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TECHNICAL STATUS OF THE RADIOACTIVE WASTE REPOSITORY – A DEMONSTRATION PROJECT FOR SOLID RADIOACTIVE WASTE DISPOSAL

F. L. Culler

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APRIL 1971

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ABSTRACT

This report reviews the work that has been done to establish the suitability of bedded salt formations for the containment of radioactive solid wastes and describes the nature of the additional investigations that are either under way or are scheduled to be carried out. It is believed that this work constitutes a valid technical justification for proceeding with the present plans to establish a demonstration repository at Lyons, Kansas.

1. INTRODUCTION

The quantity of radioactive wastes will increase by several orders of magnitude over the remainder of this century as nuclear power assumes an increasingly important role in meeting the nation's electrical energy requirements. The wastes of primary concern, designated "high-level" and "alpha" wastes, contain radioactive nuclides that decay so slowly that they will require containment outside the biosphere for hundreds of thousands of years. High-level wastes, principally the fission-product concentrates that arise in the reprocessing of spent reactor fuels, are characterized by intense, penetrating radiation and high thermal power. The alpha wastes are principally solid materials contaminated with plutonium that are generated in the fabrication of reactor fuels or in AEC production plants. They are characterized by radiations of low thermal power and penetrability, but are extremely toxic.

The practice up to the present time has been to store these wastes on a temporary basis as aqueous solutions and slurries in tanks, and as solids in near-surface land burial grounds. Recognizing that a truly permanent solution of this problem is required, the AEC has promoted the development of processes for converting the liquid wastes to stable, solid forms; further, in consultation with the National Academy of Sciences and members of the geologic community, the AEC has sponsored work over the past 15 years, principally at Oak Ridge National Laboratory, to examine

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the suitability of bedded salt formations as a final repository for these materials. As a first step in implementing an overall waste management policy, the AEC has published an appendix to 10CFR50 in the Federal Register. This appendix requires industry to solidify their high-level liquid wastes and ship them to a federal repository. In June 1970, the AEC announced the tentative selection of a site near Lyons, Kansas, for a demonstration salt-mine repository.

This report reviews the work that has been done to establish the suitability of bedded salt formations for the containment of radioactive solid wastes, and the nature of the additional investigations which are either under way or scheduled to be done. It is believed that, on assessment, this work constitutes a valid technical justification for proceeding with the present plans to establish a demonstration repository.

Acknowledgment. - This statement is based upon the work performed by many ORNL staff members since approximately 1956. I particularly wish to acknowledge the assistance of R. L. Bradshaw (deceased), J. O. Blomeke, W. C. McClain, J. P. Nichols, T. F. Lomenick, F. M. Empson, E. J. Frederick, B. F. Bottenfield, A. L. Boch, G. W. Renfro, E. Sonder, R. D. Cheverton, W. D. Turner, S. Lindenbaum, R. S. Dilion, G. H. Jenks, G. M. Watson, A. P. Malinauskas, and E. G. Bohlmann.

2. CHARACTERISTICS OF THE WASTES

The wastes vary widely in their physical, chemical, and radiation characteristics, depending on their origin and previous history of treatment. They can, however, be processed and packaged so that they can be safely shipped and stored as solids, under carefully prescribed conditions, in a salt mine.

2.1 High-Level Wastes

The source of virtually all the high-level wastes is the reprocessing plants that recover the uranium and plutonium from spent reactor fuels. As generated, these wastes are nitric acid solutions containing fission products and smaller quantities of uranium, plutonium, and other heavy elements, and they can be generally characterized by their very intense, penetrating radiation and their high heat-generation rates. The radiation characteristics of a typical waste and the identity of the significant radionuclides are presented in Tables 2.1, 2.2, and 2.3.

Four processes have been developed for converting these liquid wastes to solid forms which, we believe, can be shipped and stored safely in salt. Table 2.4 summarizes the characteristics of these solid products, which average about 3 ft^3 in volume per ton of fuel processed. The solids can be described as calcine cake for the pot calcination process, microcrystalline ceramic for the spray solidifier, glass for the phosphate glass process, and a powder for the fluidized-bed calciner and spray calciner. The pot and fluidized-bed products have higher leach rates than do the spray solidifier and phosphate glass products; however, the pot calcine material is more stable at elevated temperatures. The spray and phosphate glass melts, as well as the fluidized-bed powder, can be packaged in mild-steel containers, whereas stainless steel containers must be used in the pot calcination process.

The acceptance criteria for the Repository specify that the dry, solid high-level waste shall be chemically, thermally, and radiolytically stable, and shall be packaged in containers that will maintain their integrity for at least 90 days after they have been buried in salt.

Table 2.1. Radioactivity and Thermal Power of Typical High-Level Waste as a Function of Time Since Processing[a,b]

Time Since Processing (years)	Radioactivity (Ci/metric ton of fuel charged to the reactor)			Thermal Power (W/metric ton of fuel charged to the reactor)		
	Fission Products	Actinides	Total	Fission Products	Actinides	Total
0	4,380,000	22,700	4,440,000	19,300	811	20,100
0.5	2,390,000	12,100	2,400,000	10,800	421	11,200
1	1,680,000	7170	1,690,000	7540	240	7780
2	1,000,000	3800	1,000,000	4400	117	4520
3	691,000	3010	694,000	2890	89	2980
4	533,000	2770	536,000	2110	81	2190
5	448,000	2640	451,000	1680	77	1760
7	365,000	2460	367,000	1260	72	1330
10	311,000	2210	313,000	1010	65	1080
11	300,000	2140	302,000	959	63	1020
12	289,000	2070	291,000	919	61	980
15	264,000	1870	266,000	828	55	883
17	250,000	1750	252,000	781	52	833
20	231,000	1590	233,000	720	47	767
25	204,000	1350	205,000	634	41	675
30	181,000	1160	182,000	560	35	595
35	160,000	1010	161,000	495	30	525
40	142,000	875	143,000	438	27	465
50	112,000	679	113,000	344	21	365
75	61,500	417	61,900	188	13	201
100	34,100	309	34,400	103	9	112

[a]The fuel from a light water reactor with an initial uranium enrichment of 3.3% is assumed to have been irradiated at an average specific power of 30 MW/metric ton to an exposure of 33,000 MWd/metric ton. The waste consists of all fission products plus the actinides remaining after removal of 99.5% of the uranium and plutonium following a postirradiation decay period of 150 days.

[b]A typical container of solidified waste having inside diameter of 6 in. and an active height of 8 ft contains waste from 0.476 metric ton of fuel charged to the reactor.

Table 2.2. Radioactivity and Mass of Radionuclides in Typical High-Level Waste as a Function of Time Since Processing[a,b]

Nuclide	Radioactivity (Ci/metric ton of fuel charged to the reactor) After: 1 year	3 years	5 years	10 years
^{239}Np	18	18	18	18
^{238}Pu	91	110	109	105
^{239}Pu	2	2	2	2
^{240}Pu	3	3	3	4
^{241}Pu	483	444	404	318
^{241}Am	154	155	156	157
^{243}Am	18	18	18	18
^{242}Cm	4160	189	12	3
^{244}Cm	2220	2060	1910	1580
^{90}Sr	74,700	71,100	67,700	59,900
^{90}Y	74,800	71,200	67,700	59,900
^{95}Zr	5610	2	0	0
^{95}Nb	11,900	5	0	0
^{99}Tc	14	14	14	14
^{106}Ru	205,000	51,600	13,000	413
^{106}Rh	205,000	51,600	13,000	413
^{125}Sb	6280	3760	2250	623
^{125m}Te	2590	1560	934	258
^{127m}Te	604	6	0	0
^{127}Te	597	6	0	0
^{134}Cs	153,000	77,600	39,400	7270
^{137}Cs	104,000	99,300	94,900	84,500
^{137m}Ba	97,300	92,900	88,700	79,000
^{144}Ce	316,000	53,100	8930	104
^{144}Pr	316,000	53,100	8930	104
^{147}Pm	76,000	44,800	26,400	7030
^{151}Sm	1140	1120	1100	1060
^{154}Eu	6550	6000	5510	4430
^{155}Eu	4350	2020	939	138
Total[c]	1,690,000	694,000	451,000	313,000
	Mass (g in waste per metric ton of fuel charged to the reactor)			
Fission products	28,900	28,900	28,900	28,900
U	4780	4780	4780	4780
Np	486	486	487	487
Pu	50	53	54	56
Am	140	140	140	141
Cm	29	26	24	20
Total[c]	40,500	40,500	40,500	40,500

[a]The fuel from a light water reactor with an initial uranium enrichment of 3.3% is assumed to have been irradiated at an average specific power of 30 MW/metric ton to an exposure of 33,000 MWd/metric ton. The waste consists of all fission products plus the actinides remaining after removal of 99.5% of the uranium and plutonium following a postirradiation decay period of 150 days.

[b]A typical container of solidified waste having inside diameter of 6 in. and active height of 8 ft contains waste from 0.476 metric ton of fuel charged to the reactor.

[c]Total includes all radionuclides.

Table 2.3. Photon and Neutron Source Strength of Typical High-Level Waste as a Function of Time Since Processing[a,b]

	Photons/sec or Neutrons/sec per Metric Ton of Fuel Charged to the Reactor			
	1 year	3 years	5 years	10 years
Photons				
0.30 MeV	3.55×10^{15}	9.06×10^{14}	3.65×10^{14}	1.87×10^{14}
0.63 MeV	2.20×10^{16}	1.16×10^{16}	7.20×10^{15}	3.59×10^{15}
1.10 MeV	1.50×10^{15}	5.56×10^{14}	2.87×10^{14}	1.45×10^{14}
1.55 MeV	4.87×10^{14}	1.91×10^{14}	8.71×10^{13}	2.04×10^{13}
1.99 MeV	1.95×10^{14}	3.81×10^{13}	7.76×10^{12}	2.00×10^{11}
2.38 MeV	5.49×10^{13}	1.38×10^{13}	3.46×10^{12}	1.10×10^{11}
2.75 MeV	1.78×10^{12}	4.48×10^{11}	1.13×10^{11}	3.57×10^{9}
3.25 MeV	3.23×10^{10}	8.12×10^{9}	2.04×10^{9}	6.49×10^{7}
Neutrons	3.98×10^{8}	3.35×10^{8}	3.12×10^{8}	2.58×10^{8}

[a]The fuel from a light water reactor with an initial uranium enrichment of 3.3% is assumed to have been irradiated at an average specific power of 30 MW/metric ton to an exposure of 33,000 MWd/metric ton. The waste consists of all fission products plus the actinides remaining after removal of 99.5% of the uranium and plutonium following a postirradiation decay period of 150 days.

[b]A typical container of solidified waste having inside diameter of 6 in. and active height of 8 ft contains waste from 0.476 metric ton of fuel charged to the reactor.

Table 2.4. Characteristics of Solidified High-Level Wastes

	Pot Calcine	Spray Melt	Phosphate Glass	Fluidized-Bed Calcine
Form	Calcine cake	Monolithic	Monolithic	Granular
Description	Scale	Microcrystalline[a]	Glass	Amorphous[b]
Chemical composition, mole %				
Fission product oxides	15 to ~ 80	5 to 30	5 to 25	5 to 50[c]
Inert metal oxides	10 to 50	40 to 50	10 to 30	10 to > 90
Sulfur oxides (if in waste)	0 to 40	0 to 40	0	0 to 40
Phosphorus oxides	~ 0	25 to 40	~ 60	~ 0
Bulk density, g/ml	1.1 to 1.5	2.7 to 3.3	2.7 to 3.0	1.0 to 1.7
Thermal conductivity, Btu hr^{-1} ft^{-1} $°F^{-1}$	0.15 to 0.25	0.4 to 1.0	0.4 to 1.0	0.10 to 0.25
Maximum allowable thermal power, W/liter of solid[d]	85	205	190	70
Leachability in cold water, g cm^{-2} day^{-1}	10^{-1} to 10^{-2}	10^{-3} to 10^{-6}	10^{-4} to 10^{-7}	10^{-1} to 10^{-2}
Hardness	Soft	Hard	Very hard	Moderate
Friability	Crumbly	Tough	Brittle	Moderate
Residual nitrate, wt % of product	≤ 0.05	≤ 0.005	≤ 0.005	≤ 4.0
Volume, liters/1000 MWd (thermal)	1 to 2.5	1.2 to 3	1.5 to 5	1.5 to 5
Maximum stable temperature, °C	~ 900	Phase separation at ~ 900	Devitrifies at ~ 500	~ 600
Container material	Stainless steel	Mild steel or stainless steel	Mild steel or stainless steel	Mild steel or stainless steel

[a]Glassy products can also be made with some difficulty.

[b]Microcrystalline products can also be made.

[c]Composition ranges for fluidized bed are also for Purex waste and are estimated.

[d]Approximate values for storage in air in 8-in.-diam cylindrical pots to maintain pot center-line temperatures at less than 900°C and pot wall temperatures at less than 425°C. Average k values were used.

2.2 Alpha Wastes

The alpha wastes consist of a wide assortment of solid materials that arise principally from the preparation of reactor fuels and from AEC production plants. They include both combustible and compactable materials and may contain plutonium and other transuranium elements in amounts ranging from trace concentrations to several grams per cubic foot. Since about 60% of these wastes are combustible, substantial volume reduction by incineration is possible. Alpha wastes are packaged in tightly-sealed steel drums and crates, and since they are virtually free of penetrating radiation and evolve only a few hundredths of a watt per cubic foot, they do not require remote handling techniques.

Packages that are known to contain liquids, potentially explosive chemicals, pyrophoric materials, or oxidants (other than air), and sealed inner containers filled with gas under pressure, will not be accepted at the Repository. These materials must be converted to stable, inert, solid forms by mixing with Portland cement or an approved substitute.

2.3 Shipment of Wastes

Containers of solidified high-level waste will be shipped to the Repository in shielded casks on rail cars. Alpha wastes will be shipped to the Repository as "Type B" or "Large Quantity" radioactive materials shipments in specially designed ATMX rail cars that are equipped with returnable 8 x 8 x 20-ft-long cargo containers. All shipments will conform with regulations of the Department of Transportation (49 CFR 173.389 through 173.399) and the Atomic Energy Commission (AEC Manual, Chapter 0529 for AEC contractors or 10 CFR 71 for AEC licensees).

3. RATIONALE FOR THE USE OF SALT DEPOSITS

In 1955, an advisory committee of geologists and geophysicists was established by the National Academy of Sciences--National Research Council (NAS-NRC) to consider the possibilities for disposal of high-level radioactive wastes in geologic formations within the United States.[1] This committee proposed storage in natural salt formations as the most promising method for the near future. Three subsequent NAS-NRC committees have endorsed this recommendation following reviews of on-going research and development work.[2-4] Laboratory and field studies of the feasibility of using salt formations for this purpose were conducted by ORNL during 1957-1962,[5] and during 1963-1967 we performed extensive studies with high levels of radioactivity in the Carey Salt Company mine in Lyons, Kansas.[6] These studies, designated Project Salt Vault, yielded highly satisfactory results, and after a further estimate of the feasibility and near-term need for a repository to meet the nation's requirements,[7] we undertook preliminary conceptual design studies in 1969. In 1970, Kaiser Engineers was selected as the architect-engineers for this project.

3.1 Advantages of Salt Formations

Salt formations were first recommended for use in the disposal of radioactive wastes by the National Academy of Sciences. This recommendation was based on a number of characteristics which make salt particularly attractive for this purpose. It is widespread and abundant, underlying about 500,000 square miles in portions of 24 states; it has good structural properties, with a compressive strength and radiation-shielding characteristics similar to concrete; it is relatively inexpensive to mine; its thermal properties are better than those of most other rock types; and it occurs generally in areas of low seismicity. Most importantly, bedded salt deposits are completely free of circulating ground waters and are isolated from underground aquifers by essentially impermeable shale. This situation tends to be preserved because any fractures which might develop are healed by plastic deformation and recrystallization of the salt.

3.2 Principal Alternatives to Salt

Other possibilities that have been studied or seriously considered either require excessive surveillance, are more costly, or are more limited in their applicability. "Perpetual" storage as liquids in tanks requires continuing surveillance, mechanical means for removal of decay heat for several centuries, and periodic replacement of the tanks. Disposal as liquids in bedrock caverns would be restricted only to those sites with the proven geological characteristics necessary to ensure long-term safety, and the relative scarcity of potential sites and the extensive explorations required to verify their suitability would place a serious limitation on the siting of reprocessing plants. Disposal as solids in surface vaults and granite caverns would cost several times more than would the use of salt formations, and would result in an undesirable proliferation of disposal sites. And finally, the practical application of such advanced techniques as transmutation or disposal into space is dependent on the development of sophisticated and expensive technologies in the distant future which can be justified only for reasons other than waste disposal.

