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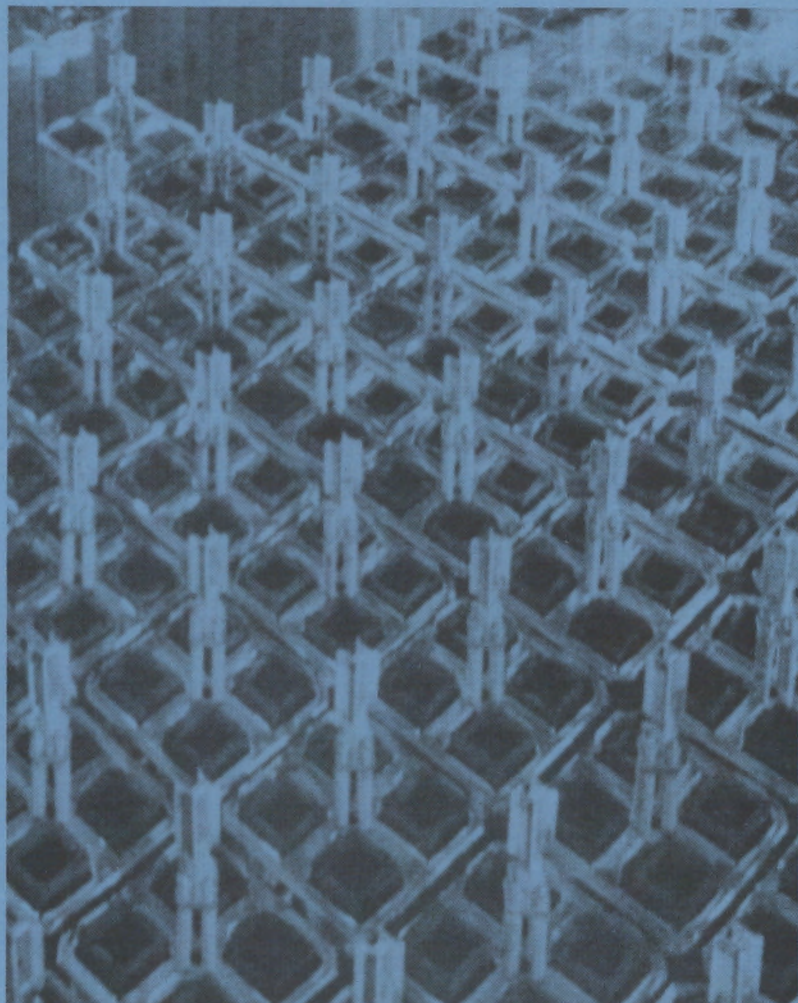
Monticello BWR Spent Fuel Assembly Decay Heat Predictions and Measurements



Prepared for the U.S. Department of Energy
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MONTICELLO BWR SPENT FUEL ASSEMBLY DECAY HEAT PREDICTIONS AND MEASUREMENTS

M. A. McKinnon
J. W. Doman[a]
C. M. Heeb
J. M. Creer

June 1986

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Pacific Northwest Laboratory
Richland, Washington 99352

(a) General Electric-Morris Operation,
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This report presents the results of analyses and experimental work conducted by the Pacific Northwest Laboratory (PNL) and General Electric-Morris Operation (GE-MO) for the U.S. Department of Energy (DOE). The authors would like to acknowledge those from DOE, GE-MO, Monticello, and Los Alamos National Laboratory (LANL) who contributed to the project.

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Appreciation is extended to Phillip Rinard and Gene Bosler of LANL for providing the ION-1 system used to obtain axial radiation profiles of each spent fuel assembly. Their assistance in setting up the instrument and in training the GE-MO staff in its operation was helpful.

Thanks are extended to Northern States Power Company's Monticello Nuclear Generating Plant for use of their spent fuel assemblies.

EXECUTIVE SUMMARY

This report compares pre-calorimetry predictions of decay heat rates of six 7x7 boiling water reactor (BWR) spent fuel assemblies with measured decay heat rates. The assemblies were from Northern States Power Company's Monticello Nuclear Generating Plant and had burnups of 9 to 21 GWd/MTU and cooling times of 9 to 10 years. This study is an extension of the decay heat work performed with Commonwealth Edison's Dresden Nuclear Power Station BWR spent fuel and Nebraska Public Power District's Cooper Nuclear Station BWR spent fuel by the Electric Power Research Institute (EPRI) and U.S. Department of Energy (DOE) to evaluate the ORIGEN2 computer code. The Dresden fuel had burnup values of 5 GWd/MTU and cooling times of 11 to 12 years, and the Cooper fuel had burnup values of 20 to 28 GWd/MTU and cooling times of 2 to 4 years.

Predictions were made prior to calorimetry to facilitate an unbiased comparison of predictions with measurements. The predictions were made with a standard version of the ORIGEN2 computer code using the standard BWR cross section library for a ^{235}U enriched fuel. Before the code was used for the decay heat predictions in this study, it was compared to another standard version of the code on a different computer and found to be operating satisfactorily.

Two types of measurements were made on the fuel assemblies: decay heat measurements and axial radiation measurements. The decay heat measurements were made using an existing in-pool calorimeter that was designed, constructed, and tested by General Electric's Morris Operation (GE-MO) for DOE. Concurrent with its use in this test program, the calorimeter was calibrated using an electric heater. Corrections were applied to spent fuel assembly measurements to account for differences in heat capacity (ratio of heat supplied to corresponding temperature rise) and radiation losses between the calibration and measurement modes.

The predicted decay heat values were found to be 0 ± 15 W less than measured values when using a 1984 calibration method, or 21 ± 2 W when a 1985 calibration method was used, as shown in Figure S.1. Precision depends on the measurement technique, and the accuracy is associated with the calibration

