SESSION 4

NUCLEAR WASTE

Monday: July 25, 1994
Co-Chairmen: R. Porco
W. Bergman

OPENING COMMENTS OF SESSION CO-CHAIRMAN PORCO

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This session is on nuclear waste. I think one of the most pertinent comments about nuclear waste made by Admiral Guimond this morning was that most of the DOE budget is for nuclear waste and nuclear cleanup, and that the largest environmental protection program in the world is the DOE projects. I think that sets the tone for the significance of this nuclear waste session.

GENERATION AND RELEASE OF RADIOACTIVE GASES IN LLW DISPOSAL FACILITIES

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Abstract

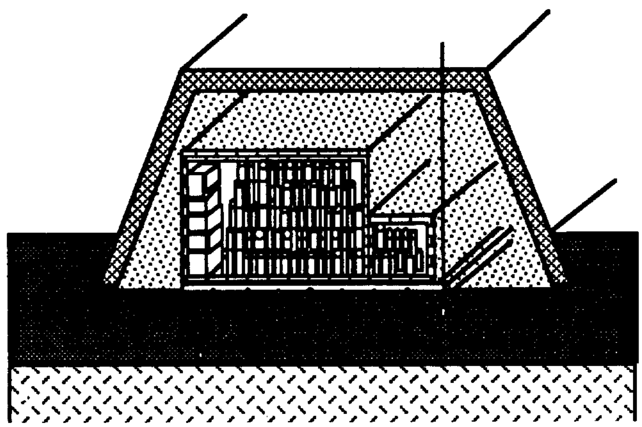
The atmospheric release of radioactive gases from a generic engineered LLW disposal facility and its radiological impacts were examined. To quantify the generation of radioactive gases, detailed characterization of source inventory for carbon-14, tritium, iodine-129, krypton-85, and radon-222, was performed in terms of their activity concentrations; their distribution within different waste classes, waste forms and containers; and their subsequent availability for release in volatile or gaseous form. The generation of gases was investigated for the processes of microbial activity, radiolysis, and corrosion of waste containers and metallic components in wastes. The release of radionuclides within these gases to the atmosphere was analyzed under the influence of atmospheric pressure changes.

1. Introduction

Within LLW disposal facilities, contaminated gases may be introduced into the containment structure as a result of the failure of disposal canisters, anaerobic corrosion of steel containers or the decomposition of organic wastes[1,2].The presence of contaminated gases in the engineered barrier structure may result in radionuclides entering the atmosphere by diffusion or other mechanisms through the containment structure and soil cover. Because some amount of gaseous and/or organic waste is expected to exist in all disposal facilities, the gas-phase release of radionuclides needs to be characterized.

Due to the public's unfavorable view of the shallow land burial of LLW, a major share of the LLW disposal facilities being planned by the States and state compacts are to be above- or near grade bunkered facilities. According to the current design of certain of these facilities, there is a flow drain in the engineered concrete structure and a standpipe, for monitoring, which is connected to the flow drain. This feature potentially provides a direct pathway for gases to be transported out to the atmosphere from within the facility (Figure 1). If a significant amount of radioactive material can be incorporated into the gas-phase, it may be available for releases at rates far greater than would be normally experienced through the liquid (groundwater) pathway. For the overall performance assessment of the newly proposed LLW disposal facilities, the potential radiological dose from the airborne release of radioactive materials through the drain system and monitoring well needs to be assessed. In this case, a major factor that is known to affect the transport and release of generated gases within a LLW disposal facility is the atmospheric pressure variation.

Figure 1. Schematic of an Earth Mounded Concrete Bunkered LLW Disposal Facility



2. Source Inventory Characterization for Gas-Phase Releases

Quantifying the generation of radioactive gases within a LLW disposal facility and the radiological impacts from the release of such gases requires understanding of the gaseous source term, i.e., the total amount and rate of gaseous radionuclide generation. This source term is influenced by the radionuclides inventory within various LLW streams and the waste forms and containers used to dispose of the inventory. In this chapter, summarized are data on these characteristics of specific radionuclides of concern along with the discussion of their subsequent availability for release to the atmosphere in volatile or gaseous form. Radionuclides of concern are ^{14}C, ^{3}H, ^{222}Rn, ^{85}Kr, and ^{129}I.

2.1 Radionuclide Concentrations in LLW

Radionuclide concentrations in various LLW streams had been estimated by several studies(3,4,5). However, the information given in these estimates was not very useful for the purpose of this study for several reasons: No information was given on the distribution of activity according to waste class (A, B, & C) which precludes the estimation of activity distributions in different waste containers; the waste stream classifications were different from the ones used in LLW shipping manifests, and; no estimates were given for the concentration of ^{85}Kr and ^{226}Ra (precursor of ^{222}Rn).

For these reasons, it was necessary to develop new estimates of activity concentrations for the radionuclides of concern in a class-specific manner for various types of LLW generated from various sources. These estimates were made based on the information given in the U.S. Nuclear Regulatory Commission report,

Table 1 Activity Distributions (%) of Radionuclides in LLW															
Radionuclides	Carbon-14			Tritium			Iodine -129			Krypton-85			Radium-226		
Class	A	B	C	A	B	C	A	B	C	A	B	C	A	B	C
Dry Solids	25.6	0.26	19.4	3.24	89.3	0.03	5.0	0	6.14	99.2	0	0	19.3	.001	67.2
Solidified Liquids	5.76	0	1.2	1.45	2.51	0	0.07	0	0	0.02	0	0	.001	0	0
Noncarcass Biological Wastes	0.12	0	0	0.02	0	0	0	0	0	0	0	0	0	0	0
Dewatered Resins	7.54	0.96	7.37	0.12	0.02	.002	41.0	5.46	12.8	0	0	0	0	0	0
Solidified Resins	0.30	2.30	0	0.03	0.02	0	0.29	0.19	0	0	0	0	0	0	0
Sorbed Aqueous Liquids	7.74	0	0	0.68	0	0	2.25	0	0	0.01	0	0	0	0	0
Sorbed Nonaqueous Liquids	.003	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Aqueous Liquids in Vials	2.10	0	0	0.06	0	0	0.01	0	0	0	0	0	0	0	0
Animal Carcasses	1.52	0	0	0.19	0	0	0	0	0	0	0	0	0	0	0
Evaporator Bottoms	0.57	0	0	0.21	0	0	3.83	0	0	0	0	0	0	0	0
Compacted Dry Active Wastes	7.69	0	0	0.26	0	0	13.9	0	5.41	0.02	0	0	0.93	0	0
Noncompacted Dry Active Wastes	0.50	0	0.16	0.07	0	.005	0.86	0	0	0.37	0	0	11.8	0	0
Cartridge-type Filter Media	1.37	0.21	2.83	0.03	0	.006	0.23	0	0.95	0	0	0	0	0	0
Noncartridge-type Filter Media	0.74	0.59	0	0.03	.006	0	0	0.05	0	0	0	0	0	0	0
Activated Reactor Hardware	0	0	2.92	0	0	1.68	0	0	0	0	0	0	0	0	0
Solidified Chelates	0.14	0	0	.001	0	0	0.06	0	0	0	0	0	0	0	0
Solidified Oils	0.16	0	0	.003	0	0	1.43	0	0	0	0	0	0	0	0
Gas	0	0	0	0	0	0	0	0	0	0.41	0	0	.003	0	0
Nonaqueous Liquids in Vials	0	0	0	0	0	0	0	0	0	0	0	0	.001	0	0
Subtotal	**61.9**	**4.3**	**33.8**	**6.4**	**91.9**	**1.7**	**69.0**	**5.7**	**25.3**	**100**	**0**	**0**	**32.8**	**.001**	**67.2**

NUREG-1418[6]. The details of this estimation and the results are given in the reference by Yim[7]. The estimates were made for 19 different types of waste (see Table 1) from 5 different LLW generators (utilities, industry, colleges, hospitals, and government).

2.2 Activity Distributions in Various Types of LLW

Based on the estimated concentrations, the distributions of the total activities of ^{14}C, ^{3}H, ^{129}I, ^{85}Kr, and ^{226}Ra in typical U.S. LLW streams were estimated as given in Table 1. For ^{14}C, 62 % of the activity was estimated to be in Class A, 34 % was estimated to be in Class C, and the remaining 4 % was estimated to be in Class B. Among various types of wastes, 45 % of the total ^{14}C activity was estimated to remain in "dry solids", followed by "dewatered resins" (16 %), "sorbed aqueous liquids" (7.7 %), "compacted dry active wastes" (7.7 %), "solidified liquids" (7.0 %), and "cartridge-type filter media" (4.4 %). The remainder (12 %) was in various types of other wastes.

For ^{3}H, most of the activity (about 90 %) was estimated to be in Class B as "dry solids". In the case of ^{129}I, 69 % of the activity was in Class A, followed by Class C (25 %) and Class B (6 %). Among various types of wastes, 60 % of the total ^{129}I activity was estimated to remain in "dewatered resins", followed by "compacted dry active wastes" (19 %) and "dry solids" (11 %). In the case of ^{85}Kr, most of all activity (99 %)

Table 2 Activity Distributions (%) of Radionuclides in Different Waste Containers and Waste Forms (a): Envirostone

	Carbon Steel Drums						Carbon Steel Liners						High Integrity Containers					
	Cement	Sorbent	None	Bitumen	E stone (a)	Blank/ Other	Cement	Sorbent	None	Bitumen	E stone (a)	Blank / Other	Cement	Sorbent	None	Bitumen	E stone (a)	Blank / Other
C-14	5.1	9.7	27.4	0.2	0.2	7.1	1.5	2.4	6.9	0.04	0.06	1.9	22.5	0.04	4.2	0	2.2	8.8
H-3	1.2	0.9	2.8	0.02	0.02	0.2	0.3	0.2	1.6	.004	.005	0.04	2.6	0.03	90.0	0	.005	0.01
I-129	0.4	3.7	47.7	2.9	0.04	0.4	0.1	1.0	12.2	0.7	0.01	0.1	0.5	6.7	23.3	0	0	0.2
Kr-85	0	19.1	60.9	0	0	0	0	4.8	15.2	0	0	0	0	0	0	0	0	0
Ra226	1.5	1.1	23.3	0	0	0	1.1	0.3	5.5	0	0.02	0	67.2	0	0	0	0	0

was estimated to remain in Class A "dry solids". For ^{226}Ra, "dry solids" contain 87% of the activity, followed by "noncompacted dry active wastes" (12 %). About 67 % of the total ^{226}Ra activity was estimated to be in Class C wastes, with the remainder being in Class A.