4. PROJECT SALT VAULT

A field-scale experiment called Project Salt Vault[6] was carried out at a disused salt mine at Lyons, Kansas, to demonstrate the feasibility and safety of this waste-disposal concept, to demonstrate the equipment and techniques for handling packages of highly radioactive solids, and to secure data for the design of an actual disposal facility. The experimental facilities at the mine (Fig. 4.1) consisted of five rooms (three 30 ft wide and two 40 ft wide) off a 30-ft-wide corridor, all newly excavated at a level about 15 ft above the existing mine. The two end rooms contained arrays of seven holes, 5 ft apart and 12 ft deep. Fourteen irradiated fuel assemblies from ETR, two each in seven containers, were used to simulate actual solidified wastes. These were placed in one of the arrays, while electrical heat only was applied to the other as a control. In the middle two rooms, electrical heaters were used to raise the temperature of a large quantity of salt in the central pillar in order to obtain information on its in situ deformational properties. The fifth room provided access to the lower end of a 20-in.-diam shaft from the surface.

There were three basic steps in the operation of the experiment which ran from 1965 to 1967 (Fig. 4.2): (1) encapsulation of the fuel assemblies into canisters and loading of the canisters into the shipping cask, which were carried out in hot-cell facilities at NRTS; (2) shipment of the 30-ton cask and its forced-cooling equipment to the mine by truck; and (3) transfer of the containers into the mine. At the mine, the shipping cask was placed over the small shaft and the canisters were lowered one at a time into an underground transporter. This transporter was then maneuvered over one of the specially lined storage holes and the canister was lowered into it by remote control.

This experiment was successful in all respects. A total of about 4 million curies of fission-product activity, in 21 containers, was transferred both into and out of the mine without incident. The maximum personnel radiation exposure during any quarter was about 200 mrems to the head and hands. During the 19-month course of the experiment,

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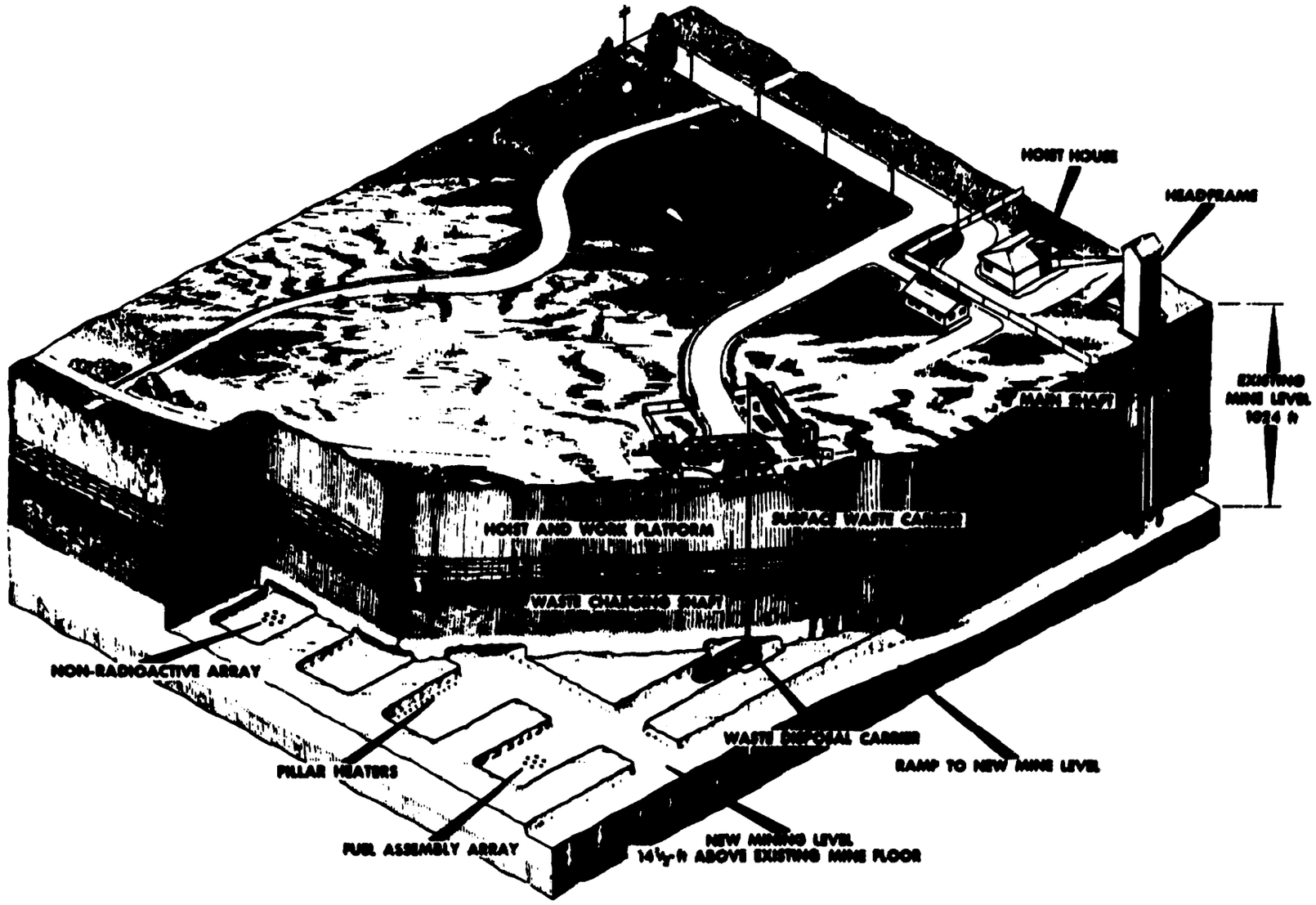

Fig. 4.1. Demonstration of Radioactive Solids Disposal in Salt.

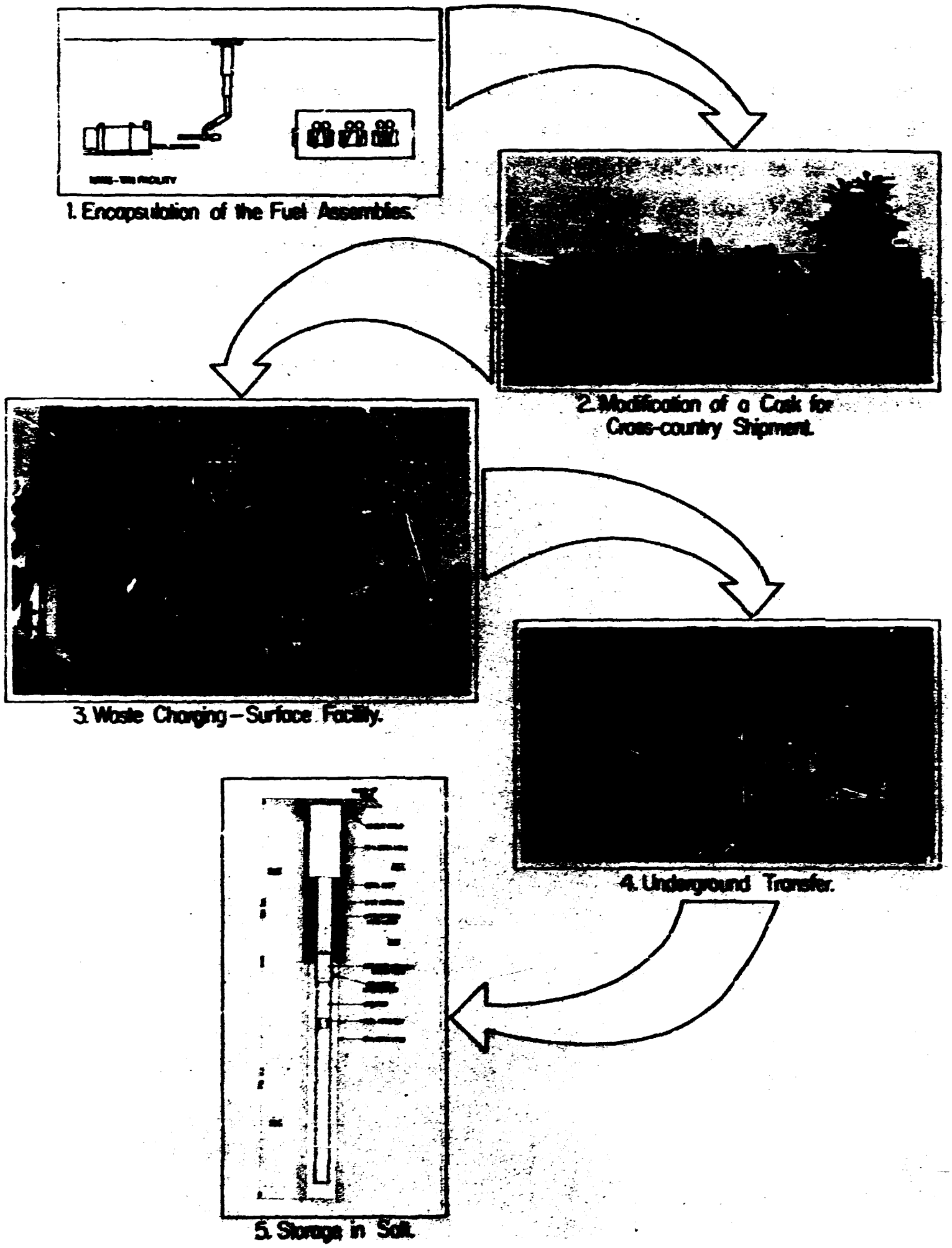

Fig. 4.2. Project Salt Vault - Major Design Phase.

the average dose delivered to the salt at the walls of the array holes was about 8×10^8 rads with a peak dose of 10^9 rads. In spite of these high doses, there were no measurable radiolytic or structural effects in the salt. In addition to the radiation, a considerable quantity of heat (both decay and electrical) was deposited in the salt. Around the arrays a circular ellipsoidal volume of salt of about 1400 ft^3 was heated to between 100°C and 200°C. Under the center (heated) pillar, a volume of salt of about 30,000 ft^3 was heated to temperatures greater than 60°C. These elevated temperatures produced a rapid acceleration in the rate of closure of the mine by modification of the plastic properties of the salt and also altered the pattern of deformations by the imposition of thermal stresses.

5. SITING CONSIDERATIONS

5.1 Site Selection for the First Repository

In selecting the site for the initial waste repository, certain geologic criteria were established consistent with considerations of long-term isolation and efficiency of operations. These requirements should not necessarily be considered applicable to the siting of future repositories. As more information and experience are gained from the operation of this first facility, some of these criteria will probably be relaxed and perhaps new ones added.

1. The salt formation should be of the bedded type, approximately horizontal, and relatively undisturbed structurally so that the previous research and development work for this disposal concept will be applicable.

2. The formation should have a considerable horizontal extent, amounting to at least several tens of miles to provide adequate isolation of the disposal area.

3. The formation should not be less than 500 ft deep and not less than 200 ft thick to provide for adequate heat dissipation as well as to provide a margin of isolation above and below the disposal plane.

4. The depth of the formation should not be greater than 2000 ft because of both the increased cost and the increased difficulty of operating at greater depths.

5. The formation should not be associated with, or be in the immediate vicinity of, potentially valuable reserves of petroleum or other mineral resources.

6. The site should be situated in a zone of tectonic stability.

Salt deposits are quite extensive in the continental United States, underlying approximately 500,000 square miles in portions of 24 states (see Fig. 5.1). The application of these site criteria automatically eliminated all but four areas from consideration. These areas were:

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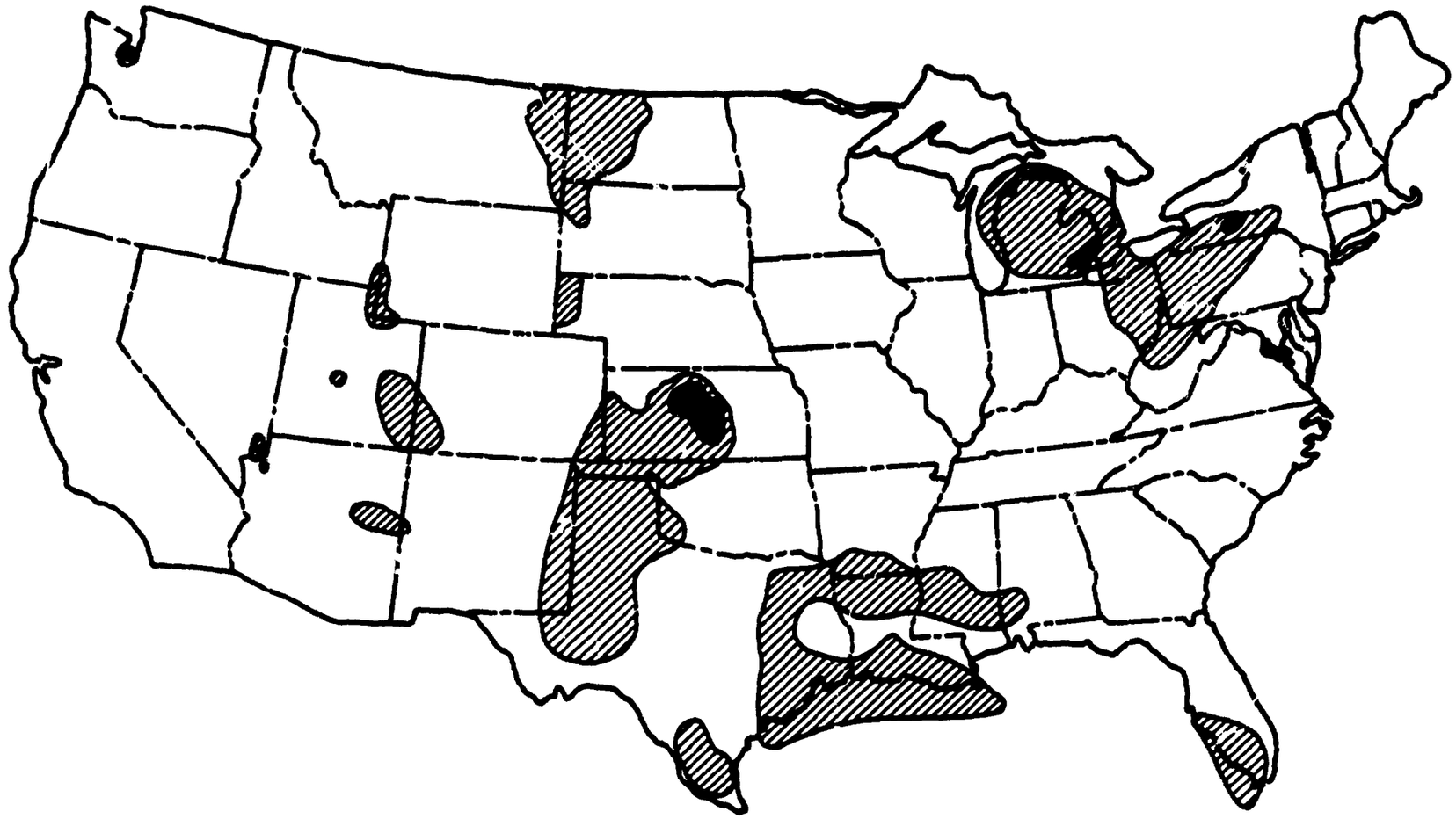

ROCK SALT DEPOSITS IN THE UNITED STATES (AFTER PIERCE AND RICH, U.S.G.S. BULL. 1148)

SALT DEPOSITS HAVING THICKNESS OF AT LEAST 200 ft AND LYING WITHIN 2000 ft OF LAND SURFACE.

Fig. 5.1. Rock Salt Deposits in the United States and Those Portions Suitable for the Initial Demonstration Repository.

(a) a 10,000-square-mile region of central Kansas; (b) a small (about 1000 square miles) area in west-central New York; (c) a similar area in southeastern Michigan, in general underlying metropolitan Detroit; and (d) a small area in northern Michigan. This last area was immediately excluded from further consideration because there have never been any salt mines in the area and (partially as a consequence of this) sufficient detailed geologic information for a rational analysis of the area was not available. The remaining three areas were examined and compared on the following basis:

A. Land Values and Population Centers. - Both the Kansas and New York areas were equally suitable from this standpoint, but the Michigan area was much less desirable, because it mostly underlies Detroit.

