Radiation Dose Rates from Commercial PWR and BWR Spent Fuel Elements

C. E. Willingham

October 1981

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ABSTRACT

Data on measurements of gamma dose rates from commercial reactor spent fuel were collected, and documented calculated gamma dose rates were reviewed. As part of this study, the gamma dose rate from spent fuel was estimated, using computational techniques similar to previous investigations into this problem. Comparison of the measured and calculated dose rates provided a recommended dose rate in air versus distance curve for PWR spent fuel.

SUMMARY

Data documented by several organizations indicate that discrepancies exist in calculated and measured radiation dose rates in air from commercial Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) spent fuel elements. Evaluation of these documented data associated with calculated spent fuel radiation dose rates is required to ensure an accurate baseline of information for spent fuel shipping, handling and storage. An improved knowledge of the dose rate would help establish the occupational risks of handling spent fuel. In addition, shielding requirements for spent fuel handling and storage can be assessed using the dose rate information generated by this study.

The objective of this study was to survey and evaluate documented data associated with calculated spent fuel radiation dose rates in air. The study focused on the gamma dose rate in air resulting from a single PWR spent fuel assembly. Reports outlining calculated dose rates were assembled and reviewed. Experimental determination of gamma dose rates was also reviewed to explain the discrepancies between the calculated and measured dose rates. For the PWR air dose rate measurements the Pacific Northwest Laboratory (PNL) calculations were 25% lower than the measured values. The PNL calculations vary significantly (30 to 50%) compared to the measured values at dose points locations toward the ends of the assembly. Calculations performed by Oak Ridge National Laboratory (ORNL) were used extensively in this study as a reference point in comparing various reported results. The ORNL calculations were more detailed and accounted for geometry and axial burnup effects. Estimates of the air dose rate extrapolated from the ORNL calculated data were in closer agreement to the measured values (~10% lower). These discrepancies and lower values can be traced primarily to gamma scattering from the air and structure of the hot cell itself and also to uncertainties in detector calibration, assembly hardware shielding effects, axial burnup effects and fuel self-shielding effects.

The results of the PNL and ORNL gamma dose rates in air for 1- and 5-year-old PWR spent fuel exposed to 33,000 MWd/MT are shown in Table S.1. The calculated dose rates in air on top (vertical direction) of a 1-year-old PWR

TABLE S.1. Dose Rate Comparison for ORNL and PNL Calculations

Distance From Assembly Midplane (m)	Dose Rate (rad/hr)			
	1-Year Decay Time		5-Year Decay Time	
	ORNL	PNL	ORNL	PNL
0	2.1×10^5	1.3×10^5	4.7×10^4	3.4×10^4
1	1.1×10^4	7.5×10^3	2.4×10^3	1.9×10^3
2	3.8×10^3	2.9×10^3	8.4×10^2	9.5×10^3

spent fuel assembly exposed to 33,000 MWd/MT are 5.8×10^2 rad/hr on the surface, 11 rad/hr 1 meter from the top and 3.6 rad/hr 2 meters from the end of the assembly. For a 5-year-old PWR spent fuel assembly the dose rates are 1.3×10^2 rad/hr on the top surface, 2.4 rad/hr 1 meter from the top and 0.8 rad/hr 2 meters from the end of the assembly.

Only a limited amount of data for actual measurement of gamma dose rates in air was found. Thermoluminescent detectors (TLDs) were placed on or near the assembly while it was in a hot cell. The results of one of these experiments can be seen in Table S.2, along with the PNL ISOSHL D calculation results for this arrangement. TLD detector measurements were not made directly above the Turkey Point spent fuel assembly.

Another experiment measured the PWR gamma dose rate for Point Beach spent fuel stored in water. The results of this experiment are shown in Table S.3.

Using ion chambers, gamma dose rate measurements for spent fuel (both PWR and BWR) in water storage at the General Electric (GE) Morris Operation were

TABLE S.2. Turkey Point Spent Fuel Dose Rates (25,000 MWd/MT, 3.7 Years Cooling Time)

Distance From Assembly (m)	Dose Rate (rad/hr)	
	TLD Measurement	PNL ISOSHL D
0	5.9×10^4	1.6×10^4
1.2	3.4×10^3	8.0×10^2
2.1	1.6×10^3	4.5×10^2

TABLE S.3. Point Beach Spent Fuel Dose Rates
(23,400 MWd/MTU, 640 Days Cooling Time)

<u>Distance From Assembly (m)</u>	<u>Dose Rate (rad/hr)</u>	
	<u>TLD Measurement</u>	<u>PNL ISOSHL D</u>
0.5	380	250
0.75	70	40
1.0	18	8

made 25-300 cm directly above a basket of four assemblies. Although the measurements were made over a single assembly, some contribution from the other assemblies occurred. For a one-year-old PWR assembly ($\sim 36,000$ MWd/MT) the gamma dose rate 25 cm directly above the top was measured at ~ 1500 rad/hr. Additional measurements of both PWR and BWR spent fuel assemblies (water and air) are being made but have not yet been published.

During the course of this study no BWR air gamma dose rate measurements were found. One study did report the measurement of total gamma activity for BWR as well as PWR spent fuel, but did not give enough information to arrive at dose rates. Documented BWR gamma dose rate calculations apparently do not exist.

Experimental uncertainties that accounted for variations from the reported calculated dose rates were identified. Also, the variations among calculated dose rates were reported. Because of the lack of sufficient gamma dose rate measurements other than the Turkey Point, Point Beach and GE-Morris experiments, the baseline dose in air versus distance curve is based on limited data. No BWR fuel dose in air versus distance curve was developed since little measurable data were available.

As a result of these studies, PNL concludes that reliable, although limited, PWR gamma dose rate information (calculated and measured) exists for use in spent fuel handling and storage studies. Additional calculations and experiments of BWR gamma dose rates are needed to complete the data base.

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1.0 INTRODUCTION

Design and operation of spent fuel storage basins, whether at a commercial reactor site, temporary storage-receiving at a reprocessing facility, or at an away-from-reactor spent fuel storage area, require a knowledge of the decay heat levels, gamma dose rates and neutron source strength, as well as many other characteristics of irradiated fuel elements. This study focuses on the gamma dose rate in air resulting from a single fuel assembly.

The Department of Energy contracted with Pacific Northwest Laboratory (PNL) to survey and evaluate documented data on spent fuel gamma dose rates and to recommend a dose rate in air curve for commercial spent fuel. This work was conducted as part of the Interim Spent Fuel Storage Program under the Savannah River Operations Office.

Data documented by several organizations indicate significant discrepancies in calculated and measured radiation dose rates in air for irradiated commercial boiling water reactor (BWR) and pressurized water reactor (PWR) spent fuel elements. The data will be studied to resolve or explain the apparent discrepancies in the documented calculated and measured dose rate results. Based on this survey, a dose rate versus distance curve in air for 1- and 5-year-old spent fuel will be recommended. The results of this study will provide a data base for gamma dose rates from spent fuel handling at commercial facilities. The results will be particularly useful in analyzing the dose rate characteristics of drywell, silo, vault or dry fuel storage concepts for storing spent reactor fuel assemblies.

This project was undertaken in three main task areas:

- Literature Survey - All relevant reports on radiation dose rates from spent fuel were collected. Documentation of measured dose rates was also obtained.
- Differences in Reported Dose Rates - Comparisons between calculated and measured dose rates were made. Factors such as spent fuel, isotopic composition, age of fuels, shielding and structure material and measurement geometry, as well as other qualities, were considered.

- Dose Versus Distance Curve - As a result of the documented data found, a dose in air versus distance curve for 1-and 5-year old PWR spent fuel assemblies was determined.

The characteristics of PWR and BWR fuel assemblies are summarized in Chapter 2.0 and the neutronics and physics assumptions required by the study are discussed. Chapter 3.0 gives the assumptions for gamma dose rates and Chapter 4.0 describes the literature search that was conducted. In Chapter 5.0 the calculations of gamma dose rates from spent fuel assemblies are examined. The computer codes, geometry models and case assumptions used in the reported studies are reviewed and the gamma dose rates are also reported. The theory of thermoluminescent detector (TLD) operation and the results of TLD measurements are reviewed in Chapter 6.0. In Chapter 7.0 experiments concerning gamma scanning nondestructive techniques and mock-up gamma dose experiments are examined. Finally, in Chapter 8.0 the gamma dose rates obtained by experiments are compared to the dose rates calculated using computer codes.

