E-02
U233	1.36000E+03
U236	4.48000E+00
PB210	2.17000E+09
TH230	6.00000E+03
PA231	1.12000E+06
TH229	4.38000E+05
TH232	1.62000E-01
AC227	2.28000E+09
RA228	0.00000E+00
TH228	0.00000E+00

Risk dose values for receptor ADULTS

RN ID	Value
C14	1.72000E+06
CS135	1.23000E+03
I129	2.05000E+02
NP237	1.31000E+05
PU239	1.24000E+04
PU240	4.56000E+04
PU242	7.25000E+02
RA226	8.79000E+06
SE79	1.10000E+05
TC99	7.72000E+02
U234	4.56000E+02
U235	1.48000E-01
U238	3.63000E-02
U233	1.36000E+03
U236	4.48000E+00
PB210	2.17000E+09
TH230	6.00000E+03
PA231	1.12000E+06
TH229	4.38000E+05
TH232	1.62000E-01
AC227	2.28000E+09
RA228	0.00000E+00
TH228	0.00000E+00

DISRUPTIVE EVENTS
Details of disruptive event 1
Parameter ID : FLOOD
Description : Flooding the repository site facility
Annual rate of occurence : 1.00000E-05
Event can reoccur

The number of descriptors is 1
Parameter defining descriptor : FLOODS
Description :

Consequences:

WPs event moved to AE : 0.0000
WPs disrupted by event : 90.0000
There are 9.0000 WPs moved to PATH1
1.0000 % of mass is moved from PATH1 to AE
The number of parameters modified is 0

Details of disruptive event 2
Parameter ID : QUAKES
Description : earthquake event affecting the repository
Annual rate of occurence : 1.00000E-07
Event can reoccur

The number of descriptors is 1
Parameter defining descriptor : QUAKE
Description : earthquake disruption of repository

Consequences:

WPs event moved to AE : 0.0000
WPs disrupted by event : 90.0000
There are 9.0000 WPs moved to PATH1
1.0000 % of mass is moved from PATH1 to AE
The number of parameters modified is 0

STRATEGY

Element ID :
Element Description :
The element is active
This element starts at 0.00
The cost of the element is UNDEF
The element takes UNDEF years to complete
There are 0 precedent elements

This element modifies 0 parameters

PARAMETER DATABASE

Parameter No. 1 of 26
Parameter ID : AAR
Description : air alteration rate
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 1.00000E-03 S.D. = 1.00000E-04

Parameter No. 2 of 26
Parameter ID : C14MTR
Description : c14 mass transfer rate
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 1.00000E+10 S.D. = 1.00000E-01

Parameter No. 3 of 26
Parameter ID : CATCH
Description : effective catchment area
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 5.76000E+02 S.D. = 1.00000E-02

Parameter No. 4 of 26
Parameter ID : DISP
Description : dispersivity
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 1.00000E+02 S.D. = 1.00000E-02

Parameter No. 5 of 26
Parameter ID : DISS
Description : matrix dissolution rate
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 5.00000E+00 S.D. = 1.00000E-01

Parameter No. 6 of 26
Parameter ID : E
Description : The natural number E
Save Time History : TRUE
The parameter is a constant: 2.71828E+00

Parameter No. 7 of 26
Parameter ID : EVENT1
Description : earthquake
Save Time History : TRUE
The parameter is an Event
The annual probability of the event : 1.00000E-07
The event can reoccur

Parameter No. 8 of 26
Parameter ID : FAIL
Description : failure of containers due general corrosion
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 7.50000E-01 S.D. = 1.00000E-03

Parameter No. 9 of 26
Parameter ID : FAIL2
Description : failure due to creep-corrosion
Save Time History : TRUE

The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 7.50000E-01 S.D. = 1.00000E-03

Parameter No. 10 of 26
Parameter ID : FLOOD
Description : Flooding the repository site facility
Save Time History : TRUE
The parameter is an Event
The annual probability of the event : 1.00000E-05
The event can reoccur

Parameter No. 11 of 26
Parameter ID : FLOODS
Description :
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 1.00000E-05 S.D. = 1.00000E-04

Parameter No. 12 of 26
Parameter ID : FLOW
Description : Repository flow rate
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 7.50000E-01 S.D. = 1.00000E-03

Parameter No. 13 of 26
Parameter ID : FREE1
Description : free fraction
Save Time History : TRUE
The parameter is stochastic
The distribution is linear

Sampling bias : Low Bias
Normal : Mean = 5.00000E-01 S.D. = 1.00000E-03

Parameter No. 14 of 26
Parameter ID : GAPS
Description : gaps fraction
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : Low Bias
Normal : Mean = 1.25000E-02 S.D. = 1.00000E-03

Parameter No. 15 of 26
Parameter ID : MAT
Description : retardation for matrix
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 5.00000E+03 S.D. = 1.00000E-04

Parameter No. 16 of 26
Parameter ID : MEANPH
Description : mean pH of the repository
Save Time History : TRUE
The parameter is a constant: 7.25000E+00

Parameter No. 17 of 26
Parameter ID : PI
Description : The value of PI
Save Time History : TRUE
The parameter is a constant: 3.14159E+00