B. Tectonic Stability. - Both the Michigan and the Kansas areas were considered to have a low seismic risk. The New York area, however, had the disadvantage of its close proximity to the earthquake zone along the St. Lawrence Seaway.

C. Waste Transportation Costs. - A study indicated that, although operations in the Kansas area will experience a higher transportation cost (initially at least), this will be almost exactly compensated by lower mine operating costs resulting from a somewhat shallower mining depth.

Based on these considerations, the central Kansas area appeared to be somewhat more favorably situated than either of the other two areas. When considering specific sites within the general areas, there was one site which exhibited significant advantages over all others in any of the three areas. This was at, and adjacent to, the Lyons Mine of the Carey Salt Company at Lyons, Kansas. In the first place, this is the only site having a nonoperating mine to which access is immediately available and which has a mined-out volume sufficient for utilization for the disposal of alpha-contaminated wastes. Secondly, this mine was the site of the Project Salt Vault[6] experiment where a considerable quantity of information related to the design of an operating facility was obtained. All of those

data would be directly applicable to the design of a facility at this location, whereas, at any other location in Kansas, a certain amount of extrapolation and confirmatory investigations would be required. At the other two general areas, a much greater uncertainty would be introduced.

5.2 Salt "Domes" as Possible Future Sites

First, it must be emphasized that salt domes have always been considered as feasible in principle as bedded salt deposits for the disposal of radioactive wastes. The 1955 National Academy of Sciences Committee on Waste Disposal made no distinction between domes and bedded formations in their recommendation that salt deposits offered the best prospect for waste disposal. Since that time, all of our considerations on the general advantages of natural salt formations for this purpose have failed to reveal any fundamental reason for excluding the possibility of a repository in domal structures. However, there are several problems in selecting a specific dome and, more importantly, in demonstrating its suitability as a site for waste-disposal operations.

It was felt that these problems were sufficient to preclude domes from possible consideration as the site of the first repository. These problems are related to:

1. The geometry of the structure. - Since domes can be very irregularly shaped, it will be necessary to accurately determine the outline of the site being considered, primarily to ensure that the excavations will not too closely approach the edge, thereby producing an inflow of water. Also, the necessary width of the barrier between the edge of the dome and the limit of the underground workings has not been established.

2. The tectonics of the structure. - The salt domes of the Gulf Coast region apparently have resulted from the upward intrusion of massive volumes of salt from source beds buried at great depths. The velocity of emplacement is apparently no more than a few millimeters per year, and there is evidence suggesting that at least some of the domes are still moving at that rate.

Therefore, before a particular dome could be considered suitable for waste disposal, it would be necessary to demonstrate that it was not currently active, that rejuvenated movement (either natural or induced by repository excavation or the resulting heating due to radioactive decay) would be impossible, and that past movements had not resulted in residual stress concentrations which would interfere with the excavation operations and the programmed closure of the openings at the completion of waste-disposal operations.

3. The hydrological regime around the structure. - Because they are intruded through overlying sediments, domes are not protected from dissolution by circulating groundwaters by thick sequences of impermeable shales. In fact, the anhydrite "cap rock" immediately overlying most Gulf Coast domes is believed to be the insoluble residue left by the dissolution of as much as several thousand feet of the dome height. Site investigations would have to define in considerable detail the characteristics of the groundwater flow around the dome and demonstrate that future extensive dissolution of the salt would be impossible.

6. DESCRIPTION OF THE REPOSITORY

The Repository will utilize the existing 180-acre Carey mine and approximately 900 acres of the adjoining salt formation. In addition, we recommend that a buffer zone at least 1700 ft wide be established around the periphery of the site through the acquisition of subsurface mineral rights. While we expect it to have the capability of storing all of the alpha and high-level wastes to be generated in this country through the remainder of the century, it will be operated as a demonstration while we verify all aspects of our design and operational procedures. The schedule of wastes to be buried at the Repository and the projected waste inventories are given in Tables 6.1 and 6.2. The mine will be designed so that within 60 to 100 years the salt, by the processes of deformation and recrystallization, will have caused reconsolidation of the entire area, and the wastes will be completely isolated from the biological environment. The detailed conceptual design is currently being developed; so only a very preliminary and rather generalized description can be presented at this time.

The principal components of the Repository are (Fig. 6.1): (1) the existing salt mine, situated about 1000 ft below the surface, which will be used for storage of the alpha wastes; (2) a building in which the alpha waste packages are received, monitored, and transferred to the existing mined area by direct handling techniques; (3) a building equipped to receive containers of high-level wastes and transfer them remotely from their shipping casks to the mine level; (4) a newly mined area for burial of the high-level wastes; and (5) various support facilities such as operations and office buildings, a warehouse, and those other services and utilities that are required for mining and underground storage operations. All facilities will be designed to ensure that the necessary confinement potential and radiation safety are maintained following exposure to any credible internal and external forces (including tornadoes and earthquakes).

An 18-ft-diam, multipurpose "men and materials" shaft with three completely independent compartments will be used for lowering alpha

Table 6.1. Schedule of Alpha and High-Level Wastes to Be Buried at the Repository

Calendar Year	Volume of Alpha Waste[a] (10^6 ft^3)	High-Level Solidified Wastes	
		Number of Containers[b]	Area of Salt Used[c] (net acres)
1975	0.350	-	-
1976	0.350	88	0.2
1977	0.350	184	0.4
1978	0.350	329	0.8
1979	0.350	561	1.3
1980	0.350	819	1.9
1981	0.700	1240	2.9
1982	0.700	2220	4.0
1983	0.700	2720	5.2
1984	0.700	2880	6.7
1985	0.700	3660	8.4
1986	0.700	4580	10.5
1987	1.000	5530	12.7
1988	1.000	6470	14.8
1989	1.000	7490	17.0
1990	1.000	8600	19.5
1991	1.000	9690	21.9
1992	1.000	10,900	24.6
1993	1.000	12,000	27.2
1994	1.000	13,200	29.8
1995	1.000	14,600	32.7
1996	1.000	15,800	35.5
1997	1.000	17,200	38.6
1998	1.000	18,400	41.4
1999	1.000	19,800	44.4
2000	-	21,300	47.7
Total	19.300	199,261	450.

[a]Assumes waste has been compacted by factors of 3 to 10.

[b]Containers are 6 in. in diameter and contain 1.6 ft^3 of solidified waste each. Half of the waste is aged 5 years and half is aged 10 years at the time of its receipt at the Repository.

[c]"Net" acres can be converted to "gross" acres by dividing by 0.5.

Table 6.2. Projected Waste Inventories at the Repository

	Calendar Year		
	1980	1990	2000
Number of annual shipments			
High-level waste[a]	23	240	590
Alpha waste[b]	420	1200	0
Accumulated high-level waste[c]			
Volume of waste, ft^3	3170	74,200	319,000
Salt area used, acres	9	200	900
Total thermal power, megawatts	1.17	24.4	94.9
Total activity, megacuries	329	7030	27,700
^{90}Sr, megacuries	59.0	1310	5290
^{137}Cs, megacuries	83.1	1850	7500
^{238}Pu, megacuries	0.102	2.34	9.88
^{239}Pu, megacuries	0.00157	0.0368	0.158
^{240}Pu, megacuries	0.00400	0.101	0.470
^{241}Am, megacuries	0.151	3.54	15.3
^{244}Cm, megacuries	1.58	34.1	133.3
Accumulated alpha waste[c,d]			
Volume of waste, 10^6 ft^3	2.1	10.3	19.3
Salt area used, acres	20	96	180
Total thermal power, megawatts	0.0142	0.170	0.476
Total activity, megacuries	14.2	151	300
Total mass of actinides, metric tons	1.40	15.8	38.3
^{238}Pu, megacuries	0.232	2.57	6.02
^{239}Pu, megacuries	0.0515	0.580	1.41
^{240}Pu, megacuries	0.0741	0.834	2.02
^{241}Pu, megacuries	13.8	146	286
^{241}Am, megacuries	0.0617	1.03	4.74

[a]Each shipment consists of 57.6 ft^3 of waste in thirty-six 6-in.-diam cylinders. Each ft^3 of waste represents 10,000 MWd (thermal) of reactor operation. Half of the waste is aged 5 years and half is aged 10 years at the time of its shipment. Last shipments are assumed to be made in the year 2000.

[b]Shipments are made in ATMX railcars; each shipment contains 832 ft^3 of waste. Last shipments are assumed to be made in the year 1999.

[c]At end of year.

[d]The isotopic composition of Pu at the time of its receipt is 1% ^{238}Pu, 60% ^{239}Pu, 24% ^{240}Pu, 11% ^{241}Pu, 4% ^{242}Pu.

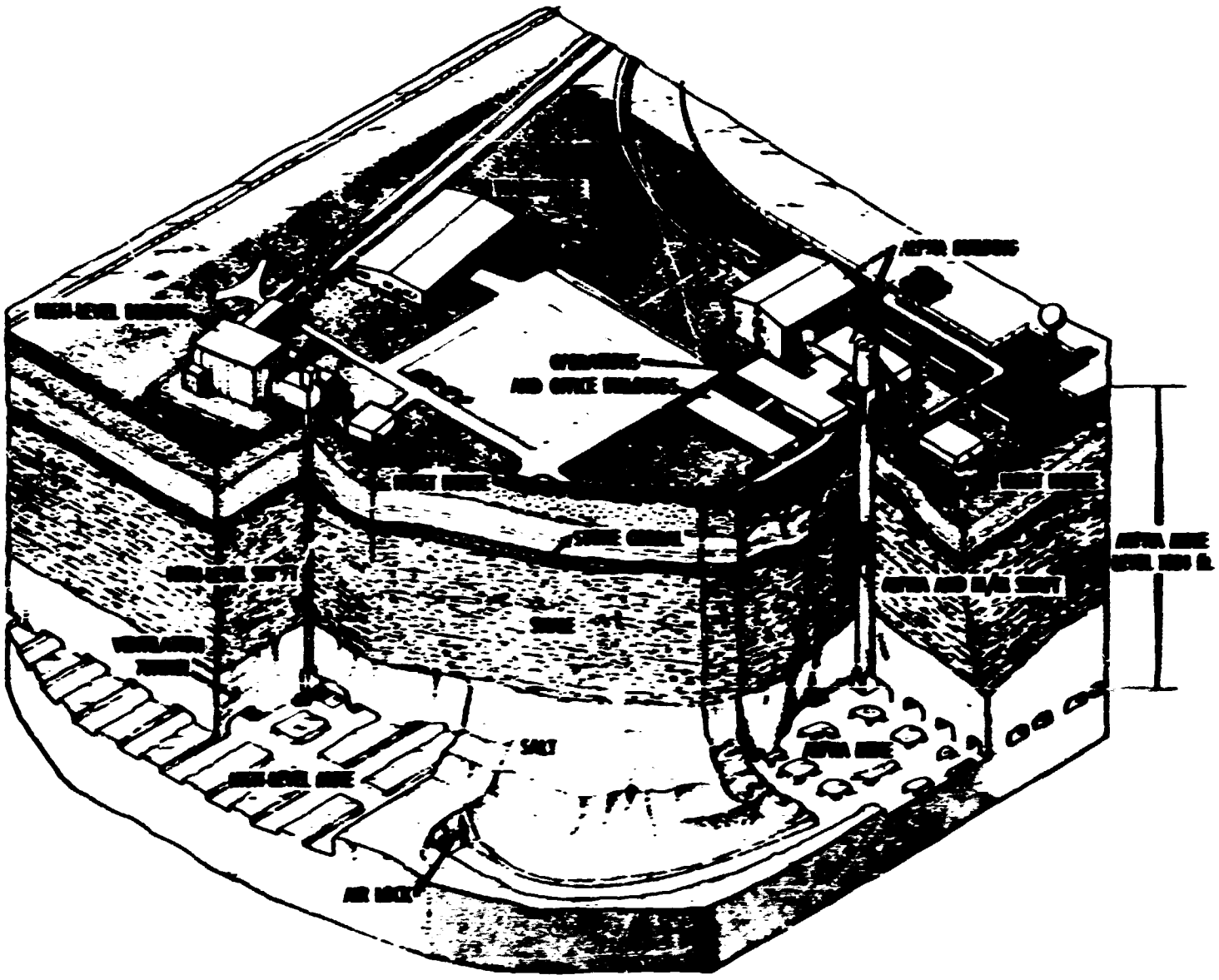

Fig. 6.1. Artist's Conception of the Demonstration Repository.

wastes into the mine; for transferring personnel and materials; and for admitting ventilation air into the mine. The alpha waste receiving building will provide earthquake- and tornado-resistant facilities in which waste packages can be removed from returnable cargo containers, assayed, recanned and repaired as necessary, and lowered into the mine. We expect that the alpha waste will be shipped to the Repository in gasketed metal drums and in metal crates with outside dimensions no greater than 4 ft x 6 ft x 6 ft high, and that individual burial units will weigh no more than 5 tons. Initially, shipments of alpha waste will be made only by rail in specially designed ATMX railcars, but it is probable that provisions will be made later to receive motor freight also.

The high-level waste receiving building will be located about 500 ft from the alpha waste building. Solidified high-level wastes will be sealed in steel containers and transferred by rail to the Repository in shipping casks weighing up to 100 tons. The containers will be unloaded here and lowered individually into the mine through a 4-ft-diam shaft. The individual waste containers may vary from 6 to 14 in. in diameter and from 2 to 10 ft in length, may weigh up to 4000 lb, and may possess up to 5000 W of thermal power from radioactive decay.

The presently mined-out area will provide space for storing about 20 million cubic feet of alpha wastes, and the alpha facility will have the capability to store 1 million cubic feet of waste per year. The alpha waste transfer shaft will terminate in an alpha-confinement room at mine level. Here, the packages of waste will be unloaded from the hoisting cage by a forklift truck and transported to the disposal rooms where they will be unloaded and stacked. When an area or room has been completely filled, it will be backfilled with salt and sealed with masonry stoppings. The mine will be ventilated with 20,000 cfm of air supplied through the 18-ft-diam multipurpose shaft, directed through the mine with brattices, and combined with the exhaust air from the high-level mine.

The high-level mine will be developed over the life of the Repository in a northeasterly direction from the alpha mine using conventional salt-mining techniques. The facility will ultimately have the capability of handling up to 300 containers of waste per week. The required rate of

mining will increase during the life of the facility from about 200,000 tons per year initially to about 1.3 million tons per year during the last several years of operation. About 50% of the salt will be used for backfilling; eventually, the remainder will be raised to the surface and offered for sale.

The containers of high-level waste will be transported to rooms mined in the salt formation and buried in holes in the salt. The spacing between containers will be such that the ralioactive decay heat will be dissipated without causing excessive temperatures in the salt or in the neighboring geological formations. After the burial area of a room has been utilized, the room will be backfilled with crushed salt and sealed from the remainder of the mine.

Ventilation air leaving the high-level mine will be drawn through ventilation tunnels above the corridors, combined with the air from the alpha mine, and exhausted to the atmosphere through the mine air exhaust shaft, filter station, and stack. The mine exhaust filter station will consist of two banks of roughing filters in series to remove most of the salt particles, followed by two banks of HEPA filters in series to remove practically all of the remaining particles.

Assuming FY-1972 funding, we expect to finish the conceptual design and preliminary safety analyses by the end of FY-1971; the Title I design of the alpha facility will begin in October 1971. Title II and some early mine cleanup and shaft work should begin in April 1972, and construction of the alpha facility will begin in May 1973. It should be ready to begin operation as a demonstration facility early in 1975. The high-level facility will proceed apace, with construction starting in 1974 and operations, again on a demonstrational basis, beginning early in 1976.

7. CONSIDERATIONS AFFECTING LONG-TERM SAFETY AND OPERABILITY OF THE REPOSITORY

A number of considerations have been of paramount importance throughout the period of establishing the technical feasibility and long-term safety of the demonstrational repository. These considerations have included the geology and hydrology of the site, thermal effects, radiation effects, radiological and physiological safety, ecology, environmental monitoring, and concepts for retrieval of waste. These topics will be discussed in the following sections.