2.0 DESCRIPTIONS OF PWR AND BWR FUEL ASSEMBLIES

In this section the characteristics of both PWR and BWR fuel assemblies are summarized.

2.1 PWR FUEL ASSEMBLY CHARACTERISTICS

The physical qualities of a PWR fuel assembly are given in Table 2.1 and Table 2.2 shows the material compositions and weights. The uranium dioxide (UO_2) fuel pellets are in the shape of a right circular cylinder with a 0.8 cm diameter and are about 1.35 cm long. These pellets are stacked in a Zircaloy tube 360-380 cm high. The tube ends are sealed by welded Zircaloy plugs. Figure 2.1 shows a schematic illustration of this fuel rod or element. These fuel rods are arranged in a square 17 x 17 array, shown in a cross-sectional view in Figure 2.2 and length-wise in Figure 2.3 and are held together by Inconel grid spacers along the lengths of the assembly and two stainless steel end sections. Of the total 289 positions 264 are occupied by fuel elements. The remaining locations are occupied by control-rod guide

TABLE 2.1. PWR Fuel Assembly Characteristics (Westinghouse 1972)

Assembly length, m	4.059
Assembly dimensions, cm	21.4 x 21.4
Active fuel height, m	3.658
Fuel element array	17 x 17
Fuel element/assembly	264
Total weight of assembly, kg	657.9
Assembly volume, m ³	0.186
Element pitch, cm	1.259
Number of grids/assembly	7
Number of guide thimbles/assembly	24
Fuel element OD, cm	0.950
Diameter gap, cm	0.0165
Fuel clad thickness, cm	0.05715

TABLE 2.2. PWR Fuel Assembly Material Compositions (Westinghouse 1972)

Fuel material	UO ₂
Clad material	Zircaloy-4
Guide thimbles	Zircaloy-4
Composition, kg	
U	461.4
UO ₂	523.4
Zircaloy	108.4
Hardware	26.1

thimbles and instrument sheaths. The PWR fuel assembly considered in this report for dose rate calculations is typical of the Westinghouse 17 x 17 (RESAR-3) designs. The PWR fuel assembly design for several of the dose rate experiments that were investigated are of a different type and will be discussed in more detail in later sections of this report.

2.2 BWR FUEL ASSEMBLY CHARACTERISTICS

Figure 2.4 shows a schematic diagram of a BWR fuel assembly. The total overall length of the fuel assembly is ~446.7 cm with 365.7 cm of active fuel zone. The BWR type fuel rods, shown in Figure 2.5, are assembled in a 7 x 7 array. Each assembly is contained in a can-like fuel channel with the fuel rods spaced and supported between upper and lower tie plates. Eight of the rods are threaded into the bottom support plate and are bolted down to the upper support plate to provide a structural tie for the assembly. This arrangement allows the plate to be inserted into or removed from the core by the upper tie plate handle. The remaining rods have end plugs that fit into holes in the support plates. A cross-sectional view in Figure 2.6 shows the arrangement of the fuel rods within the assembly structure. Control is provided by cruciform control blades. Within the cruciform sheath boron carbide powder is contained in stainless steel tubes. Supplementary reactivity control may be provided for fresh fuel by boron-bearing stainless steel curtains inserted between fuel assemblies. Some fuel rods may contain gadolinia as a

FIGURE 2.1. PWR Fuel Rod

FIGURE 2.2. PWR 17 x 17 Fuel Assembly (All Dimensions cm)

FIGURE 2.3. PWR Fuel Assembly

FIGURE 2.4. BWR Fuel Assembly (All Dimensions cm)

FIGURE 2.5. BWR Fuel Rod

FIGURE 2.6. BWR Fuel Assembly Cross Section
(All Dimensions cm)

burnable poison. As control blades are withdrawn, water in the vacated spaces provides more neutron moderation and causes local flux peaking. Several different enrichments are used in each assembly to offset the tendency for this flux peaking. Low enrichment UO_2 fuel is typically in the form of 1.24 cm diameter-pressed and sintered pellets stacked in a 365-cm column contained in

Zircaloy-2 cladding. Above the fuel a plenum having a volume of 8 to 11 per-cent of the fuel volume is provided. This plenum accommodates helium backfill plus gaseous fission products. Table 2.3 summarizes some important character-istics of a BWR fuel assembly and fuel rods.

TABLE 2.3. BWR Fuel Assembly Characteristics
(U.S. Nuclear Regulatory Commission 1976)

Overall length, m	4.34
Active fuel length, m	3.65
Fuel rods per assembly	49
Fuel rod array	7 x 7
Rod pitch, cm	1.87
Fuel rod O.D., cm	1.43
Clad I.D., cm	1.24
Clad thickness, cm	0.093
Clad material	Zircaloy-2
Fuel assembly cross section, cm	13.8 x 13.8
Composition for an assembly, kg	
UO ₂	208
Zircaloy	57.9
Hardware	9.8

3.0 ASSUMPTIONS FOR GAMMA DOSE RATES

The dose rate data generated as a result of these analyses apply to commercial light water reactor fuel. Most of the study was concerned with PWR spent fuel since the majority of information is about this type of reactor. For the dose rate and shielding computer codes used here, the PWR fuel assembly was assumed to be in the core for 3 cycles (about 3 years). Several decay times were considered, but PNL was requested to provide data for 1- and 5-year old fuel. The dose rate at various distances from the source was valid for a single spent fuel assembly in air. For the experiments and measured dose rate, the specific assembly characteristics for a particular reactor's spent fuel were also investigated. The dose rate from a spent fuel assembly is proportional to burnup for decay times more than a few years, and at short decay times (discharge) the dose rate is directly dependent on the specific power (MW/MT fuel). The dose rate depends on both burnup and specific power at intermediate times. The PWR spent fuel dose rates are not applicable to BWR spent fuel data since the BWR assemblies contain smaller amounts of uranium and have nonsymmetrical burnup distribution patterns. The dose rate calculations discussed in this report are based on gamma radiation and do not include the contribution from neutrons. The dose rate from neutrons emitted in a ^{235}U enriched unshielded fuel assembly is small compared to gamma ray doses. The neutron dose rate becomes significant if gamma shields surround the assembly and no provision is made to shield the neutrons.

In this report, the gamma dose or exposure is expressed in units of either "rem" or "rad". The "rad" is an indication of the amount of radiation energy absorbed and does not provide an adequate measure of the biological damage or account for the effectiveness of the radiation. The "rem" unit is a measure of the biological dose defined by the following:

$$\text{Dose in rems} = \text{RBE} \times \text{Dose in rad}$$

The relative biological effectiveness (RBE) for gamma rays for whole body irradiation (blood-forming organs critical) is taken to be 1. Thus, rad and rem are equivalent for gamma rays.

4.0 LITERATURE SEARCH

Available reports and data associated with calculated dose rates were collected. Each report was reviewed and the data evaluated for its usefulness to this study. Documented results for the actual dose rate measurements were difficult to obtain. Most of the references to either calculated or measured gamma activity from spent fuel can be classified in one of following categories:

- calculation of dose rates for nonproliferation measures (^{60}Co spike sources, etc.)
- safeguards and nondestructive assay of spent fuels (gamma scanning)
- dose rates from reprocessing waste.

The literature survey revealed that almost all of the available reports dealt with safeguards and nondestructive assay issues. These reports yielded information concerning gamma activity for specific isotopes but did not report gamma dose rates for an entire spent fuel assembly. Several ORNL and Los Alamos National Laboratory (LANL) reports on the subject of dose rates of spiked fuel (nonproliferation) provided some information on gamma dose rates and relative gamma activities. The gamma dose rate calculations on reprocessing waste (shielded casks) were not useful to this study.

The experimental gamma dose information was obtained primarily through contacts with other laboratories. Personnel from ORNL, Battelle Memorial Institute (BMI) in Columbus Ohio, LANL, and Hanford Engineering Development Laboratory (HEDL) were questioned about experiments measuring gamma dose rates. Gamma dose rate measurements of both PWR and BWR spent fuel (water and air) have also been made at the General Electric Morris Spent Fuel Storage Operation in Morris, Illinois. However the final results of these measurements have not yet been published. The most useful experiments were some Electric Power Research Institute (EPRI) sponsored underwater gamma dose tests and BMI hot cell experiments. These will be reviewed in detail in Chapter 6.0.

5.0 CALCULATED GAMMA DOSE RATES

In this chapter the calculations of gamma dose rates from spent fuel assemblies are examined. The computer codes, geometry models and case assumptions used in the reported studies are reviewed and gamma dose rates are reported. Gamma dose rates calculated by PNL as part of this study also are documented in this section.