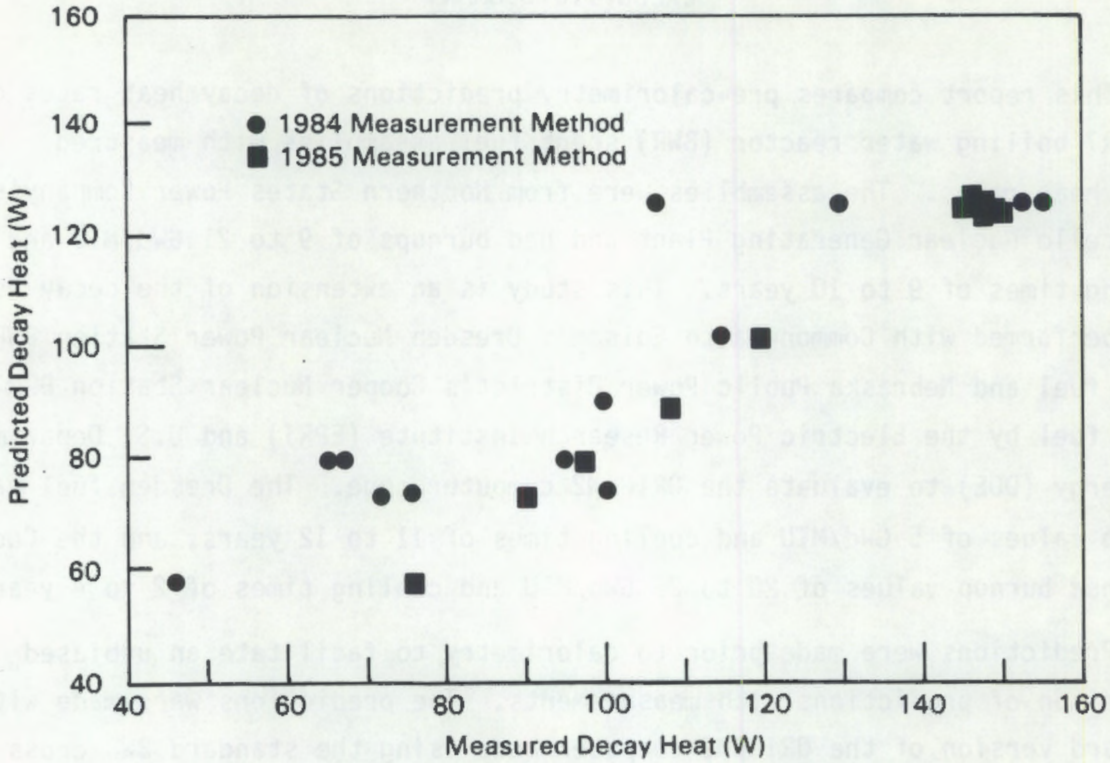

FIGURE S.1. Predicted and Measured Decay Heat Rates for Monticello Spent Fuel

process. The data collected in this study agree well with calorimetry data collected previously for Dresden and Cooper fuel assemblies, as shown in Figure S.2. This study also indicated that it may be possible to increase the precision of the calorimeter by reducing the uncertainty in the calibration process and by better defining and adhering to operating procedures.

A set of radiation measurements was made to determine the axial gamma and neutron profiles of each assembly. The measurements were made at nine pre-selected elevations with an ION-1/fork measurement system developed by Los Alamos National Laboratory and provided to GE-MO. Comparisons were made between predictions of axial decay heat profiles and gamma profile measurements. The pre-calorimetry predictions were based on core-averaged axial

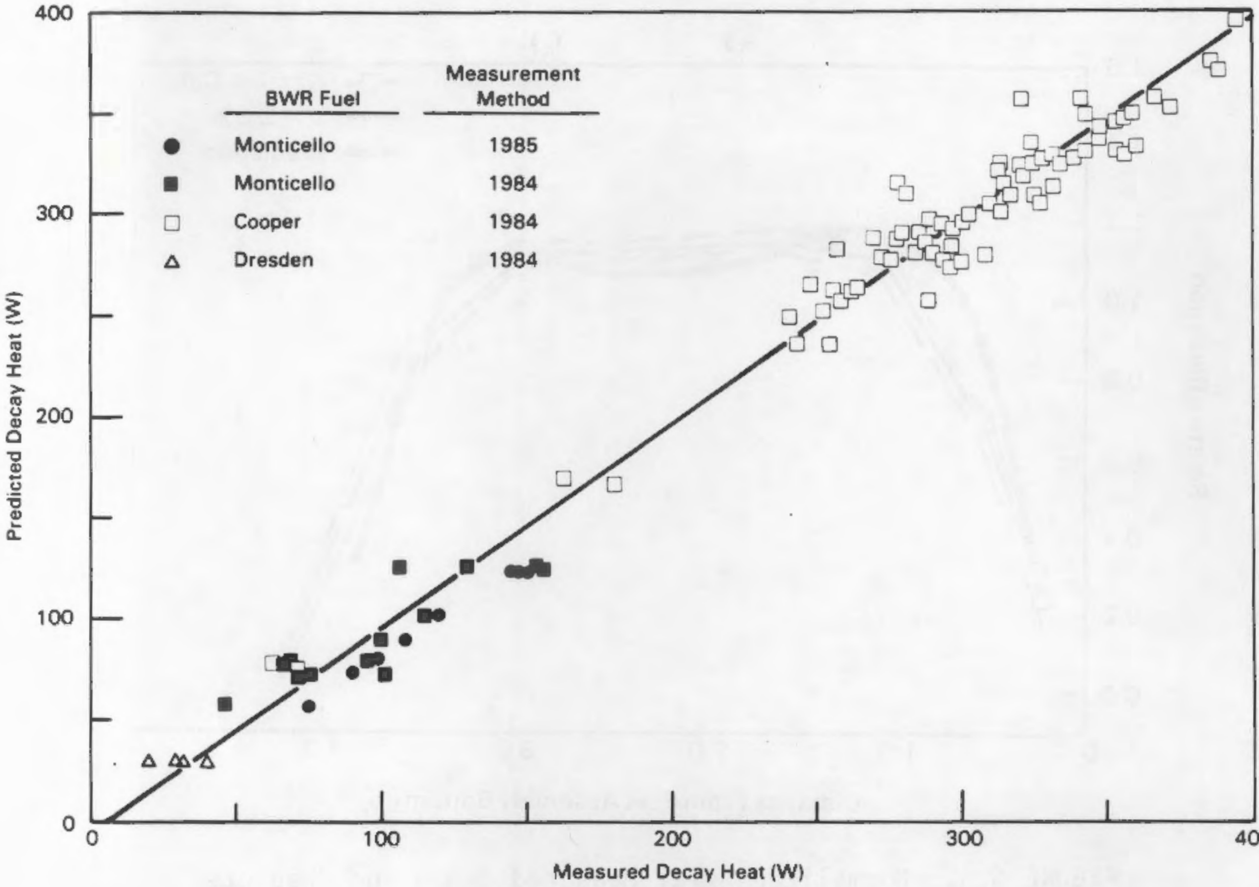

FIGURE S.2. Predicted and Measured Decay Heat Rates for BWR Spent Fuel Assemblies

burnup distributions and assumed that gamma-ray source strengths were proportional to burnup for burnups above 5.0 GWd/MTU. A comparison between predicted decay heat and measured gamma axial profiles is shown in Figure S.3. The agreement between the two curves is good.