7.1 Geology and Hydrology

7.1.1 Description of the Site

Regional Setting. - Physiographically, the proposed repository site at Lyons, as well as a large part of Kansas, lies within the Great Plains Province of the United States. This broad belt of highlands, which extends from Canada to Mexico, slopes gradually eastward from the Rocky Mountains to the Central Lowland. Within the state a relatively thin and single succession of sedimentary rocks cover the greater than 600-million-year-old (Precambrian) basement complex. In general, the sedimentary rocks, which range in age from about 600 million years (Cambrian) to about 1 million years (Quaternary), all dip gently toward the west (Fig. 7.1). This slight westward dip of the rocks, coupled with the gradual change in elevation from east to west, has resulted in the general succession of the surface outcropping of the oldest rock in the east to progressively younger ones in the western part of the state. Even though Kansas and the stable interior of the continent have not been subjected to diastrophic movements for the last 600 million years (since Precambrian time), some areas have been structurally positive (rising) in the geologic past while others have been structurally negative (subsiding). The Lyons site is situated on the eastern flank of one of these ancient structural features, known as the Central Kansas Uplift. This major subsurface feature is presently inactive, and there is good evidence to indicate that it has remained inactive for the past 200 million years.

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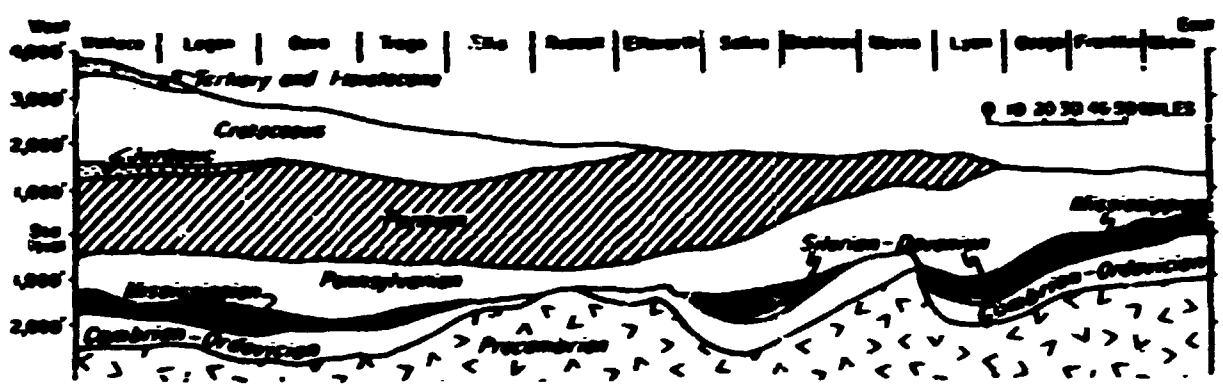

Fig. 7.1. Generalized Geologic Map and Cross Section of Kansas Rocks (after State Geological Survey of Kansas Special Distribution Publication 5, Lawrence, Kansas, December 1963).

Stratigraphy. - The rocks at the site are all of sedimentary origin and, for the most part, are flat lying. In general, the stratigraphic sequence begins at the surface with about 50 ft of unconsolidated silty terrace deposits that are principally about 10,000 to 1,000,000 years old (Pleistocene) (Fig. 7.2). Cretaceous-age shales that contain interbedded sandstones about 60 to 135 million years old comprise the section from about 50 to 120 ft below the surface, while Permian-age rocks (230 to 280 million years old) lie unconformably beneath the Cretaceous sediments. For our purposes, the most important aspect of these rocks is the presence of a thick accumulation of rock salt. The salt beds are nearly 300 ft in thickness, extending from a depth of about 800 to 1100 ft in the Lyons-Hutchinson area, and are capped with several hundred feet of red-colored shale and siltstones. Thick beds of anhydrite (i.e., gypsum - the basic ingredient of plaster of paris) lie below the salt, while the remainder of the Permian rocks consists mostly of alternating layers of shale and limestone. Pennsylvanian rocks (280 to 310 million years old) lie unconformably beneath the Permian formations and extend to a depth of about 3200 ft. This thick sequence of mostly alternating beds of shale and limestone with a few sandy members contains many oil and gas horizons, especially below about 2700 ft. Mississippian and Devonian rocks (greater than 310 million years old) are represented in the columnar section only by a relatively thin section of shale. Silurian and Devonian rocks are absent in the columnar section beneath the Repository. Ordovician-age rocks (425 to 500 million years old) at the site are predominantly shale and limestone. The "Arbuckle" group of rocks lies unconformably between the overlying Ordovician rocks and the Precambrian basement. This thick sequence of predominantly dolomitic rocks contains important oil and gas reservoirs in the general vicinity of Lyons as well as throughout central Kansas.

Structure and Tectonics. - The rocks beneath the proposed repository site have been subjected to a long and complex history of deformation with the older formations showing, as would be expected, a more complicated regime than successively younger beds. Locally, movements along the Central Kansas Uplift have been instrumental in fabricating the

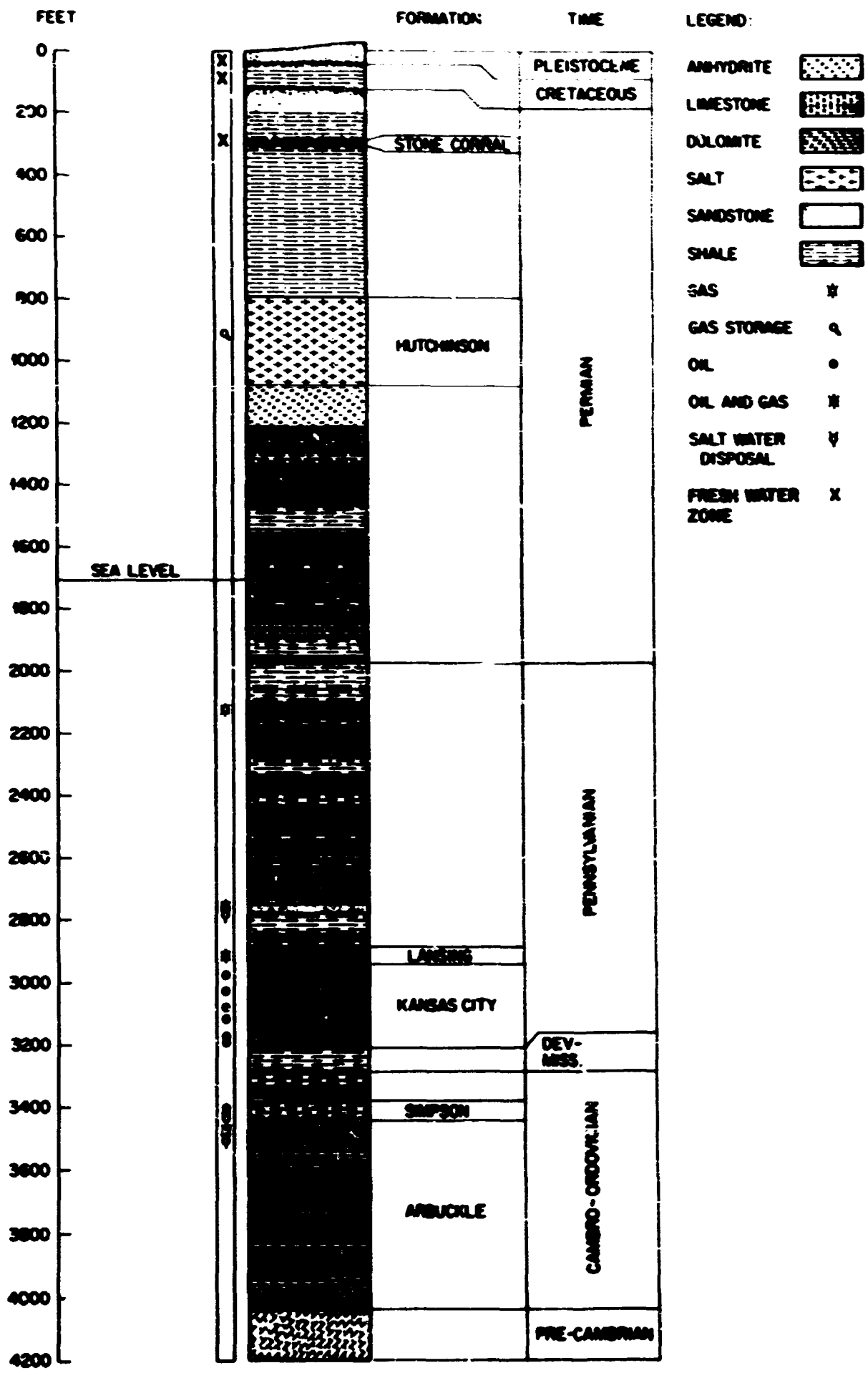

Fig. 7.2. Generalized Geologic Section, Rice County, Kansas (after Type Log for Rice County, Catalog No. 20-19-8w, Published by Kansas Geological Society).

existing structure, while the inclinations of the rocks at the site have also been affected by broad regional warping of the earth's crust. The repository site lies on the eastern flank of the Central Kansas Uplift, which extends beneath a large part of that portion of the state.

Figure 7.3 is a map of Kansas that shows the location of historic earthquakes and their relationship to the major structural features in Kansas. Note the almost complete absence of earthquakes along the Central Kansas Uplift. This structure had its major development at the end of Mississippian time some 300 million years ago and, as evidenced by the dearth of earthquake activity, is now inactive. The earthquakes that have occurred in the central Kansas area range up to a Modified-Mercalli Intensity Scale rating of VI (damage small). These shocks, as well as others in Kansas, have occurred at rather shallow depth (16 to 38 miles) and probably represent minor adjustments in the underlying granitic crustal rocks. Epicentral locations of many of the earthquakes, including the two most damaging, were centered along the Nemaha anticline in eastern Kansas about 100 miles from the Lyons site. Based on the recent seismic history of all of the other structures in Kansas, it is likely that they are tectonically inactive.

Rock Salt Deposits. - The rock salt column beneath the proposed repository consists generally of a succession of layers of rock salt, which accounts for about 75% of the section, intercalated with shales and occasionally anhydrite beds. The principal salt zones can be traced laterally from the northeast corner to the southwest corner of the site, although some irregularities in thickness and in lithology occur. In general, lateral consistency in the column is greatest in the lower to middle part of the column and least at the extreme bottom and upper parts of the formations. The thickest and most persistent salt zone lies at a depth of about 1000 ft below the surface. A 9-ft-thick bed in the lower part of the zone comprises the mined unit at The Carey Salt Company's workings at the site. The bed of salt is unquestionably the most uniform and the best quality of the entire sequence and is, therefore, the desired host bed for the waste. The proposed mined area will lie at a level of about 20 ft above the existing mine floor in the old workings

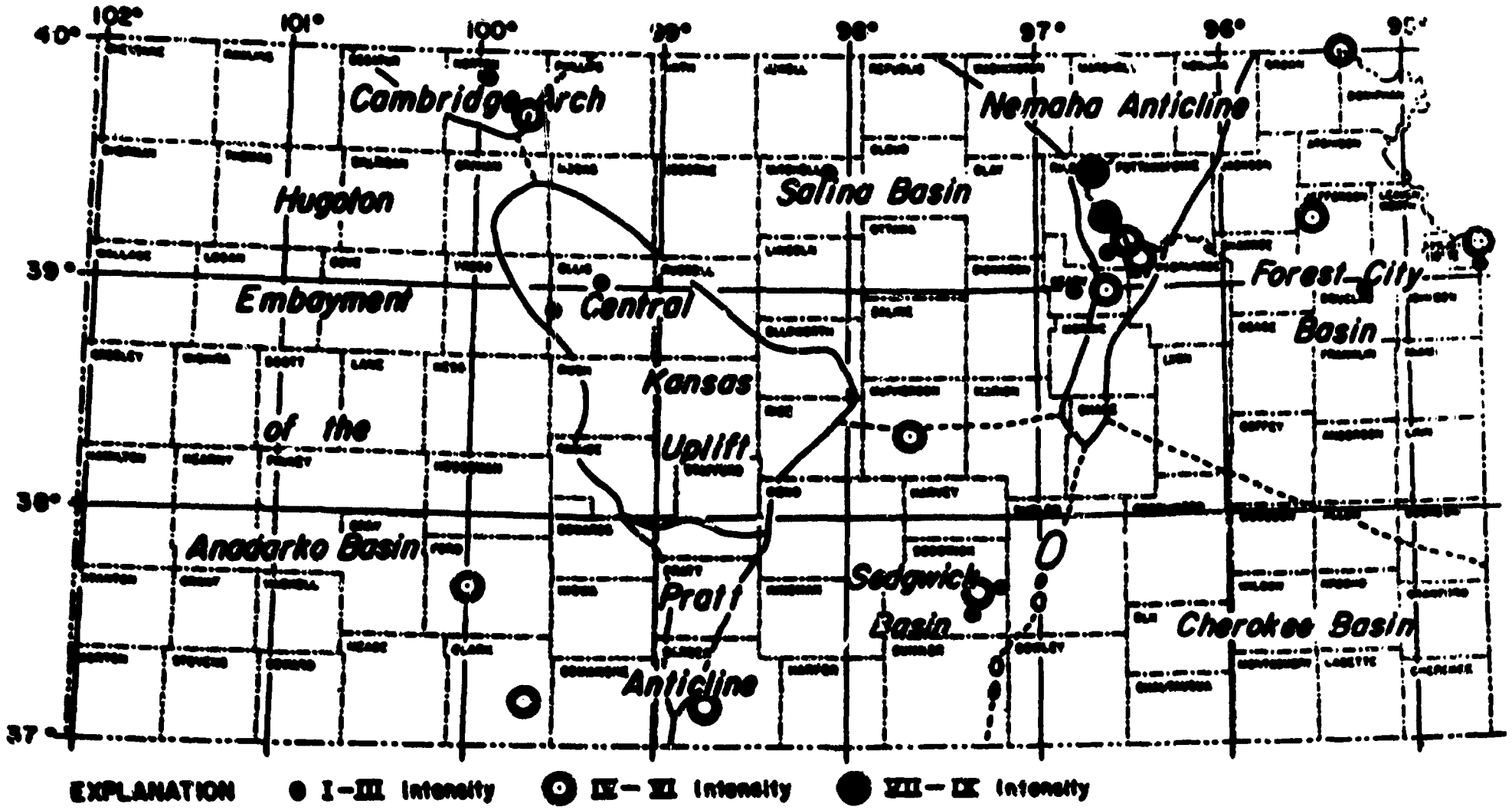

Fig. 7.3. Map of Kansas Showing Major Structural Features and Location of Earthquake Epicenters (after Merriam, D. F., 1956, History of Earthquakes in Kansas: Seismological Society of America Bulletin, Vol. 46, No. 2, pages 87-96).

and extend upward a distance of about 15 ft. The floor of the newly mined area will be rock salt, while a thin shale parting will mark the roof. The salt in the proposed excavation area contains numerous blebs of polyhalite that are commonly concentrated along bedding and give the salt a banded appearance. Due to the structural attitude of the rocks at the site, a constant rise in the mining level will be experienced from the southwest corner of the site to the southeast corner, while there will be essentially no change in the mining level along the western boundary of the site. From the southwest corner to the northeast, the mining level will rise constantly to the apex of a northwestwardly plunging anticlinal fold and then show a gradual decrease.

Hydrology. - The proposed repository site is drained by Owl Creek and an unnamed tributary of Cow Creek (Fig. 7.4). These intermittently flowing streams originate a few miles to the north of the site and flow southwestwardly through it before joining Cow Creek a few miles to the south. These streams would be expected to flow and thus carry runoff from the site only during periods of heavy rainfall, which occur most frequently during the late spring and early summer months. Perhaps the most important source beds for groundwater supplies in Rice County are the thick deposits of gravel and coarse sand fill in the Arkansas River Valley and Cow Creek a few miles south of the site. Within the immediate vicinity of the site, fresh water is confined essentially to the uppermost 300 ft of the rock column. The Stone Corral, which lies between about 270 and 285 ft, is the deepest known freshwater aquifer at the site (Fig. 7.2). It may yield water at the rate of about 1 to 3 gpm. The Cretaceous-age sandstones (45 to 100 ft below the surface) are capable of yielding water at the rate of about 5 gpm. The average depth to groundwater in the blanket of terrace deposits at the site is about 25 ft.