5.1 REPORTED GAMMA DOSE RATES

Several reports on nonproliferation and materials safeguards measures give gamma dose rate information. However, no reports that specifically address gamma dose rate in air from a spent fuel assembly were found in the open literature. The following sections of this chapter deal primarily with the total gamma dose exposure rate (rad/hr) calculations.

Oak Ridge National Laboratory Gamma Dose Rate Calculations

The ORNL report, Calculated, Two-Dimensional Dose Rates From a PWR Fuel Assembly by A. G. Croff, O. W. Hermann and C. W. Alexander (1979), probably represents the most recent gamma dose rate calculations. The purpose of the study was to report the dose rate characteristics of a fuel assembly for the following cases: (1) spent fuel fission product gamma rays, and (2) a uniformly distributed spike in two dimensions as a function of decay time and distance.

In the ORNL report the PWR fuel assembly design was simplified to make the gamma dose rate calculations practical. The fuel assembly was approximated as a right circular cylinder. The fuel rods' space and assembly structure were made uniform by mixing the fuel assembly constituents within the assembly volume. Stainless steel was substituted for Zircaloy on a gram-for-gram basis. The ORNL calculations were performed for a one-fourth model of the spent fuel assembly. The basic constituents for the model were air, cladding and fuel. All of the ORNL cases in this report were for gamma dose rates in air. The atom densities for the homogeneous model were ^{235}U - 6.8×10^{-5} atoms barn $^{-1}\text{cm}^{-1}$; ^{238}U - 6.7×10^{-3} ; O- 1.6×10^{-2} ; Cr- 1.2×10^{-3} ; Fe- 4.5×10^{-3} ; Ni- 6.2×10^{-4} and for the air, O- 1.1×10^{-5} atoms barn $^{-1}\text{cm}^{-1}$ and N- 3.4×10^{-5} .

ORNL assumed that the cylindrical fuel assembly was 360 cm long, with two end pieces 12 cm in length and 24 cm in diameter. Because the assembly was assumed to be symmetrical, the two-dimensional calculations were made for only one-fourth of the assembly. This model, shown in Figure 5.1, then consisted of the homogeneous mixture 180 cm long with a 12 cm end piece and 12 cm diameter.

ORNL calculated dose rates for several different points:

- on the side and the end surfaces of the fuel assembly
- on a spatial surface 1 m from the side and end of the fuel assembly
- on spatial surface 2 m from the side and end of the fuel assembly.

The surfaces where dose rate calculations were made are shown in Figure 5.2. The ORNL calculations were made for a fuel exposure of 33,000 MWd/MT and a fuel decay age of 10 years.

In its calculations ORNL used the DOT-IV and an internal, undocumented code called FALSTF. DOT is a discrete-ordinates two-dimensional computer code that calculates photon transport within the fuel assembly (Oak Ridge National Laboratory 1976). A multigroup gamma spectrum, shown in Table 5.1, was used in DOT-IV as a cosine distributed source. Gamma ray fluxes were converted to dose rates at the assembly surface using the conversion values given in Table 5.2. The ORNL FALSTF code used the gamma source calculated by DOT-IV and the fuel assembly geometry of Figure 5.1 to arrive at a gamma flux and the dose rate at various points a few meters away from the assembly. The gamma spectrum was calculated using ORIGEN and ENDF/B-IV data files (Bell 1973).

For absolute gamma dose rates as a function of time the code ANISN was used in the ORNL report (Engle 1967).

The ORNL study presented the calculated dose rates for 10-year-old PWR spent fuel. BWR spent fuel was not considered. Figure 5.3 shows the gamma dose rate (rem/hour) for several different surfaces on or near a 10-year-old fuel assembly.

FIGURE 5.1. PWR Fuel Assembly Model (Croff, Hermann and Alexander 1979)

The ORNL calculations were summarized by the following relationship:

$$\text{Dose Rate (rem/hr)} = 23,400 \times \text{SG} \times \text{DF} \quad (1)$$

where

SG = source shape and geometry factor from Figure 5.4

DF = decay factor from Figure 5.5, accounting for fission product decay

FIGURE 5.2. Dose Rate Surfaces for PWR Fuel Assembly
(Croff, Hermann and Alexander 1979)

TABLE 5.1. Fission Product Gamma-Ray Spectrum For 10-Year-Old PWR Fuel
(Croff, Hermann and Alexander 1979)

Energy Interval Boundaries (MeV)		Gamma-Ray Intensity (photons sec ⁻¹ assembly ⁻¹)
Upper	Lower	
4.0	3.0	2.88+08
3.0	2.5	2.22+09
2.5	2.0	3.73+10
2.0	1.5	1.58+12
1.5	1.0	5.11+13
1.0	0.70	1.30+14
0.70	0.45	1.55+15
0.45	0.30	7.36+13
0.30	0	8.20+14

The equation is based upon the midplane surface dose rate for a 10-year old PWR spent fuel assembly (23,400 rem/hr). The SG factor accounts for both the position of the dose point with respect to the fuel assembly and the axial variation in burnup within the fuel assembly. The DF factor is for fission product decay and the change in gamma ray spectrum with time. Figures 5.6 and 5.7 show the dose rates for 1- and 5-year-old fuel, respectively.

The dose rate plots for PWR spent fuel indicate that the dose rate drops suddenly from the centerline midpoint to the end of the assembly. The fuel region acts as a shield to the more intense gamma radiation released from the higher burnup fuel regions. Also, because the fuel assembly is a finite length, the dose rate decreases more rapidly than the inverse relationship for an infinite long cylindrical source. Figure 5.8 shows the dose rate in air versus distance from midplane curve for the three decay periods.

Other Calculated Gamma Dose Rates

A General Atomic (GA) report examined the decay heat and gamma dose rate source terms from High Temperature Gas-Cooled Reactor (HTGR) spent fuel elements (Sund, Strong and Engholm 1977). Although these elements are of a different composition and geometry from a PWR or BWR, the thorium and uranium

TABLE 5.2. Gamma-Ray-Flux-To-Dose-Rate Conversion Factors
(American Nuclear Society Standards Committee 1977)

<u>Photon Energy-E (MeV)</u>	<u>DFg(E) (rem/hr)/ (photons/cm²-s)</u>
0.01	3.96-06
0.03	5.82-07
0.05	2.90-07
0.07	2.58-07
0.1	2.83-07
0.15	3.79-07
0.2	5.01-07
0.25	6.31-07
0.3	7.59-07
0.35	8.78-07
0.4	9.85-07
0.45	1.08-06
0.5	1.17-06
0.55	1.27-06
0.6	1.36-06
0.65	1.44-06
0.7	1.52-06
0.8	1.68-06
1.0	1.98-06
1.4	2.51-06
1.8	2.99-06
2.2	3.42-06
2.6	3.82-06
2.8	4.01-06
3.25	4.41-06
3.75	4.83-06
4.25	5.23-06
4.75	5.60-06
5.0	5.80-06
5.25	6.01-06
5.75	6.37-06
6.25	6.74-06
6.75	7.11-06
7.5	7.66-06
9.0	8.77-06
11.0	1.03-05
13.0	1.18-05
15.0	1.33-05

FIGURE 5.3. Fission Product Dose Rate from 10-Year-Old PWR Fuel Assembly

FIGURE 5.4. Source Shape Factor for a Fission Product Source
(Croff, Hermann and Alexander 1979)

fuel fission product spectrum resembles the PWR spent fuel fission product characteristics sufficiently to allow a gamma dose rate comparison. The gamma dose rates reported in this document are based on a point source and do not account for internal self-shielding by the fuel material itself. Because the fuel is taken to be a point source, no geometry factors (axial burnup) were taken into account in the GA study. The assumptions of the GA HTGR fuel elements gamma dose are given in Table 5.3. The HTGR fuel element from the GA study contained about 9 kg of U and Th. Since the GA results are for a single fuel element, the gamma dose rate results cannot directly be compared to PWR spent fuel gamma dose rates. However, the dose rates (rad/hr) can be normalized to 1 kg of either HTGR or PWR spent fuel. The fission product source terms (Ci/MT) are similar for both cases (4.2×10^6 Ci/MTU + Th for HTGR fuel and 2.2×10^6 Ci/MTU for PWR fuel). Figure 5.9 shows the gamma dose

FIGURE 5.5. Decay Factor for Fission Products
(Croff, Hermann and Alexander 1979)

rate at one meter away from an single HTGR fuel element. If the dose rate in Figure 5.9 is normalized to a kg fuel basis, the rate becomes about 95 rad/hr kg U + TH. The equivalent dose rate for PWR spent fuel fission products is about 3 rad/hr/kg U. The high gamma dose rate for the HTGR can be attributed mainly to the fact that self shielding by the fuel itself was not considered. Other factors such as a higher Ci source content, greater burnup, and geometry also contributed to a higher dose rate for the HTGR case.