2.3 Inventory for Gaseous Releases

In modeling the performance of engineered barriers in relation to the release of radionuclides, it is necessary to know what types of waste containers and forms are used and how much of each radionuclide is disposed in these containers or forms.

Detailed characterization of these types of information have been performed(7) and the results are shown in Table 2. According to the results presented in Table 2, most of the activity disposed, except for ^{226}Ra and ^{14}C, was in the "none required" category (however, for ^{3}H these "none required" waste forms were mostly disposed in special packages thus are not easily subject to release).

In the case of ^{14}C, about 29 % of the activity was reported to be buried in a "cement" waste form, with the majority (39 %) being estimated to be buried as the "none required" waste form. About 12 % of the activity was estimated to be stored in "sorbents". For about 18 % of the total ^{14}C activity disposed, no proper information of waste form was given.

Based on current container design characteristics and anticipated lifetimes, existing waste containers do not provide any significant barrier for releases in the gas-phase(7). Even though high integrity containers (HICs) are designed to maintain structural stability for at least 300 years, the design requirement for the presence of a passive gas vent provides a direct release path for the gaseous radionuclides. This vent is designed with a charcoal filter in it and provides relief against pressure buildup. It is therefore permeable to air or water vapor flow(9).

The organic fraction of LLW will have the potential to degrade by microbially mediated processes to produce gas comprising a mixture of CH_4 and CO_2. About 31% of the total ^{14}C activity was estimated to exist in organic compounds and most of this is in Class A waste(7). Among this 31%, roughly 39% resides in a cement waste form (including Envirostone) in which the pH will remain high*. Excluding this fraction, about 20% of the total ^{14}C activity has the potential for gas-phase release through microbial attack. The estimate of 20% of the ^{14}C inventory to be biodegradable is a significant number in terms of its implications in overall LLW performance assessments. EPA has previously estimated(10) this biodegradable fraction to be about two-thirds of the total inventory, suggesting the possibility of a significant reduction in the ^{14}C inventory in the waste disposal facility through gaseous releases. However, EPA's estimate was based on a crude characterization of ^{14}C inventory in various waste streams.

Tritiated organic compounds have the potential to degrade by microbially mediated processes under anaerobic conditions. In this case, the generated gas, methane, will contain tritium. About 50% of the total ^{3}H activity was estimated to exist in organic compounds, mostly in Class B waste(7).

In the case of ^{129}I, about 30% of the activity which is in HICs may be subject to radiolysis, even though the radiation level may not be high enough to produce any significant amount of iodinated gas.

For ^{85}Kr, most of the activity was expected to remain in sealed sources. Considering the half-life of ^{85}Kr and lifetime of sealed containers, gas-phase release may not, as previously mentioned, be a concern for this radionuclide.

All of the ^{222}Rn activity, which reaches secular equilibrium with ^{226}Ra, is available for release as a gas. The actual amount of release will depend on how much time delay takes place for ^{222}Rn within the waste forms or facility before it reaches the atmosphere.

3. Release of Carbon-14

Carbon-14 is a radionuclide of major concern in LLW disposal due to its long half-life ($T_{1/2}$= 5730 years) and high mobility in the biosphere. For the groundwater pathway, ^{14}C is usually one of the most important contributors to the estimated risks. The characteristics of release of ^{14}C activity associated with these gases within a low-level radioactive waste disposal facility were analyzed in this chapter. The effects of radiolysis on the generation of ^{14}C contaminated gases were also described.

Gases that contain ^{14}C can be generated from a LLW disposal facility from either

* The optimum pH value range for methanogenesis is 6.7 to 7.4. Below 6.0 and above 8.0, very little methanogenesis occurs. The use of cement as a matrix would significantly alter the pH conditions in the waste drums. Cement is expected to serve as a pH buffer, the buffering being controlled by reactions between the waste material constituents. If sodium, potassium, calcium and silicon are removed from the contact with water, the pH will gradually decline from about 13. It will stay above 10.5 if, on average, there is more calcium than silicon in the cement's composition. In general, microbial activity virtually ceases above pH 10.5(8).

an aerobic or anaerobic process of degradation of organic components of LLW, depending on the level of oxygen in the surrounding environment. The organic fraction of LLW has the potential to degrade by microbially mediated processes that produce gas composed of a mixture of CH_4 and CO_2, depending on the nature of the waste and the type of waste forms. These microbially-mediated degradation pathways can be generalized into the following two reactions:

1) the aerobic degradation of organic carbon species,

$$(C_6H_{12}O_6)_n + 6nO_2 \xrightarrow{H_2O} 6nCO_2 + 6nH_2O \tag{1}$$

2) the anaerobic degradation of organic carbon species by fermentation

$$(C_6H_{12}O_6)_n \xrightarrow{H_2O} 3nCO_2 + 3nCH_4 \tag{2}$$

The disposal facility selected for the study was a concrete bunkered facility with a floor drain and a monitoring well* as shown in Figure 1. The fluctuations in barometric pressure, which is expected to affect the transport of gas within the facility, were investigated as a major driving mechanism for the release of gases through the monitoring well. To estimate the atmospheric pressure variation, two years of data collected in Boston (March 21, 1991 through March 29, 1993) were processed to produce the average pattern.

3.1 Modeling the Release of ^{14}C Contaminated Gases

The generation of ^{14}C contaminated gases within waste matrices by microbial activity needs to be analyzed along with the attending modeling of oxygen transport and consumption. The consumption of oxygen was calculated using the generation rate of CO_2 according to Equation (1). The transport of oxygen was treated the same as for other gases as described below.

Once ^{14}C containing gases are generated, they can leak out from waste containers and migrate through the disposal facility. Gases can be released from an undisturbed disposal facility through two main migration modes: (1) diffusion as impurities in air and water vapor, and (2) advection of air and water vapor from the disposal facility due to atmospheric pressure variations. For these investigations of gas production and transport, a computer code, called GETAR (Gas Evolution, Transport, and Reaction), was developed[11].

The mathematical model of the computer code, GETAR, includes the time-

* A monitoring well which is connected to the drainage system of the facility provides a conduit for barometric pumping.

dependent effect of gas generation, diffusion, advection, chemical reactions, and radioactive decay for each gas of concern. The resulting space- and time-dependent partial differential equations were solved through the model geometry with appropriate boundary conditions. To solve the model equation numerically, the method of lines has been applied in this study.

The facility modeled in this study was a concrete bunkered facility. It was assumed that the LLW disposal facility site is comprised of 100 cells, each cell with a dimension of 10 (ft) x25 (ft) x125 (ft). Due to the open space volume generated from the emplacement of containers, the actual volume of wastes that can be stored in a facility was assumed to be 54% of the total facility volume. The analysis of gas release was based on the modeling of one cell. The engineered cell facility was one-dimensionally modeled and the analysis tracked the migration of gases from the waste matrix to the atmosphere through a subsurface drain of a generic engineered structure. For the simulation of flow through the drain system, the model accounts for a pea gravel layer at the bottom of the vault structure, the primary drainage system and the cell drain sump monitor standpipe.

3.2 Estimates of ^{14}C Activity Releases

The generation and release of microbially-mediated gases from a LLW disposal facility were simulated using GETAR for a post-closure period. Analyses were performed before and after the assumed occurrence of a break in the waste containers.

For the purpose of modeling it was assumed that the waste containers were carbon steel*, and two scenarios were used: (1) All the waste containers fail from time zero. In this case, oxygen in the facility will immediately start to migrate into the waste matrix (Hereafter called the aerobic case); (2) Waste containers do not fail until after 23 years of service (this being the average estimated life of carbon steel containers). The gas generation for the first 23 years was assumed to occur under the as-sealed condition of the containers without any release of generated gases to the outside of a drum. This would be the most conservative case (the anaerobic case).

For the two scenarios assumed, the ^{14}C activity incorporated into generated gases was predicted by using the GETAR. The ^{14}C concentration within aerobic containers was estimated to approach about 2 pCi/L. In the anaerobic case, the concentration was predicted to approach 1000 pCi/L. These predicted values are found to be in good agreement to the range of values observed in existing or previous LLW facilities: The concentration observed at the Beatty facility was between 1.46 and 25.9 pCi/L(12); at Maxey Flats, between 0.05 and 80 pCi/L(13); and at West Valley, between 1,000 and 14,000 pCi/L(14).

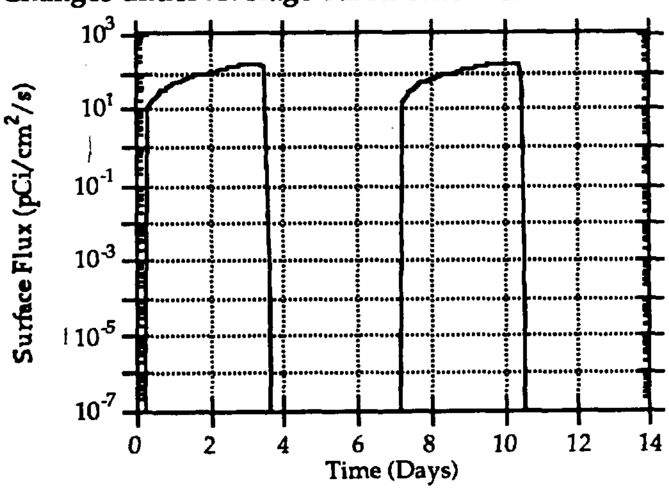
* According to the reference by Yim[7], only 5% of the biodegradable C-14 activity remains in high integrity containers.