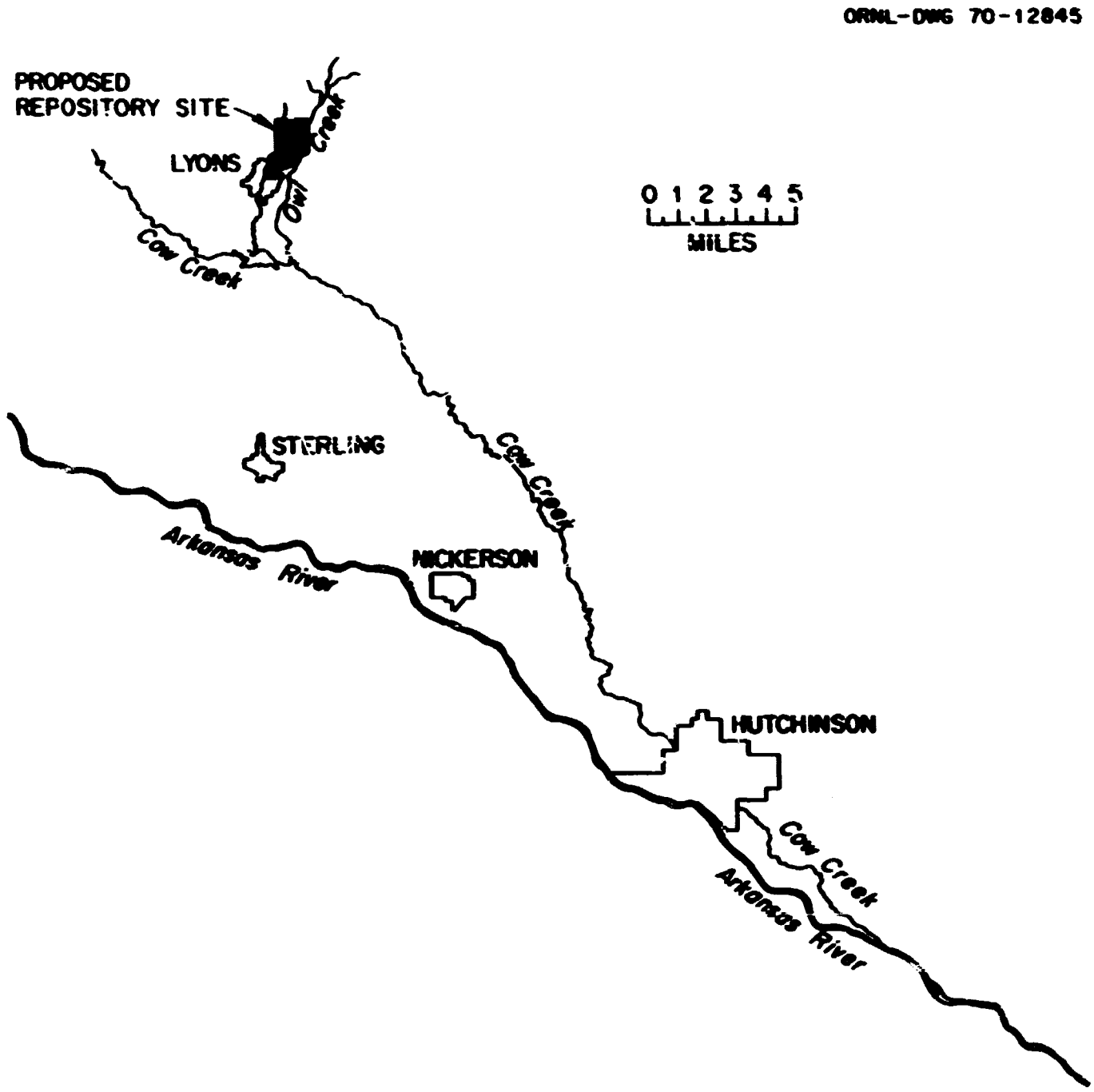

Fig. 7.4. Surface Drainage at Proposed Waste Repository.

7.1.2 Long-term Geological and Hydrological Considerations

Boundary Solutioning of Salt Formation. - Along most of its eastern boundary (about 25 miles east of Lyons), the salt deposit ends abruptly and a series of subsidence ponds and salt-water springs overlie the present stratigraphic position of the salt member; this suggests that its original limits extended somewhat farther east. In northwest Saline County, precise dating of subsidence features obtained from river channel migration studies indicates that the salt front has retreated a distance of 5 miles or less in the last 1 million years (or throughout the Pleistocene), while, in McPherson and Harvey Counties, westward migration of the salt has been 13 and 12 miles, respectively, from Delmore (Pliocene) to Kansas (a time span of about 5 million years). According to Fent,* the natural leaching of the salt body by circulating groundwater depends on the development of secondary permeability in the overlying shales, which, in turn, is brought about by erosion when the protective Cretaceous shales and clays are removed. In areas where the overlying shale section is complete, which includes the Lyons site and essentially all of the central Kansas salt area west of the eastern boundary of Rice County, it may be considered to be protected for geologic time periods of at least 1 million years or for about the time span of the Pleistocene.

Additional work using different techniques of dating the westward migration of the salt boundary and an evaluation of the possibility of drastically altered solutioning rates due to man's activities will be undertaken.

Penetration Solutioning of Salt Formation. - In the central Kansas area, a large number (probably thousands) of wells have been drilled through the salt formation in search of oil and gas. Some of these wells date from the last century. In four known cases, a situation has developed where fresh water from near-surface aquifers has drained into one of these wells, circulated through a break in the well casing at or near the depth of the salt bed, then discharged down the well into a lower formation, probably the Arbuckle. This circulating water dissolves the salt around the well, eventually causing the collapse, either sudden

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*O. S. Fent, Salina, Kansas, unpublished report to Oak Ridge National Laboratory, July 1970.

or gradual, of the overlying beds and the development of a "sink" on the surface. These four known cases of penetration solutioning have in common: (1) a shallow aquifer yielding water at the rate of several hundred gallons per minute; (2) an oil well having a history which suggests the possibility of a casing break in the salt zone; and (3) a highly permeable deeper zone having a low hydrostatic head. Three of these four wells had been used as salt-water disposal wells following their productive life.

Ten wells have been drilled on or within 660 ft of the Repository site. (The effective limit of solutioning appears to be about 400 ft.) Although these wells must be considered possible sites for solutioning, their potential does not appear to be very great because: (1) only one has been a producing well, and none of them has ever been used for salt-water disposal; (2) at least some of these wells were cased and plugged in such a manner as to meet the much higher than usual practice requirements of the holder of the salt rights; and, more importantly, (3) the combined yields from the aquifers at the site would be, at the most, only a few tens of gallons per minute.

A number of steps will be taken to preclude the possibility of solutioning by this mechanism at the site: (1) a survey at and around the site will be made using aerial remote sensing equipment and ground-based geophysical techniques in order to assure that all wells have been located; (2) each of these wells will be cleaned out and replugged using multiple layers of sulfate-resistant cement and sand-clay mixtures developed for this purpose; and (3) a portion of the long-term geological monitoring program will be designed to give early indication of possible solutioning at any place on or adjacent to the site. It should also be pointed out that any previous solutioning will be detected when the wells are opened and reentered.

Preliminary information from the Kansas Geological Survey, obtained while doing hydrological mapping of the Lyons area for the project, suggests the possibility of a circulation cell of a type that could possibly lead to eventual solutioning and development of a "sink" at the southeast corner of the proposed site. These data will be discussed

in detail in their report of this work, which we have not yet received. As we understand it, a preliminary report on the inference of the ground-water-level anomaly is based on information that is preliminary and incomplete. However, at the moment, the data must be considered valid, and the situation most definitely requires thorough investigation and complete definition. Plans are currently in progress to extend and confirm the hydrologic data as soon as funds become available (FY-1972). If this testing confirms the presence of a circulating system, the suspect well will be reentered and examined for solutioning of the salt. Depending on the results of this examination, appropriate corrective action will be taken. That action may be simply a specialized replugging of the well, or pressure grouting of the solution cavity.

Erosion and Denudation. - Since a portion of the radioactive wastes to be stored at the proposed repository will remain active for periods of hundreds of thousands of years, estimates are being made of the expected erosion by stream action and glaciation at the site and its effect on the stability of the underlying salt formation for long periods of geologic time. The present rate of denudation for the Arkansas River basin, which includes the Lyons site, has been determined to be about 1.4 in./1000 years. This value, which is a measure only of the uniform lowering of the land surface of the entire basin, suggests that perhaps a hundred feet or so of rock cover would be removed from the formations at the site during the effective lifetime of the waste. Another estimate of the expected erosion can be made from measured stream incisions in the bedrock in the vicinity of the site during the Pleistocene epoch (or the last 1 million years of geologic time). With a recorded channel incision depth of 200 ft and a projected incision having a similar depth during the next 1 million years of geologic time, it can be concluded that a 200-ft-deep channel could be cut into the present unincised bedrock at the site and/or an additional 200-ft deepening of the present gravel-filled channels in the immediate vicinity of the site could occur. In either case the salt formation would not be breached, as it lies some 800 ft below the present land surface.

To determine more precisely the nature and extent of erosion in the area of central Kansas during the next 1 million years, a study has been initiated whereby estimates of future erosion will be obtained by statistical treatment and extrapolation of modern rates of erosion in several climatic zones that would be representative of the kinds of climate that have existed in the midcontinent during the last million years and would be expected to prevail during the next equivalent period.

In order to further validate these studies, an investigation will be undertaken of the possibility of drastic changes in the drainage pattern of the area, including the Arkansas River, and their consequent effect on the predicted erosion rates at the site.

Diapirism. - Under certain conditions of stress and temperature, and after sufficient time, salt will deform plastically. In many respects, this deformation is similar to, but much slower than, the sag and droop of an apparently rigid candle left standing undisturbed in its candlestick for a long time. In general, evidence of this plastic deformation can be found in all salt deposits buried to any appreciable depth in the geologic past. Of course, the extent of the deformation varies greatly, ranging from slight postdepositional thickening and thinning, such as may have occurred in places in the central Kansas deposit (the point is still being debated by geologists), to the formation of salt domes rising through tens of thousands of feet of overlying rocks from deep-lying "mother" salt beds, as in the Gulf Coast Embayment.

Based on an examination of the conditions under which salt domes have formed at other locations throughout the world, it is clear that the rock salt in central Kansas is much too shallow, too thin, and located in a region which is too tectonically stable to permit large-scale deformations of this type. Certainly, the increase in temperature of the salt due to the decay heat generated in the waste will increase the plasticity of the salt within the confines of the Repository, but this very limited area affected (as compared to that required to produce a geologic effect) and the geologically short heating period (a few thousand years) will limit the ensuing deformation to a small zone around the mined openings.

7.2 Thermal Considerations

Thermal calculations are necessary in order to be able to establish the allowable heat generation per unit area of the high-level mine as a function of waste and container characteristics. The greater the allowable heat generation rate, the more economical the operation, since a large portion of the waste disposal cost is dependent upon the mine space requirements. By comparison, heat generation rates in the alpha mine will be extremely low; alpha wastes can be packed tightly in the mine with essentially negligible effect on the thermal environment.

Factors which influence the allowable heat generation rate per unit area of mine include: (1) the maximum temperature permissible at the center line of the waste container; (2) the release of relatively small quantities of brine contained in small cavities in the salt at temperatures above 250°C; (3) temperature and thermal-stress transfer effects on mine stability during operation; (4) possible instability in the whole formation, including such effects as breach of the containment afforded by the overlying strata; (5) temperature rise in freshwater aquifers; (6) surface heating; and (7) temperatures beyond the boundaries of the property and mineral rights belonging to the facility.

The first of the above factors has generally been found not to govern the average heat generation rate per unit area for the waste types anticipated, although it must be considered in terms of the maximum allowable heat generation per waste container. As a consequence of laboratory and field tests, the second and third factors have resulted in the establishment of two temperature criteria for the salt in the floor of the mine rooms: (1) the maximum salt temperature midway between two containers in the center of a room should not exceed 200°C; and (2) the maximum temperature 8 in. from the wall of the hottest container should not exceed 250°C. The fourth through seventh factors also appear to be satisfied by the 200 and 250°C mine-floor-temperature criteria for the geologic conditions which exist at the proposed site in Lyons, Kansas, as will be discussed later in this section.

Up to the present time, the thermal calculations have been done using literature values for the thermal properties of rocks other than salt, and an assumed value for geothermal flux. The approximate stratigraphy as determined from the original mine shaft log, the electric logs of the waste charging shaft, and a few feet of core into the roof and floor of the existing mine was also used. Since these approximations had to be made, it was felt that the preliminary calculations could be made using simplified approximations of the stratigraphy. The nature of the approximations was varied, depending on the particular effect which was being investigated in a particular calculation. Also, the mine geometry was approximated in similar fashion. Generally, most approximations and simplifications were made in a manner that would tend to produce results which were overestimates of the actual temperatures. For example, usually the waste heat generation characteristics were chosen to represent those which would produce maximum long-term temperature rises in the whole formation.

All of the calculations to date are limited primarily by the lack of actual data from the site. Some of these data are currently being developed by the USGS and various consultants to ORNL. When the stratigraphic, thermal property, and geothermal flux information becomes available, the previous calculations will be repeated using the appropriate values, together with detailed simulation of the thermally important features of the mine and the geologic formation. The necessary studies appear to be within the capabilities of existing computers but, for maximum efficiency, will require some modification of existing three-dimensional, transient computer programs. It should be noted that the results using the new data are not expected to be greatly different from those obtained to date and reported in the following paragraphs.

With the new physical property data and the knowledge of the temperature-time relationships from the planned thermal studies, new estimates will be made of rock movements due to subsidence, thermal expansion, and any other factors which are applicable. If it appears that rock temperatures anywhere in the formation will be such as to pose a potential

threat to the containment of the waste within the salt formation, the maximum allowable heat generation rates at mine level will be adjusted to an acceptable value. This will also be the case if calculated temperatures in aquifers or beyond the property boundaries are found to be excessive.

7.2.1 Brine Migration

Brine in small cavities will be released by salt decrepitation at temperatures above 250°C, but field tests indicate no hazard if small masses of salt around the containers in the floor reach 300°C or greater. Long-term, large-mass temperatures this high might be undesirable from the standpoint of the larger quantities of water released.

At temperatures below the decrepitation point these small brine-filled cavities (size range from a few millimeters down to microscopic, and representing about 1/2% of the salt volume) migrate toward a heat source. The mechanism for cavity migration is the diffusion of sodium and chlorine ions from the hot to the cold side of the cavity due to the slightly greater solubility of NaCl at the higher temperatures. Migration rates are directly proportional to temperature gradient and increase with increasing temperatures. Calculations indicate that in the high-level facility the maximum inflow of brine per disposal container hole would be about 2 to 10 liters, which would cease approximately 20 to 30 years after burial. About 60% of the total brine inflow would be expected during the first 10 years after burial. In a 50- x 300-ft room there could be about 200 container holes; so the total integrated inflow of water (brine) (if it were all to find its way into the room) would be less than 2000 liters During the period of operation, the daily ventilating air volume in the waste-disposal area will contain more water than this at its normal relative humidity. Other than its effect on radiolysis container corrosion (Sect. 7.3.1), this moisture should not be of any significance.

7.2.2 Temperature Effects on Stability

Elevated temperatures increase the flow rates of salt. (On 4-to-1 width-to-height pillars, flow rates increase with about the tenth power of absolute temperature.)

Figure 7.5 shows calculated temperatures in and around a disposal room at the time the peak temperature is reached (assuming a 10-year-old waste). Notice the large area beneath the room and in the pillar which is above 170°C. Figure 7.6 shows the temperature rise at various points as a function of time after burial of the waste. Notice that temperatures at all points are above 100°C by the end of 10 years. From these figures, it is apparent that a maximum salt temperature of 200°C is not unduly conservative from the standpoint of stability during operation. Yet, it should be adequate to achieve the desired room closure and recompaction of the backfilled salt. Also, salt starts to behave quite plastically at temperatures around 300°C. If a large area of salt were to approach 300°C while operations were still under way in a mine, that area could possibly fail to support its overburden load, thus transferring excessive loads to the current operating area.

7.2.3 Stress Transfer Effects

In Project Salt Vault, thermal stresses produced by sources in the floor were found to transfer essentially instantaneously around the room and, in this respect, to be more important in determining localized stability conditions during the initial stages of operation than the change in creep properties of salt due to rising temperatures. In the Project Salt Vault array rooms the most noticeable and significant effect was a sudden fivefold increase in the rate of sag of the 2-ft-thick salt layer in the roof of the experimental area.