Another report that gave gamma exposure for a PWR spent fuel assembly was a Safety Analysis Report for the West Valley Fuel Reprocessing Plant (Nuclear Fuel Services, Inc. 1973). The fission product gamma dose rate was from the design fuel, which was assumed to be irradiated to 30,000 MWd/MTU and then decayed for 150 days. A surface gamma dose rate of 1×10^6 rad/hr was obtained from a non-self-shielded line source model. Using Equation 1, a dose

FIGURE 5.6. Fission Product Dose Rate from 1-Year-Old PWR Fuel Assembly

FIGURE 5.7. Fission Product Dose Rate from 5-Year-Old PWR Fuel Assembly

FIGURE 5.8. Dose Rate in Air Calculation (ORNL) for 1-, 5- and 10-Year-Old Fuel

rate of 2×10^6 rad/hr can be calculated. Real comparison of these two numbers is probably not meaningful because of the differences in burnup and shielding assumptions.

An ORNL report on gamma ray skyshine from spent fuel storage basins reported a value of 2.6×10^4 rad/hr 1 cm from the top of a single fuel assembly (Pace, Cramer and Knight 1980). The decay time was 1 year with fuel

TABLE 5.3. HTGR Fuel Element Characteristics and Assumptions for Gamma Dose Rate Calculation (Sund, Strong and Engholm 1977)

Power, MW _{th}	3000
Number of Fuel Elements	3994
Average Power Density, watts/cm ³	8.5
Fuel Elements In-Core Time, years	4
Burnup, MWd/MT	94,000
Decay Periods, years	1-10
Dose Point to Source, meters	1
Shielding Medium	air

FIGURE 5.9. Gamma Dose Rate vs. Time After Discharge Curve for HTGR Fuel Elements (Sund, Strong and Engholm 1977)

exposed to 33,000 MWd/MTU. Apparently this ORNL calculation did not account for axial burnup and self-shielding effects. According to the ORNL report, discussed previously, (Equation 1) the contact dose rate (on top) would be <1000 rads/hr.

A report on irradiated fuel monitors (Dowdy and Caldwell 1979) also gave some dose rate information. This study dealt mainly with light water reactor assemblies in vertical storage racks and various methods to measure radiation from these types of sources. As part of the discussion on scintillators, several dose rates were given. The dose point is on one end with water, fuel and the assembly top structure attenuating the gamma rays. The dose rate in water was calculated at 0.077 m off the axis and 0.30 m from the end of a single UO_2 rod. In this configuration 99% of the dose rate comes from the top 0.30 m of the rod. The calculations were made for a uniform distribution of gamma ray sources, with no buildup and unit probability of emission of the gamma rays of different energies. The authors extrapolated the single rod results to arrive at a dose rate of 500 rad/hr, 0.30 m above the top of a fuel assembly for a recently exposed assembly and 2 rad/hr after long-term storage. The exact decay times were not given in the report.

A dose rate in air curve was developed in a spent fuel storage, handling and transportation baseline description report by Sandia Laboratories (Mayer and Sonnier 1978). This curve was developed for a single PWR fuel assembly in air with the dose point 1 meter from the assembly. The exact burnup history of the fuel assembly was not reported. Apparently, self-shielding effects of the fuel material were not accounted for in the dose rate results. The dose rate for 1-year spent fuel was reported to be approximately 3×10^5 rad/hr. The dose rates reported by Sandia are a factor of 10 higher than other reported values.

Several dose rate calculations results appeared in Pacific Northwest Laboratory (PNL) reports. A contact dose rate of 2×10^6 rem/hr was reported for a spent fuel assembly 100 days old with an exposure of 80,000 MWd/MTU (Dietrich, Moryson and Schmiedel 1971). Self-shielding effects and exact dose point location were not discussed in the document. Another PNL report documented gamma source strength and spectra calculations with ORIGEN and dose

calculations using ANISN for a 10 kg sphere of UO_2 fuel (Boegel et al. 1978). The dose rate in air at 1 meter from the fuel was reported to be 5×10^4 rad/hr at 0 days decay, 40 rad/hr at 300 days and <2 rad/hr at 1500 days. In each case the fuel exposure considered was only 300 MWd/MT. Comparisons to other results are difficult because of the low exposure values.

PNL Gamma Dose Rate Calculations

As part of this study, several preliminary gamma dose calculations were performed. Because of the nature of this study the calculations should be considered only as rough first order approximations. The ISHOSHL code was used to arrive at gamma dose rates (Engle and Greenborg 1966; Simmons 1967). Attenuation calculations were performed using point kernel integration by Simpson's rule. The dose at the desired point is the contribution from many individual point sources and the numerical integration carried out over the source volume to obtain the total dose. One option of the ISOSHL code is to use a subroutine to calculate the quantity of all radioisotopes in a fuel volume at various times following exposure. The fission product library contains gamma photon abundance for 450 fission products. Photon yields are calculated into 25 energy groups, which are shown in Table 5.4. ISOSHL prints gamma exposure rates (rads/hour) for each decay period. The information required to make the ISOSHL calculations is power, irradiation time or exposure, ^{235}U absorption cross section, plus other factors such as conversion ratios, fast fission factors, cooling times and shield/source densities.

A single PWR spent fuel assembly was modeled as a cylinder 360 cm long with a 12 cm radius. The parameters given previously in Tables 2.1 and 2.2 were used to determine the cross-sectional areas of the fuel, cladding and gaps and thus, the material densities. These densities are given below in Table 5.5. The gamma exposure rate at decay times of shutdown, 1, 2, 5, and 10 years after discharge were calculated. Three dose points in air were considered: (1) surface or contact, (2) 1 meter, and (3) 10 meters. A schematic drawing of the various dose points is shown in Figure 5.10. All of the dose points were perpendicular to the midplane point. Note that in comparison with the ORNL results, the PNL numbers do not account for axial burnup distribution.

TABLE 5.4. Energy Group Structure for ISOSHLD Calculations

<u>Group</u>	<u>Group Average Energy MEV</u>
1	1.500-002
2	2.500-002
3	3.500-002
4	4.500-002
5	5.500-002
6	6.500-002
7	7.500-002
8	8.500-002
9	9.500-002
10	1.500-001
11	2.500-001
12	3.500-001
13	4.750-001
14	6.500-001
15	8.250-001
16	1.000+000
17	1.225+000
18	1.475+000
19	1.700+000
20	1.900+000
21	2.100+000
22	2.300+000
23	2.500+000
24	2.700+000
25	3.000+000

TABLE 5.5. Fuel Composition for ISOSHL D Calculation

Material	Density (g/cm ³)
U	3.1
O	0.41
Zr	0.60
Air	7.56×10^{-4}

FIGURE 5.10. ISOSHL D Gamma Exposure Rate Points

The ISOSHLD type of calculation is most accurate near the midplane of the assembly. ISOSHLD calculations for dose point positions off the assembly midplane were also made. The dose rate 1 meter from a 10-year-old PWR spent fuel assembly even with the top of the assembly is ~762 rad/hr. ORNL calculated a dose rate of 340 rad/hr. The PNL calculated dose rate 1 meter from the midplane was 1274 rad/hr, while the ORNL study reported a dose rate of 1180 rad/hr. On the ends of the assembly the ISOSHLD predicts an exposure rate twice the value reported by ORNL. The dose rate can be calculated to within 30% of the ORNL results using the uniform source ISOSHLD method for the middle 200 cm of the fuel assembly. At a distance of 80 cm from either the top or bottom of the fuel assembly the ISOSHLD dose rate is considerably inaccurate.

The results of the PNL calculations can be seen in Figures 5.11 through 5.13 as a function of decay times, while Figure 5.14 gives the dose rate in air versus distance curve for three decay periods.

5.2 SUMMARY OF GAMMA DOSE CALCULATIONS

Figure 5.15 shows the comparison of the ORNL and PNL gamma dose calculation for two decay periods. All of the dose rates are for PWR fuel assemblies and not BWR's. During the course of this project, no documented reports that gave results of BWR gamma dose rate calculations were found in the literature.

