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Radiation Doses in Granite Around Emplacement Holes in the Spent Fuel Test—Climax (Final Report)

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Radiation Doses in Granite Around Emplacement Holes in the Spent Fuel Test—Climax (Final Report)

Abstract

Final comparisons are made between measured and calculated radiation doses around the holes in which the spent fuel was emplaced in the Spent Fuel Test—Climax. Neutron doses were found to be negligible compared with gamma doses. Good agreement was found between the doses predicted by Monte Carlo calculations and those measured by short-exposure thermoluminescence dosimetry. Poor agreement was found between the calculational results and doses measured by exposure of LiF optical-absorption-type dosimeters for long periods, probably because of an inability to accurately correct for fade resulting from elevated temperature exposure over several months. The maximum dose to the rock occurred at the walls of the emplacement holes, and amounted to 1.6 MGy (1.6×10^6 rad) in granite for the emplacement period of nearly 3 years. It is recommended that dose evaluations for future high-level nuclear waste storage facilities also be performed by combining calculations and dosimetry. Passive dosimetry techniques, if used, should involve short exposures, so that laboratory calibrations can be performed with duplicate time, temperature, dose rate, and dose parameters. An attractive alternative would be to use active ionization chambers, inserted only periodically. These could be calibrated under appropriate temperature and pressure conditions, and could be read directly.

Introduction

As part of the National Waste Terminal Storage Program and the Nevada Nuclear Waste Storage Investigations Project, the Lawrence Livermore National Laboratory (LLNL) conducted the Spent Fuel Test—Climax (SFT—C).¹ The overall objective of the SFT—C was to evaluate the feasibility of safe and reliable short-term storage of spent reactor fuel assemblies at a plausible repository depth in a typical granitic rock, and to retrieve the fuel afterwards. An additional objective was to evaluate the differences, if any, between the effects of actual radioactive waste sources and electrical heater simulators (having essentially the same power output) on significant properties of the rock and other components.

Accordingly, LLNL emplaced 11 canisters containing spent fuel from the Turkey Point Unit 3 reactor in holes drilled into the Climax quartz-monzonite formation at the Nevada Test

Site during April and May of 1980, and removed them during March and April of 1983. Six canisters incorporating electric heaters were also emplaced and removed at the same time. Measurements were made of temperatures, ventilation parameters, rock stresses, displacements, acoustic emission, and wave propagation. These measurements have been compared with the results of calculations where possible.^{2, 3}

In the early design of the Climax test facility, no special provisions were made for placing dosimeters in the near-field environment of the emplacement holes. As peer reviews and discussions were held, it became clear that a quantitative evaluation of the radiation dose to the rock would be desirable in interpreting any radiation effects that might later be observed. Accordingly, the Test management held discussions with personnel from LLNL's Hazards Control Department and

E.G.&G., Inc.'s, Energy Measurements Group to determine the best approach. Since the degree of accuracy of dose calculations for this application was not then known, the Test management decided that the most prudent method would be to use *in-situ* dosimeters during the entire duration of the fuel assembly emplacement in the Climax facility. They wanted to include measurements of the peak dose to the rock, which would occur at the walls of the emplacement holes. This introduced three important technical problems: first, the temperature was expected to rise significantly at these locations and to vary over the course of the test, thus presenting problems for temperature-sensitive dosimeters; second, the total doses would be beyond the range of commonly used dosimeters; and third, the only accessible locations for dosimeters were small-diameter tubes originally intended to be used for thermocouples. These would only permit insertion of very small dosimeter capsules. Nevertheless, E.G.&G. accepted the challenge and agreed to perform the work on subcontract, as is discussed later.

After this work was underway, I was contacted for assistance in developing a plan to study radiation effects, which was published in April 1980.¹⁰ Meanwhile, Hanford Engineering Development Laboratory (HEDL) had proceeded with ORIGEN2 calculations of the expected heat output of the fuel assemblies, and had reached good agreement with calorimetric measurements at EMAD.¹¹ This encouraging development suggested that these same ORIGEN2 calculations could provide the source terms for a radiation transport cal-

culation to predict the doses to the rock at Climax. I therefore suggested that this approach be used, and that the results be compared with those from *in-situ* dosimetry. This was done, as is discussed later. As the results of the long-duration optical absorption dosimetry began to come in, it became clear that they contained significant inconsistencies and that they differed from the calculated results, as is discussed below. We suspected that the problem lay chiefly in the corrections for sensitivity at the exposure temperature and for fade during the long-term heating. While E.G.&G. made efforts to improve these corrections, I suggested that the long-duration optical absorption dosimetry be supplemented with short-duration thermoluminescence dosimetry, because it would be less subject to fade and easier to calibrate. In addition, if the calculations were correct for the relative contributions of the various radionuclides, it was no longer necessary to have dosimeters in place during the entire rock exposure to determine the total dose to the rock, since the decay could be easily calculated using well-known half-lives. The neutron dose had been measured and found to be insignificant. This approach was added to the E.G.&G. work already in progress, and is discussed later.