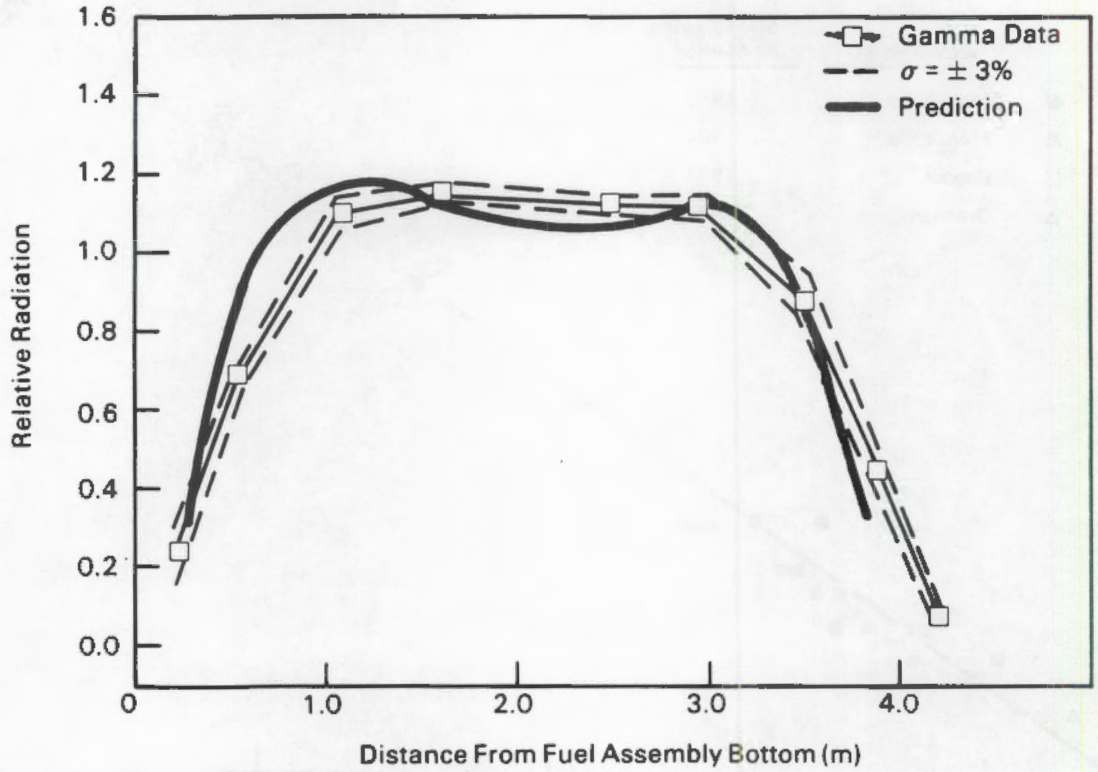

FIGURE S.3. Normalized Axial Measured Gamma and Predicted Decay Heat Profiles

CONTENTS

FIGURES

TABLES

ACRONYMS AND INITIALISMS

BWR	boiling water reactor
DOE	U.S. Department of Energy
EOC	end-of-cycle
EPRI	Electric Power Research Institute
GE-MO	General Electric-Morris Operation
H/U	hydrogen-to-uranium (ratio)
LANL	Los Alamos National Laboratory
NRC	U.S. Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act
ORNL	Oak Ridge National Laboratory
PNL	Pacific Northwest Laboratory
PWR	pressurized water reactor
RTD	resistance temperature detector

MONTICELLO BWR SPENT FUEL ASSEMBLY DECAY HEAT PREDICTIONS AND MEASUREMENTS

1.0 INTRODUCTION

No facility is currently licensed by the U.S. Nuclear Regulatory Commission (NRC) for final disposal or reprocessing of spent nuclear fuel. However, by virtue of the Nuclear Waste Policy Act of 1982 (NWPA), the federal government has made a specific commitment to accept commercially generated nuclear wastes for disposal by 1998. By the mid-1980s several reactor pool storage basins will have attained maximum capacity. As a consequence, interim storage of spent fuel must be provided. If it is not, the continued operation of nuclear-powered electric generation stations may be jeopardized. Options for dry storage of spent fuel at reactors are being developed by the utility industry and the U.S. Department of Energy (DOE) to minimize the possibility of reactor shutdowns due to lack of spent fuel storage capacity.

Because analyses and testing of dry storage systems are required to support NRC license applications, the utilities and DOE are actively pursuing research, development, and demonstration of dry storage systems. Experimental data to support at-reactor dry storage license applications could greatly reduce the effort and time required to process applications. However, the data must adequately characterize a storage system and must be obtained using well-documented procedures.

Accurate determination of spent fuel decay heat rates is a critical step in tests, demonstrations, and licensing activities associated with characterizing a storage facility because peak cladding temperatures are dependent on the heat generation rates of the spent fuel assemblies. To determine the maximum heat dissipation capacity of a spent fuel storage system, the total heat being generated must be accurately known.

For most tests and at-reactor demonstrations of dry storage systems, obtaining accurate spent fuel decay heat measurements is impractical. Only two calorimeters are known to exist that can be used to measure the decay heat output of complete spent fuel assemblies. One calorimeter is at the Engine

Maintenance and Disassembly Facility operated by Westinghouse Electric Corporation on the Nevada Test Site; the other is at General Electric's Morris Operation (GE-MO) in Morris, Illinois. Therefore, it is highly desirable to show that computer codes such as ORIGEN2 (Croff 1980a,b) can be used to accurately predict assembly decay heat generation rates.

The evaluation and determination of the accuracy of the ORIGEN2 computer program is extremely important to the success of tests and licensed at-reactor dry storage demonstrations. Also, future license applications for interim storage of spent fuel at reactors will be dependent on the accuracy of codes such as ORIGEN2 for decay heat predictions. A DOE study involving pressurized water reactor (PWR) calorimetry has demonstrated that ORIGEN2 predictions compare favorably with calorimetry data for PWR fuel (Schmittroth 1984). Similar studies sponsored by the Electric Power Research Institute (EPRI) and DOE investigated the ability to predict decay heat rates of BWR 7x7 spent fuel assemblies from Commonwealth Edison's Dresden Nuclear Power Station and Nebraska Public Power District's Cooper Nuclear Station (McKinnon et al. 1985, 1986). The Dresden spent fuel assemblies had burnup values of 5 GWd/MTU, cooling times of 11 to 12 years, and decay heat rates of less than 50 W. The Cooper assemblies had burnup values of 20 to 28 GWd/MTU, cooling times of 2 to 4 years, and decay heat rates near 300 W. The Dresden and Cooper spent fuel data did not allow the ORIGEN2 code to be adequately evaluated for intermediate decay heat values.

The objectives of the study reported herein were to perform pre-calorimetry decay heat predictions of selected Monticello BWR spent fuel assemblies with lower decay heat output than was possible with the Cooper fuel, and to compare predictions to in-pool calorimetry data. This report includes the results of the pre-calorimetry analysis, a description of the Monticello BWR spent fuel assemblies, calorimeter decay heat measurements, axial radiation scans, and a comparison of the pre-calorimetry predictions to experimental data. The results of this study show that ORIGEN2 can satisfactorily predict the decay heat of BWR fuel when the decay heat rate is 50 W or greater. The predictions were made prior to any data being available, to facilitate an unbiased comparison of predictions with measurements.

2.0 CONCLUSIONS AND RECOMMENDATIONS

The results of this study show that ORIGEN2 predictions of BWR spent fuel assembly decay heat rates agree with all experimental BWR calorimeter data within a standard deviation of ±15 W. The agreement between ORIGEN2 predictions and decay heat measurements of Monticello spent fuel is dependent on the method used to make the decay heat measurements and on the process used to calibrate the calorimeter. For the Monticello spent fuel studied, the predictions are within 0 ± 15 W and 21 ± 2 W for the 1984 and 1985 measurement methods, respectively. The accuracy of the calorimeter depends on the calibration process, whereas the precision of the measurement is related to the measurement method.

From this study and previous studies (Schmittroth 1984; McKinnon et al. 1985, 1986), it can be concluded that ORIGEN2 predicts decay heat rates of spent fuel assemblies satisfactorily when decay heat magnitudes are on the order of 50 W and greater. Spent fuel storage system tests and demonstrations simulating at-reactor or interim storage systems can be performed adequately using ORIGEN2 predictions of decay heat rates and do not absolutely require experimental calorimetry of each fuel assembly. However, to obtain satisfactory results, the ORIGEN2 predictions must be performed using detailed input information, especially burnup histories.

Results of this study are not applicable to old fuel that has very low decay heat rates. It is anticipated that decay heat predictions of actinides, where decay heats are significant in old, cold fuel, may be a problem and should be addressed. An evaluation of ORIGEN2 for predicting decay heat rates of old, cold fuel is required to verify prediction accuracies.