FIGURE 5.11. Single PWR Fuel Assembly ISOSHL D Results as a Function of Decay Time for 1 Meter Midplane Distance

FIGURE 5.12. Single PWR Fuel Assembly ISOSHL D Results as a Function of Decay Time for the Assembly Surface

FIGURE 5.13. Single PWR Fuel Assembly ISOSHLD Results as a Function of Decay Time for 10 Meters Midplane Distance

FIGURE 5.14. PWR Fuel Assembly Dose Rate in Air as a Function of Distance for 1-, 5-, and 10-Year-Old Fuel

FIGURE 5.15. Dose Rate Comparison of ORNL and PNL Calculations

6.0 MEASURED GAMMA DOSE RATES

Documented, available experimental data on gamma dose measurements were very limited. Numerous reports detail gamma measurements for nondestructive assay or burnup verification reasons, but few reports give details of gamma dose measurements in air or even total gamma dose rate measurements of spent fuel assemblies in water basins. The major emphasis of this chapter is on gamma dose measurements using TLDs made during the course of nondestructive assay tests within a hot cell at Battelle-Columbus Laboratories. Lithium fluoride is the most widely used TLD material. When exposed to gamma radiation, secondary electrons are produced by photon interactions in the medium surrounding the detector, as well as those electrons generated by photon interaction within the detector itself. The secondary electrons are trapped in the material. The TLD reading process consists of heating the crystalline material in the presence of a photosensitive detector, which converts the light emission into electric current. This readout can be calibrated in terms of the absorbed dose for a known source such as ^{60}Co and ^{137}Cs in a standard dose calibrator device. Two EPRI sponsored experiments, the first involving TLD measurements of a spent fuel assembly submerged in water and the second involving mockup PWR and BWR fuel assemblies, are also examined in this chapter. Finally, experiments reporting relative gamma activities (not dose) for safeguards and nondestructive/verification analyses will be reviewed because they reveal important information about photon transmission and geometry factors within PWR or BWR spent fuel assemblies.

6.1 GAMMA DOSE RATE MEASUREMENTS OF TURKEY POINT (PWR) SPENT FUEL ASSEMBLIES

In a joint experiment involving staff from Westinghouse Hanford Company, PNL and Battelle-Columbus Laboratories, five assemblies from the Turkey Point Unit 3, a Westinghouse PWR design operated by Florida Power and Light, were examined (in air) in a hot cell at Battelle-Columbus Laboratories (Davis 1980a). Assembly B03 (see Figure 6.1) was examined for gamma activity. The characteristics of the B03 fuel assembly are listed in Table 6.1 and the history of the Turkey Point discharged fuel is given in Table 6.2.

FIGURE 6.1. Core Map Showing Location of Gamma Dose Measurement Assembly During the Second Cycle of Reactor Operation of Turkey Point Unit 3 (Davis, 1980a)

TLDs were used in this experiment to measure gamma exposure rates. A total of 315 TLDs were exposed in this experiment. Capsules containing five TLDs were stacked side by side, wrapped in tissue paper and then in foil to minimize contamination and placed inside a gelatin container. The TLD capsules were divided into 4 groups of 13 for axial irradiations, one group of 7 capsules for perpendicular irradiations and 4 individual capsules for background and control. Figure 6.2 illustrates the TLD locations for gamma dose measurements of assembly B03. Each group of capsules was placed inside an

TABLE 6.1. Turkey Point Fuel Assembly Characteristics (Davis 1980a)

Vendor	Westinghouse
Type (Rod Array)	15 x 15
Assembly Characteristics	
Cross Section (cm)	21.4 x 21.4
Weight (kg)	644
Length (cm)	409.7
Guide Tubes	
Number	20
Material	Zr-4
Instrument Tubes	
Number	1
Material	Zr-4
Fuel Rods	
Number	204
Length (cm)	386.1
OD (cm)	1.07
Wall Thickness (cm)	0.062
Material	Zr-4
Active Fuel Length (cm)	365.7
Fuel	
Material	UO ₂
Enrichment	2.55% U ²³⁵
Density	92% TD
Weight/Assembly (kg)	448

TABLE 6.2. First Two Cycles for Turkey Point Unit 3 (Davis 1980a)

Date Changed	January 12, 1972
Date Discharged	November 25, 1975
Irradiation Time	827 full power days
Average Thermal Power	13.96 MW/Assembly
Operating Power Density	82.6 kW/liter
Core Average Linear Power	182 w/cm
Core Average Burnup	25,665 MWd/MTU

1. 30.48cm ABOVE ASSEMBLY FLAT
2. CONTACT IN MIDDLE OF FLAT
3. GUIDE THIMBLE TUBE
4. CENTRAL INSTRUMENT TUBE
5. PERPENDICULAR TO FUEL ASSEMBLY

FIGURE 6.2. Gamma Dosimetry Measurement Locations for Turkey Point Fuel Assembly

aluminum tube with a 30 cm distance between adjacent capsules. Four 400 cm tubes containing 13 TLD packets and the one 213 cm tube containing 7 TLD packets were placed in the hot cell and exposed for 10 minutes. As Figure 6.2 indicates, two of the tubes were exposed within the fuel assembly in a guide tube and instrument tube. Another aluminum tube containing TLDs was exposed resting on the surface of the assembly, and the fourth was exposed 30 cm above the fuel assembly. The 213 cm tube was irradiated perpendicular to the axis at the midplane. All measurements were made while the spent fuel assembly

rested horizontally on the hot cell table. The axial gamma exposure rates are given in Table 6.3 and Figure 6.3 and the perpendicular gamma dose rates are shown in Table 6.4 and Figure 6.4.

Another set of experiments to measure gamma dose rates from Turkey Point spent fuel was also performed on assembly D04 in a hot cell at Battelle-Columbus Laboratories. The assembly physical characteristics are identical to those given in Table 6.1 for assembly B03. However, because the burn history, cycle number and decay time are different for B03, the measured gamma dose rate data provide additional information about dose rates. The spent fuel data for assembly D04 are given in Table 6.5. The measurements were accomplished using TLD's placed in thin-wall aluminum tubes in the same configuration as shown in Figure 6.2. The axial gamma dose rates are given in Table 6.6 and the perpendicular gamma dose rates are listed in Table 6.7 and shown graphically in Figure 6.5.

6.2 GAMMA DOSE RATE MEASUREMENTS IN WATER OF POINT BEACH (PWR) SPENT FUEL ASSEMBLIES

Gamma dose rate measurements of Point Beach Fuel assembly D-14 were taken December 1977 at Battelle-Columbus Laboratories as part of an EPRI sponsored

TABLE 6.3. Axial Gamma Exposure Rates From Assembly B03
(Rad/hr) (Davis 1980a)

<u>Distance From Bottom of Fuel Assembly (m)</u>	<u>Instrument Tube Exposure Rate</u>	<u>Guide Tube Exposure Rate</u>	<u>Contact Exposure Rate</u>	<u>30.48 cm Away Exposure Rate</u>
0.3	3.87×10^4	1.75×10^5	1.32×10^3	1.74×10^3
0.61	1.33×10^5	1.79×10^5	3.26×10^3	3.25×10^3
0.91	1.74×10^5	1.69×10^5	2.24×10^4	6.02×10^3
1.22	1.80×10^5	1.67×10^5	3.81×10^4	8.39×10^3
1.52	1.81×10^5	1.74×10^5	4.29×10^4	8.03×10^3
1.83	1.71×10^5	1.65×10^5	5.67×10^4	1.18×10^4
2.13	1.68×10^5	1.72×10^5	5.88×10^4	1.23×10^4
2.44	1.68×10^5	1.76×10^5	4.86×10^4	1.20×10^4
2.74	1.72×10^5	1.65×10^5	6.33×10^4	1.24×10^4
3.05	1.67×10^5	1.25×10^5	6.23×10^4	1.31×10^4
3.35	1.63×10^5	4.77×10^4	6.57×10^4	1.29×10^4
3.66	1.27×10^5	7.18×10^3	6.26×10^4	1.17×10^4
3.96	6.29×10^4	1.16×10^3	5.79×10^4	1.03×10^4

FIGURE 6.4. Turkey Point Assembly B03 Dose Rate

project (Pasupathi, Pennin and Roberts 1978). Gamma dose data are not directly applicable to the dose-in-air cases discussed previously, but they can be used as a basis of comparison and to reveal additional gamma dose information. The Point Beach 1 reactor is a Westinghouse designed 14 x 14 PWR and is owned and operated by Wisconsin Electric Power Company. The gamma dose measurements were made in a water storage basin while the assembly was being prepared for