The first results from the measurements were compared with the calculated results in an earlier report.¹² This report compares all the measured results with the calculated ones, gives final estimates of the doses for each emplacement hole, and recommends techniques for evaluating doses in future waste storage facilities.

Monte Carlo Calculations

Monte Carlo calculations¹³ using the MORSE-I code¹⁴ were carried out at LLNL, using source terms provided by HEDL.¹¹ These source terms had been calculated using the ORIGEN2 code,¹² and were the same as those used by HEDL in conjunction with the calorimetry they performed,⁷ which provided the heat source term for the Climax thermal calculations. A later HEDL report¹³ has shown very good agreement between results of ORIGEN2 calculations and measured heat output of fuel assemblies for longer cooling times and for other pressurized water reactors in addition to Turkey Point Unit 3.

For the Monte Carlo calculations, only the four principal gamma-emitting radionuclides were considered. Neutron emission was assumed to

have a spontaneous fission spectrum, and the fuel-element geometry was modelled as shown in Fig. 1. We assumed the emplacement geometry to be perfectly concentric.

Calculations showed the neutron dose to a radius of 0.76 m to be negligible compared with the gamma-ray dose. Within the statistical uncertainty of the calculation, the dose was independent of vertical location over the central 2.44 m of the fuel assembly.

The absorbed dose-rate results, as a function of radius in the rock for the central 2.44 m of the fuel assembly, turned out to be essentially straight lines on semilog plots. We found that less than 3% of the total heat output of the assemblies was deposited in the rock as gamma rays.

Figure 1. Approximate geometry of spent-fuel emplacement.

The dose rates decreased with time as a result of radioactive decay and exhibited more negative slopes with radius as time progressed, presumably as a result of the softening of the spectrum as the shorter-lived emitters of higher energy gamma rays decay more rapidly.

The highest dose rate calculated for the rock was at the surface of the emplacement hole at a time of 2.45 years out-of-reactor, and amounted to

42.5 Gy/h in granite (4,250 rad/h). The dose over a 3-year period at this location was calculated to be 1.6 MGy in granite (1.6×10^6 rad). The dose on a granite basis came out about 9% higher than the dose on an LiF basis because of the differences in gamma-ray absorption cross sections. The overall uncertainty of the calculated results was estimated at $\pm 25\%$. The details of the calculations were described in an earlier report.⁴

Pre- and Post-Test Thermoluminescence Dosimetry

R. B. Davis of HEDL reported measurements of the exposure dose rate taken near assembly DO4 in a hot cell at Battelle Columbus Laboratory on September 12, 1979, before the assembly was shipped to the Nevada Test Site for use in the Climax test.¹⁴ These measurements made use of thermoluminescence dosimeters, which had been prepared and calibrated by the Pacific Northwest Laboratory Dosimetry Technology Section. The exposure time was 10 min. The average of the results for eight positions spaced along the central fueled portion of the assembly at contact was 9.11×10^4 R/h.

After the fuel assemblies were removed from the Climax facility and brought to the E-MAD (engine maintenance, assembly, and disassembly) facility, another set of thermoluminescence dosimetry measurements was performed, this time on assembly D34.¹⁵ These latter exposures were performed by the Westinghouse E-MAD staff on March 28, 1984, using dosimeters supplied and later read by Carl Sundbeck of the LLNL Hazards Control Department, Special Dosimetry Section.¹⁶ The dosimeters were Harshaw TLD-700 LiF chips, with an advertised ⁷Li isotopic composition of 99.93%. Three of these chips were mounted in each of 24 standard LLNL plastic dosimetry packets (disks 36 mm in diameter by 3 mm thick, containing three recessed wells for the chips, which are held in place by thin plastic windows).

The dosimeters were shipped and stored in lead shields before exposure, and were hand-carried from E-MAD to LLNL after exposure. Six of the packets were exposed to the fuel assembly without beta shields, and six were exposed with 1.6-mm-thick aluminum beta shields. These 12 were individually wrapped in a thin polyethylene sheet to prevent contamination, and were then individually taped to an aluminum block to facilitate remote handling. They were exposed separately for 10 minutes each at contact with the

center of one of the four flat sides of the assembly, and then rapidly moved to a shielded box and taken out of the hot bay. They were spaced vertically at intervals of 305 ± 25 mm (12 ± 1 in.), with the shielded and unshielded dosimeters alternating. See Fig. 2.