To investigate the effects of fluctuations in atmospheric pressure, analyses of the potential release were performed. In describing the flow of gas by advection, the effects of atmospheric pressure variation on flow velocity (so-called atmospheric pressure pumping) was determined *a priori* and utilized in the model: The flow velocity in the open drainage system was calculated based on the ideal gas law considering the total volume of air affected under barometric pressure variation.

Figure 2 shows estimates of changes in the flux of ^{14}C activity at the surface of the soil cover (outside the standpipe) for the anaerobic generation of CO_2. As soon as the pressure change starts pumping out the gases within the facility, the flux increases significantly which was estimated to approach 200 pCi/cm^2/sec. The predicted cumulative release of ^{14}C during the simulated two weeks was about 2.8 mCi from one cell. Adding the contributions from other cells within a facility (100 cells assumed), the yearly estimated release of ^{14}C was about 7.3 Ci.

In the case of aerobic gas generation, the surface flux was estimated to be less than 2 pCi/cm^2/sec. The cumulative release of ^{14}C was about 2.6 μCi in two weeks. For the whole facility, the yearly ^{14}C release would be about 7 mCi.

Figure 2 Predicted Maximum C-14 (CO_2) Surface Flux Changes under Average Barometric Pressure Variations



The results of ^{14}C activity release were calculated based on many parameters in the model. These parameters are subject to different degrees of uncertainties and it is difficult to judge the confidence in the results of activity release without considering the effects of uncertainties. According to an uncertainty analysis for the release of ^{14}C(7), the release of activity varies with a median of 42 mCi. The 95th percentile of the ^{14}C release was estimated to be 141 mCi per cell.

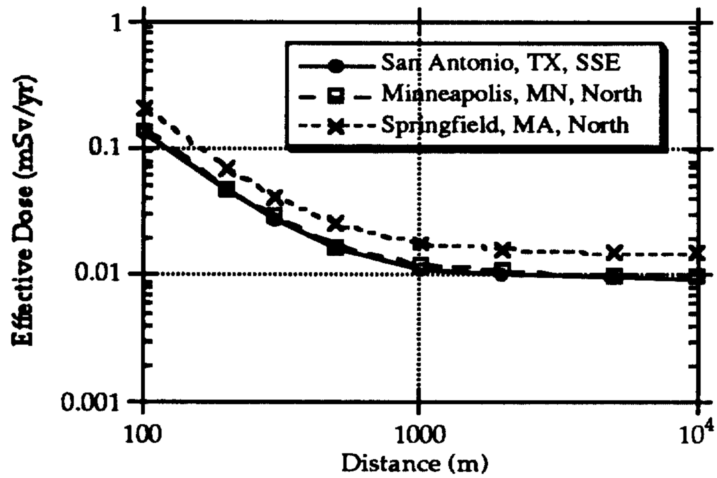
3.3 Radiological Impacts from the Release of ^{14}C

For the radiological assessments of airborne releases of ^{14}C, the CAP88-PC computer code(15) was used in this study. For the assessment of dose, major exposure pathways considered were: (1) inhalation of airborne ^{14}C; and (2) ingestion of ^{14}C which become incorporated into soil and is subsequently taken up by animals and plants. To reduce the uncertainties associated with using hypothetical site data, actual geographical locations were chosen and site specific data were used. Three sites were chosen to represent characteristics in different geographical locations in the U.S. These sites were Springfield, MA for the Northeast, Minneapolis, MN for the Midwest, and San Antonio, TX for the South. The site specific data were used for annual average meteorological conditions (atmospheric stability classes, and wind speeds and frequencies) and agricultural productivity (cattle and crop densities, etc.). For the sources of agricultural food consumed by the exposed individuals, nominal values expected in a rural setting(15) were used.

Figure 3 represents the effective dose estimates for the maximally exposed individual (MEI) due to the release of 7.3 Ci of ^{14}C per year. The 7.3 Ci of yearly ^{14}C release was about the 75th percentile of the estimated release.

According to the results of dose estimates, the ingestion pathway was of primary importance. In fact, inhalation is known as a minor exposure pathway for ^{14}C because dose rate factors of ^{14}C for inhalation are about 1% of those for ingestion(16). Based on a separate sensitivity analysis, the most important route of exposure was found to be the ingestion of cereals, followed by the ingestion of milk and vegetables.

Figure 3 Dose for the Maximally Exposed Individual from the Release of C-14 (7.3 Ci/yr; Rural setting)



3.4 The Effects of Radiolysis on the Release of ^{14}C

Radiolysis, a decomposition mechanism of compounds by the deposition of energy from ionizing radiation, can lead to the release of ^{14}C. Using the dose rate, 161.5 R/hr, estimated for the class B/C wastes(7), and the composition of organic wastes estimated based on the characterization of LLW, the amount of ^{14}C entering the gas-phase was estimated. The results were about 0.16 mCi as CO_2 and about 0.04 mCi as CH_4 per year within one cell. For the assumed 100 cells within a facility, the total ^{14}C becoming incorporated into the gas-phase by radiolysis would be about 20 mCi per year. These estimates are in the same order as the estimated release of ^{14}C by microbial decomposition of organic wastes under aerobic conditions. According to the dose rate estimates for the release of ^{14}C generated from aerobic microbial activity, the resulting dose rate from the release of 20 mCi of ^{14}C is expected to be negligible.

The G-values used for the radiolysis of organic compounds and the composition of generated gases used are given in the reference by Molecke(17). The assumed composition of organic wastes in volume was 19% cellulose, 81% plastics and about 0.1 % organic liquids.

4. Release of Tritium and Other Radionuclides

Other than ^{14}C, the radionuclides of concern for gaseous releases from LLW facilities include tritium, ^{222}Rn, ^{129}I, and ^{85}Kr.

The major process of tritiated gas generation is known to be the anaerobic corrosion of metal, and the tritiated water vapor generation and tritiated methane generation from anaerobic decomposition of organic waste. By assuming 100 cells present in one facility and considering all these processes, total ^{3}H activity entering the gas-phase was conservatively estimated to be about 80 Ci per year. The effective dose rate to the MEI from this release was estimated to be below 10^{-2} mSv (1 mrem) per year.

Due to the presence of ^{226}Ra within LLW streams, ^{222}Rn gas will be generated within LLW disposal facilities from the decay of ^{226}Ra. The total yearly ^{222}Rn release from the assumed facility was estimated to be in the order of Ci. The annual effective dose to the MEI from this release was estimated to be less than 10^{-2} mSv .

The release of ^{129}I as a gas from LLW depends on the methods used to capture and store the ^{129}I in the wastes and the mechanism for the release is known to be radiolysis(18,19). The activity of ^{129}I to be released by radiolysis from a facility is estimated to be less than 2 μCi per year(7). The resulting radiological dose are expected to be negligible (less than 10^{-4} mSv/year).

For ^{85}Kr, most all of the activity is expected to remain in sealed containers. Considering the half-life of ^{85}Kr and the anticipated lifetimes of the sealed containers, gas-phase release is expected to be negligible. However, further study on the characterization of sealed containers used to store ^{85}Kr would be desirable.

Acknowledgment

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DISCUSSION

WREN: Was the microbial rate assumed constant in your C-14 model? If there is dependence, what kind of dependence did you include in your model? You briefly mentioned that it depends very strongly on the hydrolysis of organics, and that it may be very sensitive to the temperature as well.

YIM: Our carbon-14 model is based on a diaphasic description of aerobic and anaerobic digestion. The key parameter is the rate of hydrolysis, which is dependant on the type of organic wastes and temperature. Also there is a lot of data existing currently from civil engineers and environmental engineers who are trying to figure out what is going on with municipal waste dump sites. Some of their efforts are directed to producing biogases for fuel. Therefore, they want to maximize the gas generation and they maintain high temperature conditions, above 30° C. Our condition at a typical low level nuclear waste facility is 10° C. We tried to extrapolate the number down using the Arrhenius equation, which should be adequate for this purpose. But that needs to be verified through some of the experiments. Currently there are no experimental data to back up the results. That is something that has to be done. The effect of pH could be described in the growth kinetics model of microbial population. However, pH values do not change much in a dry engineered facility. Also in a high pH environment, such as in cement waste forms, microbial activity ceases.

KIKUCHI: C-14 will be mostly adsorbed chemically by concrete, by forming $Ca^{14}CO_3$. Did you consider this effect in your calculation? If so, how?

YIM: Current calculation of C-14 release did not consider the formation of calcium carbonate to be conservative. The computer model is capable of considering chemical reactions in the release pathway, however.

HINTENLANG: Site-specific barometric variations with time resolutions on the scale of hours should be used for dose predictions. In some parts of the country, periodic variations of 12 hr (for example) are superimposed on longer term meteorological patterns. Atmospheric stability classes do not include these small amplitude variations. Wouldn't these affect dose calculations dramatically?

YIM: In this study, two years of measured barometric pressure data (in Boston) were processed to represent the effect of atmospheric pressure pumping on the release. These data had hourly resolution. The release calculations were done for two cases: yearly average and severe storm case. The effects of small amplitude variations are included in the yearly average calculation and were not too important for cumulative yearly releases. Another thing that needs to be taken into account in release calculation is the rate of radioactive gas generation. In other words, even if there is a rapid pressure pumping going on in a facility, the eventual release of radionuclide will be dependant on the inventory of gaseous radionuclides from various generation mechanisms.

NUMERICAL ANALYSIS OF A NATURAL CONVECTION COOLING SYSTEM FOR RADIOACTIVE CANISTERS STORAGE

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ABSTRACT

This paper describes the use of numerical analysis for studying natural convection cooling systems for long term storage of heat producing radioactive materials, including special nuclear materials and nuclear waste. The paper explains the major design philosophy, and shares the experience of numerical modeling.

The strategy of storing radioactive material is to immobilize nuclear high-level waste by a vitrification process, converting it into borosilicate glass, and cast the glass into stainless steel canisters. These canisters are seal welded, decontaminated, inspected, and temporarily stored in an underground vault until they can be sent to a geologic repository for permanent storage. These canisters generate heat by nuclear decay of radioactive isotopes. The function of the storage facility ventilation system is to ensure that the glass centerline temperature does not exceed the glass transition temperature during storage and the vault concrete temperatures remain within the specified limits. A natural convection cooling system was proposed to meet these functions.

