Or. 2446

3/25/81 T.S.

②

ORNL/TM-7597

R-2961

ornl

OAK RIDGE NATIONAL LABORATORY

UNION CARBIDE

MASTER

Expected Environments for a Defense High-Level Waste Repository in Salt

L. D. Rickertsen
H. C. Claiborne

OPERATED BY
UNION CARBIDE CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL/TM-7597
Dist. Category UC-70

Contract No. W-7405-eng-26

CHEMICAL TECHNOLOGY DIVISION

NUCLEAR WASTE PROGRAMS

Expected Repository Environments for Commercial Waste,
Savannah River Wastes, and Other Wastes
(Activity No. AP 05 15 15; 189 No. ONW-2807)

EXPECTED ENVIRONMENTS FOR A DEFENSE HIGH-LEVEL WASTE REPOSITORY IN SALT

L. D. Rickertsen* and H. C. Claiborne

*Science Applications, Inc., Oak Ridge, Tennessee 37830.

Date Published: March 1981

This report was prepared by Oak Ridge National Laboratory under Contract No. W-7405-eng-26 with the U.S. Department of Energy. The project was administered by the Office of Nuclear Waste Isolation, Battelle Memorial Institute.

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
U.S. DEPARTMENT OF ENERGY

CONTENTS

LIST OF FIGURES

LIST OF TABLES

ORNL/TM-7597
Dist. Category UC-70

EXPECTED ENVIRONMENTS FOR A DEFENSE HIGH-LEVEL WASTE REPOSITORY IN SALT

L. D. Rickertsen* and H. C. Claiborne

ABSTRACT

Expected environments for a defense high-level waste (DHLW) repository in salt have been predicted analogously to previous analyses for spent fuel (SF) and reprocessed commercial high-level wastes (CHLW). Environments predicted include near-field and far-field temperatures, fluid, pressure, and nuclear radiation fields. Some sensitivity studies have also been performed. The main results of the calculations reported here include the following:

1. Rock temperatures, canister wall temperatures, and waste temperatures do not exceed 86, 94, and 101°C, respectively.

2. The maximum brine inflow rate to an emplacement hole is 0.015 L/yr, occurring in the first 30 yr after emplacement. The total accumulation of brine migrating to the emplacement hole after 1000 yr is <0.5 L.

3. Gas pressures encountered by the waste package do not exceed 0.36 MPa prior to mine closure. After this time, it is conceivable that stress on the canister could approach the lithostatic rock stresses.

4. Maximum dose rates in the salt are <1400 rads/h.

1. INTRODUCTION

High-level radioactive materials are being generated in the United States as a result of weapons programs and associated projects which are operated in the interests of national defense. Significant quantities of defense high-level waste (DHLW) are currently being stored at a number of sites, with most of the waste at the Savannah River Plant (SRP), the Hanford Reservation, and the Idaho National Engineering Laboratory. The waste at these sites is currently stored as salts, sludges, and liquids in large steel-lined tanks. Although this procedure generally provides completely adequate and safe storage, it has always been viewed as a

*Science Applications, Inc., Oak Ridge, Tennessee 37830.

temporary measure, and other means for permanent disposal have been under consideration.

High-level wastes will also be generated from the operation of nuclear reactors. These wastes may include spent fuel assemblies (SF) and reprocessed commercial high-level wastes (CHLW). The most promising candidate for disposal of these wastes at present appears to be a repository in a deep geologic formation.[1] Serious consideration has been given to such a repository for SF and CHLW in salt, granite, shale, basalt, and tuff formations. The suitability of such a concept, however, depends upon the conditions that will occur in the repository as a result of the emplacement of these wastes. The wastes will generate heat, which will affect thermal, mechanical, chemical, and hydrologic environments in the formation and could modify the waste isolation properties of the repository. Therefore, considerable effort has been made to understand these environments. Studies have been initiated for the expected conditions associated with the emplacement of SF and CHLW in a salt repository,[2] which may serve as a basis for conceptual designs of such a repository.

The geologic disposal concept has also been suggested for the defense wastes, including DHLW. The DHLW could be removed from the tanks and immobilized in some satisfactory matrix, if necessary, and canistered. Possible waste forms include borosilicate or other glass, various ceramics, a metal matrix, or calcined waste. Waste canisters similar to those suggested for CHLW could be used, and a specific canister conceptual design has already been offered.[3]

The DHLW is expected to have radiation and thermal characteristics different from CHLW or SF, however, so that the conditions in the repository will not be the same as those predicted for these cases. Therefore, analysis of the environments for a DHLW repository in salt analogous to the studies for CHLW and SF are reported here. Estimates of the temperatures in the repository due to the time-dependent heat generation of the emplaced waste are given in Sect. 2. Expected fluid environments due to the migration of brine trapped in the salt formation induced by the thermal gradient are given in Sect. 3. Section 4 gives the gas pressures that might occur in the emplacement hole which the waste package could encounter. Finally, the nuclear radiation field in the vicinity of the waste canisters is predicted, and the results are shown in Sect. 6.

It is well to realize that the DHLW is not simply characterized. The density, composition, and nuclide content varies from site to site as well as at a given site. The age of the waste varies considerably as does its composition, and it is difficult to characterize the waste in a straightforward manner. A solution to this problem has been to specify the most conservative waste possible based on current inventories and projected waste streams. Such a waste description has been provided by workers at the Savannah River Plant (SRP).[4] The waste is assumed to be 15 yr old at emplacement (10 yr after reprocessing, which occurs 5 yr after reactor discharge). This is the freshest SRP waste that would be placed in a repository, is more concentrated than the DHLW at other sites, and is likely to provide conservative estimates of all waste-induced perturbations to the repository environments. It is unlikely that the actual defense waste emplaced into a repository would have impacts as large as these predicted for this reference waste.

Table 1 gives the isotopic content for the chosen DHLW at emplacement provided by SRP for this work and is taken from ref. 5.

2. THERMAL ENVIRONMENT

The mined repository concept for disposal of high-level radioactive waste in a salt formation has been described elsewhere.[2] Briefly, storage rooms are excavated in the rock at a depth of 600 m below the surface of the earth, and emplacement holes are drilled in the floor of the storage rooms for the waste canisters. The holes can then be backfilled with crushed rock or other material and plugged with a shield plug of concrete or other suitable material.

Table 1. Isotopic content (Ci/kg) of 15-yr-old waste glass[a]

Isotope	Concentration	Isotope	Concentration	Isotope	Concentration	Isotope	Concentration
C-60	3.20E-02	Sn-123	5.58E-13	Nd-144	3.40E-13	Np-237	6.19E-06
Se-79	9.52E-05	Sn-126	1.05E-05	Pm-147	1.20E-00	Pu-236	3.88E-06
Rb-87	6.31E-09	Sb-124	2.73E-29	Pm-148	1.16E-40	Pu-237	2.56E-39
Sr-89	2.56E-29	Sb-125	4.67E-02	Pm-148m	1.68E-39	Pu-238	4.87E-01
Sr-90	1.67E 01	Sb-126	1.48E-06	Sm-147	1.72E-09	Pu-239	4.98E-03
Y-90	1.67E 01	Sb-126m	1.06E-05	Sm-148	3.92E-15	Pu-240	3.13E-03
Y-91	1.13E-25	Te-125m	1.09E-02	Sm-149	1.22E-15	Pu-241	3.68E-01
Zr-93	1.27E-03	Te-127	4.94E-15	Sm-151	1.53E-01	Pu-242	4.17E-06
Zr-95	1.13E-22	Te-127m	5.05E-15	Eu-152	1.56E-03	Am-241	1.49E-02
Nb-95	2.45E-22	Te-129	1.84E-48	Eu-154	1.95E-01	Am-242	9.55E-06
Nb-95m	1.44E-24	Te-129m	2.91E-48	Eu-155	8.13E-02	Am-242m	9.59E-06
Tc-99	1.72E-03	Cs-134	8.38E-02	Tb-160	4.85E-25	Am-243	4.03E-06
Ru-103	1.37E-39	Cs-135	4.17E-05	Tl-208	1.58E-06	Cm-242	7.91E-06
Ru-106	1.09E-03	Cs-137	1.78E 01	U-232	4.76E-06	Cm-243	3.04E-06
Rh-103m	2.73E-39	Ba-137m	1.68E 01	U-234	3.24E-06	Cm-244	7.76E-05
Rh-106	1.09E-03	Ce-141	3.99E-48	U-235	3.77E-08	Cm-245	4.65E-09
Po-107	6.39E-06	Ce-142	6.66E-09	U-236	8.13E-07	Cm-246	3.70E-10
Ag-110	3.84E-07	Ce-144	9.39E-04	U-238	2.03E-07	Cm-247	4.56E-16
Cd-115m	1.43E-37	Pr-144	9.39E-04	Np-236	1.21E-11	Cm-248	4.76E-16
Sn-121m	1.99E-05	Pr-144m	1.13E-05				

[a]Reference canister contains 1480 kg of glass waste. Total activity, 70.5 Ci/kg.