The assembly was held in a vertical position by a crane during the measurements, away from large amounts of material that could scatter gamma rays. The exposures were carefully timed with a stopwatch. Three additional wrapped dosimeter packets were brought into only momentary contact with the assembly and then retracted, to measure the exposure that resulted from handling within the hot bay. The maximum exposure observed from handling was only 18 R, compared with a minimum of 2133 R observed for the 24 dosimeters receiving both handling and a 10-min exposure. For the majority of the dosimeters, the handling correction was less than 0.5%.

Two other packets were fitted with thermocouples and placed in contact with the assembly for 11 min, one with and the other without a beta shield, while the temperature was monitored. The maximum temperature was reached by the unshielded packet, and amounted to less than 30°C, compared with the hot bay ambient temperature of about 23°C. No temperature corrections were made because of these low values.

The other, unused packets were shipped and stored along with the exposed packets in order to detect any unplanned exposure. Only small background doses were found on these unexposed dosimeters, and no corrections were found to be necessary.

The average of the results from the four shielded packets in the central, fueled portion of the assembly was an exposure dose rate of 2.33×10^4 R/h on March 28, 1984, after correction for the handling dose but before correction for gamma attenuation in the aluminum shields.

Figure 2. Locations of dosimeter holders.

Long-Duration Optical Absorption and Neutron Activation Dosimetry

As mentioned above, this work was performed by E.G.&G., Inc.'s Energy Measurements Group of Goleta, California.¹⁷⁻¹⁹ It included optical-absorption measurements after *in-situ* exposure of ^7LiF crystals, and gamma counting of metal foils and capsule parts after *in-situ* neutron activation.

The ^7LiF dosimeters and neutron-activation foils were encapsulated and inserted into stainless steel tubes located on the inside surface of the emplacement holes in the rock (radius = 0.30 m), as well as at radii of 0.51 and 0.66 m in the rock. They were left in position for periods ranging from about 7.5 to 16.5 months, and were then removed. The optical absorption of the ^7LiF chips was measured at four wavelengths, corresponding to peaks produced by the various color centers. Since the peaks grow sequentially as the absorbed dose is increased, the plan was to obtain continuous coverage over the relevant range of doses using different peaks for different dose ranges.

Because it was anticipated that the peaks would fade as the dosimeters were held at elevated temperatures over these long periods, E.G.&G. performed a series of laboratory experiments that involved furnace anneals of previously room-temperature-irradiated dosimeters and periodic optical-absorption measurements. They hoped that this procedure would provide fade-correction factors that could be applied to the results from the *in-situ* dosimeters once their temperatures had been measured. We recognized that laboratory experiments would not provide an exact simulation of the *in-situ* conditions, since the laboratory dose was delivered beforehand at room temperature rather than continuously at rock temperature, but this was not thought to be important, based on published information, which indicated the dosimeter response to be independent of irradiation temperature.²⁰

However, since the initial *in-situ* results were found to be self-inconsistent (i.e., different peaks indicated different doses for the same dosimeter), this assumption had to be re-examined. Laboratory experiments were then performed that indicated that the peak height for a given dose and color center was indeed sensitive to the irradiation temperature. This led to the laboratory generation of 60°C calibration curves and the elimination of all but two absorption peaks from consideration.

Improvement of the spectrophotometry equipment made it possible to obtain continuous coverage over the relevant dose range with these two peaks, and it was hoped that use of the 60°C calibration curves, combined with the fade corrections determined earlier, would provide a means of compensating for the time-temperature history of the *in-situ* dosimeters.

The neutron-activation dosimetry involved measurement of four isotopes: ^{60}Co in cobalt foils, ^{110m}Ag in silver sealing rings, and ^{54}Mn and ^{58}Co in stainless-steel parts. Neutron activation does not suffer from temperature effects, but in this case, there were overall uncertainties of at least $\pm 30\%$ because of statistical uncertainties resulting from low counting rates and spectrum unfolding.

When the previous report⁸ was written comparing the measured with the calculated results (October 1982), there were two sets of long-duration optical absorption data and two sets of neutron activation data, representing the periods from April 1980 to January 12, 1981, and from January 12, 1981, to October 23, 1981.