The effectiveness of a natural convection cooling system is dependent on two major factors that affect air movement through the vault for cooling the canisters: (1) thermal buoyancy forces inside the vault which create a stack effect, and (2) external wind forces, that may assist or oppose airflow through the vault.

Several numerical computer models were developed to analyze the thermal and hydraulic regimes in the storage vault. The **Site Model** is used to simulate the airflow around the building and to analyze different air inlet/outlet devices. The **Airflow Model** simulates the natural convection, thermal regime, and hydraulic resistance in the vault. The **Vault Model**, based on computational fluid dynamics techniques, is the main mechanism for analyzing internal vault temperature stratification; and, finally, the **Hot Area Model** is used for modeling concrete temperatures within the vault.

The end result of design and numerical modeling is a viable natural convection cooling system that performs its intended functions and is a safe practical alternative for the storage of radioactive canisters.

I. INTRODUCTION

4-2

The strategy of nuclear high-level waste is to immobilize by vitrification, converting it into borosilicate glass, then cast the glass into stainless steel canisters. The canisters are seal welded, decontaminated, inspected, and temporarily stored in an underground vault at an on-site canister storage facility until they can be sent to a geologic repository for permanent storage. The canister storage facility provides space for safe handling and temporary storage of canisters. The storage facility in this analyses is a two-level structure consisting of an above grade operating area, and a below-grade canister storage vault (Figure 1). The storage vault contains steel tubes to hold the canisters firmly in place. The loading and unloading of the canisters is done through cover plugs in the operating area floor. The canisters provide primary confinement; and the storage tubes (along with the associated cover plugs) form a secondary confinement system.

A natural ventilation system has many advantages as compared to a forced air ventilation system, such as high reliability, reduced operating and maintenance costs and no requirements for normal and emergency power due to the absence of active components. Radioactive decay heat, generated by the canisters in storage, is removed by the natural convective air cooling ventilation system. The air for cooling is drawn from the atmosphere through a number of air inlet towers, down through an inlet plenum, directed through underground vaults across the bank of vertical tubes, and exits through an exhaust stack. As an example, a plan view of the storage vault configuration is presented in Figure 2. The air is drawn through the vault by natural convection due to the buoyancy forces created by the density difference between the hot air inside the vault and cold air outside. HEPA filtration for the storage vault is unnecessary because potential contamination from a leaked canister will be confined to individual storage tubes and cannot enter the vault area.

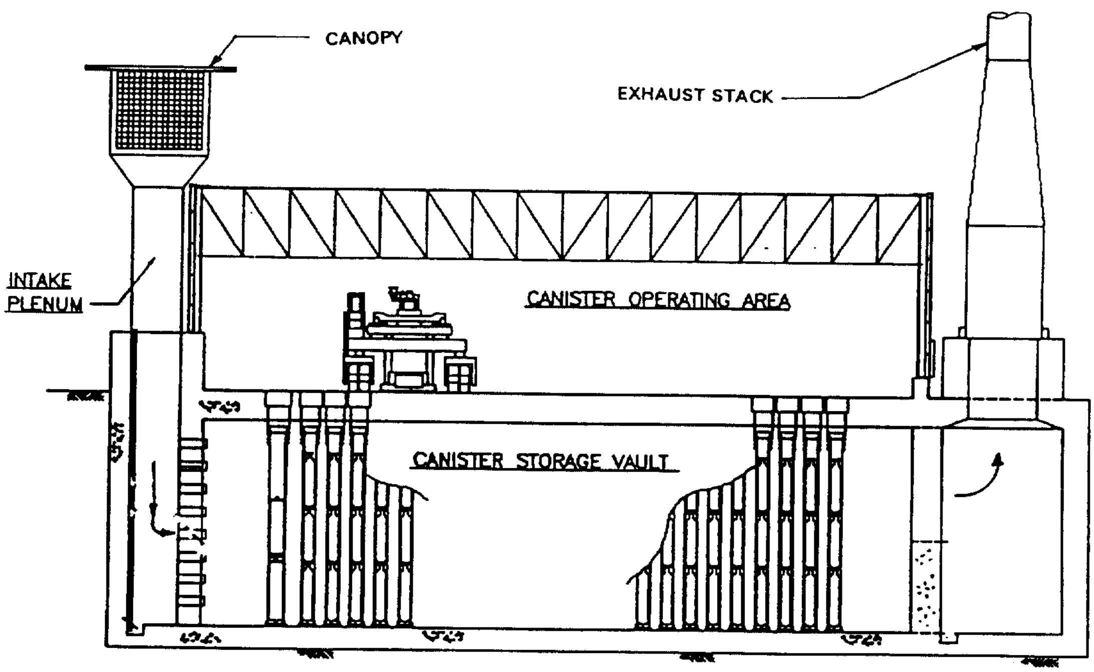

Figure 1. Section view of canister storage building.

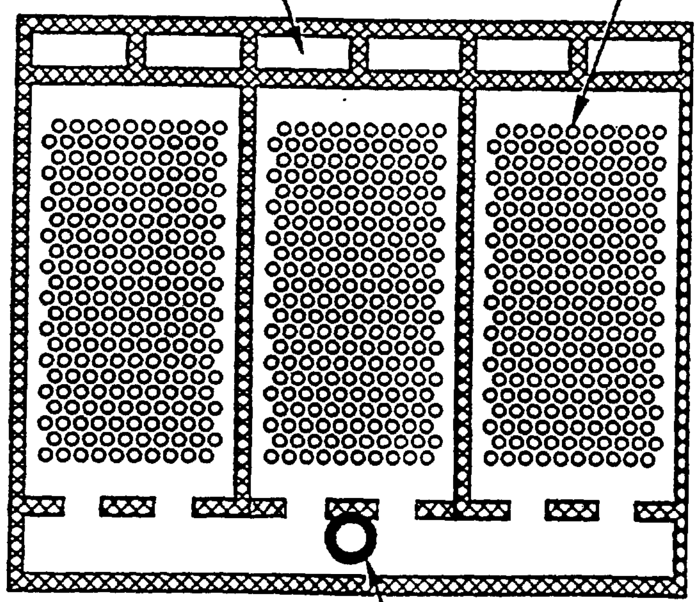

Figure 2. Plan view of canister vault.

Air movement through the vault cools the canisters and the surrounding concrete structure. The amount of airflow depends on the vault and ambient air temperature difference, stack height and diameter, and airflow resistances. Airflow resistances depend on air inlet configuration, internal vault air path geometry, stack size and height; and external wind forces, that may assist or oppose the airflow.

Several numerical models were developed to analyze the thermal and hydraulic regimes in the storage vault including the wind effect and the impact of volcanic ashfall on the overall efficiency of the passive cooling system.

II. DESIGN PHILOSOPHY

II.1 Objective

The primary objective of this study is to analyze the concept of natural convection cooling to assure that the glass centerline temperature does not exceed the glass transition temperature. Also, the vault concrete temperature must remain within specified temperature limits under varying vault loading conditions during normal, abnormal (high wind and ashfall), and accident conditions.

II.2 Description

The natural convection system consists of three major parts: inlet towers and plenums, vault loaded with canisters, and exhaust stack. Numerical modeling shows that wind pressure is one of the major factors affecting the performance of the cooling system. Therefore a number of design studies were conducted with high concern to utilize wind pressure: inlet towers contain wind braking vertical baffles and aerodynamic canopy, outlet stacks are equipped with wind deflectors. Analysis indicates that wind may assist or oppose the airflow. Wind effect is simulated using Computation Fluid Dynamics (CFD) techniques. The simulation results show the necessity of using aerodynamic canopies on the top of inlet towers in order to eliminate the negative pressure at high wind in any direction (Figure 1). The inlet tower and outlet stack performance were verified through a wind tunnel test conducted by Colorado State University[1] showing the effectiveness of the proposed construction, which includes wind braking internal baffles, aerodynamic canopy, and wind deflector. The wind tunnel test has proven that for any wind direction and speed, total pressure of wind and stack effect will stay positive.

III. ANALYSIS

III.1 General

Theoretically, if negative and positive pressures that control air movement are equal, a choking effect may occur and stop the airflow. The result of numerical analysis shows that, without specific modifications to the air inlet towers and the exhaust stack, wind may partially stop airflow, which results in a corresponding increase in vault temperature. If this choking effect continues for an extended period the canister centerline temperature may exceed its limitation.

Numerical modeling proved that canisters storage tubes should be arranged in staggered configuration because it provides greater thermal efficiency than an in-line configuration (Figure 2). Also, due to the low air velocity through the vault, the increase in pressure loss due to staggered configuration is not significant when compared to a in-line storage tube arrangement.

Another concern addressed in the numerical study is the possibility that the volcanic ash may adhere to the storage tube surfaces which could create a thermal resistance layer around the tube surface, reducing heat transfer, thus causing the canister centerline temperature to rise above the limit. In this analysis the design of the air inlet towers limits ash from being drawn into the vault and also prevents entry screens from being blocked.

The performance of the passive ventilation system has been analyzed by simulating the thermal and hydraulic regimes in the vault area through the use of the following numerical models:

- **Site Model** for simulating airflow around relevant buildings on the site and air inlet/outlet devices at different wind speeds and directions.

- **Airflow Model** for simulating the natural convection cooling system by simultaneously considering the stack, wind, and vault thermal and hydraulic effects.

- **Vault Model** for simulating internal vault conditions, such as the air temperatures stratification and air velocity distribution.

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[1]M.Poreh, J.E.Cermak, H.G.C. Woo "Canister Storage Building Wind-tunnel Test", Colorado State University, Fort Collins, Colorado, 1992

Hot Area Model for simulating the maximum temperature in the vault environment. This information is used to design the vault concrete structure.

III.2 Site Model

The objectives of site modeling are: (1) to simulate airflow around the canister storage building in order to analyze static pressure at air entry openings, and (2) to study the effectiveness of different air inlet and outlet configurations.