The cansiter design selected for this study is one that was considered earlier at SRL[3] and is shown in Fig. 1. The canister itself is a cylinder composed of 304L stainless steel, with a wall thickness of 9.5 mm and an ID of 0.591 m. Transportation or other factors may impose a different diameter, but these variables have not been taken into account here. The canister is filled to a height of 2.28 m, and the waste form fills 0.625 m^3 of the canister. The nuclides are assumed to be uniformly distributed throughout the waste matrix. This matrix is assumed to be glass; the waste form density is chosen to be 2370 kg/m^3, and the waste form in the canister weighs 1480 kg. A thermal conductivity of 1 W/m K has been used for the waste form, which is typical for the conductivity of a glass waste form in the temperature range of interest.

The emplacement of the DHLW canister is assumed to take place 10 yr after reprocessing or 15 yr after discharge of the wastes from the reactor - conditions that provide conservative estimates of the temperatures. The thermal power generation of the reference DHLW is given in Table 2. Comparison with analogous thermal decay rates for SF and CHLW[2] shows that after the important fission products have been depleted in the first 30 to 50 yr, the thermal decay is quite rapid and heat generation is relatively much smaller than for CHLW.

Based upon projected SRP waste streams in the reprocessing plant, the loading of the reference waste in a single canister amounts to 310 W per canister at emplacement 15 yr after discharge from the reactor. This loading is less than for SF and CHLW, with registers at emplacement for 10-yr-old SF and CHLW that have commonly been considered for emplacement.[2]

The repository utilizes the room and pillar design that was used in the SF and CHLW studies. Storage rooms are assumed to be 5.5 m wide, and the pillar width between rooms used in the thermal calculations is assumed to be 18 m. This design provides a 23% mining extraction ratio, which is believed to be adequate for these openings in salt.[6] The baseline DHLW repository is loaded to 11.6 W/m^2 at emplacement in a two-row-per-room

ORNL-DWG 80-1802R

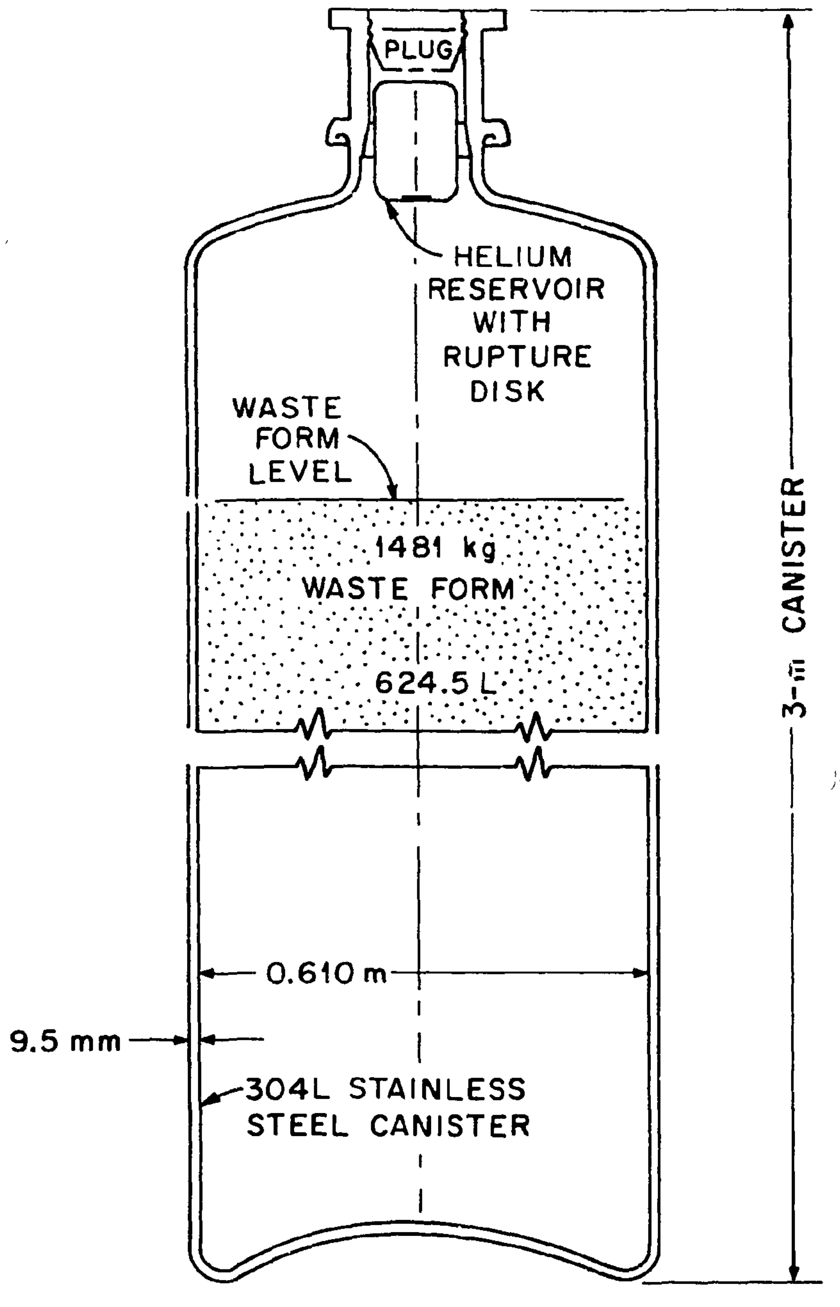

Fig. 1. Glass canister configuration.

Table 2. Thermal power generation for reference SRP-DHLW[a]

Time out of reactor	W/canister	Relative power generation from emplacement
5 (reprocessing)	540.6	-
10	358.5	-
15 (emplacement)	310.4	1.0
20	275.0	0.886
25	245.0	0.789
30	218.8	0.705
35	195.6	0.630
40	175.1	0.564
50	140.7	0.453
75	82.83	0.267
100	50.22	0.162
200	10.26	0.033
300	4.057	0.013
400	2.289	7.3E-03
500	1.521	4.9E-03
1,000	0.656	2.1E-03
1,700	0.476	1.5E-03
2,000	0.402	1.3E-03
5,000	0.298	9.6E-04
10,000	0.239	7.7E-04
20,000	0.167	5.4E-04
50,000	0.0866	2.8E-04
100,000	0.0550	1.8E-04
200,000	0.0460	1.5E-04
500,000	0.0264	8.5E-05
1,000,000	0.0090	2.9E-05

[a]E.I. duPont deNemours and Company, DPSTD-77-13-3 (1980). Further information found in ref. 5.

configuration. This loading is considerably lower than for either the SF or the CHLW case and is due to the physical limitation of a minimum separation distance between emplacement holes. The holes in the baseline repository are assumed to be 0.76 m in diameter to allow for the large DHLW cansiters. A separation between closely spaced drill hoels of at least two hole diameters would reasonably assure integrity of the rock between the holes. Therefore, the minimum center-to-center separation distance is 2.28 m, leading to a maximum local thermal loading of 11.6 W/m^2 at emplacement in the specified configuration. A canister loading of 310 W has been used in this estimate. A three-row configuration could be used to increase this loading to 17.4 W/m^2, but this design would require larger storage rooms and give an extraction ratio of nearly 40% for this loading. The three-row configuration is therefore not considered further here.

The canisters are inserted into the emplacement hole; the hole is then backfilled with crushed rock salt and plugged with concrete. The hole is sufficiently large to hold overpacked canisters should an overpack be necessary. Since the thermal conductivity of the backfill is low, the overpack has not been taken into account in order to provide conservative estimates of the waste temperatures. In addition, reconsolidation of the backfill has not been considered here to provide an additional element of conservatism. The baseline repository design utilized in this analysis is summarized in Tables 3 and 4. A comparison, with typical designs for repository for SF and CHLW, is also made in these tables. The thermal properties are given in Table 5. The assumed ambient temperatures in the rock are those indicated in Table 6.