TABLE 6.5. Turkey Point Unit 3 Assembly D04 Operation Characteristics (Davis 1980b)

First Cycle Start	December 12, 1974
Discharge Date	November 19, 1977
Irradiation Time	851 full power days
Decay Time	1.8 years
Specific Power	31.2 MW/MTU

FIGURE 6.3. Axial Gamma Dose Rates For Variable Positions Along Turkey Point Assembly B03

TABLE 6.4. Gamma Dose Rates as Function of Distance From Midplane (Davis 1980a)

Distance (m)	Exposure Rate (R/h)
0	5.88×10^4
0.31	1.55×10^4
0.61	7.99×10^3
0.91	4.62×10^3
1.22	3.37×10^3
1.52	2.55×10^3
1.83	1.90×10^3
2.13	1.56×10^3

TABLE 6.6. Axial Gamma Exposure Rates From Turkey Point Fuel Assembly D04 (Rad/hr) (Davis 1980b)

<u>Distance From Bottom of Fuel Assembly (m)</u>	<u>Assembly Surface Exposure Rate (Rad/hr)</u>	<u>30.48 cm Away Exposure Rate (Rad/hr)</u>
0.3	8.81×10^4	7.42×10^3
0.61	8.91×10^4	1.21×10^4
0.91	9.30×10^4	1.49×10^4
1.22	6.57×10^4	1.62×10^4
1.52	9.25×10^4	1.62×10^4
1.83	9.00×10^4	1.71×10^4
2.13	9.52×10^4	1.63×10^4
2.44	9.37×10^4	1.58×10^4
2.74	9.27×10^4	1.61×10^4
3.05	8.25×10^4	1.54×10^4
3.35	5.48×10^4	1.42×10^4
3.66	1.29×10^4	1.10×10^4
3.96	2.75×10^3	7.29×10^3

TABLE 6.7. Gamma Dose Rate as Function of Distance From Midplane of Turkey Point Assembly D04 (Davis 1980b)

<u>Distance (m)</u>	<u>Exposure Rate (Rad/hr)</u>
0	6.30×10^4
0.31	1.86×10^4
0.61	9.88×10^3
0.91	6.81×10^3
1.22	4.91×10^3
1.52	3.92×10^3
1.83	3.34×10^3
2.13	2.85×10^3
2.44	2.24×10^3

FIGURE 6.5. Turkey Point Assembly D04 Dose Rate

shipment to the General Electric Facility for permanent storage. The fuel assembly and reactor data pertinent to the Point Beach gamma measurement are given in Table 6.8.

While fuel assembly D-14 was being moved from the pool storage rack to a shipping cask, it was suspended in an open area of the transfer pool. Individual TLDs were inserted into and removed from positions near one face of the assembly.

Individual fuel rods from assembly D-14 had been removed and replaced prior to gamma measurements. The assembly also had 12 rods from another assembly inserted into it. However, these variations are probably not significant in terms of gamma dose rates.

The dose rates were determined by reading TLD activation and comparing these readings to a TLD calibration curve of known activation. Gamma dose points were obtained at distances of 50, 75 and 100 cm from the assembly face and at vertical locations of 30.5, 45.7, and 266.7 cm from the fuel stack bottom.

TABLE 6.8. Point Beach Fuel Assembly Characteristics
(Pasupathi, Pennin and Roberts 1978)

Fuel Rod Array	14 x 14
Number of Fuel Assemblies	121
Fuel Rod O.D. (cm)	1.07
Fuel Rod Cladding Thickness (cm)	0.06
Overall Rod Length (cm)	386
Number of Grids/Assembly	7
Number of Fuel Rods/Assembly	179
Number of Guide Thimbles/Assembly	16
Active Fuel Weight (cm)	365.7
Total Fuel Weight/Assembly (kg)	391 (U)
Burnup at End-of-Cycle 3 (moved to spent fuel pit at this time)	23,400 MWd/MTU

The results of the gamma dose measurements are presented in Table 6.9 for several distances from the assembly. Shown in Figure 6.6 is a plot of the

TABLE 6.9. Point Beach Spent Fuel Dose Rates In Water
(Klingensmith 1978)

<u>Distance From Element Face (cm)</u>	<u>Location From Fuel Stack Bottom (cm)</u>	<u>Dose Rate rad/hr</u>
50	266.70	380
50	45.72	290
50	30.48	300
75	266.70	-
75	45.72	69.3
75	30.48	58.5
100	266.70	18.19
100	45.72	14.10
100	30.48	-

FIGURE 6.6. Point Beach Assembly D-14 Underwater Dose Rate (Klingensmith 1978)

underwater gamma dose rate as a function of assembly position and distance to the dose point. All of these measurements are for spent fuel cooling times of approximately 640 days.

6.3 GAMMA DOSE RATE MEASUREMENTS AT GENERAL ELECTRIC MORRIS OPERATIONS

Measurements of gamma dose rates in water for PWR and BWR spent fuel were made by General Electric at their Morris, Illinois, storage basin (Egar and Zima 1979). Spent commercial fuel is stored at the Morris facility in

subcritical arrays under 3 to 4 meters of water. PWR fuel is stored in arrays of four assemblies to a basket while nine BWR fuel assemblies are in a typical basket. The gamma dose measurements were made with ion chambers situated directly above the end of one assembly. Because a gamma dose component from the other assemblies is in the basket, the dose rates reported are high for a single assembly. Additional experiments to measure gamma dose rates both in air and water for a single assembly are under way; the results have not been published yet. The preliminary results of PWR and BWR fuel dose rates reported are reproduced in Figure 6.7.

FIGURE 6.7. Measured Gamma Exposure Rates Above Spent Fuel at General Electric Morris Operations (Egar and Zima 1979)

7.0 GAMMA SCANNING, NONDESTRUCTIVE TECHNIQUES AND MOCK-UP GAMMA DOSE EXPERIMENTS

Two experiments are the primary focus of this chapter. The first deals with an experimental mockup of a PWR and BWR fuel assembly and determination of gamma dose factors within the assembly itself. This research was conducted by the University of Virginia and sponsored by EPRI. The second experiment, conducted by LANL, deals with the safeguards aspect of gamma doses from irradiated assemblies stored under water. Several other reports on gamma activity and nondestructive testing are also reviewed.

7.1 GAMMA-RAY ENERGY DEPOSITION IN LIGHT WATER REACTOR FUEL ELEMENTS

As shown in Figure 7.1, in this project a mock-up experimental array was constructed consisting of a 10 x 10 array of Zircaloy cladding tubes about 61 cm long (Bass and Johnson 1978). This construction closely approximates a section of a typical PWR fuel assembly. If a BWR configuration had been needed, a central row and column of fuel pins could have been removed, thus simulating the cruciform gap between fuel elements. Monoisotopic sources (thin disks) were inserted between the fuel pellets to simulate an actual fission product source. Detectors (TLDs) were placed in and around the experimental array. The homogenous PWR array was the basic geometry. Control rod holes and instrumentation tubes were simulated by removing "fuel" pins or "fuel" in selected pins. The sources used were ^{60}Co and ^{137}Cs , typical of the major fission product sources. The dose measurements along each water channel or air gap were used to arrive at an energy absorption distribution, which revealed the percent of source energy absorbed in various fuel pin lattice positions. The energy distribution pattern for the 0.66 MeV ^{137}Cs source is shown in Figure 7.2.

7.2 NONDESTRUCTIVE EXPERIMENTAL MEASUREMENTS OF GAMMA ACTIVITIES IN SPENT FUEL

Another report that provided some useful information on gamma doses and photon transmission within fuel assemblies was a LANL document (Rinard 1979).

FIGURE 7.1. Experimental Assembly Mock-Up (Bass and Johnson 1978)

Gamma doses in an ionization chamber near an irradiated assembly under water were calculated and compared to measurements of PWR and BWR assemblies. The author of the report approximated the spent fuel assembly as a line source parallel to the assembly axis capped with end disk sources.

BWR irradiated fuel assemblies were examined using a Ge detector, gross gamma detector and $\text{Be}(\sigma, n)$ detector. The gross gamma detector was positioned

SOURCE PIN (0.66 MEV ^{137}Cs)

FIGURE 7.2. Percent of Source Energy Absorption Pattern in Fuel Pins for the 0.66 MeV ^{137}Cs Source (Bass and Johnson 1978)

33 cm under water from the centerline of the assembly. The BWR assemblies used were 9 x 9 arrays of pins with 16.6 cm sides. Fuel length was 177.8 cm and the calculational model source had a 9.27 cm radius. Figure 7.3 shows the relative dose rate slope for the BWR fuel assembly. According to the LANL document the dose rate under water is halved every 9 to 10 cm. The study also showed a comparison of the calculated line source and measured values for various distances (x-plane) away from the assembly. The calculated and measured dose rates showed close agreement.