The following subsections present the specific conclusions and recommendations developed during this study.

2.1 CONCLUSIONS

The results of the decay heat predictions and measurements permit the following conclusions:

- The agreement between ORIGEN2 predictions and decay heat measurements of Monticello spent fuel is dependent on the method used to calibrate the calorimeter and to make the decay heat measurements.
- The agreement between predictions and measurements of decay heat rates of Monticello fuel is the same as that for Cooper and Oresden fuel if the same measurement method is used. The predictions are within a standard deviation of ±15 W of the measurements.
- Using a different measurement method, ORIGEN2 underpredicts the measured decay heat output of Monticello fuel assemblies by a constant 20 ± 2 W. The 20-W offset appears to be an artifact of the calibration procedure.
- The constant term in the calibration curve (i.e., $q_{DH} = mx + b$) can account for measurement differences of 40 W based on the 1983, 1984, and 1985 calibration curves.
- The difference between ORIGEN2 predictions and calorimeter decay heat measurements does not appear to be dependent on the magnitude of decay heat output.
- Predicted axial decay heat profiles are in good agreement with measured axial gamma radiation profiles.

2.2 RECOMMENOATIONS

The results and conclusions of this study led to the following recommendations:

- Predictions using other decay heat codes should be compared to experimental data contained in this report, to evaluate prediction capabilities.

- The source of the differences that exist among calorimeter calibration curves needs to be determined.
- Calorimeter operational methods need to be investigated further to determine cause and effect relationships between operational method and calorimeter precision and accuracy.

3.0 FUEL ASSEMBLY DESCRIPTION

The BWR spent fuel assemblies used in this study were from Northern States Power Company's Monticello Nuclear Generating Plant. The assemblies were of the 7x7 GE design. The design details are given in Table 3.1 and illustrated schematically in Figure 3.1. The upper and lower tie plates are 304 stainless steel castings. The lower tie plates have nose-pieces that support the fuel assemblies in the reactor. The upper tie plates have lifting bails for handling the fuel assemblies.

In addition to the standard fuel rods, each assembly has eight fuel rods that are used as tie rods that thread into the lower tie plate casting. The upper ends of the fuel/tie rods extend through and are fastened to the upper tie plate with stainless steel nuts and locking tabs. These fuel/tie rods support the weight of an assembly during fuel-handling operations when the assembly hangs by the bail. The center rod of each fuel assembly has been designed to maintain the position of the fuel rod spacers. It is inserted into the fuel assembly and rotated to lock the spacers into their respective locations. The spacers have Inconel springs to maintain rod-to-rod spacing. The fuel rods were pressurized with helium and sealed by welding end plugs on each end.

TABLE 3.1. Monticello Fuel Assembly Design Data

Fuel rods per assembly	49	
Active fuel length	3.658 m	(144 in.)
Assembly length	4.354 m	(171.4 in.)
Rod-to-rod pitch	18.7 mm	(0.738 in.)
Cladding outside diameter	14.30 mm	(0.563 in.)
Cladding thickness	0.813 mm	(0.032 in.)
Pellet outside diameter	12.4 mm	(0.488 in.)
Initial plenum pressure	1.0 atm	
Initial ^{235}U	2.25 wt%	
Zircaloy-2 weight	42.000 kg/ass.	(92.59 lb/ass.)
304 stainless steel weight	8.600 kg/ass.	(18.96 lb/ass.)

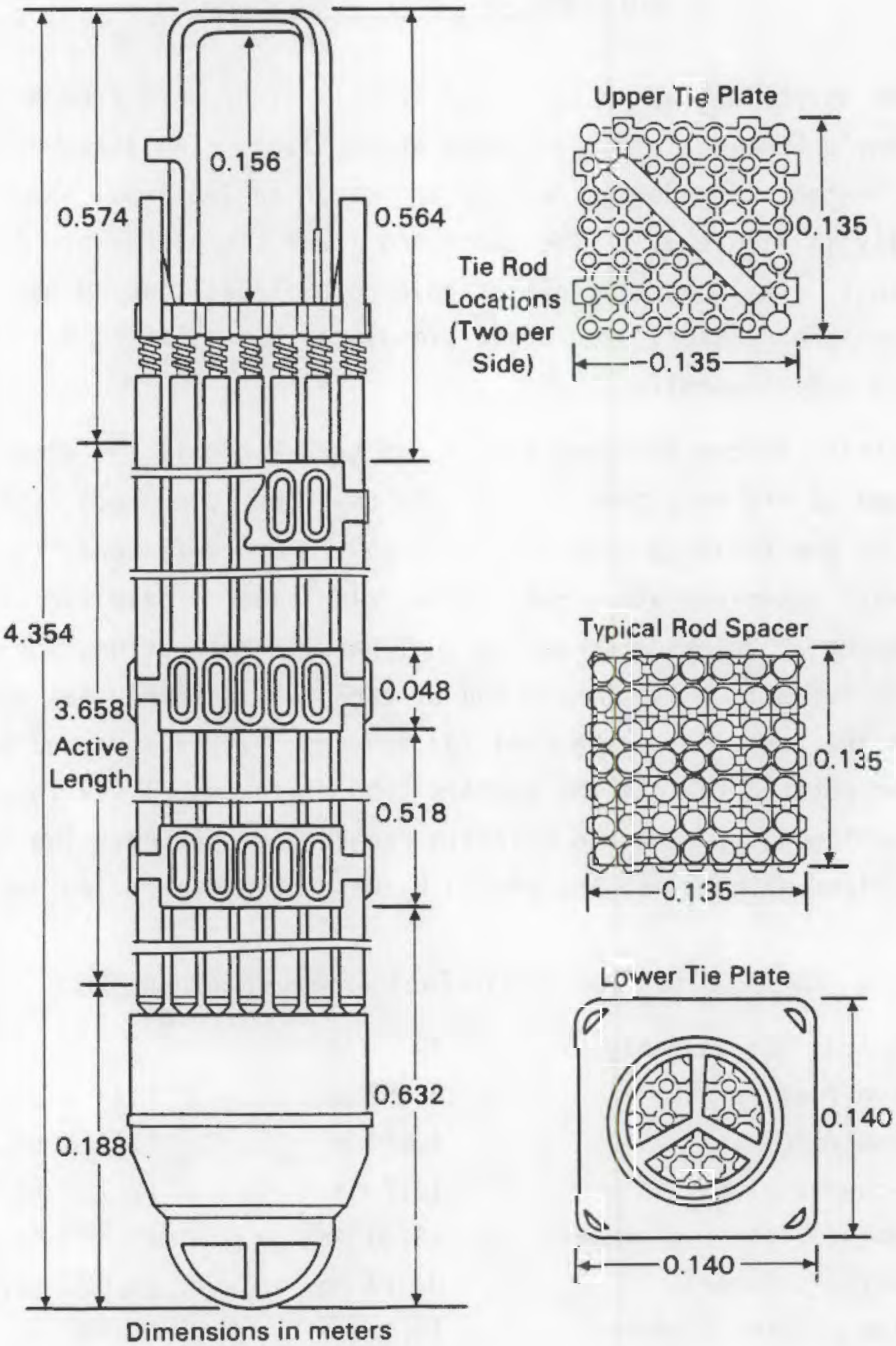

FIGURE 3.1. Monticello Fuel Assembly

The Monticello spent fuel assemblies had been out of the reactor for 9 to 10 years prior to calorimetry and had burnup values ranging from 9 to 21 GWd/MTU as indicated in Table 3.2. Burnup values were provided from two different utility accounting methods. One method, termed Form 30 reporting, is used by the utility to meet fuel storage requirements and lists only the final total burnup. The other method, referred to as Cycle Summary reporting, contains end-of-cycle (EOC) burnup values. Previous studies (McKinnon et al. 1985, 1986) have shown that cycle burnup values are required to make reasonable predictions of decay heat with the ORIGEN2 code. The information in Table 3.2 gives one source of cycle burnups and two sources of total burnups.