The models used for the thermal analysis have been described elsewhere.[2] Briefly, the thermal environments are categorized into three separate regimes: the far-field, the near-field and the very-near-field. Temperatures in the far-field are calculated in a model where the heat

Table 3. Salt repository characteristics

Characteristics	CHLW	SF	DHLW
Repository depth below surface, m	600	600	600
Thermal loading (at emplacement)[a]			
Local areal thermal loading, W/m^2	25	25	11.6
Average areal thermal loading, W/m^2	<25	15	<11.6
Canister thermal power, kW	2.16	0.55	0.31
Room description			
Room length, m	Very long	165	165
Room width, m	5.5	5.5	5.5
Room height, m	5.5	7.6	5.5
Adjacent pillar thickness, m	18.3	18.3	18.
Canister emplacement holes			
Rows per room	1	2	2
Row separation, m	-	1.67	2.28
Hole pitch (along row), m	3.66	1.67	2.28
Emplacement hole diameter, m	0.54	0.54	0.76
Canisters per hole	1	1	1

[a]Waste is 10 yr old at emplacement for CHLW and SF, and 15 yr old at emplacement for DHLW.

Table 4. Waste package characteristics

Characteristics	CHLW	SF	DHLW
Waste description			
Active length, m	2.4	3.7	2.28
Active volume, m^3	0.18	NA	0.625
Mass loading (MTHM equivalent)	2.1	0.46	NA[a]
Canister dimensions			
Outer diameter, m	0.324	0.356	0.610
Inner diameter, m	0.305	0.337	0.591
Length, m	3.0	4.7	3.0
Overpack dimensions			
Outer diameter, m	0.406	0.406	-
Inner diameter, m	0.381	0.381	-
Length, m	3.4	5.1	-
Backfill thickness, m	0.051	0.051	0.076
Liner dimensions			
Outer diamter, m	0.535	0.533	-
Inner diameter, m	0.508	0.508	-
Length (m)	5.5	6.25	-
Materials			
Waste	Glass	UO_2	Glass
Filler in canister	Air	Helium	Air
Canister	SS[b]	CS[c]	SS
Overpack	CS	CS	-
Backfill	CRS[d]	CRS	CRS
Hole liner	CS	CS	-
Emplacement hole plug	Concrete	Concrete	Concrete

[a]NA = not applicable or not available.
[b]SS = 304L stainless steel.
[c]CS = carbon steel.
[d]CRS = crushed rock salt.

Table 5. Thermal properties of materials

	Volumetric heat capacity (J/m^3 K)	Conductivity (W/m K)
DHLW waste form	2.0	1.0
304L stainless steel	3.6	16.4
Concrete	1.9	0.935
Salt	1.8	f(T)[a]
Crushed salt	1.5	0.347

[a]Thermal conductivity for salt is a function of temperature as follows:

Temperature (°C)	Thermal conductivity (W/m K)
0	6.11
50	5.02
100	4.20
150	3.60
200	3.11
250	2.77
300	2.49
350	2.30

Table 6. Ambient formation temperatures

Depth (m)	Temperature (°C)
0	15
600	38
3000	130

source is assumed to be distributed uniformly throughout the repository. The resulting temperatures are used as boundary conditions for the near-field model, where the detailed distribution of canisters in the repository is taken into account. This model is used to calculate temperatures in the rock near to the waste canisters. These temperatures are used as boundary conditions for the very-near-field model, which explicitly includes details of the waste package design and provides temperature predictions within the canister itself. All thermal calculations have been performed with the finite-differences conduction code, HEATING-5.[8]

The predicted maximum temperatures in the rock, in the canister wall, and in the waste for the baseline repository design are shown in Fig. 2. Maximum temperatures in the rock are ∿86°C and occur ∿30 yr after emplacement. The temperature history curve in Fig. 2 is that predicted at the edge of the emplacement hole at the midwaste level. The maximum canister wall temperature was calculated to be 94°C, and the maximum waste centerline temperature was 101°C.

If the restriction on canister separation is removed, the local areal thermal loading can be increased by changing the pitch of canisters in the row. Figure 3 shows the maximum rock, canister, and waste temperatures that are predicted to occur as the areal loading is changed in this way. As can be seen from this figure, the maximum rock temperature would be 140°C for a loading of 25 W/m^2 and is close to what would be obtained for SF in a two-row configuration and CHLW in a one-row configuration at this loading.[2]

For a fixed configuration and fixed average areal loading, the canister separation can still be modified by changing the separation between row in the room or by staggering the canisters in the two rows. This modification does not produce a large change in the maximum rock temperature, as can be seen in Fig. 4 (e.g., when the row spacing is reduced from 2.28 m to the 1.67-m spacing used in the SF design, the temperature varies <3%). This weak dependence is due to the small

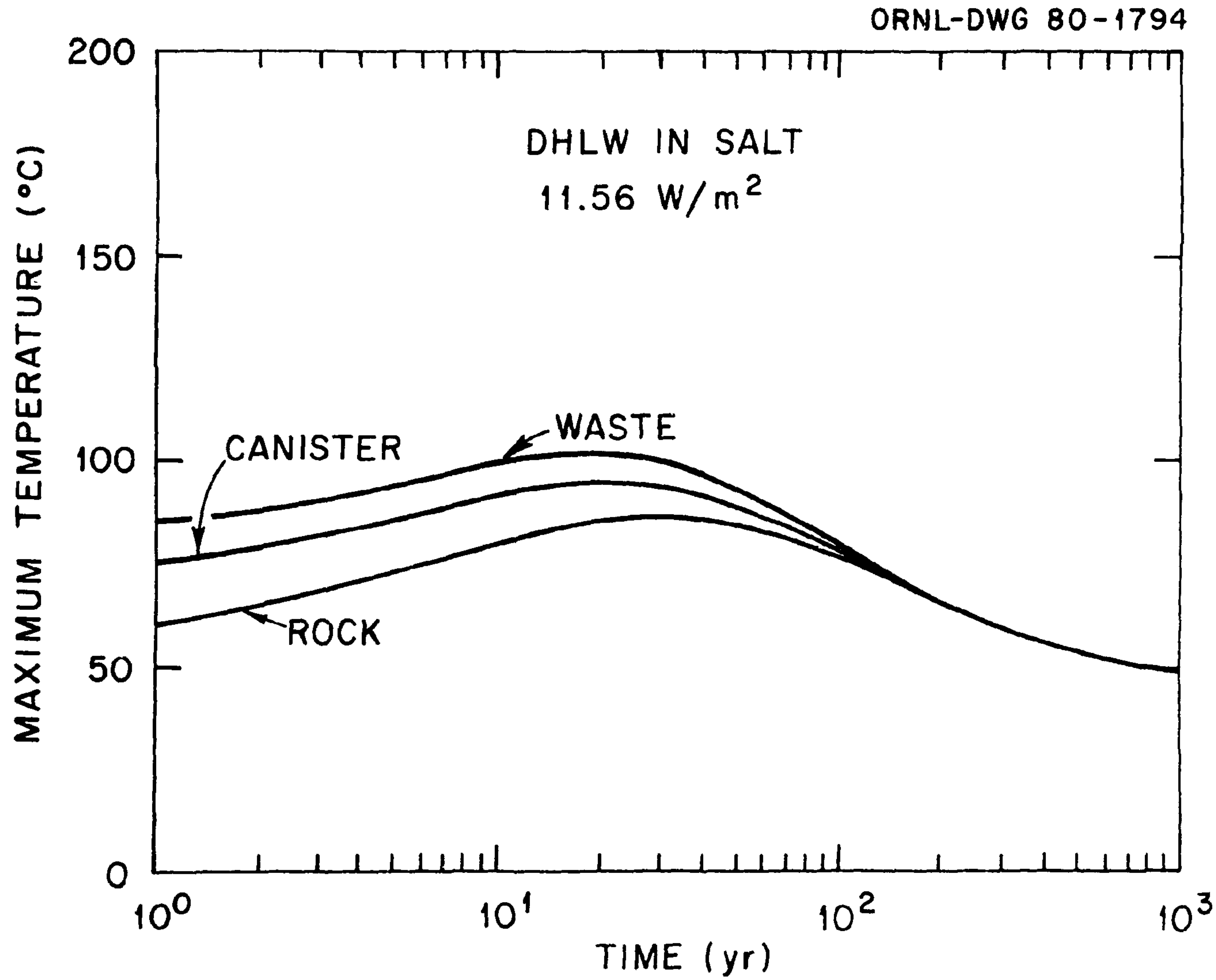

Fig. 2. Maximum near-field temperature histories for baseline repository.