In general, the spatial and time dependencies of the optical absorption data were found to be in good agreement with the calculated results. However, the absolute values in some cases were outside of the expected error limits ($\pm 25\%$). I suggested that the agreement might be improved if the 60°C calibration curve used by E.G.&G. were extended into the relevant dose range. They had been unable to do this previously because of the dose rate limitation of the available calibration source, and had used what were believed to be data for 25–30°C. The E.G.&G. experimenters agreed to improve this.

The neutron activation data showed the neutron dose rate to be about five orders of magnitude less than the gamma dose rate at the wall of the hole. This was in general agreement with the calculations, which indicated that the neutron dose rates were negligible compared with the gamma dose rates.

Since the previous comparison report⁸ was written, E.G.&G. has recovered and read out the final set of optical absorption dosimeters from the Climax facility, performed a new calibration exposure to take account of temperature effects, and recalculated the doses for the previous dosimeter sets.¹⁹ The results for all three dosimetry periods are compared with the calculated results in a later section of this report.

Short-Duration Thermoluminescence Dosimetry

The E.G.&G. experimenters decided to use both CaF_2 and MgB_2O_7 thermoluminescence dosimeters initially.²¹ They were calibrated at 60°C over the dose range expected for a one-hour exposure, and were incorporated into the same assemblies used for the long-duration dosimetry. They were exposed for 59-60 minutes on August 13, 1982. The results were compared with calculated values in the previous report.² The measured doses were below the calculated doses at the 0.51-m radius, but were above them at the wall position and at the 0.66-m position. There was considerable spread in the data, and some of them lay outside the $\pm 25\%$ error limit. I suggested that

more careful duplication of dose rate, dose, temperature, and waiting-time histories of the calibration and *in-situ* measurement runs could improve the agreement in the future.

Since the writing of the previous report, E.G.&G. has performed another set of TLD exposures, this time on March 8, 1983, using only CaF_2 dosimeters.²² (No significant differences had been seen between the results from the CaF_2 and the MgB_2O_7 dosimeters in the first set of exposures, and some of the MgB_2O_7 dosimeters were found to show large changes in sensitivity with continued use.) Special care was taken to duplicate the calibration and *in-situ* runs in every respect believed to be important. A comparison of results with calculated predictions is given in the next section.

Comparison of Calculations and Measurements

Pre- and Post-Test Thermoluminescence Dosimetry

As noted above, the measurements by HEDL and Battelle¹⁴ on September 12, 1979, produced an average value at contact with assembly DO4 of 9.11×10^4 R/h, which is equivalent to an absorbed dose rate of about 0.21 Gy/s in LiF. The value predicted by our Monte Carlo calculations⁹ for the same fuel assembly and time out-of-reactor was 0.211 Gy/s (7.6×10^4 rad/h). Considering the uncertainties involved and the fact that the scattering environments were different, this excellent agreement, though encouraging, must be regarded as partly fortuitous.

As was also noted above, the measurements performed by LLNL and Westinghouse E-MAD¹⁵ on March 28, 1984, produced an average value of 2.33×10^4 R/h at contact with assembly D34. Correcting for the gamma attenuation in the aluminum beta shields and for the differing mass energy absorption coefficients of LiF and air, we find this to be equivalent to 0.054 Gy/s in LiF. This is to be compared with a value of 0.054 Gy/s determined from the radiation transport calculations.⁹ This agreement is again excellent, especially considering the estimated uncertainties ($\pm 25\%$ for

the calculations and $\pm 11\%$ for the dosimetry) and the different scattering environments.

In-Situ Long-Duration Optical Absorption Dosimetry

The E.G.&G. experiments have provided three sets of measurements, covering the periods from April 1980 to January 12, 1981, from January 12, 1981, to October 23, 1981, and from October 23, 1981, to March 8, 1983. Because the calculated dose-rate results had not been integrated over these same time periods, it was necessary to perform appropriate integrations before comparisons could be made. This was done by fitting the calculated dose rates to functions of the form $D = D_0 t^n$ for each location of interest. These functions were then integrated analytically over the appropriate time periods, and corrections were applied for the fuel loading and burn-up of each assembly. This procedure produced calculated doses on an LiF basis for the emplacement geometry without dosimetry tubes.

Because the E.G.&G. results were given for locations inside the dosimetry tubes, whereas the calculations were made for undisturbed rock without tubes, it was necessary to adjust the

E.G.&G. results to place them on the same basis as the calculations. For the wall position, it was only necessary to correct for attenuation in the 0.5-mm-thick stainless steel dosimetry tube. This produced a multiplying factor of 1.034. For the 0.51- and 0.66-m positions, it was necessary to correct for this attenuation as well as for the lack of attenuation of the granite that would have been present if the dosimeter tubes had not been installed. The combined multiplying factor was 0.962 for these locations.