TABLE 3.2. Monticello Fuel Assembly Burnup Data

Assembly ID	Cycle Summary Burnup Values, MWd/MTU: Cycle 1	Cycle 2	Cycle 3	Cycle 4	Total	Form 30 Burnup, MWd/MTU	Ratio, Form 30 To Cycle Summary
MT116	8,294	4,583	1,389	3,215	17,482	18,040	1.03
MT123	9,074	5,078	-	-	14,152	13,030	0.92
MT133	8,552	4,786	3,452	3,398	20,189	21,000	1.04
MT190	5,054	2,927	3,495	3,836	15,312	15,150	0.99
MT228	3,652	1,936	3,458	3,524	12,570	12,130	0.97
MT264	4,059	2,068	1,915	2,047	10,089	9,160	0.91

(a) End of Cycle 2 - March 16, 1974.
(b) End of Cycle 4 - September 13, 1975.

4.0 MEASUREMENT EQUIPMENT AND EXPERIMENTAL DATA

In this section, the work performed at the General Electric-Morris Operation (GE-MO) facility is described. This work consisted of measuring the decay heat rates and axial radiation profiles of six Monticello BWR spent fuel assemblies. Decay heat rates were measured using an existing in-pool calorimeter previously designed, built, and tested by GE-MO in 1981 for DOE (Judson et al. 1982). Radiation profiles were obtained with a combined gamma and neutron measurement system developed at the Los Alamos National Laboratory (LANL) and referred to as the ION-1/fork measurement system (Halbig and Caine 1985).

4.1 CALORIMETER AND DECAY HEAT DATA

The in-pool calorimeter used at GE-MO for decay heat measurements is depicted schematically in Figure 4.1. Basically, the calorimeter is composed of two concentric pipes with an insulated annular space. The calorimeter is 4.6 m (15 ft) long and has a 0.4-m (16-in.) inner diameter. Fuel is placed in the calorimeter using a method very similar to that for loading a fuel transfer cask. The calorimeter cavity contains a fixed insert for PWR fuel and a removable insert for BWR fuel. These calorimeter inserts maintain fuel assemblies in centered vertical positions. The calorimeter utilizes resistance temperature detectors (RTDs) to measure temperatures and gamma sensors to quantify radiation losses.

During calorimetry, the system utilizes a Digistrip datalogger, a calibration tank, a sample pump, a purge system, a valve control panel, and gamma sensor readout devices. A heater power controller and a digital wattmeter are used during calibrations, but are not part of the normal equipment used during calorimetry. The calibration tank is used to prevent pressurization of the calorimeter, to leak-check the calorimeter after the fuel is loaded, and to collect calorimeter water samples.

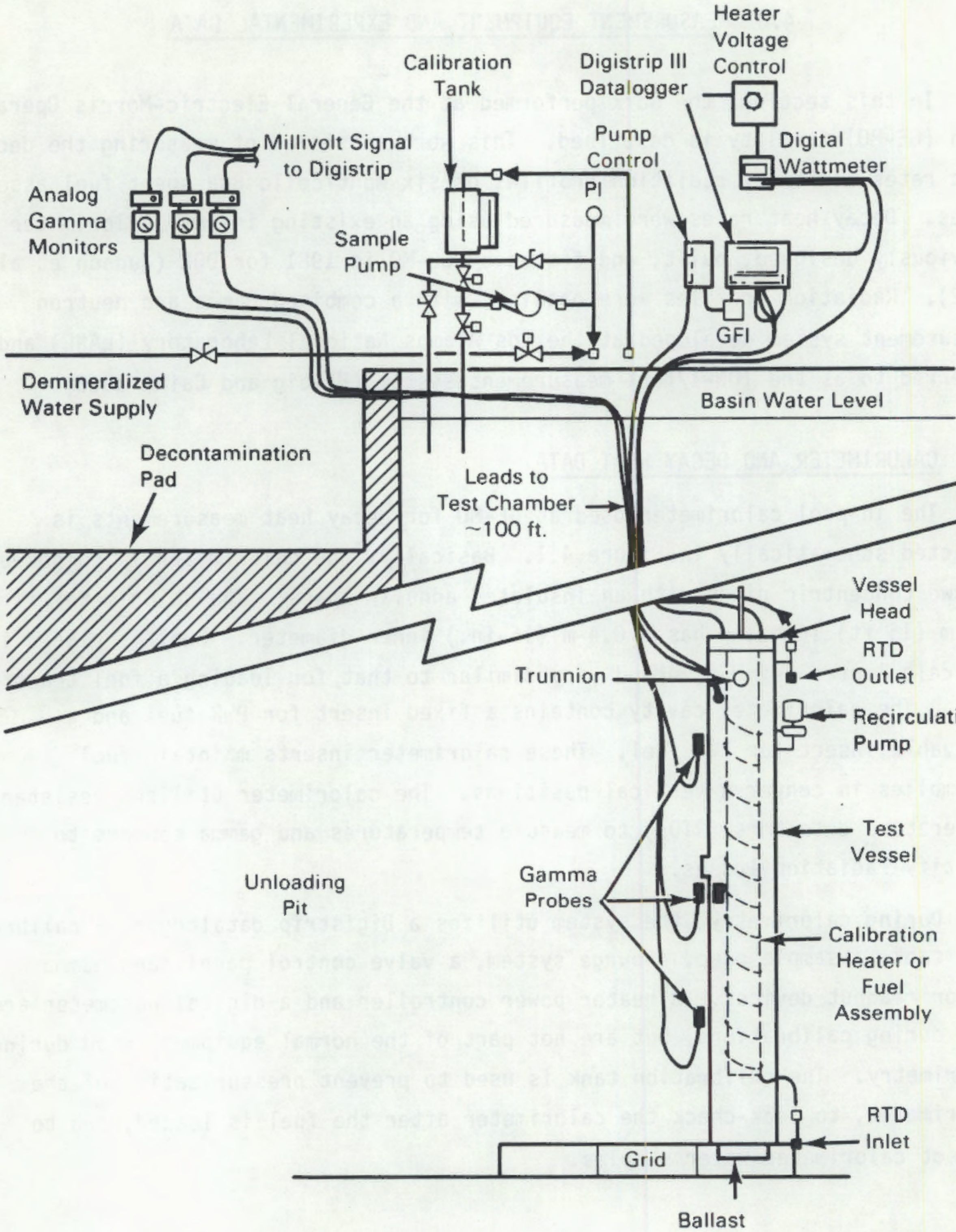

FIGURE 4.1. General Electric-Morris Operation In-Pool Calorimeter and Associated Equipment
Source: McKinnon et al. (1985, p. 3-2)

Two methods of calorimeter operation were used to obtain decay heat measurements of the Monticello spent fuel. The first method consisted of the following steps:

1. The vessel was purged to remove hot water.
2. The vessel was allowed to equilibrate with the surrounding basin water.
3. A delta T across the calorimeter of 0.0 ± 0.055°C was maintained for 1 hour.
4. A fuel assembly was lowered into place.
5. The head was torqued down and leak-checked.
6. Automatic data acquisition was initiated when the datalogger showed a temperature difference of 0.55°C between the interior vessel surface and outside skin.