Input parameters to the site model include climatological data on outside air temperature, wind speed and direction. It was suggested to analyze the extreme climatological conditions. For example, this would be an outside air temperature of 115°F (46°C) and wind speed of 31 mph (14 m/s) when the affect of these parameters is computed separately. When their combined affect is considered, the outside air temperature and wind speed assumed are 101°F (38°C) and 15 mph (6.7 m/s) respectively.

Site simulation is performed by several two and three-dimensional Computational Fluid Dynamics models using the PHOENICS[2] computer code. The results of the numerical analysis show that wind is one of the major factors effecting passive cooling system, particularly when storage vault is only partially loaded and stack effect is minimal. It was found that the wind is capable of reducing flow, reversing flow, or even completely stopping the airflow within the vault. Simulation was also performed for two kind of air inlets: tower inlets and traditional side inlets at ground level. The analysis indicates that tower inlets with internal baffles and outer aerodynamic canopy have significant pressure advantages over side inlets.

Evaluation of the numerical analysis was performed at Colorado State University by using large scale wind tunnel modeling. This evaluation includes the effect of wind pressure on the storage building at different wind speeds and directions, study of the effectiveness of the inlet structure, and selection of an efficient wind deflector. The wind tunnel test confirmed the performance of the natural convection system and helped to validate the site model, and to confirm the wind effects.

III.3 Airflow Model

The objective of the Airflow Model is to simulate temperature, airflow, pressure, and buoyancy effects in the vault in order to identify canister center line and structure temperatures at varying canister loading scenarios, air conditions, and hydraulic resistances. The Airflow Model considers the hydraulic resistance along the entire airflow path within the vault, the wind effect, and the stack effect. The airflow hydraulic resistances are determined as functions of air entry geometry, internal vault, and stack size and height using the T-DUCT[3] computer code. The amount of airflow ultimately depends on the vault and ambient air temperature difference, the stack height and diameter, and the airflow hydraulic resistances. Additional pressure created by wind can contribute to or hinder the natural convection effect. For a conservative approach the positive effect of wind is not considered, however for the same reasons, any negative effect of wind is carefully analyzed and deducted from stack

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[2]PHOENICS computer code developed by CHAM, Huntsville, Alabama

[3]T-DUCT computer code developed by NETSAL & Associates, Fountain Valley, California

pressure.

The Airflow Model assumes that airflow in the vault is moving in a horizontal direction from inlet to outlet with a limited upward temperature gradient. Later this approach is corrected in the Hot Area Model by accounting for inside air circulation and "hot spots". For a loaded vault the air is gradually heated as it passes through rows of canisters. As the first guess, air temperature at the outlet stack is assumed as the highest in the vault. Later this assumption is analyzed by using the Hot Area Model. The Airflow Model calculates maximum centerline canister temperatures for each sleeve in a row with a staggered or in-line arrangement and defines exhaust air temperature and airflow. Mostly, physical properties of nuclear materials (density, heat transfer coefficient) vary with temperature. The Airflow Model solves numerically the differential equation that describes the heat transfer process for each elemental layer in a canister. Then it does pressure balancing between stack effect and hydraulic resistance by iterations. Air densities, stack effect, and flow resistance calculations are the parts of the iteration process. The iteration process is considered converged when stack effect and total airflow resistances are matched.

As an example, the Airflow Model has been used to simulate a vault loaded with 2000 canisters. The heat loss is expected to be 1,500 kW. For this heat load, with assumptions of maximum air inlet temperature of 115°F (46°C) and zero wind resistance, the simulation predicts the vault air temperature distribution presented in Figure 3. The maximum temperature in the vault is estimated to be 187°F (87°C). The outlet air temperature is 171°F (77°C). For the above values of air inlet temperature and wind resistance, a series of simulations has been performed at various heat loads to determine the

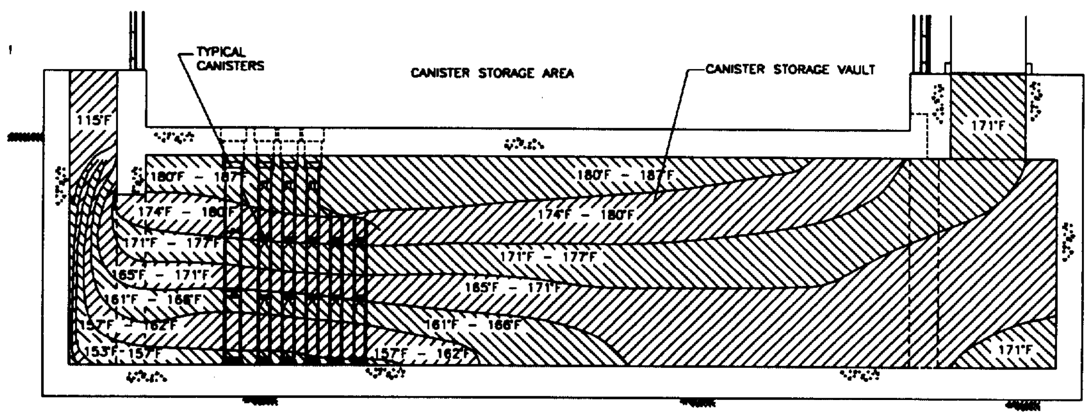

Figure 3. Temperature distribution in the vault.

total airflow through the vault and the outlet air temperatures as functions of the number of stored canisters. The results of these simulations are presented in Figures 4 and 5.

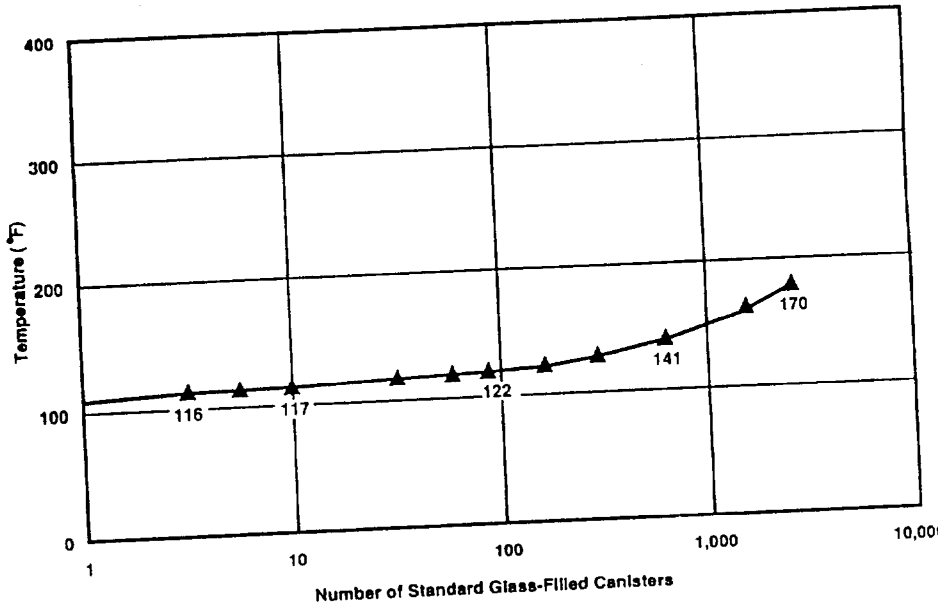

Figure 4. Outlet air temperature as a function of the number of stored canisters.

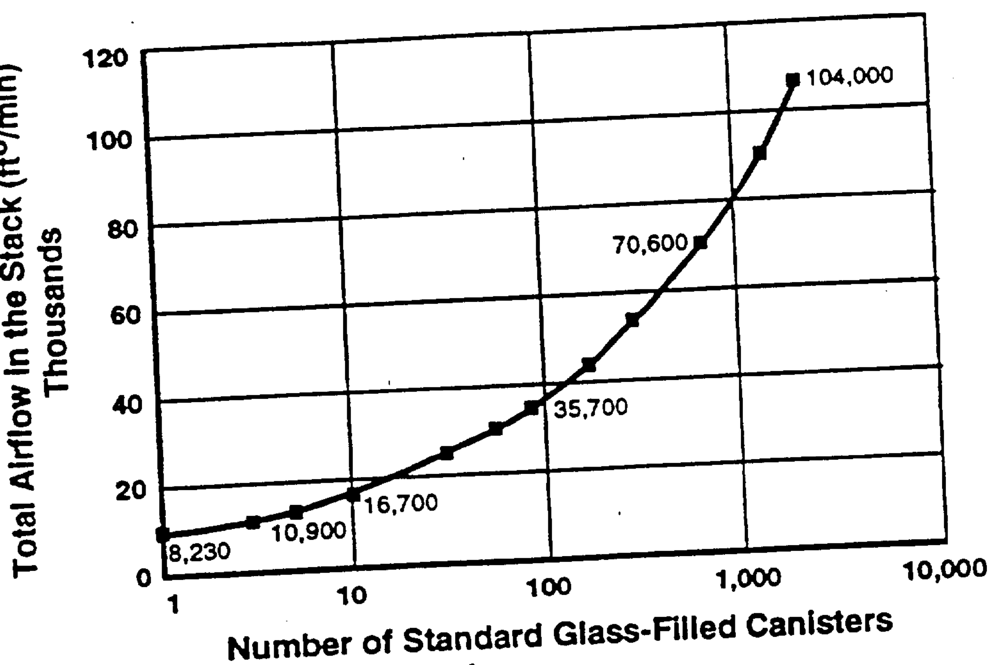

Figure 5. Total airflow in the outlet stack as a function of the number of stored canisters.

III.4 Vault Model

The objective of the **Vault Model**, based on CFD technique, is to simulate actual flow and temperature distribution inside the vault using the airflow calculated previously by the Airflow Model. The Vault Model locates the hottest temperature areas in the vault, called "hot spots." These temperatures are used: (1) to adjust temperature for the surrounding air in the Airflow Model when calculating canister center line temperatures, and (2) to calculate maximum concrete ceiling temperature.