The comparisons between the measurements and the calculations are shown in Tables 1-3 for all locations where comparisons could be made. In addition, the comparative results as a function of radius are plotted for Hole CEH3 in Figs. 3-5. In these figures, the dashed lines indicate the $\pm 25\%$ error limits estimated for the calculations. Error bars are not shown for the measured data because the uncertainties in these data are not well-characterized, in my opinion, for the reasons described below. Only the data points for the central 2.44 m of the fuel-assembly length have been plotted, since the calculations indicated constant dose rates as a function of height only over this section.

In-Situ Short-Duration Thermoluminescence Dosimetry

The two sets of E.G.&G. data for August 13, 1982, and March 8, 1983, have been compared with calculated values by interpolating, again using functions of the form $D = D_0 t^n$ to describe the calculated results. As was done for the optical absorption dosimetry, the measured data were adjusted to account for attenuation by the stainless steel tubes and lack of attenuation by the removed granite. The comparisons are shown in Tables 4 and 5, and are plotted as a function of radius in Figs. 6 and 7. Again, the dashed lines represent $\pm 25\%$ error limits on the calculated values. The data for the August 13, 1982 exposures in Table 4 and Fig. 6 were obtained by using straight lines connecting the two bordering calibration points, rather than using the straight lines that the experimenters fitted to all the calibration data. This was done because the calibration data showed curvature, and straight lines over the entire dose range did not appear to represent the data adequately.

Table 1. Comparison of calculated doses and doses measured by optical absorption for the period from April 1980 to January 12, 1981.

Hole number, fuel assembly number, dosimeter location (m)	Vertical distance from midplane (m)	Measured absorbed dose (Gy in LiF)		Calculated absorbed dose (Gy in LiF)	Ratio meas. calc.
		247 nm	374 nm		
CEH1, D34, 0.30 (wall)	+ 1.22		2.7×10^5	4.79×10^5	0.56
	0		3.2×10^5	4.79×10^5	0.67
	1.22		2.7×10^5	4.79×10^5	0.56
CEH3, D40, 0.30 (wall)	+ 1.22		3.3×10^5	5.08×10^5	0.65
	+ 0.61		3.5×10^5	5.08×10^5	0.69
	0		1.8×10^5	5.05×10^5	0.75
	0.61		3.6×10^5	5.08×10^5	0.71
	1.22		3.7×10^5	5.08×10^5	0.73
CEH3, D40, 0.51	+ 1.22	9.3×10^5	8.0×10^5	1.69×10^6	0.55, 0.47
	0	6.7×10^5		1.69×10^6	0.40
	1.22	6.6×10^5	6.7×10^5	1.69×10^6	0.39, 0.40
CEH3, D40, 0.66	+ 1.22	7.2×10^5		1.40×10^6	0.51
	0	8.4×10^5		1.40×10^6	0.60
	1.22	9.3×10^5		1.40×10^6	0.66
CEH7, D06, 0.30 (wall)	+ 1.22		2.9×10^5	4.53×10^5	0.64
	0		3.0×10^5	4.53×10^5	0.66
	- 1.22		3.4×10^5	4.53×10^5	0.75
CEH11, D04, 0.30 (wall)	+ 1.22		2.8×10^5	4.24×10^5	0.66
	0		3.1×10^5	4.24×10^5	0.74
	1.22		2.6×10^5	4.24×10^5	0.62

Table 2. Comparison of calculated doses and doses measured by optical absorption for the period from January 12, 1981 to October 23, 1981.

Hole number, fuel assembly number, dosimeter location (m)	Vertical distance from midplane (m)	Measured absorbed dose (Gy in LiF)		Calculated absorbed dose (Gy in LiF)	Ratio meas. calc.
		247 nm	374 nm		
CEH1, D34, 0.30 (wall)	+ 1.22		2.2×10^5	4.06×10^5	0.54
	0		2.5×10^5	4.06×10^5	0.62
	1.22		2.3×10^5	4.66×10^5	0.57
CEH3, D40, 0.30 (wall)	+ 1.22		2.2×10^5	4.18×10^5	0.53
	- 0.61		2.2×10^5	4.18×10^5	0.53
	0		2.8×10^5	4.18×10^5	0.67
	- 0.61		2.9×10^5	4.18×10^5	0.69
	- 1.22		2.7×10^5	4.18×10^5	0.64
CEH3, D40, 0.51	+ 1.22	8.5×10^5	6.7×10^5	1.31×10^6	0.65, 0.51
	0	7.6×10^5		1.31×10^6	0.58
	1.22	8.3×10^5		1.31×10^6	0.63
CEH3, D40, 0.66	+ 1.22	6.8×10^5		1.00×10^6	0.68
	0	6.6×10^5		1.00×10^6	0.66
	1.22	5.5×10^5		1.00×10^6	0.55
CEH7, D06, 0.30 (wall)	+ 1.22		2.3×10^5	4.16×10^5	0.55
	0		2.3×10^5	4.16×10^5	0.55
	1.22		2.4×10^5	4.16×10^5	0.58
CEH11, D04, 0.30 (wall)	+ 1.22		2.2×10^5	4.18×10^5	0.53
	0		2.5×10^5	4.18×10^5	0.60
	1.22		2.4×10^5	4.18×10^5	0.57