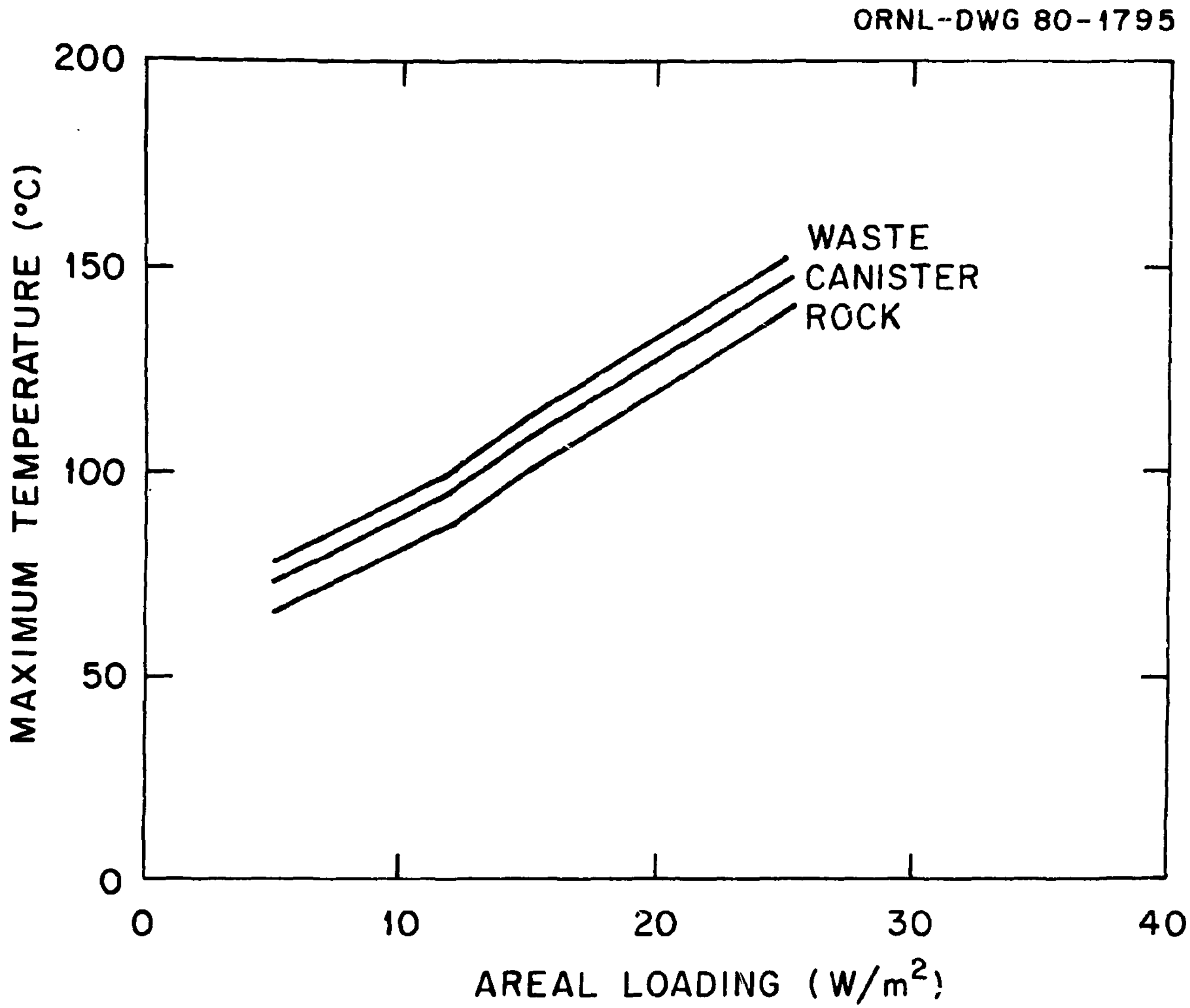

Fig. 3. Maximum near-field temperatures vs area thermal loading.

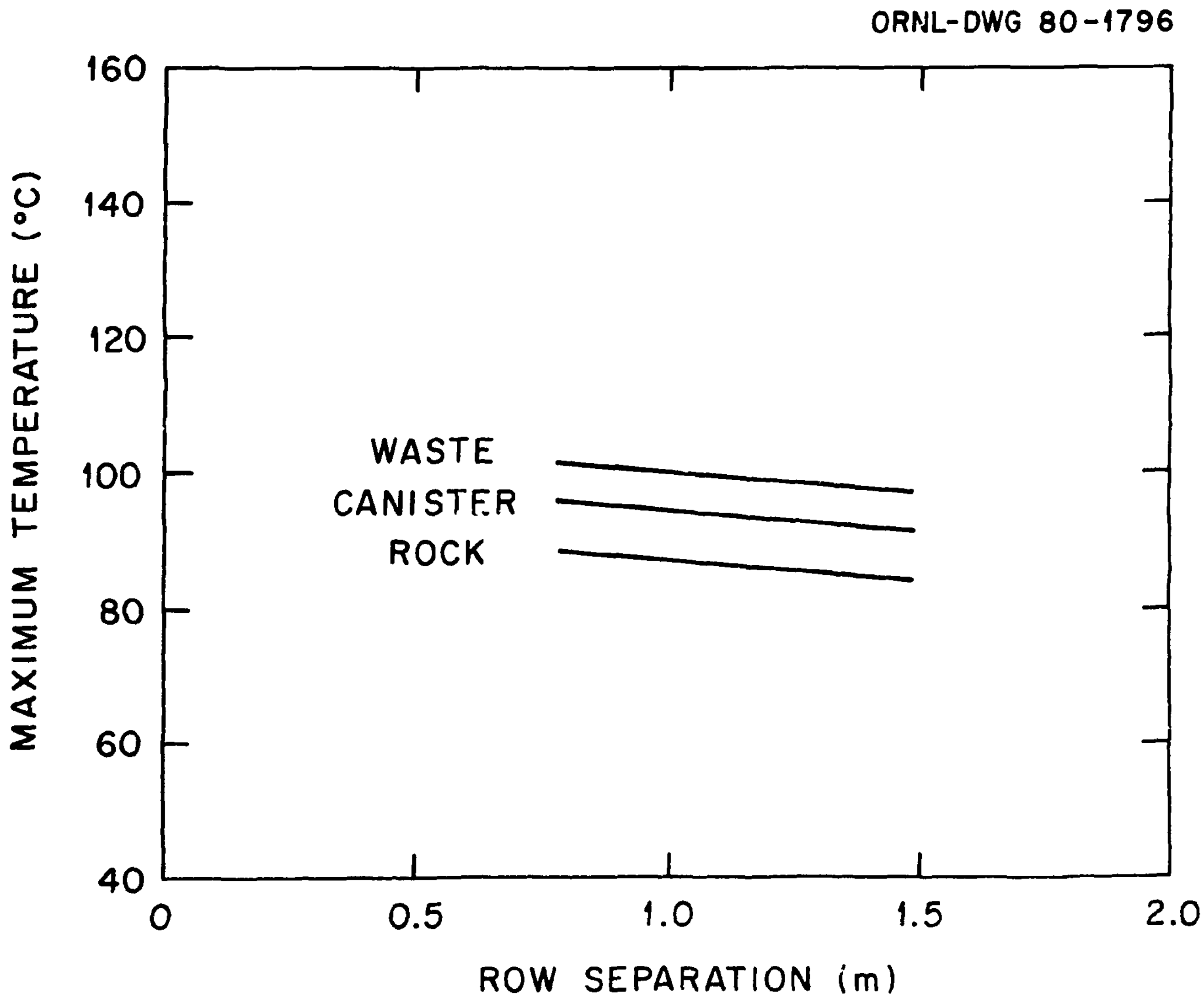

Fig. 4. Maximum near-field temperatures vs row separation.

canister loading for the DHLW. Greater sensitivity has been observed for the more heavily loaded SF and CHLW canisters.[2,9]

If other materials are used for backfilling the emplacement hole, the conductivity could be different and waste package temperatures could be affected. The dependence of the canister wall and waste centerline temperatures on the conductivity of the backfill in the emplacement hole is shown in Fig. 5. For a fixed areal loading, the thermal resistivity of this backfill has esentially a linear dependence on the temperature. For a backfill with the conductivity of consolidated salt, the resistivity is about 0.3 m K/W. In this case the maximum canister wall and waste centerline temperatures are reduced to about 87°C and 92°C respectively. As a comparison, the result of having no backfill material but which takes into account radiation between the canister and the emplacement hole wall is calculated to be 93°C and 99°C respectively.