There are three major airflow regions in the vault: (1) the beginning of the stream where entering air is moving down creating first entry vortex, (2) the middle of the stream divided into a lower part moving toward the outlet and an upper part moving toward the inlet, and (3) a third part which is a vortex located below the vault outlet opening.

Air buoyancy and slower air movement in the third flow region create higher temperatures at the top of the vault. This is called the "hot area". To calculate the actual air temperature in "hot area" is important for finding the surface temperature of the concrete. The numerical simulation, using the PHOENICS computer code, shows internal air temperatures at each cell including temperatures of inlet and outlet air. Then, the maximum difference between outlet and "hot area" air temperatures is added to the outlet temperature calculated by the Airflow Model. The result of this combined air temperature is considered as the "hot area" air temperature in the vault.

For the example above, it shows the maximum temperature difference between outlet air and air in the "hot area" is 9°F (equivalent to 5°C). This temperature difference was added to the highest air temperature in Airflow Model for identifying "hot area" temperature in the vault. It should be noted that by adjusting the geometry of the inlet and outlet plenums a minimum temperature difference between the "hot area" and the outlet air temperature can be attained.

For a partially loaded vault the outlet air temperature, the stack effect pressure as well as the airflow are lower. This may create hot areas at a lower heat load. A separate study indicates that lower heat load reduces air temperatures all over the vault. This means that the only hot area critical conditions that should be analyzed are at the highest canister loading.

III. 5 Hot Area Model

The objective of the Hot Area Model is to determine the maximum concrete structure temperature. The "Hot area" is a part of vault's inner space where, due to the buoyancy effect, the air temperature is highest. Calculation of temperatures and heat fluxes at the hot area is based on the results obtained by previous airflows and temperatures calculated in the Vault Model. The heat from hot area is transferred: (1) to the natural convection cooling system, and (2) to the operating area above the storage vault (see Figure 1).

The Hot Area Model is used for simulating temperatures in the upper vault area and to determine the concrete temperature profile. These temperatures are functions of many parameters such as outside air temperature, wind resistance, canister load, insulation thickness, etc. It is evident that concrete temperature rises with the increase in outside air temperature, however, stack effect moderates the temperature rise.

III.6 Model Interactions

A flowchart of model interaction is presented in Figure 6. Initially, the Site Model calculates static pressures differences at air inlet openings for different wind directions and velocities. In the event that wind may create a pressure resistance at inlet openings, this pressure is carried over to the following Airflow Model. If the wind creates a negative effect on the inlet, wind pressure is deducted from the total pressure balance which includes inlet wind pressure, outlet wind pressure, stack effect, and hydraulic resistance. If the wind creates a positive pressure force at inlet openings, conservative approach is taken and no wind pressure is assumed. The Airflow Model calculates the canister surface and centerline temperatures. Factors effecting these calculations are the stack effect and hydraulic resistances at the inlet towers and outlet stack calculated by the Site Model. Airflow is calculated for an equilibrium between path hydraulic resistance and the stack effect pressure.

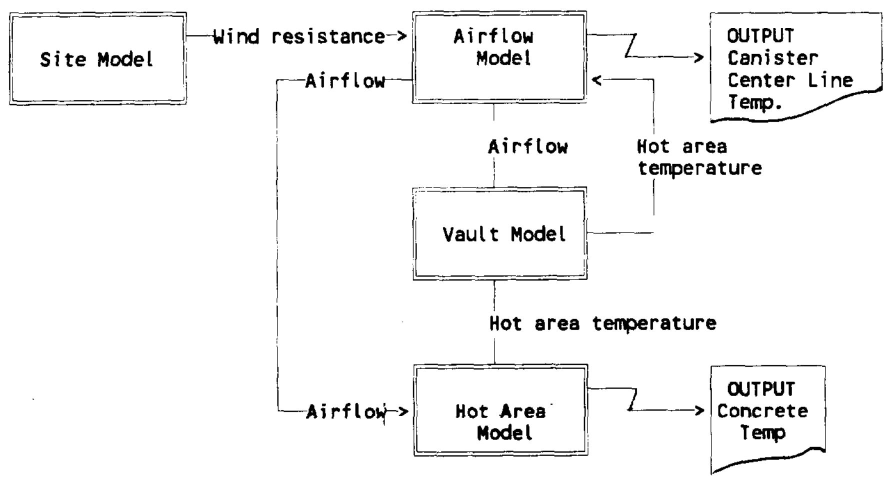

Figure 6. Computation flowchart.

The Vault Model simulates distribution of air temperatures in the vault using the amount of flow calculated by the Airflow Model. The CFD technique used in this model allows the user to identify the "hot areas" in the vault where air temperatures are the highest. It iterates into the Airflow Model and becomes a base for calculating canister temperatures in the most critical "hot areas." After several iterations between the Airflow Model and the Vault Model, the maximum "hot area" temperature is carried over to the Hot Area Model where it is used as a base for calculating concrete temperature.

IV. CONCLUSION AND RECOMMENDATIONS

The results of this engineering analysis, computer modeling, and wind tunnel testing show that a natural convection cooling system performs its intended function, and is a safe alternative design for the storage of radioactive materials.

The following are the recommendations based on the results of this study:

- The natural convection system for canister storage must have an exhaust stack with a wind deflector to reduce hydraulic resistance.
- Air inlet should be through one or more inlet towers designed and tested to prevent or minimize the negative wind pressure effect for all wind directions. Installation of the internal wind braking baffles, and aerodynamic canopy (Figure 1), are recommended.

DISCUSSION

MURTHY: Is your answer, yes; your study has now been accepted by the authorities, the DOE, or is it a new study and nobody knows about it?

TSAL: Mostly yes, because, first of all, we do have a utility in Colorado. There are several utilities in the world working on natural convection, for example in France. I know there are French people here and maybe they will help me find out where those utilities are. Also there is a utility in the United Kingdom. Unfortunately, in the United States we have to penetrate the wall of incomprehension and this is one of the reasons for my presentation. After working all my life in the HVAC field, dealing with natural convection, I truly believe that this is possible, legitimate, and that we have to go ahead with natural convection. It is much cheaper and more effective than the forced convection. But we are still going through the walls of incomprehension, it has not been easy to convince people. Fluor Daniel is in the stage of preconceptual design of natural convection (NC) for both nuclear waste and SNM storages for DOE.

DISPOSAL OF SLIGHTLY CONTAMINATED RADIOACTIVE WASTES FROM NUCLEAR POWER PLANTS

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ABSTRACT

With regard to the disposal of solid wastes, nuclear power plants basically have two options, disposal in a Part 61 licensed low-level waste site, or receive approval pursuant to 20.2002 for disposal in a manner not otherwise authorized by the NRC. Since 1981, the staff has reviewed and approved 30 requests for disposal of slightly contaminated radioactive materials pursuant to Section 20.2002 (formerly 20.302) for nuclear power plants located in non-Agreement States. NRC Agreement States have been delegated the authority for reviewing and approving such disposals (whether onsite or offsite) for nuclear power plants within their borders. This paper describes the characteristics of the waste disposed of, the review process, and the staff's guidelines.

INTRODUCTION

NRC regulations, at 10 CFR Part 20.2001, authorize four general alternatives for nuclear power plant licensees to dispose of slightly contaminated radioactive wastes from nuclear power plants: (1) by transfer to an authorized recipient as provided in 20.2006, or in Parts 30, 40, 60, 61, 70, or 72 of the NRC regulations or, (2) by decay in storage, (3) by release in effluent within the limits in 20.1301, or (4) as authorized under 20.2002, 20.2003, 20.2004, or 20.2005. In February 1983, the NRC staff published Information Notice No. 83-05 entitled "Obtaining Approval for Disposing of Very-Low-Level Radioactive Waste - 10 CFR Section 20.302." This Information Notice reminded nuclear power plant licensees that they could apply on a case-by-case basis for permission to use alternative methods for disposing of slightly contaminated radioactive materials (i.e., methods other than disposal at commercial wastes sites).

CHARACTERISTICS OF DISPOSED WASTES

During the 13-year period between 1981 and 1994, the NRC staff has reviewed and approved 30 requests for disposal of slightly contaminated radioactive materials from nuclear power plants (see Figure 1). Experience has shown that the review process can take from 2 weeks to 1 year. Table 1-5 contains a list of applications processed by the staff, as well as the general physical characteristics of the wastes. The types of waste disposed of are the following slightly contaminated materials: soil, sand, sediment from onsite settling ponds, sewage sludge, wood, spent resins used for cleaning the secondary side of pressurized water reactors (PWRs), roofing materials, and scrap metal from feedwater heaters used in the secondary side of PWRs (see Figure 2).

The principal nuclides in the wastes disposed are Co-58, Co-60, Cs-134, and Cs-137, with total activity concentrations in the range of about 1 to 50 pCi/g.

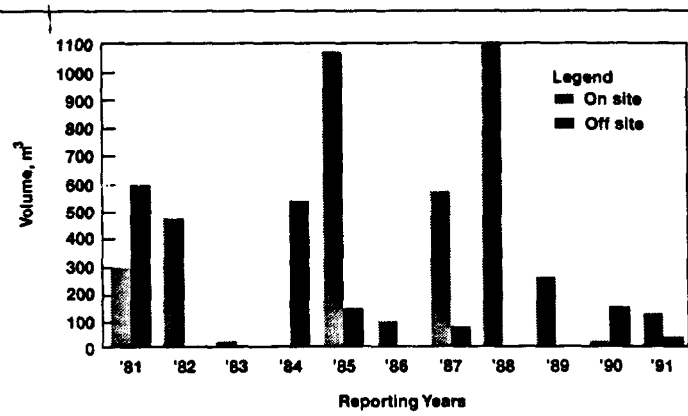

Figure 1. Requests for disposal of slightly contaminated radioactive material from nuclear power plants.