Table 3. Comparison of calculated doses and doses measured by optical absorption for the period from October 23, 1981 to March 8, 1983.

Hole number, fuel assembly number, dosimeter location (m)	Vertical distance from midplane (m)	Measured absorbed dose (Gy in LiF)		Calculated absorbed dose (Gy in LiF)	Ratio meas. calc.
		247 nm	374 nm		
CEH1, D34, 0.30 (wall)	+ 1.22		2.6×10^5	5.31×10^5	0.49
	0		3.0×10^5	5.31×10^5	0.56
	1.22		2.9×10^5	5.31×10^5	0.55
CEH3, D40, 0.30 (wall)	+ 1.22		3.0×10^5	5.52×10^5	0.54
	- 0.61		3.4×10^5	5.52×10^5	0.62
	0		3.6×10^5	5.52×10^5	0.65
	- 0.61		3.6×10^5	5.52×10^5	0.65
	1.22		3.4×10^5	5.52×10^5	0.62
CEH3, D40, 0.51	+ 1.22	6.1×10^5	6.7×10^5	1.62×10^6	0.38, 0.41
	0	4.8×10^5	4.0×10^5	1.62×10^6	0.30, 0.25
	- 1.22	5.7×10^5	4.0×10^5	1.62×10^6	0.35, 0.25
CEH3, D40, 0.66	+ 1.22	9.1×10^5		1.15×10^6	0.79
	0	8.3×10^5		1.15×10^6	0.72
	- 1.22	6.0×10^5		1.15×10^6	0.52
CEH7, D09, 0.30 (wall)	+ 1.22		3.0×10^5	5.35×10^5	0.56
	0		3.2×10^5	5.35×10^5	0.60
	+ 1.22		3.4×10^5	5.35×10^5	0.64
CEH11, D04 and D18 0.30 (wall)	+ 1.22		3.0×10^5	5.48×10^5	0.55
	0		3.2×10^5	5.48×10^5	0.58
	- 1.22		3.0×10^5	5.48×10^5	0.55

Figure 3. Comparison of calculated and optical-absorption-measured doses for hole CEH3 for the period from April 1980 to January 12, 1981.

Figure 4. Comparison of calculated and optical-absorption-measured doses for hole CEH3 for the period from January 12, 1981 to October 23, 1981.

Figure 5. Comparison of calculated and optical-absorption-measured doses for hole CEH3 for the period from October 23, 1981 to March 8, 1983.

Table 4. Comparison of calculated dose rates and dose rates measured by TLDs for hole CEH3 on August 13, 1982.

Dosimeter location (m)	Vertical distance from midplane (m)	MgB ₂ O ₃ data (Gy in LiF/h)	CaF ₂ (Gy in LiF/h)	Calculated dose rates (Gy in LiF/h)	Ratio meas. calc.
0.30 (wall)	- 1.22	54.0	54.0	44.1	1.22, 1.22
		66.0	51.0	44.1	1.50, 1.16
	0	71.0	59.0	44.1	1.61, 1.34
		70.0	59.0	44.1	1.59, 1.34
	- 1.22	66.0	72.0	44.1	1.50, 1.63
		66.0	67.0	44.1	1.50, 1.57
0.51	- 1.22	1.23	1.12	1.29	0.95, 0.87
		1.14	1.18	1.29	0.88, 0.91
	0	1.06	0.98	1.29	0.82, 0.76
		1.02	1.01	1.29	0.79, 0.78
	1.22	1.14	1.12	1.29	0.98, 0.87
		1.07	1.11	1.29	0.83, 0.86
0.66	- 1.22	0.136	0.109	0.0901	1.51, 1.21
		0.126	0.127	0.0901	1.40, 1.41
	0	0.092	0.086	0.0901	1.02, 0.95
			0.084	0.0901	—, 0.93
	1.22	0.118	0.096	0.0901	1.31, 1.06
		0.118	0.100	0.0901	1.31, 1.11

Table 5. Comparison of calculated dose rates and dose rates measured by TLDs for hole CEH3 on March 8, 1983.