Additional care was exercised to maintain the unloading pit water at a constant temperature during the runs. This method was used also in previous studies (McKinnon et al. 1985, 1986) involving Dresden and Cooper reactor spent fuel.

The second method was identical to the first except for the third step. In the modified third step a delta T across the calorimeter of 0.2°C was obtained. This modification relaxed the time requirement and did not involve such strict control of the temperature.

Two separate calibrations of the calorimeter were performed using the above operating methods and by replacing the spent fuel assembly with an electric heater. A known amount of energy was put into the calorimeter, and the increase in internal temperature with time was observed. The power delivered to the calibration heater was controlled by a variable power transformer and measured by a precision wattmeter. The datalogger continuously monitored the signal and printed out average power at 15-minute intervals. The datalogger power printouts were then averaged over a 5-hour interval to arrive at the "actual" power delivered to the calorimeter. This power was then corrected for the power lost in the heater leads external to the calorimeter.

Each run lasted 5 hours from the time the datalogger was put into automatic operation. In 1984, calibration runs were made at 0, 50, 100, 200, 300, 400, and 500 W, with a repeat run at 200 W. Calibration runs in 1985 were obtained at 0, 50, 100, 150, 200, and 300 W, with repeat runs at 50, 100, and 150 W. Two additional calibration runs were performed in 1985 using the 1984 method of operation to see if the 1984 calibration had shifted with time. The temperature-versus-time curves for the two calibration runs are shown in Figure 4.2.

The calibration curves shown in Figure 4.2 were converted to heat output in watts using the following technique: 1) a polynomial equation of the form $y = ax^2 + bx + c$ was determined for each heat-up curve; 2) the slope of each line at $t = 0$ was calculated; and 3) the relationship between "slope value" at $t = 0$ and power was determined. This relationship is expressed by the calibration curves shown in Figure 4.3. The relationships are linear with correlation coefficients of 0.9996 for the 1984 and 1985 calibration curves determined from linear regression analysis of the data. The calibration equations are

$$\text{Decay Heat} = 372.545*\text{Slope} - 85.975 \text{ for } 1984$$

and

$$\text{Decay Heat} = 381.66*\text{Slope} - 55.177 \text{ for } 1985$$

where the slope values are taken at time zero. The two calibration points taken in 1985 using the 1984 method of operation are also plotted in Figure 4.3. They do not fall on either of the calibration curves.

Table 4.1 lists the slope values and index of determination values for the 1984 and 1985 calibration runs. As can be seen from Table 4.1, the polynomial equations gave very good fits to the data above 50 W (index of determination values generally greater than 0.9999). The repeatability of the individual calibrations based on repeat runs at 50, 100, 150, and 200 W was about 1%.