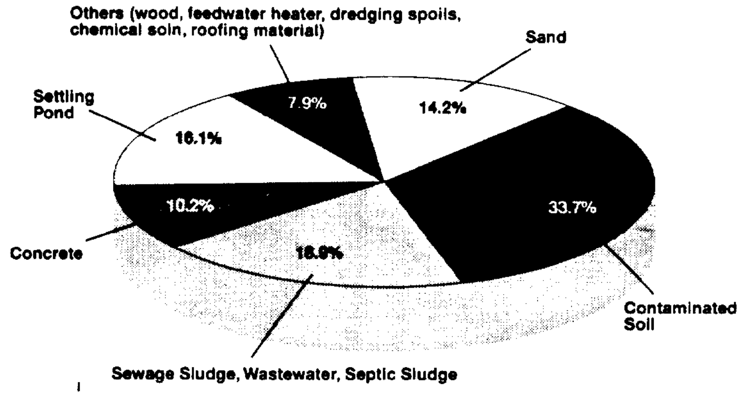

Figure 2. Characteristics of disposed wastes.

Figure 3 shows the total activity (mCi) versus volume of disposed waste.

Disposal methods have included municipal landfills, on-site burial, and processing at a hazardous waste disposal site. Although most of the requests have been for a one-time disposal, the NRC staff has approved requests for disposal of limited quantities contaminated materials on a repetitive basis, e.g., annual disposals of slightly contaminated wood, and disposals of contaminated sewage sludges every 1-2 years. For these repetitive disposals, the licensee must reapply to the NRC when a particular disposal would exceed the boundary conditions imposed by the staff's analysis (see Figure 4).

REGULATORY REVIEW PROCESS

Applications from reactor licensees in non-Agreement States should be submitted to the Office of Nuclear Reactor Regulation (NRR). Under 10 CFR 20.2002, licensees may request disposal of specific material on a case-by-case basis or licensees may request permission for routine disposal of specific types of wastes on a repetitive basis using approved procedures and systems. For disposal of radioactive materials in non-Agreement States, the application is reviewed solely by NRC. For disposal of radioactive material in an agreement State on or away from the reactor site, approval is needed from the Agreement State.

In their submittals licensees should describe the waste, the principal pathways of exposure at the disposal site, and the estimated dose to the maximally exposed individual from these pathways. For each planned request, the information regarding the waste for each planned request should briefly describe: (1) the item to be disposed, including the approximate volume or mass; (2) the principal nuclides expected to be in the waste; (3) the estimated radionuclide concentrations in the waste; (4) the estimated total activity of nuclides in the waste; and (5) the basis for the estimated concentrations and total activities (i.e., the number of samples measured, the representative nature of the samples, and the appropriateness of the instruments used to measure the activity in samples). Information regarding the disposal site should include: (1) the method of disposal (e.g., diluted with other sludge, burial in deep trenches, or spread over land and cover with "clean" soil); (2) the location of the disposal site (e.g., a legible map of the disposal site with compass direction and scale); (3) local land use (e.g., nearby residences and wells); and (4) physical or administrative barriers to prevent present or future use of the site for other than its intended purpose.[1]

Licensee should also briefly discuss the potential pathways of exposure and estimated doses to individuals from the principal pathways of exposure. Doses should be estimated for both a maximally exposed member of the public and a non-occupationally exposed worker. If a particular pathway is not of concern (e.g., inhalation of resuspended radionuclides) then this should be stated and the basis for the statement should also be stated (e.g., the nuclides are in an immobile form or the material is isolated from surface winds by several feet of earth cover.) Among pathways that are typically of concern include: 1) external exposure from standing or living above the disposal site (2)

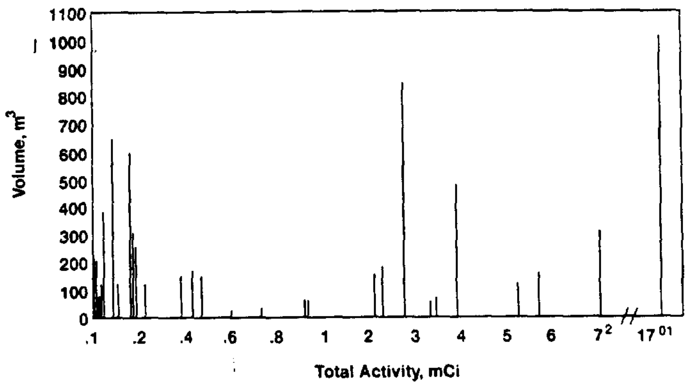

Figure 3. Activity of disposed of waste.

inhalation of resuspended radionuclides if the radioactive material is not covered promptly or effectively, (3) external and internal exposure to an inadvertent intruder, (4) external and internal exposure of an individual from assumed recycling of the material disposed of at the time the disposal site is released from regulatory control, (5) internal exposure from the ingestion of ground water, and (6) internal exposure from ingestion of food grown on the disposal site (see Figure 5).

DOSE GUIDELINES

In performing its safety evaluations of licensee submittals, the NRR staff has developed radiation dose guidelines for its staff. NRR is developing additional review guidelines such as standard review plan and staff report are under development to ensure that potential radiation doses that may result from the proposed method of disposal are calculated in a consistent manner. These guidelines are intended to apply to solid wastes (from reactor facilities) slightly contaminated with radionuclides with halflives less than 35 years. NRR reviews the licensee's waste stream description, radiological properties, and proposed disposal method, as well as radiological impacts, calculational methods, and assumptions, to ensure that the public health and safety is adequately protected. The staff will seek additional information from the licensee to further justify acceptance of the licensee's assessment methodology.

NRR developed its guidelines according to the following principles: first, the annual dose to a member of the public from exposure to the material disposed of should be a small fraction of annual exposure to natural background radiation. Second, the annual dose to a member of the public from exposure to the material disposed of should be no greater than the annual dose a maximally exposed individual would receive from exposure to radioactive effluents from normal operations at light water reactors. Third, concentrated sources of radioactive materials that might pose a health hazard before or after the time of release of the disposal site from all regulatory controls should not be permitted to be disposed of under 10 CFR 20.2002.

The NRR guidelines follow:

1. The radioactive material should be disposed of in such a manner that it is unlikely that the material would be recycled.

2. Doses to the whole body and to any body organ of a maximally exposed individual (a member of the general public or a non-occupationally exposed worker) from the probable pathways of exposure to the material disposed of should be less than 1 mrem/yr.

3. Doses to the whole body and to any body organ of an inadvertent intruder from the probable pathways of exposure should be less than 5 mrem/yr.

4. For onsite disposal, the dose to the whole body and to any body organ of an individual from assumed recycling of the material disposed of at the time the disposal site is released from regulatory control from all likely pathways of exposure should be less than 1 mrem/yr.

5. For disposal in a sanitary landfill, the dose to the whole body and to any body organ of an individual

from assumed recycling of the material disposed of at the time of disposal from all likely pathways of exposure should be less than 5 mrem/yr.

In conclusion, NRR believes that the use of 10 CFR 20.2002 for case-specific situations involving slightly contaminated radioactive wastes has been appropriate and any potential radiological impact on public health and safety or the environment has been minimized.

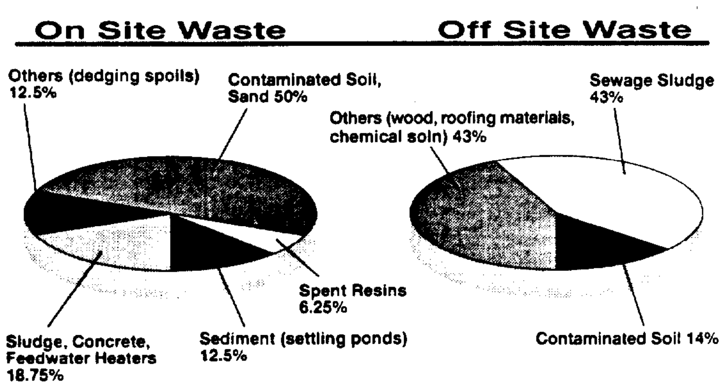

Figure 4. Percent composition of disposal of waste.

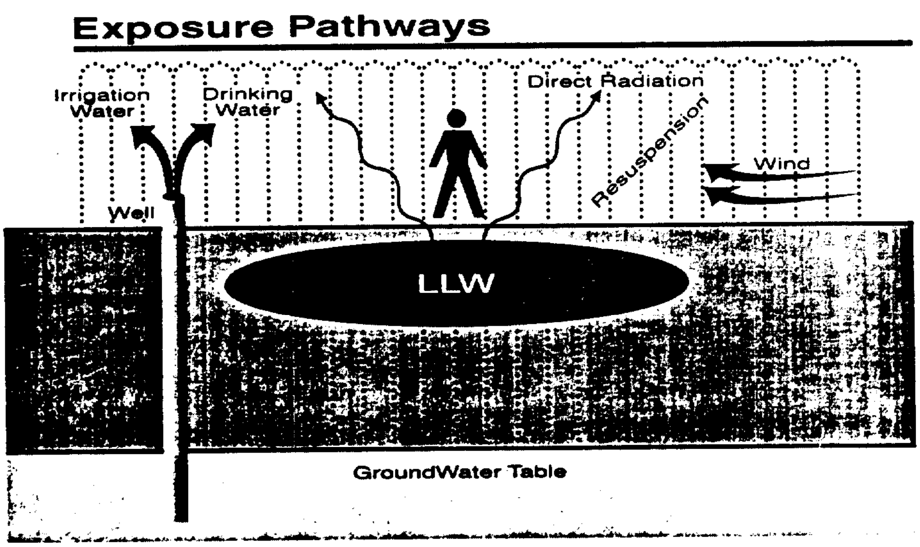

Figure 5. Potential pathways of exposure.