Dosimeter location (m)	Vertical distance from midplane (m)	CaF ₂ data (Gy in LiF/h)	Calculated dose rates (Gy in LiF/h)	Ratio meas. calc.
0.30 (wall)	- 1.22	37.6	38.9	0.97
	0	43.0	38.9	1.10
	- 1.22	43.3	38.9	1.11
0.51	- 1.22	1.04	1.11	0.94
	0	0.857	1.11	0.77
	- 1.22	0.984	1.11	0.89
0.66	- 1.22	0.0944	0.0753	1.25
	0	0.0664	0.0753	0.88
	1.22	0.0806	0.0753	1.07

Discussion

Of all the *in-situ* data, the most reliable are expected to be the thermoluminescence results for March 8, 1983, given in Table 5 and Fig. 7, because they were taken under the best-controlled conditions and were the most carefully calibrated. Happily, they are also in best agreement with the results of the Monte Carlo calculations, all the data points shown falling within the $\pm 25\%$ error limits estimated for the calculations. This, combined with the excellent agreement between the calculations and the pre- and post-test thermoluminescence dosimetry, which was also well-calibrated

and well-controlled, indicates that the Monte Carlo results are most likely valid within the estimated uncertainty.

The thermoluminescence results for August 13, 1982, exhibit more spread, and some of the data are well outside the estimated uncertainty, as shown in Table 4 and Fig. 6. These measurements were also not as well controlled and calibrated as those taken on March 8, 1983.

Finally, all the long-term optical absorption results appear to be systematically lower than the calculated results, in some cases by as much as

Figure 6. Comparison of calculated and thermoluminescence-measured dose rates for hole CEH3 on August 13, 1982.

Figure 7. Comparison of calculated and thermoluminescence-measured dose rates for hole CEH3 on March 8, 1983.

75%. These large differences probably result from the inability to accurately correct for fade in the long-duration, elevated-temperature irradiations. Because of this inherent problem and because these data are in systematic disagreement with

calculational results that are supported by the more carefully-controlled and calibrated measurements, the optical absorption data are regarded as less reliable than the calculations.

Conclusions and Estimated Total Doses for Climax Emplacement Holes

In general, the spatial and time dependencies of the measured and calculated doses are in good agreement. The absolute values are in good agreement in the case of the thermoluminescence measurements, whereas the optical absorption results differ significantly from the calculated results. As noted, we believe the thermoluminescence results to be more reliable because the calibration conditions are more nearly matched to the field conditions. Therefore, it appears that the Monte Carlo calculations provide a sound basis for determining the total doses to the emplacement holes, within an estimated uncertainty of $\pm 25\%$. Hence, we have used them to calculate the total dose for each emplacement hole.

During the course of the Climax test, some of the fuel assemblies were exchanged from one emplacement hole to another for various reasons, such as to inspect canisters and to demonstrate the capability to perform transfers. Because the fuel assemblies differed slightly in their initial fuel loading and in their degree of burnup, their gamma ray source strengths also differed slightly. To determine the total doses to the rock around the emplacement holes, it was therefore necessary to account for the exchanges. Table 6 gives a history of these exchanges for the 11 holes that contained spent fuel.

The maximum doses to the rock, which occurred at the walls of the emplacement holes at

Table 6. Canister emplacement hole history.

Canister emplacement hole	Initial emplacement	Exchange No. 1	Exchange No. 2	Exchange No. 3	Final retrieval
1	D-34 (4/25/80)	—	—	D-34 removed for inspection and replaced 8/17/82 ^a	D-34 (3/3/83)
3	D-40 (4/18/80)	—	—	—	D-40 (3/10/83)
5	D-46 (4/29/80)	—	—	—	D-46 (3/14/83)
7	D-06 (5/13/80)	—	D-06 out (10/26/81) D-09 in (10/28/81)	—	D-09 (3/29/83)
8	D-16 (5/20/80)	—	—	—	D-16 (4/4/83)
9	D-47 (5/8/80)	—	—	—	D-47 (3/24/83)
10	D-01 (5/15/80)	—	—	—	D-01 (4/6/83)
11	D-40 in (4/15/80, 1600 h) ^b D-40 out (4/17/80, 1130 h) ^b D-04 in (5/28/80)	—	—	D-04 out (8/16/82) D-18 in (8/16/82)	D-18 (3/31/83)
12	D-09 (5/1/80)	D-09 out (1/12/81) D-15 in (1/14/81)	—	—	D-15 (3/22/83)
14	D-35 (5/6/80)	—	—	—	D-35 (3/16/83)
16	D-18 (4/22/80)	—	—	D-18 out (8/16/82) D-22 in (8/18/82)	D-22 (3/7/83)

^a Out of hole 5.3 h.