Consideration of all of the effects in both the array rooms and the heated pillar rooms tends to indicate that the effects are roughly proportional to the heat input rate per unit of floor area. The rates in the array rooms were initially about 20 Btu/hr·ft^2, and in the heated pillar rooms about 23-1/2 Btu/hr·ft^2. For 10-year-old waste, the heat

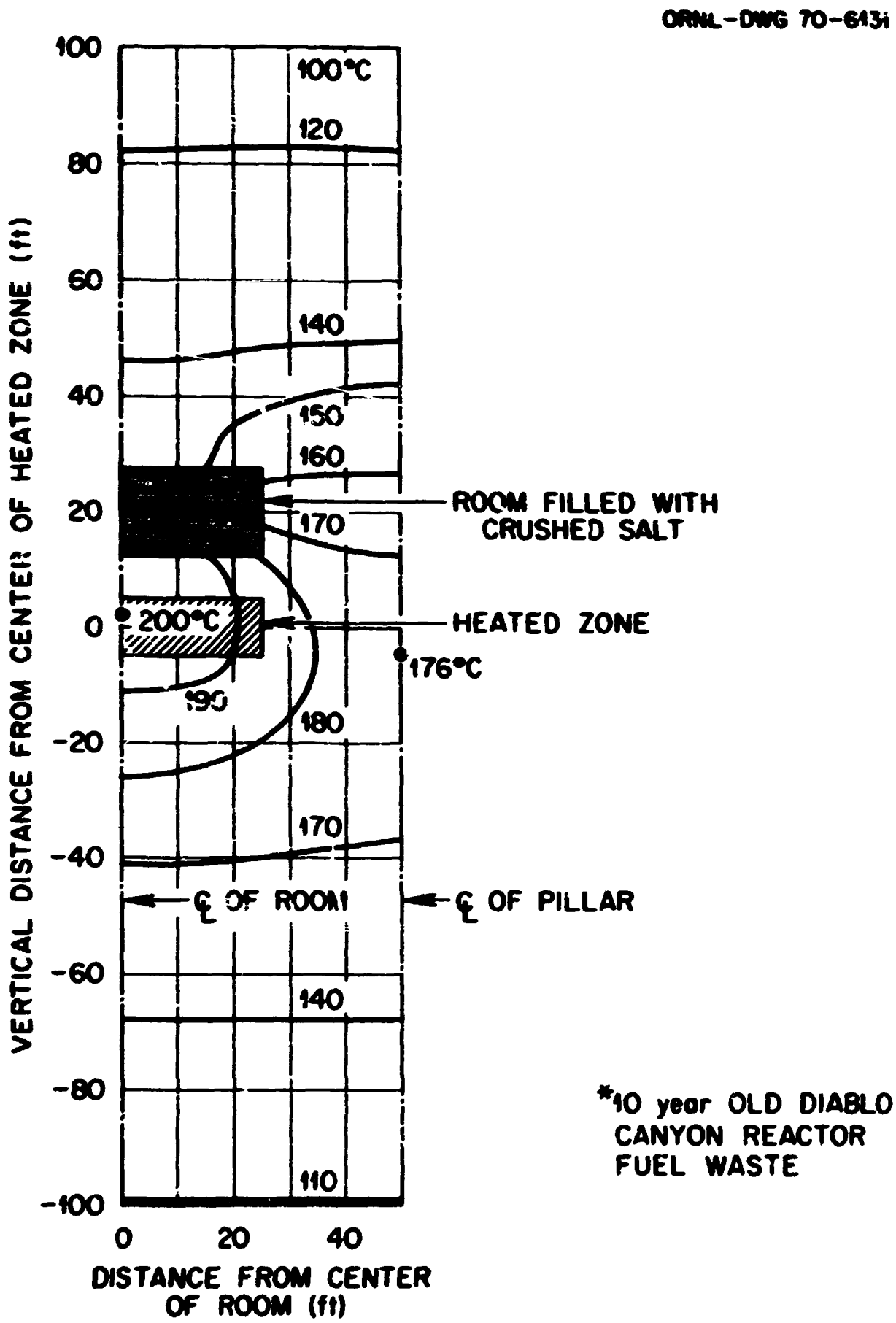

Fig. 7.5. Temperatures in Formation at Time of Peak in Heated Zone (About 3¼ Years After Burial).

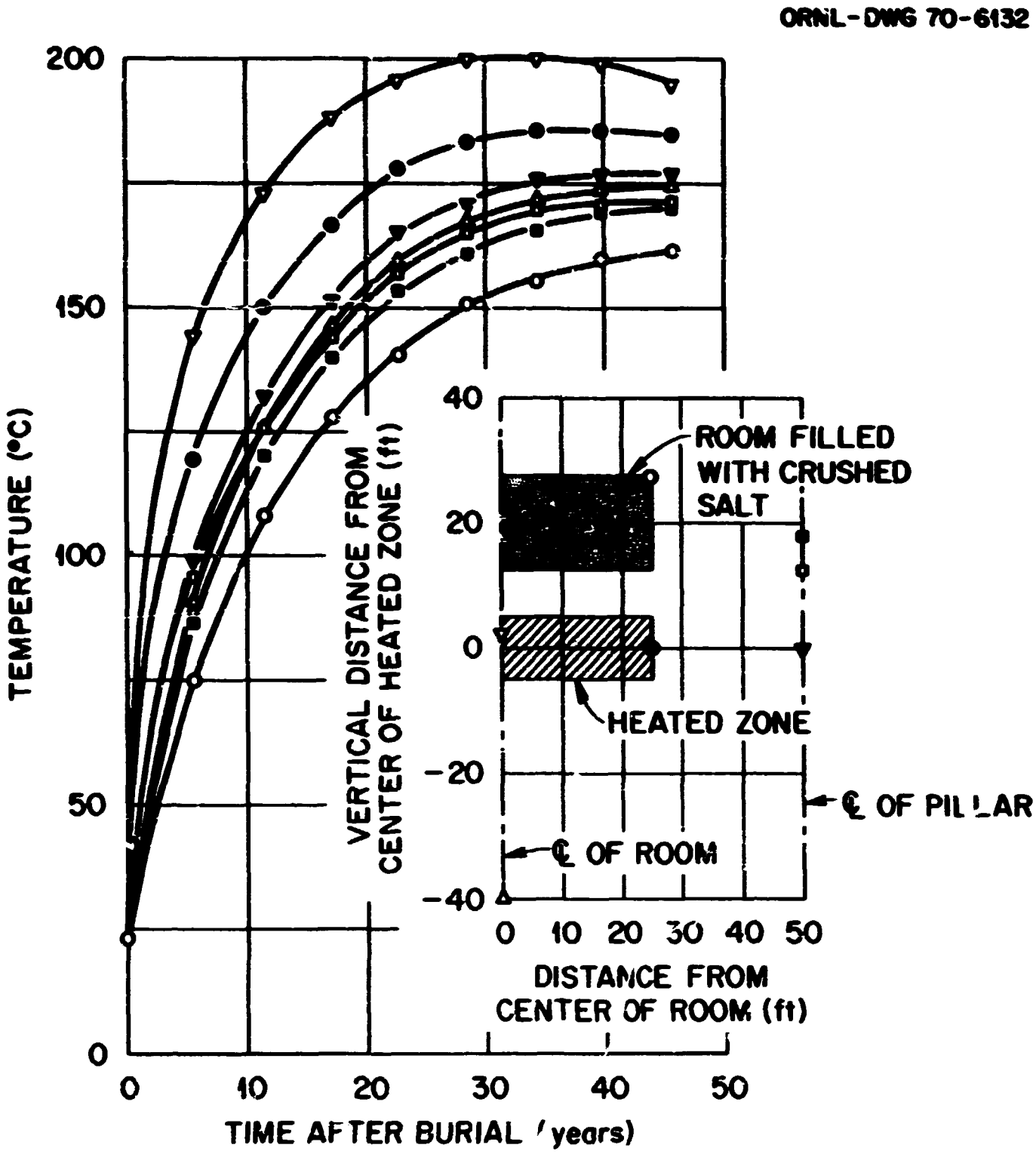

Fig. 7.6. Temperatures in Floor and Pillar.

input rate to reach 200°C maximum salt temperature in the center of a room will also be about 20 Btu/hr·ft²; so it appears that the thermal-stress transfer effects will be at least as great as in Project Salt Vault. Thus, based on the experimental results, it would appear that an initial heat generation rate per unit area which will ultimately produce a maximum salt temperature of 200°C is about as high as desirable (at least for the initial stages of operation) for the first disposal facility.

7.2.4 Formation Stability

As may be noted in Fig. 7.5, at the time of peak salt temperature in the floor, about 100 ft of formation above and below the center of the heat-generating zone will be heated to in excess of 100°C. Thus, some shale beds will reach temperatures in excess of 100°C. At these temperatures, it can be concluded that the shale, and the small quantity of water in its pores, may expand somewhat, but the net effect should be no more than an increase in the rate of closure of the rooms. On the other hand, if the temperatures were allowed to go much higher, it is conceivable that some type of unknown behavior could be encountered.

7.2.5 Aquifer Temperature Rises

Figure 7.7 shows temperatures (in °F) in the entire formation for a condition where the peak salt temperature in the floor of a room reaches 200°C with a 10-year-old waste. The peak is reached at 25 to 50 years after burial, but this type of calculation is not adequate for temperatures in and near the heat-generating zone, since the heat input is assumed to be distributed beneath both the room and the adjacent pillar. This figure indicates that the heat front will reach the Stone Corral formation (depth, about 285 ft) after about 100 years, and that the peak rise would reach a maximum of about 32°F after 800 years (assuming no flow in the aquifer). Portions of similarly stagnant aquifers at depths near 100 ft would experience temperature rises of about 14°F at about the same time. The aquifers in question are of limited yield and use in the area near the proposed facility location. Therefore, it appears that the projected temperature rises would be acceptable.

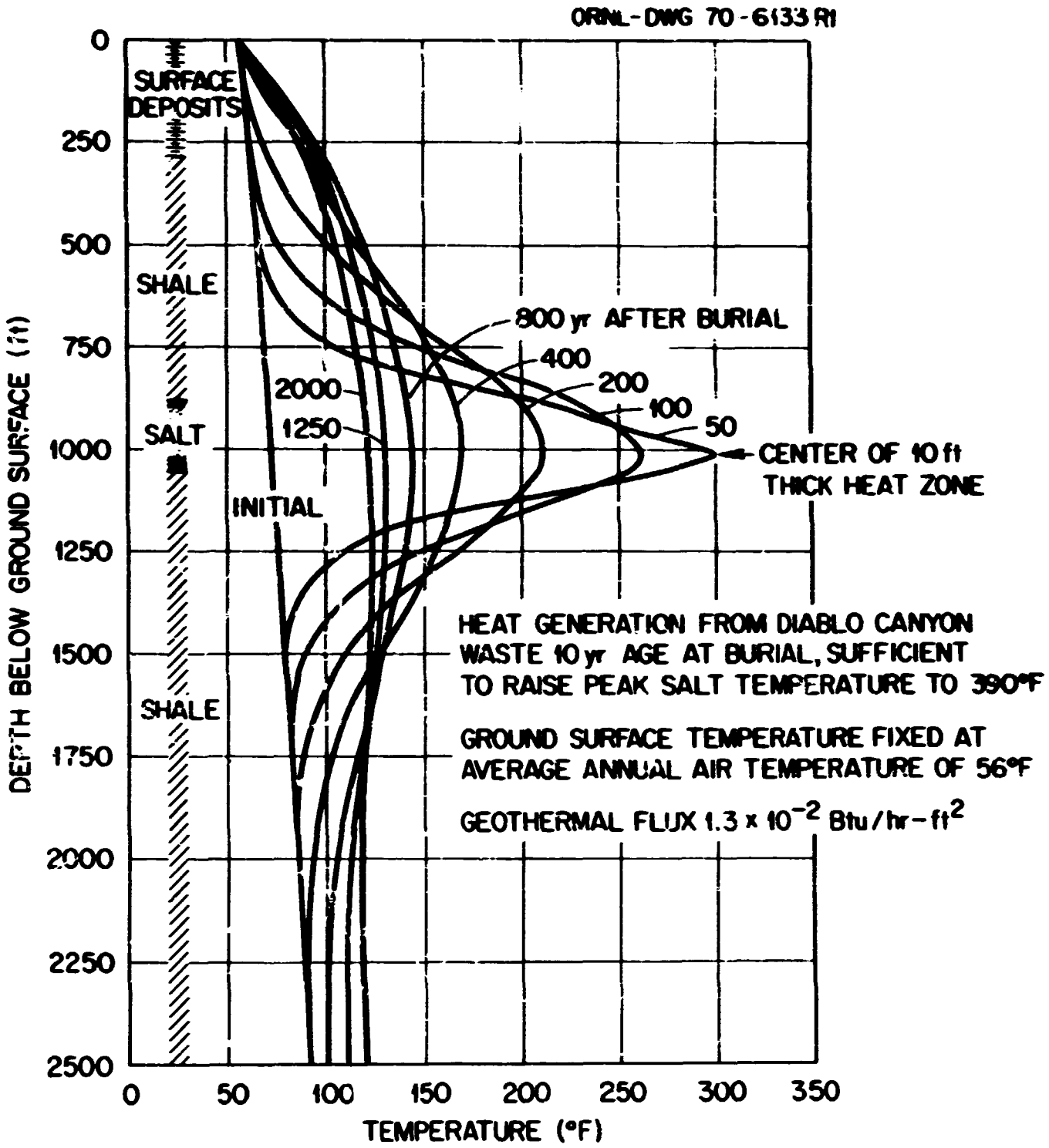

Fig. 7.7. Estimated Temperature Distributions in the Demonstrational Salt Repository.

7.2.6 Surface Heating

Figure 7.7 also indicates that the peak flux at the ground surface, due to the decay heat, will be no more than six or seven times the normal geothermal flux. On the other hand, the average annual solar flux reaching the ground is three to four orders of magnitude greater than the geothermal flux; so the decay heat should have essentially no effect on surface temperature and thus on the local climate, meteorology, or ecology. The maximum increase in surface temperature is estimated as less than 0.1°F.

7.2.7 Temperatures Around Mine

The temperatures in the formation around the mine were computed by fixing the ground surface temperature at 56°F and imposing the geothermal flux at a depth of 1000 ft below the mine. The peak temperature rise at a distance of 500 ft from the edge of the mine is about 10°F, occurring about 1000 years after placement of the waste. Another point of interest is the temperature rise at 1700 ft out from the edge of the heated zone. The maximum rise of about 1°F occurs at about 3000 years after placement of the waste.

7.2.8 Movement of Containers After Burial

After the containers of waste have been deposited in the floor of the mine, they will be contained in a material (the salt) which behaves more like a very viscous fluid than a rigid solid. Plastic deformation of the floor will contribute to the closure of mined openings brought about by the effects of the overburden stress and elevated temperatures. The waste containers will be carried along with the deformation of the floor salt and may experience translational motions certainly no greater than 2 ft vertically and 6 in. horizontally. Of course, the relative motion between the wastes and the salt in this case is zero.

Since the waste containers can be visualized as being suspended in a fluid, the maximum possible rates of floating upward or sinking downward can be estimated by calculations which neglect many of the resisting drag

forces, specifically the frictional drag resistance along the sides of the containers. For extreme values of density difference between the salt and the waste containers, these calculations indicate that the maximum velocity of vertical relative motion between the salt and the waste containers is about 10^{-10} in./year or 1/10,000 of an inch in 1 million years.

If it is assumed that the waste containers cease to exist entirely and the solidified waste becomes an agglomeration of discrete smaller particles, the velocity of vertical motion becomes much smaller as the size of the particles becomes smaller. The situation here is approximately analogous to the settling velocity of a marble and a grain of sand in water.