Parameter No. 18 of 26
Parameter ID : POISON
Description : poisson transition rate

Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 3.10000E-01 S.D. = 1.00000E-03

Parameter No. 19 of 26
Parameter ID : QUAKE
Description : earthquake disruption of repository
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 1.00000E-05 S.D. = 1.00000E-05

Parameter No. 20 of 26
Parameter ID : QUAKES
Description : earthquake event affecting the repository
Save Time History : TRUE
The parameter is an Event
The annual probability of the event : 1.00000E-05
The event can reoccur

Parameter No. 21 of 26
Parameter ID : SOL
Description : element solubility
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 1.00000E-02 S.D. = 1.00000E-04

Parameter No. 22 of 26
Parameter ID : SORP
Description : retardation for sorption
Save Time History : TRUE
The parameter is stochastic

The distribution is linear
Sampling bias : No Bias
Normal : Mean = 1.00000E+02 S.D. = 1.00000E-03

Parameter No. 23 of 26
Parameter ID : TEDGE
Description : uncertainty in edge temperature
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 5.00000E+01 S.D. = 1.00000E-02

Parameter No. 24 of 26
Parameter ID : TEMPA
Description : temperature uncertainty
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 9.00000E+01 S.D. = 1.00000E-02

Parameter No. 25 of 26
Parameter ID : TIMEA
Description : time uncertainty
Save Time History : TRUE
The parameter is stochastic
The distribution is linear
Sampling bias : No Bias
Normal : Mean = 5.00000E+01 S.D. = 1.00000E-02

Parameter No. 26 of 26
Parameter ID : VEL
Description : input velocity
Save Time History : TRUE
The parameter is stochastic
The distribution is linear

Sampling bias : No Bias
Normal : Mean = 1.00000E+02 S.D. = 1.00000E-02

APPENDIX C. DESCRIPTION OF MODEL/PARAMETERS DEVELOPMENT AND ASSUMPTIONS/LIMITATIONS

Genting Island repository facility model includes three different categories: waste package, pathways, and disruptive event scenarios. The waste package encompasses near-field conditions, and the waste package description. Pathways include groundwater parameters as the primary pathway in this model. The disruptive event scenarios include earthquakes and flooding the repository.

The near-field conditions include water contact mode, and temperature conditions. The diffusive transport mode is the only mode used in the water contact mode category. Temperature conditions require two types of temperatures: Repository mean temperature, and Incremental temperature at the edge of the waste package. In this case, these two inputs have been translated into high-temperature thermal loading, and ambient-temperature thermal loading regimes. Table C.1 shows these incremental temperatures. Temperature variability is assumed to be uniform with the symmetric variability about one being 0.5. The rewetting temperature is 100°C.

Waste package parameters include the number of waste packages (832), the MTHM per package (13.4), and the assumed waste burnup (33,000 MWd/MTHM). The mass transfer/bound exposure parameter is not independent of failure mode.

The mass transfer description in this study includes a repository flow rate is 0.75 m/yr and a water volume contacting matrix of 0.1 m^3. In the exposure category, the air alteration rate is assumed to be 0.001/year. The matrix dissolution rate is assumed to be 5 $g/m^2/yr$ and the surface area of matrix is 0.1 m^2/g.

The primary container failure mode is assumed to be general corrosion with the probability of failure of 0.75. The failure distribution type is the Weibull distribution. The secondary container failure mode is creep rupture of cladding. This failure distribution type is also Weibull.

Table C.1. Repository Mean and Edge Temperature of Waste Packages

Repository Mean Temperature		Incremental Temperature at edge of WP	
Time (Years)	Temperature (°C)	Time (Years)	Temperature (°C)
0	35	0	25
1	80	1	30
10	125	10	50
100	135	100	70
1000	160	1000	60
10000	85	10000	40

The transport pathway considered in these analyses is shallow groundwater. The volumetric flow rate is 315.58 m^3/yr.

Two types of disruptive event scenario were evaluated in this study. These events included earthquakes with an annual rate of occurrence of $1x10^{-7}$ and flooding the repository with an annual rate of occurrence of $1x10^{-5}$. When either of these events occurs, the event consequences are:

1. Number of waste packages moved to the AE is zero.
2. Number of waste packages disrupted is ninety (approximately 10% of the total number of containers).
3. Number of waste packages moved to the transport pathway is one (approximately 1% of number of waste packages disrupted).
4. Movement of the contents of waste packages from one pathway to another is zero, since only one pathway is considered.

Another assumption made in addition to those described in chapter 5 is that for the first 1000 years after initial emplacement, none of the waste packages fails. The other limitation of this preliminary study is that the exclusion zone that was employed includes not only the land portion on the east-west sides (or the width of the island), but also the seawater surrounding the island. Therefore, the transport pathway length of the groundwater (1000 m) may not necessarily reflect the true length of the groundwater pathway.

Using the assumptions above, the calculation of the total releases of radionuclides to the AE by the RIP computer code may not represent the most accurate analyses, since significant dilution will occur once the radionuclides reach the seawater. Therefore, the results of these calculations may be an overestimate. This, in turn, suggests that calculations for the AEDE, CEDE and CCEDE would be significantly less than those presented in this preliminary study.