The PWR fuel assemblies in the LANL study had approximately 200 fuel pins in a 21.6 cm square array. For the line source calculation a cylinder 365.8 cm long with a radius of 10.8 cm was used. Total gamma intensities and doses at several axial positions were obtained from the Ge detector in a similar manner to the BWR experiments. As was the case with the BWR fuel assemblies, the line source adequately described the dose rate from a PWR fuel assembly. Figure 7.4 shows axial scans for different positions for the Ge detector and the calculated dose rate values. In the case of the PWR, the line source accounts for

FIGURE 7.3. Relative Dose Rates for BWR Fuel Assembly (Rinard 1979)

the dose at and beyond the assembly edge. The more uniform axial flux for the PWR caused the end of the line source to dominate the disk source. Because of the large self-absorption of the UO_2 fuel, disk source approximations are needed to account for end dose rates. Most of the assembly away from the detector may be expected to make only minor contributions to the total dose rate. Only the upper $3/\mu_s$ contribute to the dose rate (μ_s is the linear attenuation coefficient (cm^{-1}) for UO_2 : ^{137}Cs - 1.24, ^{95}Zr - 1.11, ^{144}Pr - 0.561).

The LANL study also examined the relative contribution from gamma rays of several major fission products. The average burnup for the BWR assemblies was approximately 12,000 MWd/MTU with a cooling time of about 900 days. The PWR fuel assemblies were exposed an average of 25,000 MWd/MTU and cooled for about 500 days. For decay times up to 2 years the most important contributors for

FIGURE 7.4. Comparison of Measured and Calculated Relative Dose Rates for PWR Spent Fuel at Several Distances from the Assembly (Rinard 1979)

both PWR and BWR fuel are ^{95}Nb , ^{95}Zr and ^{144}Pr . At longer decay times the most important contributors are ^{134}Cs , ^{137}Cs and ^{154}Eu .

The attenuation of gamma rays within a fuel pin array depends on the energy level and activity distribution. For the 0.661 MeV ^{137}Cs gamma ray the attenuation of 50% of the photons occurs in about 0.5 cm of UO_2 . For a 1 cm diameter fuel pin ~30% of the photons are attenuated within the fuel material itself. Void formation in the fuel complicates the determination of the attenuation factor. Fuel temperature and burnup history affect the degree of void formation. The actual attenuation of gamma rays in a whole assembly is even more difficult to estimate because of nonuniform burnup characteristics. Corner measurements are preferable to side measurements for whole square

assembly assays. According to LANL, a 30% variation in the 1.6 MeV ^{140}La gamma activity was produced by a 10° angular variation of the detector. For corner measurements this same deviation caused a 1.5% variation of the ^{140}La activity. Only fuel rods nearest to the corner in front of the detector contribute to the measured gamma activity. A self-absorbing source can be approximated by

$$T = - \frac{1}{\mu_s} \ln (1-F) \quad (2)$$

where T cm is the thickness of the source of linear absorption coefficient μ_s (cm^{-1}), which absorbs a fraction F of the gamma rays present in the source. Shown in Table 7.1 are μ_s values for several nuclides.

The total gamma absorption and leakage from a spent fuel assembly can be estimated by comparing the photon flux values at the surface of the assembly to the total photon source rate within the assembly. The energy flux on the midplane surface of a two year PWR fuel assembly was found to be 3.11×10^{10} MeV/ cm^2 -sec. Using a radial surface area of 2.7×10^4 cm^2 , the gamma energy generation rate is 8.4×10^4 MeV/sec. The major contribution to the dose rate is from 0.63 MeV. At this energy the gamma source strength is 1.44×10^{16} MeV/sec. Thus, for the homogeneous cylindrical model only 6% of the gamma energy escapes or 94% of the gamma energy is absorbed within the assembly.

In a BWR assembly the rods at the edge of the array experience a higher burnup since they are close to a water gap. In an LANL document that reviewed nondestructive spent fuel assay methods, the corner rods were assumed to have 20% higher burnup than the central rods. Actual measurements showed burnup variations as high as 40%. The LANL study used a Monte Carlo gamma transport code to calculate the gamma ray transmission from the assembly. The collimator was considered to be perpendicular to the length of the fuel assembly submerged in water. Gamma transmission was defined as the ratio of the number of gamma rays that reach the detector to the number of gamma rays from the fuel assembly with a trajectory that would pass through the collimator and reach the detector. Figure 7.5 shows the activity distribution used in the LANL study.

TABLE 7.1. Linear Attenuation Coefficients for a Range of Gamma Energies and Fuel Types

<u>Nuclide</u>	<u>Gamma Energy (MeV)</u>	<u>Fuel</u>	<u>Density (gr/cm³)</u>	<u>μ_{s1} (cm⁻¹)</u>
¹⁴⁴ Ce	0.134	UO ₂	10.6	31.8
		UC	13.6	43.9
		U	18.9	64.1
¹³⁷ Cs	0.662	UO ₂		1.24
		UC		1.64
		U		2.32
⁹⁵ Zr	0.724	UO ₂		1.11
		UC		1.46
		U		2.06
¹⁴⁴ Pr	2.18	UO ₂		0.482
		UC		0.623
		U		0.870

Another experiment that contained some useful information on dose rate measurements from spent fuel assemblies was a study conducted by the Japan Atomic Energy Research Institute (JAERI) (Ohno and Matsuura 1980). In this experiment TLDs were placed in and around several fuel rod positions within a BWR spent fuel assembly. The 6 x 6 fuel assemblies were exposed an average of 5639 MWd/MT and had a decay time of 8 years. Table 7.2 gives some of the characteristics of the BWR fuel assembly used in these experiments. The distribution of relative gamma dose rate in the BWR spent fuel assembly is shown in Figure 7.6. The TLD measurement positions were about 1.12 m from the bottom of the active fuel. At this position the highest reported gamma dose rate was measured at 4×10^3 rad/hr. Using the relative dose rate curve as an approximation, the contact dose rate would be around 2500 rad/hr.

FIGURE 7.5. ^{137}Cs Activity Distribution in BWR Fuel Assembly
(Hsue et al. 1978)

TABLE 7.2. BWR Fuel Rod and Assembly Data for JAERI Experiments
(Ohna and Matsura 1980)

Fuel Material	UO ₂
Enrichment	2.6 wt%
Pellet Diameter, mm	12.5
Pellet Density, g/cm ³	10.4
Cladding Material	Zircaloy-2
Cladding O.D., mm	14.12
Cladding Thickness, mm	0.76
Active Fuel Length, mm	721
Number of Fuel Rods/Assembly	36

FIGURE 7.6. Relative Gamma Dose Rate Measured by TLDs in JAERI Experiments (Height of these measured positions is 116 cm from bottom of active fuel) (Ohno and Matsuura 1980)

8.0 MEASURED AND CALCULATED GAMMA DOSE RATE COMPARISON

In this chapter the gamma dose rates obtained by experiments are compared to the dose rates calculated using computer codes. The original request-for-proposal specified that the dose rates in air be calculated for 1- and 5-year-old BWR and PWR spent fuel. The only documented experimental gamma dose rates in air were those done at the BMI hot cell and the fuel age in this case was about two years. No experimentally determined gamma dose rates for BWR fuel assemblies were found. In addition, no documented calculations of BWR gamma dose rates were reported. Several documented measurements of gamma dose rates under water exist, however, and these results are compared to values calculated by PNL. For the air gamma dose rates, calculations by ORNL and PNL are used as examples.

Figure 8.1 shows the comparison of the measured gamma dose rates and the calculations by ORNL and PNL for Turkey Point assembly B03. The PNL calculations were ISOSHLTD runs for the specific fuel assembly characteristics (dimensions, number of fuel pins, fuel-to-air or water ratio) and its burnup history (exposure, power and decay time). Figure 8.2 shows the comparison of Turkey Point assembly D04 TLD measurements and ORNL and PNL calculations. The ORNL data were taken from an empirical relationship and are only approximate. These data are for a gamma dose point along the midplane. Figure 8.1 indicates that good agreement exists among the values cited.

A similar curve that compares the Point Beach gamma dose rates in water for experimental and calculational methods can be constructed. The PWR fuel is about 1 3/4 years old and these experimental data come from measurements sponsored by EPRI. PNL ISOSHLTD cases serve as the calculated data points (Figure 8.3).