7.2.9 Induced Geophysical Deformations

In the high-level waste disposal area, the sizes of the disposal rooms and the support pillars between them will be designed in such a way that, in time, the underground openings will completely close, causing a consolidation and recrystallization of the crushed salt used to backfill the rooms. The pillar deformation necessary to effect this closure results from the overburden load and will occur much more rapidly (65 to 100 years) because of the elevated temperatures than is ordinarily the case. In the immediate vicinity of the mined rooms (out to a distance of perhaps 10 to 20 ft from the walls and ceiling), fracturing, bed separation, localized crushing, and other extreme effects will occur. At distances greater than about 20 ft, plastic deformation of the salt will accommodate imposed motions without any of these effects. As the mine closes, which represents a net decrease in the volume of the geologic system, the rocks begin to become heated from the radioactive decay with the resulting thermal expansion representing a net increase in the total volume. Preliminary calculations of the rates of these two processes indicate that they almost exactly cancel each other at all locations in the system including the surface, except, of course, in and immediately around the openings. This means that the overlying protective rocks (and the surface) will be subjected to only very small deformations and

motions for approximately the first 200 years following waste burial. These deformations and motions will be only slightly greater than those resulting from normal salt mining operations. Beginning about 200 years after waste burial, the heat loss at the surface exceeds the heat input from the waste, and the rocks begin to contract, that is, begin a recovery of the previous thermal expansion. This process will be very slow, requiring several thousand years to complete, and will eventually result in the development of a very broad, flat, and shallow (about 3-1/2 ft) subsidence basin over the area. The rate of development of this basin will be so slow and gradual that the shale rocks overlying the salt formation can easily accommodate it without even approaching conditions leading to fracturing or other adverse effects. Likewise, the surface and shallow aquifers will easily adjust to these motions, which are only slightly greater than the current rate of surface erosion in the area. It should also be pointed out that the rate of this subsidence over the waste-disposal area will be approximately the same as the rate of subsidence over an equally sized producing salt mine and, furthermore, that the total amount of subsidence will be approximately half that expected in operating mines because of the backfilled rooms.

Current best estimates of the chronology of the various thermal effects are presented in Fig. 7.8 and Table 7.1.

7.2.10 Future Plans for Thermal Effect Studies

Data to Be Obtained. - In order to perform the calculations and studies which, we feel, are needed to establish allowable heat generation rates in the mine (as governed by the factors mentioned previously), a number of items of information are needed from the proposed facility site. Some of this information is currently being developed by the USGS and various consultants to ORNL.

Cores (to about 1300-ft depth in one hole, and through the salt member in the other) have been taken at two corners of the proposed facility. Representative samples from more than 80 locations in the full-depth core are being furnished to the USGS. From these samples,

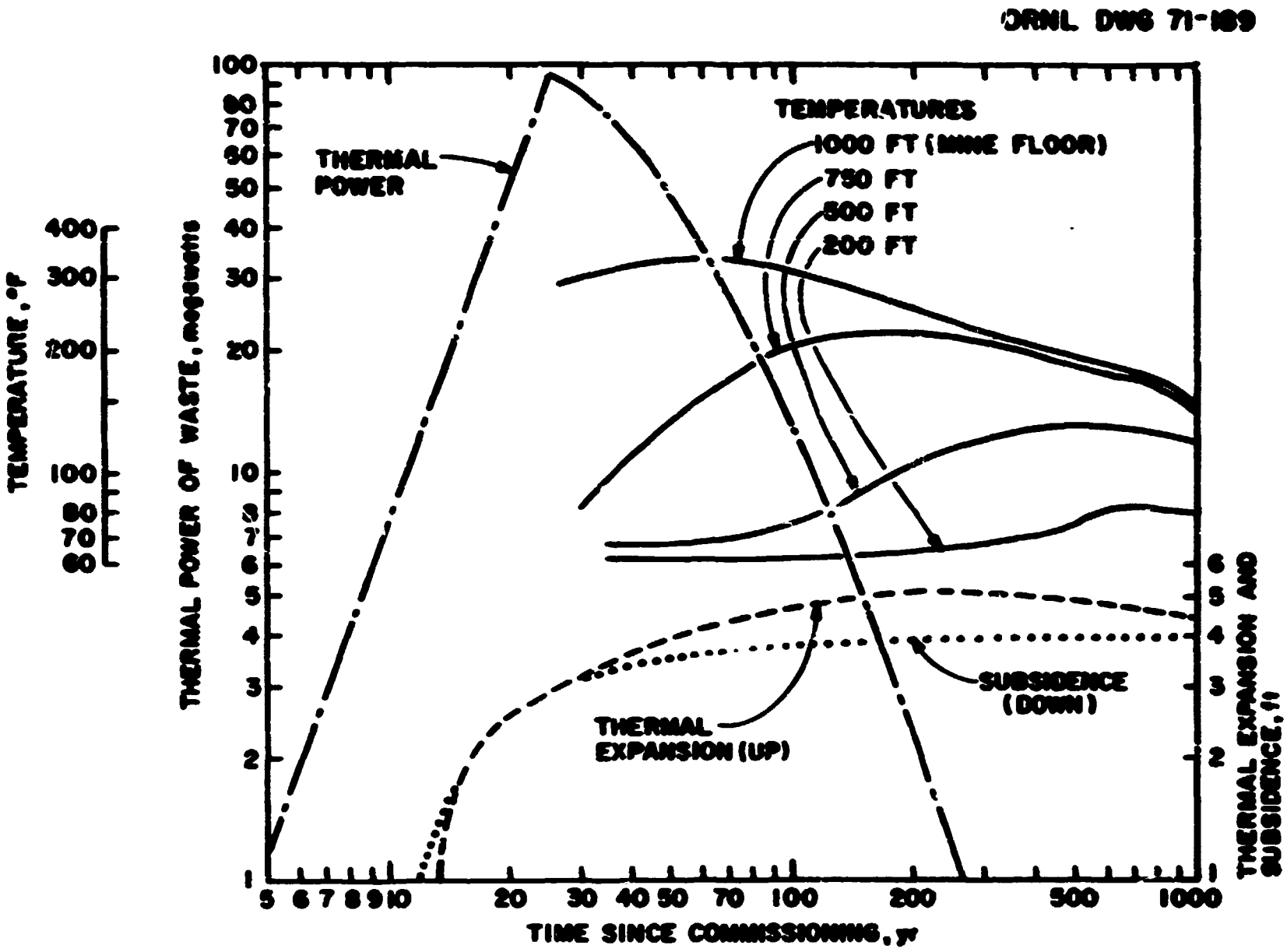

Fig. 7.8. The Effect of Time Since Commissioning on the Projected Total Thermal Power, Surface Rise Due to Thermal Expansion, Surface Fall Due to Subsidence, and Temperatures Within the Formation.

Table 7.1. Chronology of Thermal Effects in the Repository

Years	Effect
25	Decommissioning; power peaks at 95 MW.
30	400°F peak temperature rise at 8 in. from cylinders; 10 liters total brine inflow per container.
100	95% closure of voids by subsidence; 0.9-ft increase in typical surface level.
150	110°F peak temperature rise at 750-ft depth.
200	5.2-ft peak in net thermal expansion; 1.3-ft peak in typical surface level.
500	60°F peak temperature rise at 500-ft depth.
700	20°F peak temperature rise at 200-ft depth; 0.1°F peak temperature rise at surface.
3,000	1°F peak temperature rise at 1700 ft from edge of mine at 1000-ft depth.
~15,000	Total ~4-ft subsidence essentially complete; all temperatures within 5°F of initial condition

they will determine thermal conductivity; and, combined with temperature measurements which they have made in the holes, the USGS will supply us with the calculated geothermal flux. The temperature measurements and sample conductivity data will also be supplied to us.

In addition to these data, we will need to obtain from representative core samples sufficient data to provide estimates of the following thermal and physical properties (as a function of temperature where appropriate): (1) mineral composition; (2) water content; (3) thermal conductivity; (4) specific heat (per unit weight); (5) density or thermal diffusivity; (6) thermal expansion coefficient; (7) Poisson's ratio; (8) elastic modulus; (9) possible phase changes; and (10) effects of stress on physical properties. For rocks within about 50 ft above and below the center of the heat generating zone in the mine floor, the temperature range investigated should be about 20 to 200°C. For the next 50-ft increment above and below, the range should be to about 150°C. From 100 to 200 ft above and below, it should be to about 125°C; from 200 to 400 ft distance, to 100°C; and beyond that, to about 50°C.

At this time, it has not been decided how, or by whom, some of the above information will be obtained.

Thermal Calculations. - When the stratigraphic, thermal property, and geothermal flux information (discussed above) becomes available, the previous calculations will be repeated using the appropriate values. In addition, three-dimensional calculations will be run to ascertain such effects as thickening and thinning or disappearance of salt and/or other rock strata. Also, since some stratigraphic information is available from various electric logs on five other holes within the proposed site, it should be possible to construct a three-dimensional model which will be a reasonable approximation to the entire high-level disposal facility and to accurately estimate temperatures in the strata surrounding the proposed actual mine layout.

Rheological Studies. - With the physical property data (outlined above) and the knowledge of the temperature-time relationships from the planned thermal studies, new estimates will be made of rock movements due to subsidence, thermal expansion, and any other factors (such as mineral alteration resulting from changes of phase, hydration, or pore volume) which are applicable. If it appears that rock temperatures anywhere in the formation are such as to cause a potential threat to the containment of the waste within the salt formation, the maximum allowable heat generation rates at mine level will be adjusted to an acceptable level. This will also be the case if temperatures in aquifers or beyond the property boundaries are found to be excessive.

7.3 Radiation Effects

7.3.1 Chemical Environment Around the Waste Containers

We have recently completed a detailed review of the radiolytic and chemical effects on the salt in the vicinity of the wastes and on the waste containers. Our conclusions and recommendations for additional experimental work follow.

Radiolytic Effects. - Radiation dosages in excess of 10^{11} rads induce no chemical effects of consequence to the salt; however, they do produce changes in the brine intrusions that constitute about 0.5 vol % of the salt and in the very small amount of air initially in the voids in crushed salt immediately surrounding the containers. This brine has the following composition: Mg^{2+}, 2.3 to 3 M: K^+, 0.4 M; Na^+, 1 to 2 M; Cl^-, 7 to 7.5 M; Br^-, 0.05 M. It is slightly alkaline.

The value of $G(H_2)$ in this brine may be as high as 2.1 molecules of H_2 per 100 eV of absorbed radiation energy, and the oxidized species will be O_2 and/or $ClO_3^- + BrO_3^-$. Based on this upper-limit rate, we estimate that the radiolytic hydrogen can be produced at a maximum rate of only about 1 ft^3/min throughout the entire Repository when the thermal power from radioactive decay is about 100 MW. The bromates and chlorates around the containers will decompose thermally to halides and oxygen. Ozone and nitrogen oxide formation in the gas-vapor mixture will not be a problem at the low partial pressure of air and at the temperatures which will prevail.

All $MgCl_2$ in solution which gets into the open spaces around the containers will hydrolyze to form HCl, provided the location of the $MgCl_2$ is such that the HCl can move away from the site of the reaction. The quantity of HCl produced in this manner is unimportant except from the standpoint of its possible influence on container life (see below).

Although our evaluations of these effects, based on very pessimistic assumptions, are most favorable, we nevertheless believe that measurements should be made of the actual generation rates and resultant concentrations of radiolytic and hydrolysis products that may exist under steady-state conditions around the containers in the mine.

Corrosion of the Containers. - Rather early in the Salt Vault program, a number of scouting studies were conducted on the corrosion of various alloys in salt compacts such as were anticipated in salt-mine storage of fission-product wastes. These laboratory capsule studies indicated that type 304L stainless-steel containers would be subject to relatively minor generalized and pitting attack which diminished with continuing exposure. One capsule exposed in a ^{60}Co gamma source showed no enhanced corrosive attack because of the radiation exposure.

Subsequent exposure of welded stainless-steel alloy specimens attached to pots of simulated waste buried in the floor of the Carey mine and electrically heated to 575°C for 31 days gave results comparable to those obtained in the laboratory studies.

Unfortunately, the tendency of the brine inclusions in the salt beds to migrate up the thermal gradient toward the hot containers and thereby provide a long-term source of corrosive solution was not known when these investigations were carried out. This brine migration produced a number of instances of stress corrosion cracking of stainless-steel equipment used in Project Salt Vault. In practically all cases, however, the attack occurred in regions which were sufficiently cool for condensation of moisture, with resultant formation of acidic, concentrated brine solutions known to cause active stress-corrosion of stainless steels. Such low-temperature stainless-steel container surfaces would not be expected in the mine. Unfortunately, stress-corrosion cracks penetrating halfway through the wall (0.237 in. nominal) were found in the center of a type 304L stainless-steel modified pillar heater which was continuously maintained between 150 and 200°C for ~ 350 days. Conceivably the crack could have been produced during heatup or cooldown, but this seems unlikely; so other mechanisms than formation of aggressive solutions by condensate must be sought. It seems likely that contact with salt in the presence of an HCl-steam atmosphere would be sufficient to cause failure.

In general, the experience with carbon-steel equipment in Project Salt Vault was more favorable. Moderate generalized attack with no evidence of localized, severe penetration like the stress cracking of the stainless steel was observed.

The required lifetime of the containers is still a moot question; however, consideration of the problems of retrieval of the wastes in the unlikely event that some unexpected development forces abandonment of the concept suggests that it would be advantageous to prolong container life for at least a decade or so. Studies which could lead to such lifetimes include:

1. Addition of CaO to the crushed salt backfill to react with the water of the brine accretion to form stable $Ca(OH)_2$.
2. Investigation of new, stress-corrosion-resistant alloys.
3. The behavior of the carbon-steel equipment suggests that a duplex container with type 304L stainless steel on the inside and carbon steel or aluminum on the outside should perform very well. The carbon steel should give long-term protection both because of the slow, generalized attack and also by acting as a sacrificial anode to the stainless steel after the carbon steel or aluminum has been penetrated.

7.3.2 Stored Energy

Stored energy refers to energy that may be stored in irradiated solids due to the introduction of defects and other disorder. When the solid is heated subsequent to its irradiation, energy is released as the solid passes through various stages of reordering. In some cases (particularly in strongly covalent crystals in which large amounts of energy can be stored), all of the energy can be released with a relatively low initiating temperature since the initial energy release can heat the solid sufficiently to trigger energy releases at successively higher transition temperatures. In the case of long-term storage of radioactive wastes, it is important to know how much energy can be stored at saturation in the materials that will be in the radiation field and thermal environment of the Repository. (We define saturation as that level of damage at which the radiation removes as much energy by radiation annealing as it supplies by radiation damage.) The materials in question are the glass or ceramic boule containing the radionuclides, the metal container, and the surrounding salt. The temperature of the wastes and their containers will be from 600 to 900°C initially

and will decrease to less than 500°C in about 10 years, while the temperature of the salt will range between 20 and 350°C, depending on its proximity to the waste and the thermal power of the source. The maximum accumulated radiation dose in salt will range from 2×10^{11} rads adjacent to the wastes to about 82 rads at a distance of 5 ft.

Some work with stored energy in NaCl has been reported in the literature. Kobayashi[8] measured an energy release of 2 to 2-1/2 cal/g between room temperature and 400°C in NaCl that was heavily irradiated with protons at room temperature. Bunch and Pearlstein[9] have compared the F-center concentration in x-ray-irradiated NaCl with the stored energy, and report a value of 12.4 eV per F center. For a saturation concentration of 2×10^{19} F centers/cm^3 (ref. 10), this would yield a saturation stored energy of 4 cal/g.

At ORNL, we have carried out an experimental investigation designed to yield upper-limit values of the amount of energy that can be stored in salt under the conditions that will exist in the Repository. In the course of this investigation, we have measured stored energy by solution calorimetry and by differential scanning calorimetry of samples of salt taken from the experimental holes of Project Salt Vault, and of samples of salt taken from the Lyons mine and irradiated with an electron accelerator at different temperatures and dose rates (Table 7.2). Also, we supplied E. J. Zeller of the Kansas State Geological Survey with specimens of salt from the PSV holes, as well as with other salt from Lyons for proton irradiation experiments. We conclude from both our own results and those of Professor Zeller, which he finds to be in substantial agreement with ours,[11] that energy storage in the salt where the containers will be buried cannot exceed 15 to 20 cal/g at ambient temperatures; and at temperatures above 100°C (such as will exist in the vicinity of the high-level wastes), no more than 1 or 2 cal/g can be stored.