Finally, the far-field rock temperatures are shown in Figs. 6 and 7. These predicted temperatures are the result of a repository with an average loading of 11.6 W/m^2. When the passive areas in the repository are taken into account, the average loading would be less than the local loading, but this variable has not been considered here. Figure 6 shows the predicted temperature profiles at selected times in the formation. Figure 7 shows the temperature histories at selected depths in the rock. The far-field temperatures are below those for SF and CHLW due largely to the rapid decay of the waste in the long term. For example, the temperature rise at a depth <500 m never exceeds 17°C.

3. BRINE MIGRATION

Brine inclusions in salt may migrate toward a heat source introduced into the rock. This migration has been estimated for a DHLW repository with the single-crystal model previously used for the estimates for SF and CHLW repositories.[2] In this model, single-phase brine inclusions migrate up the temperature gradient according to the Jenks equation.[10] Trapping along grain boundaries or multiphase effects that could reduce the flow rates to

ORNL-DWG 80-1797

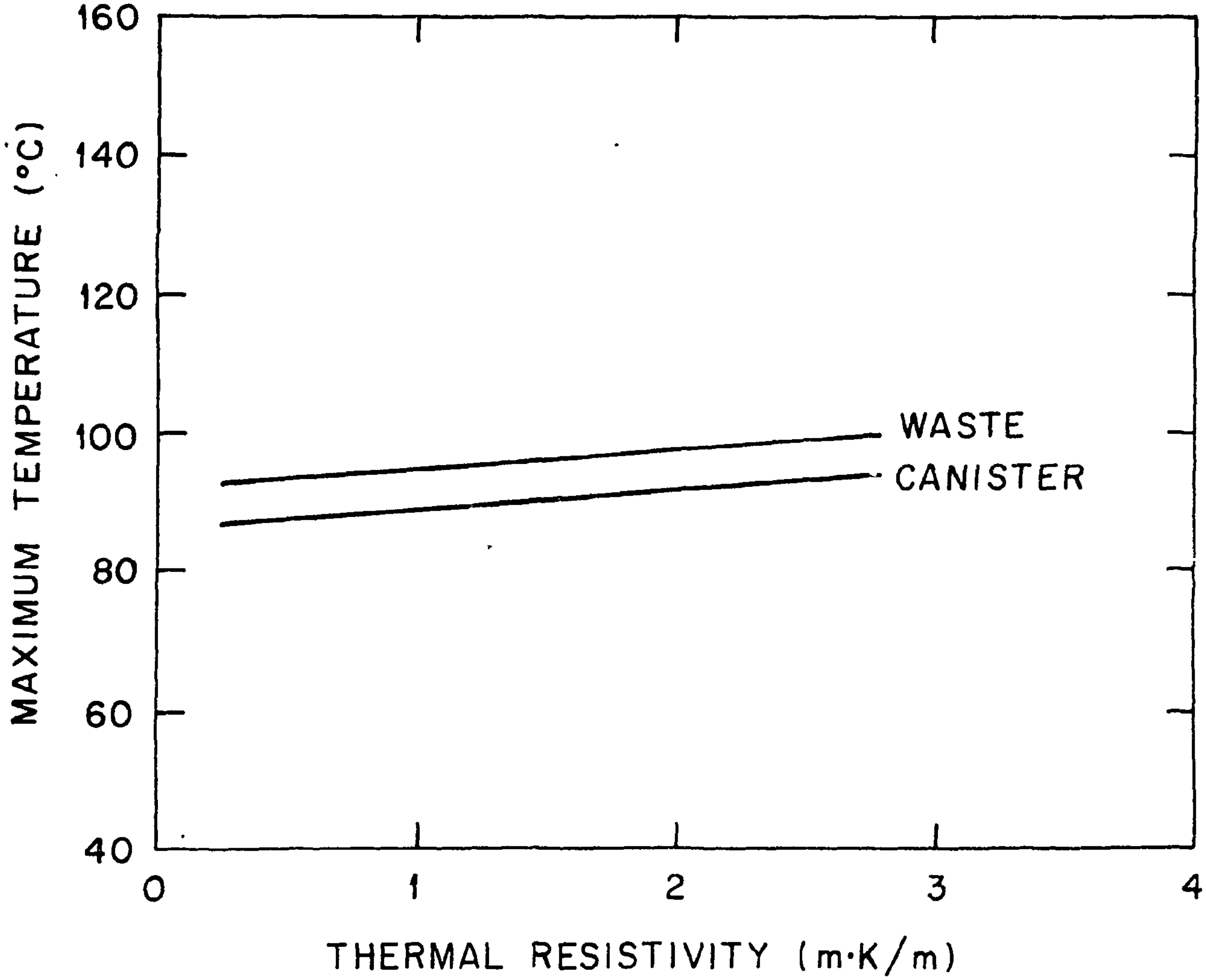

Fig. 5. Canister and waste temperatures vs backfill thermal resistivity.

ORNL-DWG 80-1798

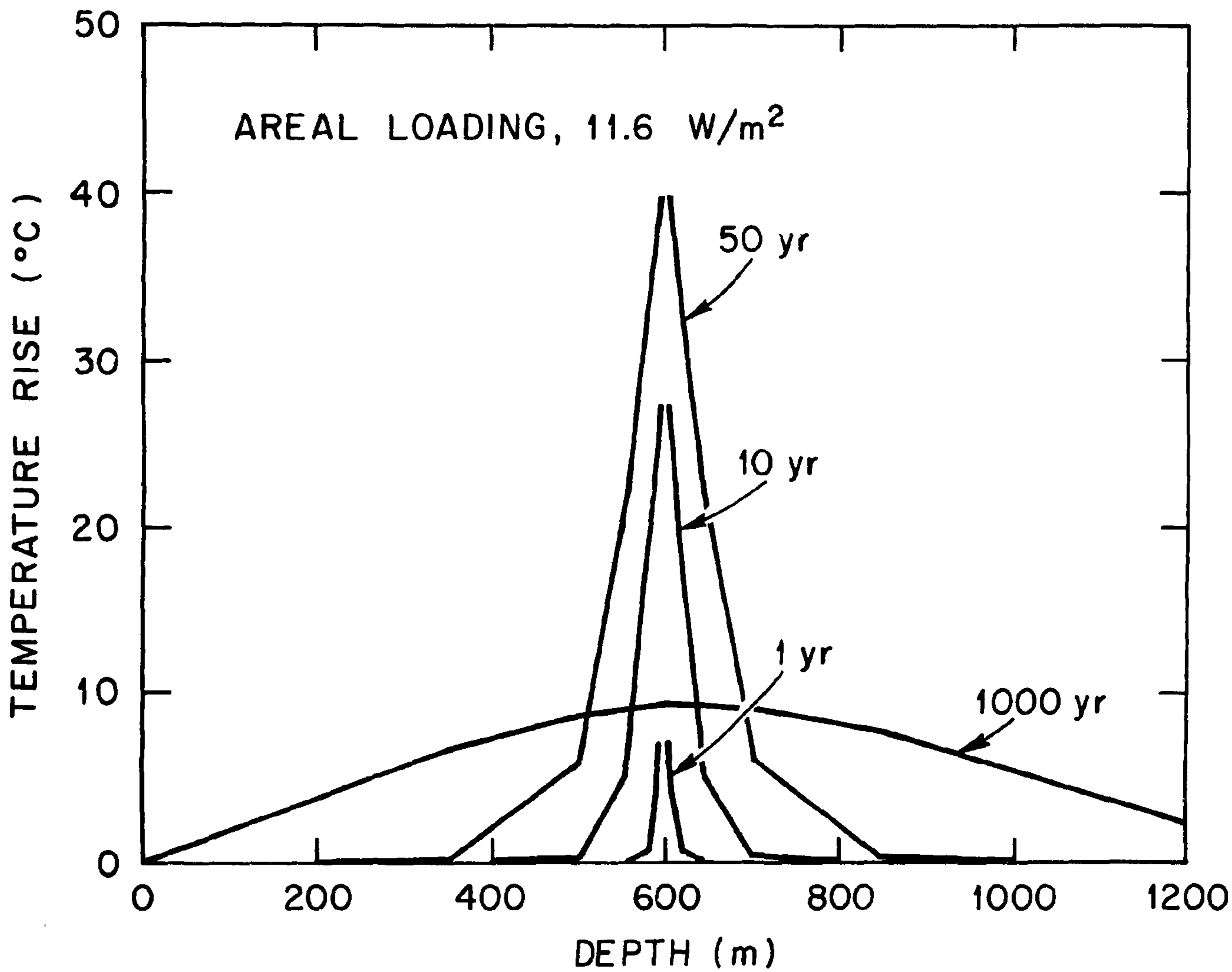

Fig. 6. Far-field temperature profiles.