Table 1-5 Approved Requests for Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants PUrsuant to 10 CFR 20.302.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
San Onofre	sand 300	onsite	Cs-137	0.2 mCi	APPROVAL: 1 mrem/yr whole body
Oyster Creek	contaminated. soil 480	onsite	Co-60 Cs-137 Mn-54 Cs-134	5 mCi	APPROVAL: 3 mrem/yr whole body
DC Cook	contaminated concrete, steam generator replacement 653	onsite	Co-60 Cs-134 Cs-137	0.1 mCi	APPROVAL: Insignificant impact because it involves pathways less significant than those considered in the FES.
Vermont Yankee	septic waste 262	onsite	Co-60 Mn-54 Cs-137 Cs-134 Zn-65	0.2 mCi per acre	APPROVAL: 0.2 mrem/yr maximally. exposed individual/organ:, 3.91 mrem/yr inadvertent intruder licensee evaluation..
Yankee Rowe	sewage 200 once every 1 to 2 years for 30 years.	offsite	Co-60 Mn-54 Cs-134 Cs-137	0.2 mCi	APPROVAL: 0.12 mrem/yr maximally exposed individual/whole body (child), ground irradiation,inhalation,stored vegetables,leafy vegetables,milk ingestion.
Big Rock Point	dredging spoils 15 yr	onsite	Co-60 Mn-54 Cs-137 Cs-134 Sr-90	0.9 mCi	APPROVAL: 0.03 mrem/yr whole body dose maximally exposed individual (groundshine,inhalation,groundwater)& 0.857 mrem/10 yr.
Palisades	soil 170	onsite	Co-60 Cs-137	0.03 mCi	APPROVAL: <1 mrem/yr whole body dose maximally exposed individual (groundshine, inhalation, groundwater ingestion)

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Maine Yankee	Hazardous chemical solution 40	offsite	Co-60 Zn-65 Cs-137	0.1 mCi	APPROVAL; 0.1 mrem/yr (lung) inhalation resuspended activity/ingestion of crops grown onsite; 0.0064 mrem/yr (whole body) ingestion drinking water; 0.34 mrem/yr (whole body) direct radiation buried activity; 0.12 (whole body) direct radiation waste transportation; 0.005 (whole body) direct radiation waste handling.
Sequoyah	trash 750	offsite		200 mCi	REJECTED: high-level of specific activity of less than 2 nCi/gm, and total activity proposed/ year of 200 mCi, are each orders of magnitude higher than similar parameters of any 20.302 proposal approved for offsite, also non-homogeneity of the trash
Fermi-2	cont soil 850	onsite	Cr-51 Mn-54 Co-58 Co-60	0.3 mCi	APPROVAL: 0.044 mrem/yr whole body public water and fishing ingestion and shoreline sediments, 0.0674 mrem/yr direct exposure to contaminated soil, resuspension of soil into air: 0.3 mrem/yr direct exposure.
Kewaunee	waste sludge 454	onsite	Co-60 Cs-137	0.2 mCi	APPROVAL; 0.034 mrem/yr (whole body) groundshine; 0.008 mrem/yr (whole body) inhalation; 0.007 mrem/yr (whole body) groundwater ingestion.

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Brunswick	dredging sediments, sand	onsite	Mn-54 Co-60 Cs-137		NON- APPROVAL: Meet NRC acceptance criteria, but the licensee is regulated by State. State approval is needed
Point Beach 1,2	sewage sludge 113	onsite	Co-60 Cs-137	0.003 mCi	APPROVAL: Based on licensee pathway analysis and the licensed materials <3% of the primarily nuclides, already acceptable in the FES,site-speciffic application to be insignificant radiological impact.
Surry 1,2	soil 300	onsite	Co-60 Cs-134 Cs-137 Mn-54	72	APPROVAL: Groundshine, inhalation dose breathing resuspended airborne radioactivity, and ingestion of radioactivity from contaminated water to maxiimally exposed member of the public <1 mrem/yr; inadvertent intruder <5 mrem/yr.
H.B. Robinson	sediment 170	Fossil plant ash pond in licensees controll area	Co-68	75	APPROVAL: <5 mrem (teenager, total body of person working 400 hr/yr above contaminated surface of soil cover zone.
H.B. Robinson	Soil (1.5)	Onsite along the bottom of a drainage ditch	Co-58 Co-60 Co-134 (10%) Cs-137 (23%) Mn-54 all nuclides	0.014	APPROVAL: <5mrem (direct radiation whole body of person working 400 hr/yr above contaminated surface of soil cover zone.

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Humboldt Bay 3	Sludge (36.8)	Offsite RCRA chemical waste disposal landfill (Martinez , Calif.)	Co-60, Cs-134 Cs-137 Th-234	267.8 3.1 155 19.3	APPROVAL: 1.5 mrem (direct radiation to worker standing on uncovered dried sludge)
Oconee Units 1,2,3	Sewage sludge 113	Offsite sanitary landfill	Co-58, Co-60 (27%) Cs-134 Cs-137 (45%) all nuclides	0.07	APPROVAL: 0.6 mrem (whole body direct radiation standing on uncovered dried sludge) 0.2 mrem(whole body ingestion of vegetable grown on landfill w/sludge). 0.3 mrem (highest dose to any organ, ingestion of vegetable).
H.B. Robinson	Setting pond sediment 60,000 cu meters	Onsite fossil ash pond	C0-60	1700 (over life of pond)	APPROVAL: 5 mrem (direct radiation) to teenager 67 hr/yr at ash pond
R.E.Ginna 1	Roofing materials <100 tons	Offsite municipal landfill	Co-60 Cs-134 Cs-137	0.30 0.23 0.92	APPROVAL: 4 mrem (1st year) 9 mrem (thereafter)
McGuire 1 and 2	Wastewater residue sludge <368	Onsite	Co-58 Co-60	0.05 0.05	APPROVAL: <1 mrem (whole body direct radiation standing 2000 hr/yr on soil cover); <0.1 mrem (lung dose, worker inhaling dust 2000 hr/yr); <0.1 mrem (highest dose to any organ, ingestion of vegetable).

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Oconee 1, 2,& 3	Feedwater heater a high activity tube bundls. 160 tons. b. very low activity heater shells 100 tons	Company controlled area (outside sec fence) Onsite	Co-60 (79%) Cs-137 (15%) Co-60 (80%)	6.5	APPROVAL: 0.01 mrem (whole body direct radiation standing on soil cover) <2mrem to maximally exposed individual
Oconee 1,2,3	Sand (45)	Onsite Company controlled area (outside security fence)	Cs-134 Cs-137 Co-60 Mn-54 all nuclides	1.2 3 0.1 0.005 <12.3	APPROVAL: <1mrem(groundshine to whole body, standing 2000 hr on soil cover); <0.1 mrem (10 days inhalation of dust from disposal process; <2 mrem (future ingestion of crops grown on burial site, whole body or any organ.)
Big Rock Point	Cont. soil leak in comdensate process monitor	Onsite retain soil in place			APPROVAL
Davis Besse	Secondary side resins 142 or 150 every 5 yrs.	Offsite company owned	Co-58 (34%) Co-60 (3%) Cs-134 (27%) Cs-134 (36%)	8.5/ every 5 yrs.	APPROVAL: 0.7 mrem (direct radiation standing on uncovered basin dredgings); adult eating vegetable. grown on disposal site <3 mrem (whole body); <4 mrem highest dose to any organ; <0.1 mrem (drinking ground water (licensee estimate)

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m³)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Oconee 1,2,3 McGuire 1&2 Catawaba 1&2	Wood 12-21	Offsite sanitary landfill	Assume Cs-137 100%	0.4 to 0.7 per yr per station	APPROVAL: <1 mrem (resident on decommissioned. landfill direct radiation. whole body or any organ of adult eating vegetable grown on soil cover); <0.1 mrem (nuclear station workers, direct radiation. or inhalation).
Pilgrim	Contaminated soil (63	Onsite inplace	Co-60 Cs-137	0.19 mCi 0.4	APPROVAL: <1 mrem external exposure < 0.02 ground shine; <0.5 internal exposure inhalation.resuspended nuclide, (2000 hr/yr), direct radiation0 2000 hr/yr.
D.C. Cook	Contaminated sludge	Onsite pre-burial	Cs-137 Cs-134 Co-60 I-131	8.89 (1982) 5.02 (1991)	APPROVAL: <1 mrem external exposure from disposal site; <1 mrem internal form inhalation of resuspended radionuclide; (3) <0.7 mrem internal exposure ingesting ground water

1. For wastes containing mobile radionuclides (e.g., H-3) detailed information on geology and hydrology may be necessary.

*Disposal of Waste from Nuclear Power Plants.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN BERGMAN

We have had three presentations, the first on the Generation and Release of Radioactive Gases in Low Level Waste Disposal Facilities, by Dr. Yim. He basically stated that the waste is not a static entity, it is dynamic. Just because you have it buried somewhere doesn't mean it is not going generate some surprises. Gases will be generated by a variety of mechanisms. He reviewed microbial, radiolysis, and corrosion. Although, from his analysis, a significant amount of gas would be released, it would be within regulatory limits. It may be within regulatory limits at the present time but looking at the historical picture, when people make measurements twenty or fifty years from now, I can guarantee that regulatory limits will have been decreased. So it is a very important topic and I think the information should be folded into the decision of what kind of solid waste form we should have.

The second paper was by Dr. Tsal on Numerical Analysis of Natural Convection Cooling Systems for Radioactive Canister Storage. He described a natural convective cooling system consisting of two parts: (1) thermal buoyancy inside a vault that creates a stack effect, raising the gas, and (2) external wind forces that may either assist or oppose the air flow through the vault. He used several numerical models to analyze temperature and airflow within the storage vault and showed that natural convective will not only work, but is a safe alternative to the design of current systems.

The third presentation, by Mr. Minns, on the Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants, described methods for disposal, both on site and off site. He also described current regulations that cover disposal of these wastes. People fail to consider these wastes when they are designing the new facilities. I know DOE's EM section is preoccupied with the initial processes for treatment of waste, with less and less thought and attention given to the end. What they do not realize is that after all their hard work, it may not be acceptable (as was pointed out this morning), if they do not consider the full cycle. Waste disposal is one of the most critical components. Current practice is to do more and more burial on site or wherever people do not get excited about it. A lot of facilities, such as the DOE weapons complex, are leaving items in storage because they can't find a waste site. Also, we have critical facilities within the DOE complex that contain HEPA filters that are 20-30 years old; they are kept in place because no one can get rid of them. To borrow the human analogy of Dr. Tsal, just think what would happen if you kept on eating and eating and never disposed of your waste. It doesn't take much imagination to see the consequence. So consideration of waste is a very important part of nuclear activities.