8.1 DOSE RATE VERSUS DISTANCE IN AIR CURVE FOR PWR SPENT FUEL

Based on the limited experimental gamma dose rate measurements available, a dose versus distance curve is recommended for both 1-year-old and 5-year-old fuel (see Figure 8.4). No measured BWR dose rates were found and thus a dose rate in air curve could not be determined.

FIGURE 8.1. Comparison of Measured and Calculated Gamma Dose Rates for Turkey Point Assembly B03

FIGURE 8.2. Comparison of Measured and Calculated Gamma Dose Rates for Turkey Point Assembly D04

FIGURE 8.3. Comparison of Measured and Calculated Gamma Dose Rates for Point Beach Assembly D-14 Under Water

FIGURE 8.4. Dose Rate in Air Curve for 1- and 5-Year-Old PWR Spent Fuel

9.0 CONCLUSIONS

Good agreement exists between the calculated gamma dose rates and measured dose rate values reported in experiments. For the two experiments of PWR air dose rate measurements, PNL calculation results were 25% lower than the measured values. The PNL calculations vary significantly (30 to 50%) compared to the measured values at dose point locations toward the ends of the assembly. Estimates of the air dose rates extrapolated from the ORNL calculated data were in closer agreement to the measured values (~10% lower). Several possible reasons could account for the differences between the calculated and measured results.

- Uncertainties existed in calibrating the TLDs.
- During the measurements, TLDs were placed within thin metal tubes.
- The spent fuel assembly was resting on a hot cell table so some gamma scattering occurred.
- The effect of assembly hardware shielding on some of the TLDs was difficult to assess.

Each batch of TLDs had a slightly different calibration factor because of the material properties and annealing process. In the case of the measurements reviewed in this study the uncertainty associated with the measured gamma dose rates was at the most 9%. This uncertainty is based on statistical counting errors and does not account for experimental errors.

All of the calculated gamma dose rates were for direct gamma contribution from the assembly and did not account for scattering. In actuality, the spent fuel assembly was resting on a table inside a hot cell. Gamma radiation emerging from the assembly could possibly collide with the hot cell wall and if not absorbed within the wall, scattered into the detector. The scattered photon flux depends on many factors such as photon energy, scattering angles and electron density of the hot cell wall. The complex geometry makes estimating the scattered gamma flux difficult. As the detector point moves away from the fuel assembly, the scattering component becomes more important.

Air scattering of the gamma rays was also neglected in the calculations. The wall scattered gamma dose rate contribution could be as high as 10% of the direct beam gamma dose rate.

Significant discrepancies occurred between calculated dose rate studies. However, almost all of these differences can be traced to assumptions made in the studies. Listed below are some of the factors associated with reported dose rate calculations.

- Self-shielding by the spent fuel itself was not taken into account in some studies.
- Some dose rate results are valid for a few fission products only and not the entire inventory of possible fission product gamma emitters.
- Variation in the axial burnup of a spent fuel assembly was taken into account in only one or two studies. Axial burnup is a significant factor in dose rate determination.
- Many of the reported dose rates are for a point directly above a spent fuel assembly as in a spent fuel pool or basin. End effects such as self-shielding and burnup were not estimated correctly.
- The calculated results considered a fuel assembly in air and neglected scattering from walls of the hot cell or surrounding structure.
- Variations in the calculated dose rates can also occur if the spent fuel assemblies are modeled differently (right circular cylinder or line source, heterogeneity effects, smeared densities and power and burnup levels).

In the ISOSHL D calculations performed by PNL the accuracy of the information in the photon probability library is also a factor in dose rate determination. For decay times greater than about eight hours the ISOSHL D accurately calculates the gamma intensity and shield requirements. At decay times less than eight hours the calculated results are low. Some of the short-lived isotopes in the ISOSHL D photon library do not have gamma energies or emission probabilities. The error is largest at zero decay time (~25%). All of the

buildup factors and dose conversion factors used in the calculations are based on the latest information to minimize the error associated with these factors. Homogenizing the assembly from square to circular cross section and end hardware simplification have little effect on the results. Also, uncertainties in the composition of the fuel assembly have a negligible contribution for gamma dose calculations. Uncertainties in densities of the materials are due to composition generalizations used in the calculations. These generalizations are introduced with methods such as homogenizing the fuel assemblies while ignoring steel support structure.

No dose rate calculations and no documented gamma dose rate measurements were found for a single BWR spent fuel assembly in air. Therefore, no baseline dose rate curve for BWR spent fuel can be recommended. The PWR dose rate curve was shown in Figure 8.4.

As a result of this study, the need for additional information has been identified in the following areas:

- Additional TLD gamma dose rate measurements are needed for PWR spent fuel.
- Dose rate calculations for BWR spent fuel are needed.
- Hot cell experiments to measure the gamma dose rate for BWR spent fuel are required.

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APPENDIX A

DOSE RATE FROM SPENT FUEL IN STORAGE BASIN

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DOSE RATE FROM SPENT FUEL IN STORAGE BASIN

The gamma dose rate from a single PWR spent fuel assembly in water was calculated using the ISOSHL code. The dose point was taken to be directly above the fuel assembly approximately 1 meter above the water level. Figure A.1 shows the arrangement of the fuel assembly in the storage pond and the close point.

FIGURE A.1. Spent Fuel Basin Geometry

The fuel assembly was modeled as a right circular cylinder with a 12.0 cm radius and 360 cm height. The fuel was homogenized and burned for 1095 full power days and 33,000 MWd/MT. The results of the ISOSHL D calculation are below in Table A.1. The actual dose rates are less than those given in Table A.1 because axial burnup was not accounted for in this ISOSHL D case.

TABLE A.1. Gamma Dose Rates from PWR Fuel Assembly Stored in a Basin

<u>Depth</u>	<u>Decay Period</u>	<u>Dose Rate (rad/hr)</u>
10' (3 m)	0	9.4×10^{-2}
	1 hour	2.2×10^{-2}
	4 days	8.6×10^{-4}
15' (4.6 m)	0	1.4×10^{-4}
	1 hour	2.6×10^{-5}
	4 days	8.1×10^{-7}
25.5' (7.7 m)	0	3.3×10^{-7}
	1 hour	4.7×10^{-11}
	4 days	8.3×10^{-13}

APPENDIX B

SHIELDING CALCULATIONS FOR PWR SPENT FUEL ASSEMBLY

APPENDIX B

SHIELDING CALCULATIONS FOR PWR SPENT FUEL ASSEMBLY

Gamma shielding calculations were performed for a single PWR spent fuel assembly with both water and concrete as the shielding medium. The arrangement of the fuel assembly and location of dose points is shown in Figure B.1. The assumptions made concerning the spent fuel assembly are listed in Table B.1. The fuel was homogenized as fuel, structure and air (or water) with slab shield for the water or concrete zone. The concrete was considered to be ordinary concrete (2.35 gm/cm^3) or magnetite high-density concrete (3.68 gm/cm^3). The results of the ISOSHLD calculations are given in Tables B.2 and B.3.

FIGURE B.1. Dose Point Location for PWR Spent Fuel Assembly

TABLE B.1. ISOSHL D PWR Spent Fuel Assumptions

Fuel Exposure	33,000 MWd/MT
Decay Period	90 days and 5 years
Fuel Weight	460 kg/assembly
Flux	4.25×10^{13} n/cm ² -sec
Core Residence Time	3 years

TABLE B.2. Results of Water Pool Calculations

<u>Fuel Decay</u>	<u>Water Depth</u>	<u>Dose Rate (mrem/hr) at Water Surface</u>
90 Days	1.5 m	6.22 + 3
	3 m	1.25 + 0
	4 m	1.54 - 2
5 Years	1.5 m	2.29 + 2
	3 m	3.54 - 2
	4 m	3.82 - 4

TABLE B.3. Results of Concrete Slab Calculations

<u>Fuel Decay</u>	<u>Concrete Thickness & Type</u>	<u>Dose Rate vs. Distance (mrem/hr)</u>		
		<u>At Surface</u>	<u>At 1 m.</u>	<u>At 5 m.</u>
90 Days	152.4 cm - magnetite	2.12 - 4	1.33 - 4	4.70 - 5
5 Years	152.4 cm - magnetite	2.86 - 6	1.80 - 6	6.38 - 7
5 Years	152.4 cm - ordinary	8.82 - 3	5.51 - 3	1.79 - 3

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