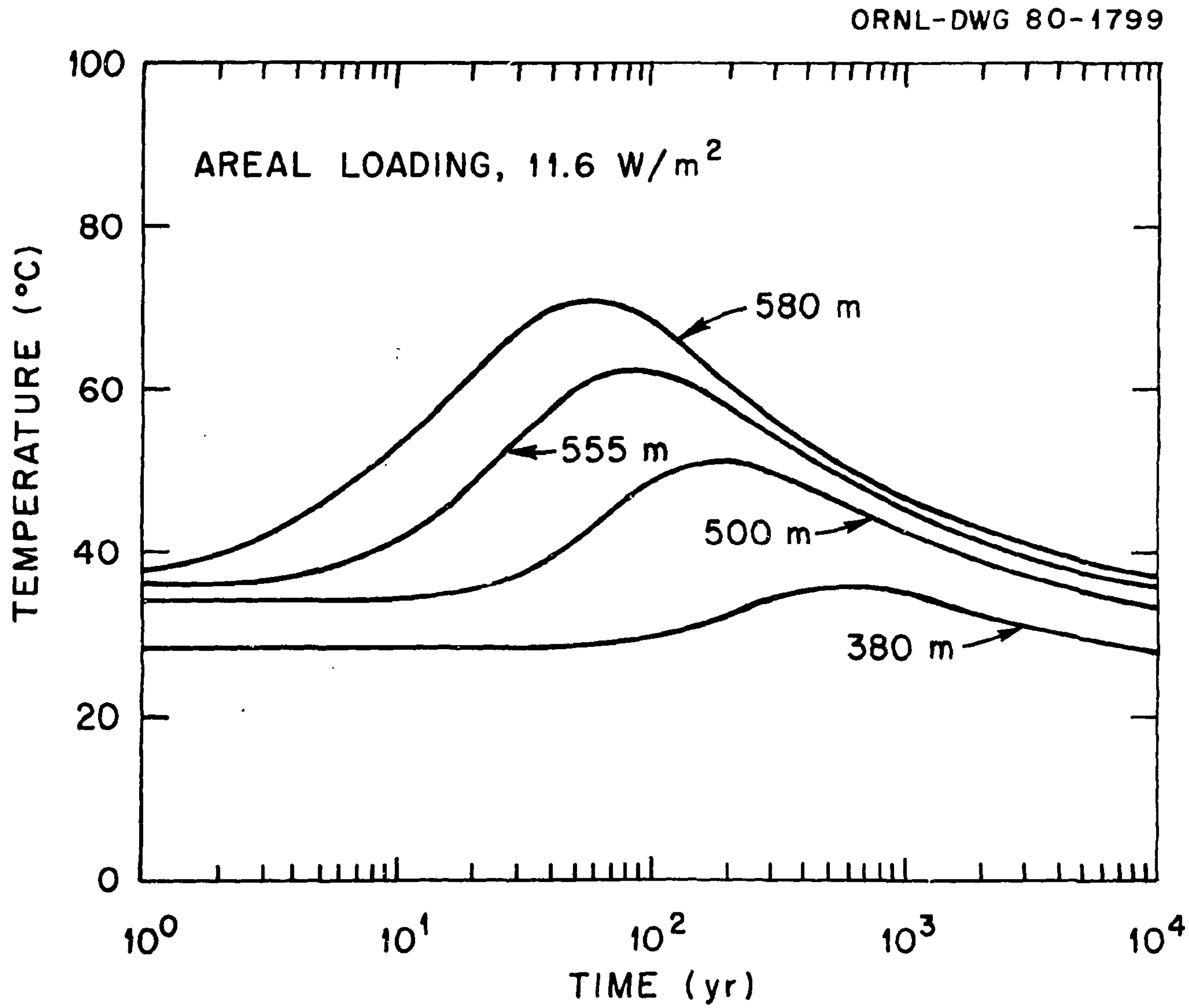

Fig. 7. Far-field temperature histories.

the emplacement hole are not considered here. The MIGRAIN code[2] has been used to predict the brine inflow to a given emplacement hole for an assumed 0.5 vol % of brine inclusions. The results are shown in Fig. 8. It can be seen that <0.5 L of brine is predicted to flow into the emplacement hole over a period of 1000 yr.

4. GAS PRESSURE

The increased temperature in the vicinity of the waste canister and the accumulation of brine in the emplacement hole can modify the gas pressure encountered by the waste package. The REPRESS model[2] has been used to estimate this pressure in the emplacement hole. Two scenarios have been assumed for the calculation. In the first scenario, it is assumed that the emplacement hole is sealed so that vapor cannot escape. Considering the 20% void space in the backfill, the volume available to the water is 190 L. In the second scenario, it is assumed that the hole is unsealed, and vapor is free to escape the emplacement hole and move to the backfilled storage room. In both cases, the hole is backfilled with crushed salt. Figure 9 shows the expected gas pressure histories for the emplacement hole in the baseline repository for the two scenarios. It can be seen that the maximum predicted pressure is only 3.6 atm, or 0.36 MPa, even for the most extreme scenario. From the analysis for CHLW, closure of the hole due to creep and reconsolidation of the salt is not expected to result in gas pressures greater than these values mentioned above. In both cases, it can be assumed that at some point mine closure will result in lithostatic stresses on the waste package.

5. NUGLEAR RADIATION ENVIRONMENT

The radiation emitted from the DHLW is absorbed as energy in the salt. The gamma radiation dose has been calculated using the one-dimensional ANISN shielding code.[11] The neutron flux would be negligible and has not been calculated. The gamma-ray energy groups used

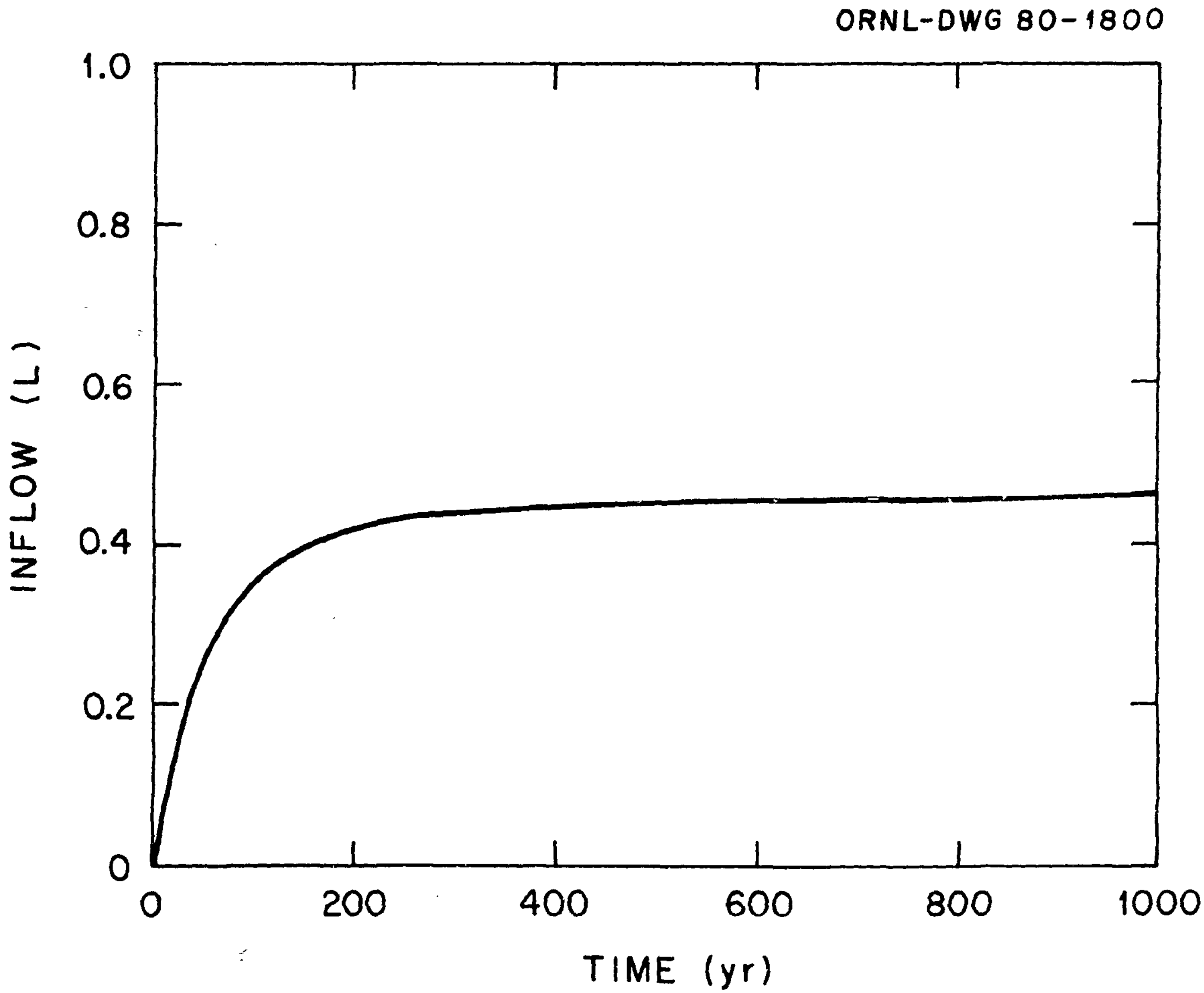

Fig. 8. Brine inflow to DHLW emplacement hole.