^b Hot dry run.

Table 7. Calculated total absorbed doses to wall of emplacement holes for entire period of emplacement.

Hole No.	Absorbed dose in granite (MGy)
1	1.55
3	1.62
5	1.59
7	1.55
8	1.53
9	1.58
10	1.58
11	1.60
12	1.57
14	1.57
16	1.55

the height of the central, fueled portion of the fuel assemblies, were calculated from the Monte Carlo results. The actual dates of emplacement and retrieval were used, and the total doses were calculated by fitting the Monte Carlo dose rate results to functions of the form $D = D_0 t^n$ and integrating over time. The results are given in Table 7 on a granite basis. Doses as a function of radius in the rock are given for hole CEH1 in Table 8. Values for other holes can be determined by multiplying

Table 8. Calculated total absorbed dose in granite around hole CEH1 as a function of radius.

Radius		Absorbed dose in granite (Gy)
(in.)	(m)	
9	0.229 (outside of liner)	1.96×10^0
12	0.305 (wall of granite)	1.55×10^0
13	0.330	1.10×10^0
14	0.356	6.82×10^{-1}
15	0.381	4.49×10^{-1}
16	0.406	2.90×10^{-1}
18	0.457	1.20×10^{-1}
20	0.508	4.85×10^{-2}
22	0.559	2.04×10^{-2}
24	0.610	8.18×10^{-3}
26	0.660	3.67×10^{-3}
30	0.762	6.04×10^{-3}

these results by the ratio of the dose at the wall of the hole in question to the dose at the wall of hole CEH1.

Because of the good agreement between the thermoluminescence measurements and the Monte Carlo calculations, it appears that these estimated total doses are accurate to within the estimated total uncertainty of the calculations, which is $\pm 25\%$.

Recommendations for Determining Doses Around Emplacements in Future Nuclear Waste Storage Facilities

It seems certain that there will be a continuing need to evaluate radiation doses and dose rates near high-level nuclear waste in storage facilities and repositories. This type of information is needed to design shielding for personnel protection and to predict and interpret radiation effects. Possible radiation effects include radiation-induced chemical changes in the gaseous, liquid, or solid media surrounding the nuclear waste and effects on corrosion of metal parts. At very high gamma ray doses, changes in the structural, mechanical, or thermal properties of rock may occur. (Recent work²³ shows no observable changes in compressive strength, Young's modulus, or Poisson's ratio of Climax quartz monzonite and West-ely granite for a dose of 9.2 MGy.)

As a result of this work, it appears that the

doses and dose rates for nuclear waste facilities are best determined by a combination of calculations and measurements. The calculations offer the advantages of producing results throughout a complicated geometry, rather than at only a few points, and giving insight into the temporal and spatial variations and the reasons underlying them. The measurements, on the other hand, provide an independent check that does not depend on the many input parameters and modeling assumptions that go into the calculation of the source term and radiation transport. The combination of these two methods provides both trustworthiness and general results.

The ORIGEN2 source term calculation coupled with the MORSE-L Monte Carlo radiation transport analysis used here proved to be in good

agreement with the most carefully calibrated measurements. The fact that HEDL also found the ORIGEN2 predictions of spent fuel heat output to agree well with calorimetric measurements adds further confidence in the calculational approach. As Schmittrath¹³ pointed out, it would also be helpful to test the ORIGEN2 predictions for boiling-water-reactor spent fuel and for high-burnup, long-decay-time conditions. For future waste storage facilities, discrete-ordinates-type transport calculations should also be considered, since they are very amenable to simple geometries and are often less expensive in computer time. We used the Monte Carlo approach in our work because it is more commonly used at our Laboratory and was therefore more readily available.

Concerning measurement techniques, this work has shown that if passive dosimetry is used,

it should involve short exposures and be combined with accurate temperature measurement so that laboratory calibrations can be performed readily with the same time, temperature, dose rate, and dose parameters. An attractive alternative approach would be to use active ionization chambers coupled with temperature monitors, which are inserted only for short times. These could be calibrated under the same temperature and pressure conditions that are present in the measurement location. They could be read directly, in real time; they would not be subject to degradation by radiation effects or corrosion; they could be small in size, i.e., less than about 1.5 cm in diameter and 10 cm long. It would be necessary to design access tubes into the facility that would allow such chambers to be inserted while preventing excessive radiation leakage.

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