Although there have been no measurements to date of energy storage in solidified wastes, it is quite likely that energy can be stored in some of these materials, and over a wider temperature range than is possible in salt. Zeller _et al._[12] have measured up to 200 cal/g in olivine (a mineral containing about the same amount of silicate as is present in

Table 7.2. Stored Energy Measurements in Rock Salt by Solution Calorimetry

Sample	Irradiation Temperature (°C)	Dose (rads)	Stored Energy (cal/g)
		Electron-Irradiated Samples	
V_oA	30	1.7×10^{10}	14.1
V_oB	30	1.7×10^{10}	1.6[a]
V_oC	30	1.7×10^{10}	6.9[b]
V_1	80-100	3×10^{10}	3.0
V_2	80-100	8.3×10^{10}	2.1
V_3	80-100	1.1×10^{11}	4.3
		Project Salt Vault Samples	
Hole 10 (7-1/2-ft depth)	125	8×10^{7}	0
Hole 4A (11-ft depth)	165-200	9×10^{8}	0
Hole 4B (9-1/2 to 11-ft depth)	165-200	9×10^{8}	0

[a]Heated at 250°C for 7 hr.

[b]Stored at 25°C for 78 days.

some of the solidified waste products) when it was irradiated with protons at liquid nitrogen temperatures. All of this energy was released below 450°C, however.

Thermal Effects. - A variety of calculations have been made to evaluate the maximum transient temperatures in the salt that could result from an instantaneous release of stored energy. In this study we have evaluated, primarily, the effect of initial age of the waste, initial thermal power, and the length of time that the cylinder has been stored in the salt. The energy storage has been conservatively overestimated as 1% of the total deposited photon-neutron energy (Fig. 7.9) up to a maximum saturation value of 200 cal/g in the calcined waste and 20 cal/g in the salt.

The transient temperature in the vicinity of a waste cylinder before and after a stored energy release was calculated using numerical analysis techniques of the type described in Sect. 7.2. Of the cases studied, the maximum thermal effect of a stored energy release occurred under the assumption that a cylinder having initial thermal power of 5000 W and initial waste age of 10 years resulted in a stored energy release 30 years after burial in the salt (Table 7.3). The effect of this release was to cause no melting of the salt and a temperature rise of less than 2°F at 13.5 ft from the center of the cylinder.

Mechanical Effects. - Rapid thermal expansion of the waste and nearby salt, resulting from a release of stored energy would cause some plastic deformation of the salt in the region near the container. Much of the volume increase would be accommodated by the voids in the crushed salt used for backfill. The remainder of the deformation would be expressed as a slight bulge in the floor above the container.

Table 7.3 presents results of estimates of the maximum increase in volume for the several assumed conditions of stored energy release. Materials properties that were assumed for these calculations are given in Table 7.4. Thermal expansion of the brine, which constitutes 0.5% by volume of the salt, was overestimated by assuming that the brine is pure water, using thermodynamic data from steam tables.

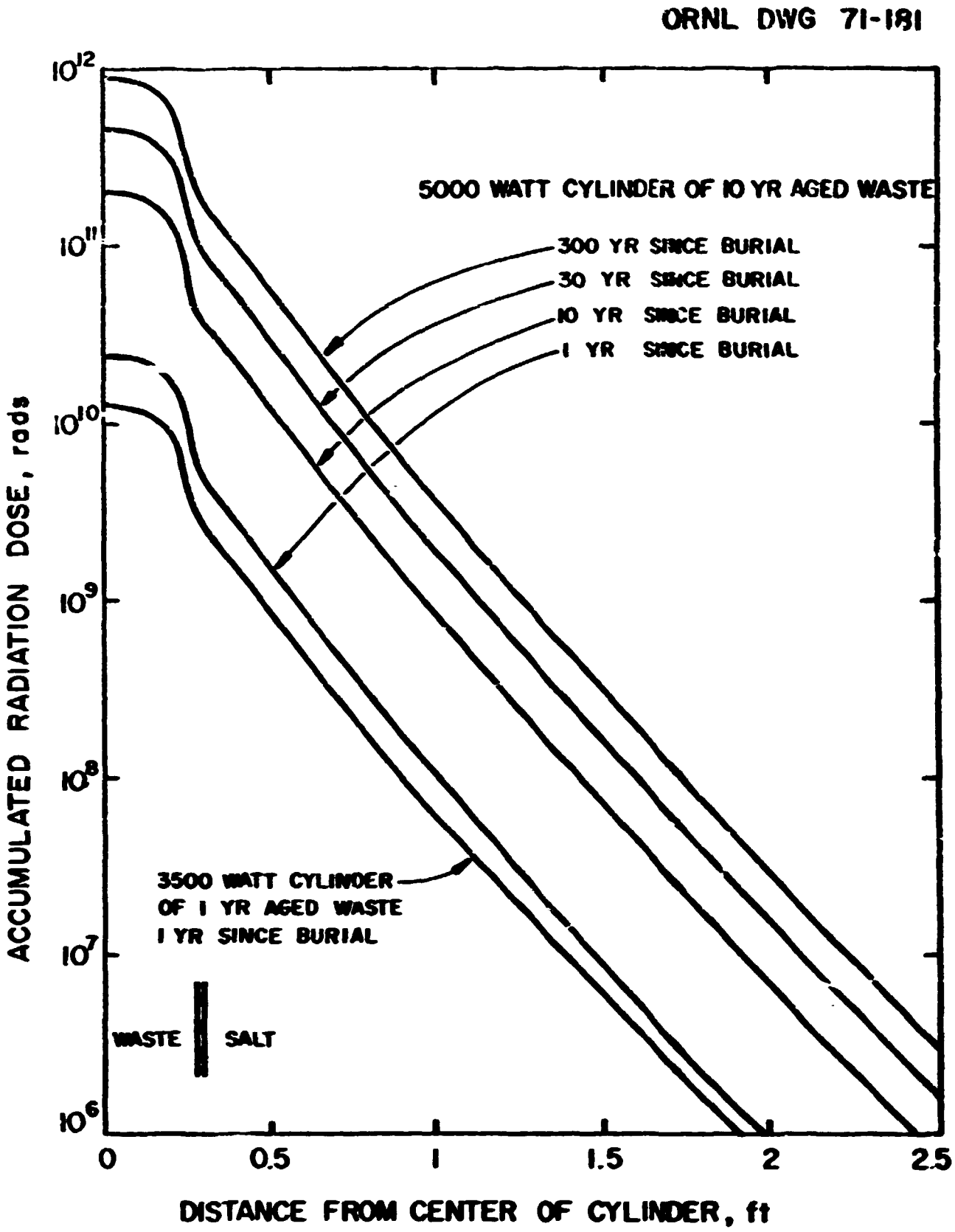

Fig. 7.9. Accumulated Photon-Neutron Dose in the Horizontal Midplane of Cylinders of Solidified High-level Waste Buried in Salt. Waste cylinders have inside diameter of 6 in., steel wall thickness of 0.25 in., and active height of 8 ft.

Table 7.3. Estimated Maximum Temperature and Volume Increases That Could Result from Release of Stored Energy Developed by a Waste Cylinder Stored in Salt

Properties of Waste Cylinder[a]				
Initial thermal power, W	5000	5000	520	3500
Initial age of waste, years	10	10	10	1
Time of storage in salt, years	1	30	30	1
Distance between cylinders, ft	27	27	8	12
Volume Change due to Thermal Expansion, ft^3				
Waste and salt	0.12	0.36	0.13	0.08
Brine	1.53	5.62	1.79	0.88
Total	1.65	5.98	1.92	0.96
Maximum Temperature Rise, °F				
1 ft from center of cylinder	56	170	150	49
4 ft from center of cylinder	4	7	4	3
Midpoint between adjacent cylinders	0.9	2	4	2

[a]Cylinders are assumed to have diameter of 6 in. and active length of 8 ft.

Table 7.4. Assumed Properties of Calcined Waste and Salt Used for Analysis of Stored Energy Release

Calcined Waste	
Density, lb/ft^3	113
Thermal conductivity, Btu/hr·ft·°F	0.25
Heat capacity, Btu/lb·°F	0.22
Coefficient of cubical expansion, $°F^{-1}$	0.88×10^{-6}
Rock Salt	
Density, lb/ft^3	135
Brine content, vol %	0.5
Thermal conductivity, Btu/hr·ft·°F	1.81
Heat capacity, Btu/lb·°F	0.218
Coefficient of cubical expansion, $°F^{-1}$	0.67×10^{-4}
Melting point, °F	1470
Crushed Salt	
Density, lb/ft^3	95
Brine content, vol %	0.5
Thermal conductivity, Btu/hr·ft·°F	0.36
Heat capacity, Btu/lb	0.218

Conclusions. - On the basis of the foregoing, we are confident that energy storage under the conditions that will prevail in the Repository does not carry any serious implications. We expect to conclude our experimental work related to NaCl by establishing saturation values of stored energy as a function of temperature, radiation dose, and chemical purity. Also, some additional work is required to establish upper-limit values for energy storage in the solidified wastes, and the conditions under which it can be released.

7.3.3 Isotope Transport

We have made analyses of the various mechanisms by which radioisotopes can migrate within the mine, and, while there have been no indications of problems caused by these mechanisms, we plan to perform additional work that will provide a firmer basis for planning the operational safety of the Repository and devising contingency plans for possible retrieval of the wastes.

An analysis of solid-state diffusion of plutonium in NaCl indicates that this isotope should not migrate more than about 25 cm in 1 million years at a temperature of 350°C; however, there is little information available on surface diffusion of isotopes along grain boundaries. We have begun experimental investigations of this mechanism.

Recognizing that the containers will probably deteriorate in a relatively brief period of time after burial, we have looked at the prospects for enhancement of the volatility of isotopes by steam arising from the brine within the salt. This appears to be a very remote possibility; however, we believe that some observations with unpackaged wastes under actual or simulated burial conditions are justified.

We have investigations under way on the permeability of crushed salt to gases and to nonvolatile radioisotopes, and on the formation and fate of PuO_2 aerosols. This information is needed in the design of the mine ventilation system and for the analysis of the possible consequences of accidents.

7.4 Radiological and Physiological Safety

Repository systems for protection against exposure to radioactive and physiologically hazardous materials are being designed to comply with all federal regulations that apply to radiation safety, mine safety, and industrial safety. The principal regulations that dictate safety provisions of the Repository are AECM 0524 and 10CFR20 for radiation safety; 30CFR57 for mine safety; AECM 0550 and 41CFR50-204 for industrial safety; and AECM 0529 and 10CFR71 for transportation safety.

The only significant source of off-site exposure to radioactive or other materials originating from the Repository will be from very small quantities of airborne radioactive materials that escape through high-efficiency particulate air (HEPA) filters. It is planned that there will be no discharge of liquid waste from the Repository. Solid wastes generated at the Repository will be packaged and buried in the mine. Aqueous wastes will be recycled, with any excess water being evaporated to the atmosphere.

Instrument systems will be provided for continuous monitoring of radiation levels and concentrations of airborne materials in work areas, in stack effluents, and at the boundary of the site.

The materials that will escape to the atmosphere in small concentrations include: (1) solid waste particles resulting from resuspension of surface contamination; (2) gases (^{85}Kr, ^{3}H, He) that result from decay of the waste; (3) radon gas that occurs naturally in the mine; (4) non-radioactive gases (H_2, HCl) that result from decomposition of the salt, corrosion, electrolysis, and radiolysis; (5) exhaust fumes from diesel equipment; and (6) salt particles. Concentrations in work areas of the Repository will be maintained well below the applicable standards for occupational exposure.

Present estimates of the rate of release of these materials to the atmosphere, and the maximum off-site concentrations that will result at the southern boundary of the site, are presented in Table 7.5. These maximum concentrations are less than 0.1% of the applicable standards for exposure of the public and, in general, are well below the concen-

Table 7.5. Estimated Release Rates and Average Annual Off-site Concentrations of Radioactive and Other Materials Resulting from Full-Scale Operation of the Repository

Material	Average Annual Rate of Release to Atmosphere	Average Annual Off-site Concentrations Resulting from Repository Effluents[a]	Percent of Applicable Standards for Exposure of the Public[b]	Percent of Estimated Present Concentration in the Atmosphere
High-level waste particles	0.007 Ci/year	2×10^{-15} Ci/m³	0.02	5.0
Alpha waste particles	0.04 μCi/year	1×10^{-20} Ci/m³	0.0001	0.02
^{85}Kr (spontaneous fission)	0.014 Ci/year	4×10^{-15} Ci/m³	0.000004	0.04
^{3}H (spontaneous fission)	0.0009 Ci/year	3×10^{-16} Ci/m³	0.0000004	0.002
^{222}Rn (natural sources)	0.9 Ci/year	2×10^{-13} Ci/m³	0.02	0.2
^{220}Rn (natural sources)	0.04 Ci/year	1×10^{-14} Ci/m³	0.0003	0.2
H_2 (corrosion, radiolysis, electrolysis)	37 scfm	0.2 ppm	0.002	40.
He (alpha decay)	0.001 scfm	0.000004 ppm	4×10^{-8}	0.00008
HCl (brine decomposition)	0.07 scfm	0.0003 ppm	0.006	100.
CO_2 (diesel exhaust)	50 scfm	0.2 ppm	0.004	0.07
CO (diesel exhaust)	0.05 scfm	0.0002 ppm	0.0004	0.2
NO_2 (diesel exhaust)	0.05 scfm	0.0002 ppm	0.004	7
SO_2 (diesel exhaust)	0.03 scfm	0.0001 ppm	0.002	1
CH_2O (diesel exhaust)	0.0007 scfm	0.000003 ppm	0.00003	0.05
Soot (diesel exhaust)	2 lb/year	0.1 μg/m³	0.0007	0.2
Salt particles	5 lb/year	0.2 μg/m³	0.001	0.8

[a]These are the maximum concentrations which result at the southern boundary of the site.

[b]Based on one-third of limits in 10CFR20, Table II, Column 1 for radionuclides and threshold limit values (American Conference of Governmental Industrial Hygienists) for other materials.

trations of the same materials that exist at present due to natural and man-made sources.

It is estimated that the buildup of off-site surface contamination of various species over the life of the Repository will ultimately result in a level that is less than 1% of the level that presently exists for the same species (including such radioisotopes as ^{90}Sr, ^{137}Cs, and ^{239}Pu which have been deposited by fallout). At these levels of surface contamination, the experience has been that the materials cannot be reconcentrated to hazardous levels in food chains.

Additional experimental work is planned to provide a more precise preoperational background of the radiation environment of the site and surroundings. In addition, continuing studies of the chemical interactions of the waste containers and salt will provide a better definition of the concentrations of the materials that may be released to the atmosphere.

7.5 Ecology and Environmental Monitoring

The ecological investigations of the site to date have been: (1) a broadly based survey of the biotic resources of the several ecosystems on, and immediately adjacent to, the site; (2) collection and compilation of published data applicable at the site on climate and meteorological factors, as well as available information on the behavior of radionuclides in those environments; and (3) formulation of a model for the dynamics of strontium-90 in a grassland system. This model provided information for the preliminary safety analysis and illustrates the use of the technique for predicting the distribution of contaminants, from either long-term chronic releases or short-term acute releases resulting from an accident. Future work will be directed primarily at establishing similar models and their transfer functions for other nuclides and in the other ecosystems (aquatic, agriculture, etc.).

Instrumentation for radiological monitoring of the facility operations at the site is now included in the facility design; and, as the project develops, a monitoring program will be established in conjunction with appropriate state and federal agencies. We are now planning discussions with the Kansas Department of Health, representatives of area educational institutions, Kansas Geological Survey, Kansas Advisory Council on Ecology, and the Kansas Nuclear Energy Council after the project is authorized. We will establish a cooperative monitoring program designed to resurvey existing levels of radioactivity in the region and establish a monitoring network. Similarly, surveillance of both radiological and thermal effects on the geologic and hydrologic formations will be established.

7.6 Retrieval of the Wastes

While we believe that the deposition of radioactive wastes 1000 ft underground in salt in the manner that we now plan is a safe and responsible course of action, we also believe that contingency plans for their removal should be developed in the event some unforeseen and highly improbable event should indicate the necessity for doing so. The Commission has publicly stated its position on this subject as follows:

> "Once radioactive wastes are emplaced in the salt repository, they will be regarded as in permanent storage. It is not planned that they will be relocated elsewhere. Retrieval would only be considered in the light of an objective safety problem under circumstances which to date have not been postulated. However, the facility will be designed so as not to preclude retrieval. Further, the burial locations for each container will be accurately surveyed and recorded so that the precise location of the wastes will be known. Retrieval through use of remote mining equipment would thus be possible. Concepts for retrieving the stored wastes will be examined in the course of the design of the facility."[13]

We concur in this statement. As the design of the facility progresses and after the more important overall aspects of the design are decided upon, we will consider, in some detail, the concepts and methods that could be applied for recovery of the wastes within existing or foreseeable technology.

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