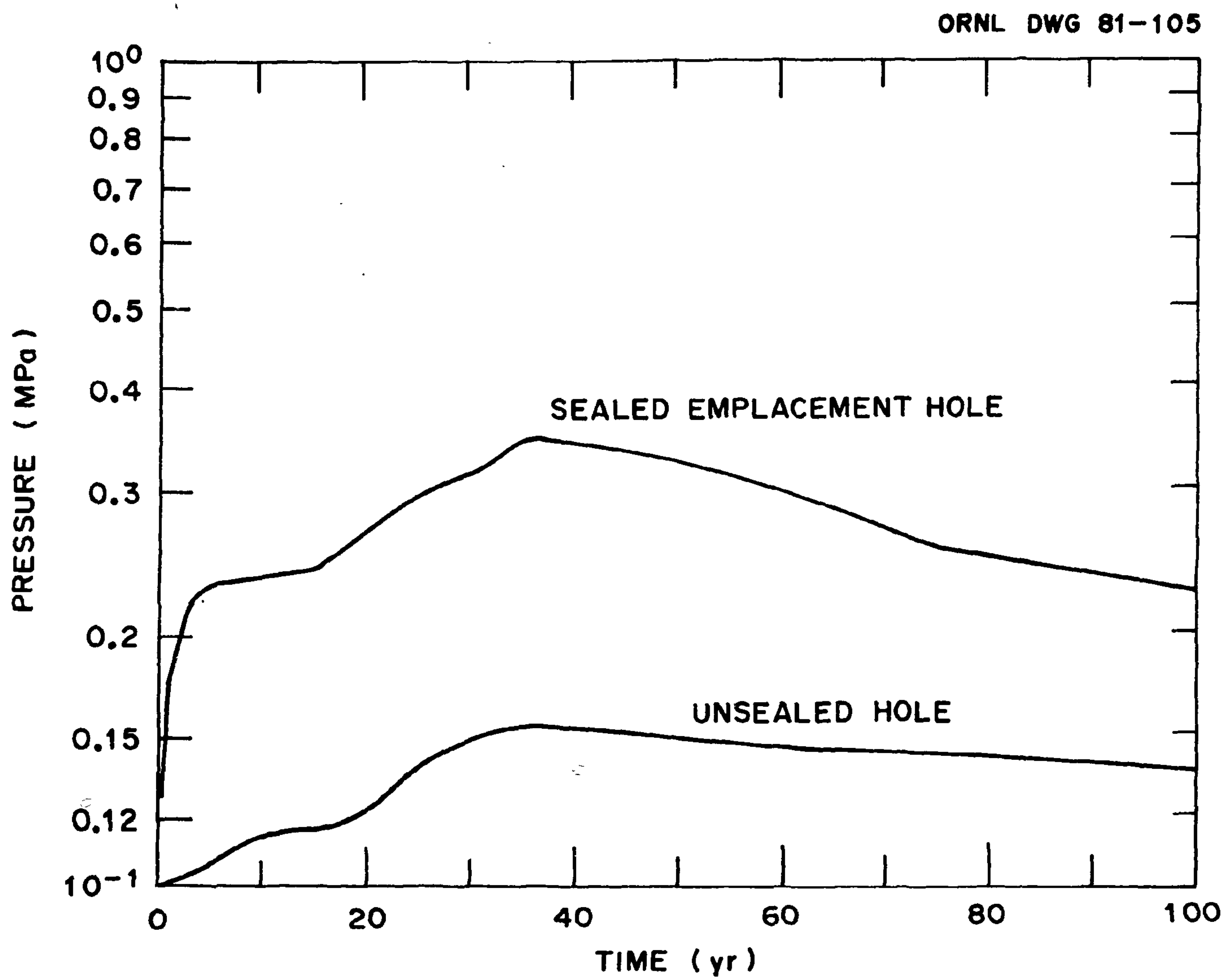

Fig. 9. Gas pressure history for emplacement hole.

in this calculation are given in Table 7 and were taken from ref. 5. The calculated dose rates are shown in Fig. 10 and indicate values for selected times after emplacement of the waste. The maximum absorbed dose rate at the edge of the emplacement hole is estimated to be 1400 rads/h, which is considerably less than the 10^5 and 4×10^3 values predicted for SF and CHLW, respectively.[2]

Table 7. X-ray energy groups (photons/cm^3-s)

Group	Energy range (MeV)	5.0 yr	10.0 yr	15.0 yr emplacement	20.0 yr	25.0 yr	30.0 yr	35.0 yr	40.0 yr
7	4.0000 - 3.0000	5.24369E-36	0.0	0.0	0.0	0.0	0.0	0.0	0.0
8	3.0000 - 2.5000	9.14690E 03	4.08492E 02	1.52609E 02	1.46710E 02	1.42481E 02	1.36630E 02	1.30443E 02	1.24365E 02
9	2.5000 - 2.0000	4.55107E 06	5.48706E 04	7.18133E 02	2.15443E 01	6.18180E 00	3.11086E 00	1.60699E 00	8.31072E-01
10	2.0000 - 1.5000	2.32510E 05	7.53314E 03	2.51968E 02	1.63948E 01	8.52409E 00	7.92965E 00	7.55592E 00	7.19743E 00
11	1.5000 - 1.0000	3.80742E 07	1.34262E 07	6.07912E 06	2.98981E 06	1.51781E 06	7.79707E 05	4.02276E 05	2.07873E 05
12	1.0000 - 0.7000	2.06284E 08	3.82974E 07	7.11883E 06	1.32360E 06	2.46159E 05	4.58321E 04	8.58357 03	1.65591E 03
13	0.7000 - 0.4500	1.98782E 09	1.54212E 09	1.33535E 09	1.18250E 09	1.05247E 09	9.37694[illegible] 08	8.35644E 08	7.44745E 08
14	0.4500 - 0.3000	1.69085E 07	4.75298E 06	1.33753E 06	3.77895E 06	1.08264E 05	3.25065E 04	1.12217E 04	5.24201E 03
15	0.3000 - 0.1500	3.99714E 06	1.12444E 06	3.17242E 05	9.03885E 04	2.66048E 04	8.64414E 03	3.56344E 03	2.10559E 03
16	0.1500 - 0.1000	6.66114E 07	1.15333E 065	1.25025E 05	4.06260E 04	1.91845E 04	1.26567E 04	1.02735E 04	9.14818E 03
17	0.1000 - 0.0700	9.53268E 06	1.17215E 05	6.96756E 03	5.08779E 03	5.08779E 03	4.86714E 03	4.67649E 03	4.50798E 03
18	0.0700 - 0.0450	8.34798E 06	4.86189E 05	4.94034E 05	5.71373E 05	6.31899E 05	6.78318E 05	7.13595E 05	7.40081E 05
19	0.0450 - 0.0300	6.73822E 06	9.28642E 05	2.66718E 05	9.19884E 05	4.28188E 04	2.87576E 04	2.45604E 04	2.31026E 04
20	0.0300 - 0.0200	1.63125E 04	2.51694E 04	3.20678E 04	3.74201E 04	4.15519E 04	4.47203E 04	4.71284E 04	4.89367E 04
21	0.0200 - 0.0100	5.29929E 06	5.20623E 06	5.09793E 06	4.97872E 06	4.85213E 06	4.72093E 06	4.58726E 06	4.45276E 06