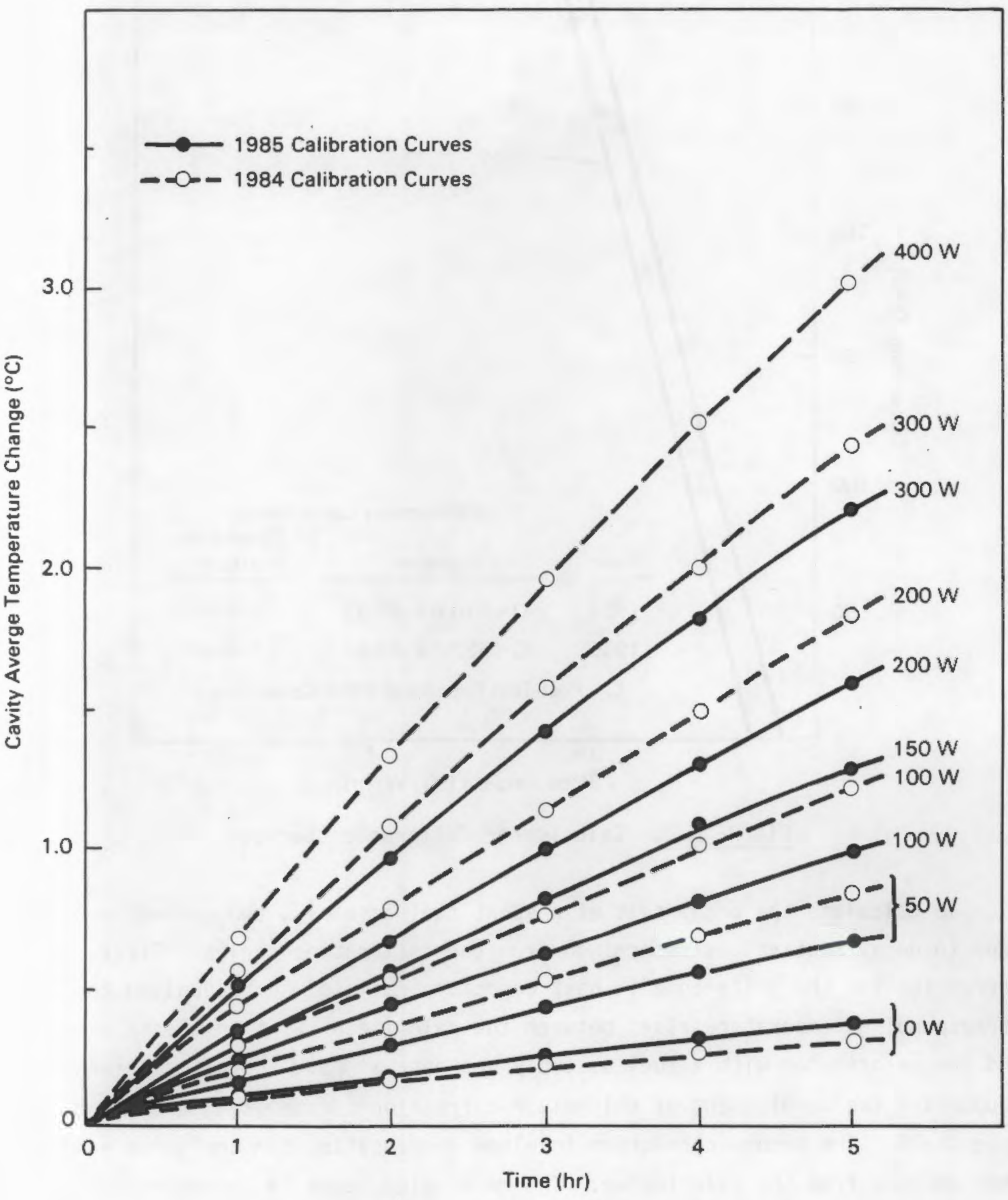

FIGURE 4.2. Calorimeter Time-Versus-Temperature Calibration Curves

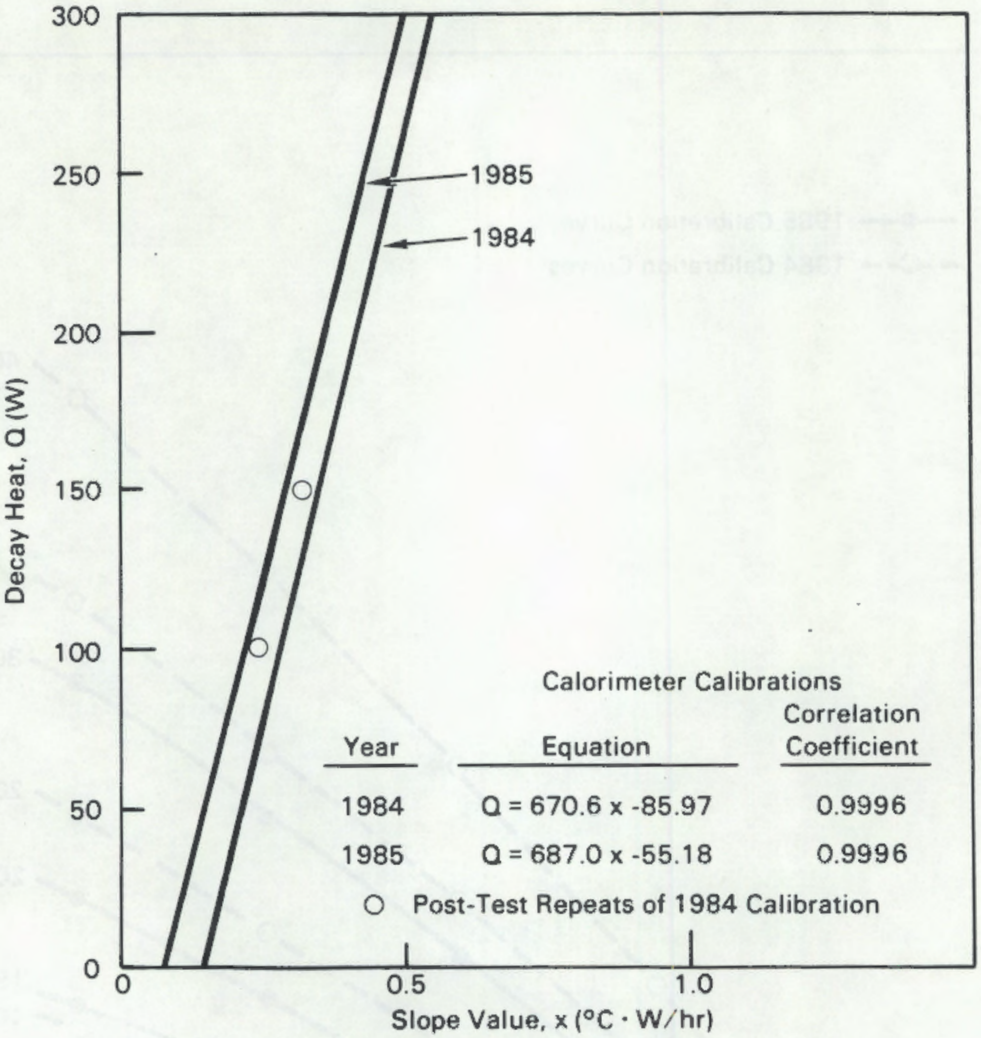

FIGURE 4.3. Calorimeter Calibration Curves

To calculate the decay heat of a spent fuel assembly, two corrections were made to decay heat values determined from the calibration curves. First, a correction for the difference in heat capacity (ratio of heat supplied to corresponding temperature rise) between the calorimeter with an electric heater and the calorimeter with a fuel assembly was determined. The calculations supporting the development of this minor correction factor (<2%) are shown in Appendix A. The second correction involves compensating for the gamma energy that escapes from the calorimeter. The methodology used in determining the correction factor was developed in 1981 during initial calorimetry work for DOE (Judson et al. 1982), and then modified based upon measurements of the actual

TABLE 4.1. Calorimeter Calibration Data

Date Calibration Run	Design Power, W	Measured[a] Power, W	Slope Value	Index of Determination
1984 Method				
10/16/84	0	0.0	0.156464	0.99959
10/15/84	50	49.2	0.352357	0.99996
10/14/84	100	99.7	0.522143	0.999984
09/20/85[b]	100	99.4	0.43786	0.99998
09/19/85[b]	150	151.8	0.58150	0.99997
10/13/84	200	199.6	0.765893	0.999998
10/15/84	200	199.7	0.758893	0.999995
10/14/85	300	298.8	1.028286	0.999993
10/13/84	400	398.9	1.291536	0.999999
1985 Method				
09/15/85	0	0.0	0.19643	0.96554
09/18/85	0	0.0	0.16375	0.99970
09/16/85	50	52.9	0.28025	0.99995
09/17/85	50	50.2	0.27721	0.99995
09/18/85	50	50.3	0.27486	0.99985
09/10/85	100	98.5	0.41061	0.99989
09/14/85	100	100.6	0.40332	0.99996
09/06/85	150	149.2	0.53850	0.99998
09/15/85	150	149.0	0.54243	0.99999
09/05/85	200	199.6	0.65496	0.99998
09/06/85	300	300.4	0.93400	0.99993

(a) Actual watts are 5-hour averages, corrected for power loss in lines to calibration heater.
(b) Check on 1984 calibration.

amount of absorption that occurred in the outer wall of the calorimeter (McKinnon et al. 1985). The specifics relating to the development of this correction factor (<12%) are contained in Appendix A.

Table 4.2 is a summary of the calorimetry decay heat data. A more complete listing of the data is found in Appendix A. Repeatability of the data using the 1985 method can be assessed from the runs made on fuel assembly MT133. The repeatability of these measurements corrected for decay rate (i.e., a same-day measurement comparison) is indicated by a standard deviation of ±1.7 W. The repeatability of the data collected using the 1984 method of operation is no better than ±15 W, based on repeat measurements on assemblies MT123, MT133, and MT228. The reason the 1985 method of calorimeter operation gave better repeatability than the 1984 method was not obvious. Its evaluation was outside with the scope of this study.

TABLE 4.2. Monticello Spent Fuel Assembly Calorimetry Results

Assembly ID	1984 Measurement Method		1985 Measurement Method	
	Date	Decay Heat, W	Date	Decay Heat, W
MT116	06/10/85	114.9	08/27/85	119.6
MT123	06/05/85	66.8	08/27/85	97.2
	06/08/85	95.3		
	06/11/85	65.9		
MT133	05/29/85	152.6	06/13/85	146.0
	06/06/85	129.0	06/13/85	145.4
	06/09/85	154.8	08/20/85	146.0
	06/12/85	106.7	08/21/85	146.8
			08/29/85	149.9
			08/30/85	144.7
			08/31/85	147.0
			09/01/85	147.8
MT190	06/08/85	99.2	08/28/85	107.6
MT228	05/30/85	101.0	08/20/85	90.3
	06/07/85	71.2		
	06/11/85	76.4		
MT264	06/05/85	46.1	08/28/85	76.2