Group	Energy range (MeV)	50.0 yr	75.0 yr	100.0 yr	200.0 yr	300.0 yr	400.0 yr	500.0 yr	1000.0 yr
8	3.0000 - 2.5000	1.12933E 02	8.86886E 01	6.96472E 01	2.64880E 01	1.00748E 01	3.83126E 00	1.45711E 00	1.16282E 02
9	2.5000 - 2.0000	2.22302E 01	8.22642E 03	3.04423E 04	5.70884E 10	1.07057E 15	2.00760E 21	3.76481E 27	8.73107E 56
10	2.0000 - 1.5000	6.52587E 00	5.10978E 00	4.00455E 00	1.51769E 00	5.76754E 01	2.19312E 01	8.34058E 02	6.65601E 04
11	1.5000 - 1.0000	5.55820E 04	2.05888E 03	7.79131E 01	7.24738E 01	2.97097E 01	1.27385E 01	5.95919E 02	1.40416E 02
12	1.0000 - 0.7000	1.23929E 02	5.82229E 01	4.92589E 01	2.80815E 01	1.93293E 01	1.55441E 01	1.37701E 01	1.13171E 01
13	0.7000 - 0.4500	5.91555E 08	3.32636E 08	1.87045E 08	1.87004E 07	1.86973E 06	1.87044E 05	1.88105E 04	1.16603E 02
14	0.4500 - 0.3000	3.09122E 03	2.91005E 03	2.92056E 03	2.92866E 03	2.92866E 03	2.93526E 03	2.94010E 03	2.94293E 03
15	0.3000 - 0.1500	1.52213E 03	1.34534E 03	1.25483E 03	1.06165E 03	9.80646E 02	9.43315E 02	9.24513E 02	8.93972E 02
16	0.1500 - 0.1000	7.95784E 03	6.70688E 03	6.29684E 03	5.99690E 03	5.88402E 03	5.78784E 03	5.70330E 03	5.39229E 03
17	0.1000 - 0.0700	4.21855E 03	3.76583E 03	3.25103E 03	2.25489E 03	1.78664E 03	1.55368E 03	1.42966E 03	1.23001E 03
18	0.0700 - 0.0450	7.73720E 05	7.95197E 05	7.80522E 05	6.74434E 05	5.77873E 05	4.95545E 05	4.25396E 05	2.02754E 05
19	0.0450 - 0.0300	2.19169E 04	1.96420E 04	1.79685E 04	1.28405E 04	1.01925E 04	8.73328E 03	7.85070E 03	6.00114E 03
20	0.0300 - 0.0200	5.12339E 04	5.27043E 04	5.17079E 04	4.44878E 04	3.79166E 04	3.23156E 04	2.75446E 04	1.24202E 04
21	0.0200 - 0.0100	4.18603E 06	3.56295E 06	3.02478E 06	1.62349E 06	9.48361E 05	6.10824E 05	4.31149E 05	1.51262E 05

ORNL-DWG 80-1801

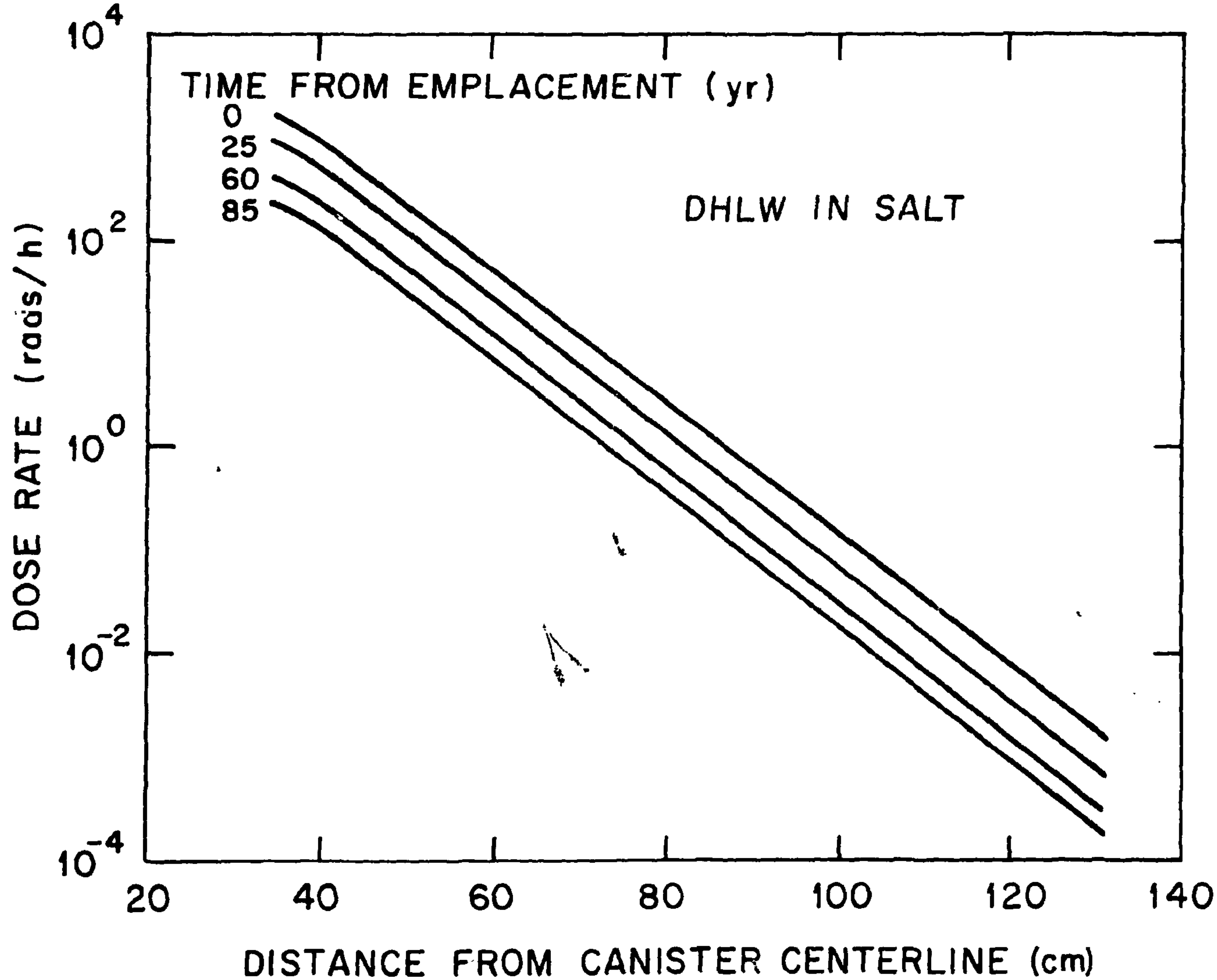

Fig. 10. Salt dose rate for DHLW.

6. REFERENCES

1. Interagency Review Group, Report fo the President on Nuclear Waste Management, TID-294442 (1979).

2. H. C. Claiborne, L. D. Rickertsen, and R. F. Graham, Expected Environments in High-Level Nuclear Waste and Spent Fuel Repositories in Salt, ORNL/TM-7201 (August 1980).

3. R. A. Gregg, Savannah River Laboratory, letter to H. C. Burkholder, Office of Nuclear Waste Isolation, Oct. 3, 1979.

4. N. E. Bibler, Savannah River Laboratory, letter to Reference Repository Conditions Committee members, June 11, 1980.

5. E. I. duPont deNemours and Company, Preliminary Technical Data Summary No. 3 for the Defense Waste Processing Facility, DPSTD-77-13-3 (April 1980).

6. Dames and Moore, Inc., Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations. vol. 20, Y/OWI/TM-36/20 (April 1978).

7. L. Obert and W. I. Duvall, Rock Mechanics and the Design of Structures in Rock, Chap. 16, Wiley, New York, 1967.

8. W. D. Turner, D. C. Elrod, and I. I. Simon-Tov, HEATING-5, An IBM 360 Heat Conductivity Program, ORNL/CSD/TM-15 (March 1977).

9. Science Applications, Inc., Storage Room Design Parameters for the ONWL Spent Fuel Disposal Study: Nine Case Studies in Salt, ONWI/78/SUB-E513-00415-1 (March 1979).

10. G. H. Jenks, Effects of Temperature, Temperature Gradients, Stress and Irradiations on Brine Inclusions in a Salt Repository, ORNL-5526 (July 1979).

11. W. W. Engle, Jr., A Users Manual for ANISN - A One-Dimensional Discrete Ordinates Transport Code with Anisotopic Scattering, UCND-K-1693 (1967).