4.2 ION-1 SYSTEM AND AXIAL RADIATION DATA

Gamma and neutron axial profile data were taken on each of the fuel assemblies subjected to calorimetry. The LANL portable spent-fuel detector, known as the ION-1/fork measurement system, was used at GE-MO to make these radiation readings. Basically, the LANL ION-1 system shown in Figure 4.4 consists of underwater sensors and an above-water electronics unit that monitors and displays the measured radiation. The underwater unit consists of two cylindrical forked tines made of polyethylene. Each tine contains a cadmium-covered fission chamber, a noncovered fission chamber, and an ion chamber. The opening between the tines was about 1.3 cm (0.5 in.) greater than the width of a

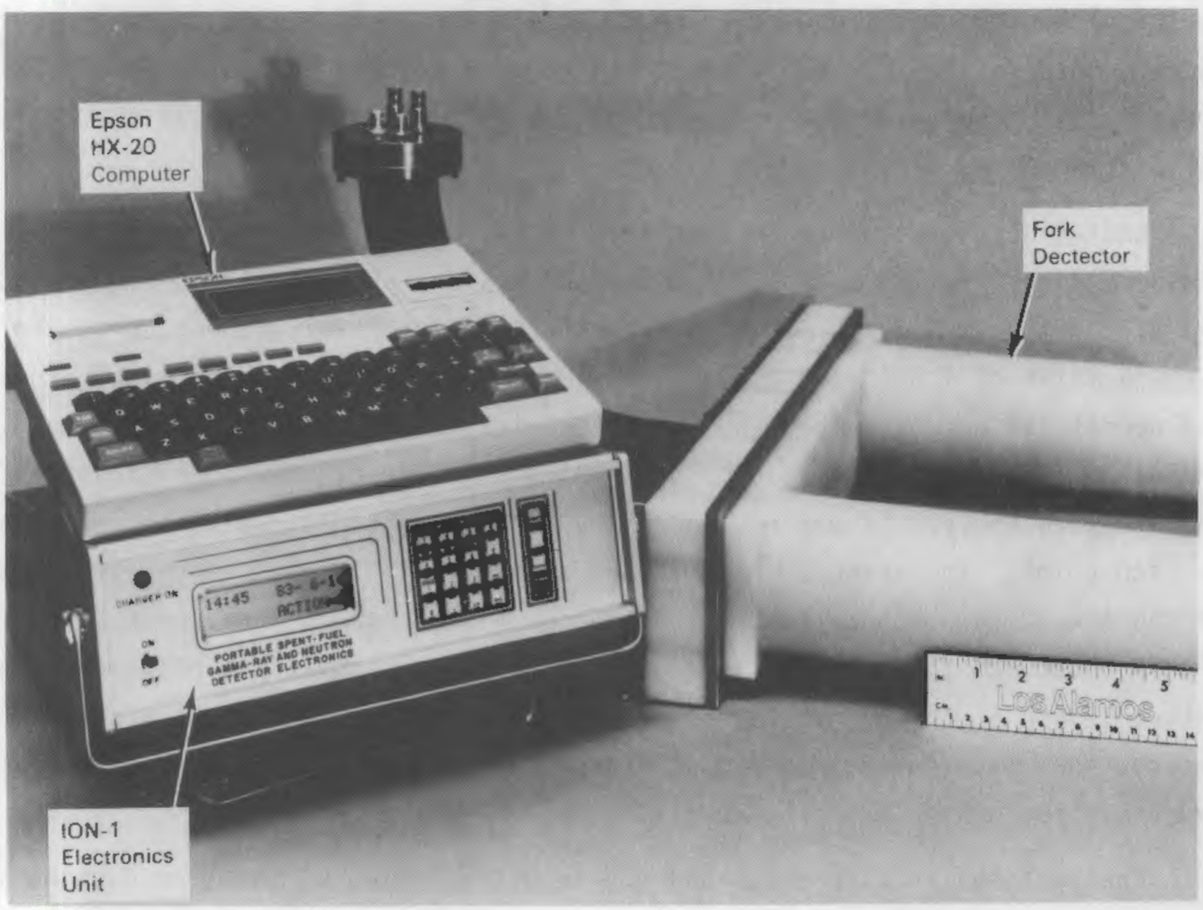

FIGURE 4.4. Los Alamos ION-1 Spent Fuel Radiation Measurement Equipment
Source: McKinnon et al. (1985, p. 3-12)

typical 7x7 BWR spent fuel assembly. The above-water electronics unit provides hard copies of the data output and has a magnetic tape interface.

In preparation for using the ION-1 at GE-MO, various preliminary neutron and gamma measurements were taken to test the equipment and to determine the detector response for well-defined source-detector configurations. Neutron measurements were made in water with a ^{252}Cf source of known strength centered between the tines. For a lower-level discriminator setting of 25 on the ION-1, the efficiency for the cadmium-covered detector was 2.68×10^{-5}; for the bare detectors the efficiency was 5.2×10^{-5}. The ion chambers were checked using a ^{60}Co source in air. The average linear response of the two ion chambers is 70 R/hr per ION-1 reading. These measurements verified that the equipment was functioning properly. However, direct correlations between the source measurements and actual fuel assembly measurements were not within the scope of this project. Such correlations may be developed by performing detailed neutronics calculations.

Radiation was measured at nine axial locations on each fuel assembly. These readings are listed in Tables 4.3 and 4.4 and are graphed in Figures 4.5 and 4.6. Figures 4.7 and 4.8 show the effect of normalizing these curves to an average value of 1 over the active fuel length and then taking an average of the normalized curves. The readings taken with the ION-1 were intended to give radiation information sufficient to establish axial decay heat profiles for each fuel assembly. It was not necessary to have absolute radiation readings at each point. The measurements made on assembly MT133 show the repeatability of the ION-1 measurements to be within about ±1% for gamma measurements and within about ±2% for the neutron measurements for a significant portion of the active length of the assemblies. This repeatability is consistent with observations made during a previous study (McKinnon et al. 1985) where the repeatability was observed to be ±1%.

The profiles in Figure 4.5 and 4.6 show the effect of local conditions in the reactor on the profiles. It is apparent from these profiles that localized conditions (proximity to control rods, void fracture, and total burnup) do affect the gamma and neutron profiles. Figures 4.7 and 4.8 give a representative average profile for the six fuel assemblies.

TABLE 4.3. Monticello Spent Fuel Axial Gamma Radiation Profiles

Assembly ID	Elevation, m								
	0.19[a]	0.55	1.06	1.57	2.47	2.98	3.49	3.85	4.19[b]
MT123	5.3	43.5	71.1	73.0	70.2	67.1	49.3	24.1	5.8
MT116	19.3	54.4	87.5	92.6	92.6	92.6	73.9	37.3	3.6
MT133	27.1	66.6	107.8	115.1	111.7	113.2	93.0	46.6	10.3
MT133	27.1	65.8	106.5	114.0	111.1	112.3	92.6	46.5	10.2
MT133	28.1	67.5	107.3	113.6	110.4	111.1	90.3	44.9	10.0
MT190	10.1	47.4	77.7	80.6	82.2	85.2	72.8	37.4	5.6
MT228	16.6	43.5	66.1	66.6	62.4	60.2	46.9	23.2	4.2
MT264	11.4	32.2	51.1	54.0	53.0	50.7	39.3	20.0	3.1

(a) Elevation of lower tie plate.
(b) Elevation of upper tie plate.

TABLE 4.4. Monticello Spent Fuel Axial Neutron Radiation Profiles

Assembly ID	Elevation, m								
	0.19[a]	0.55	1.06	1.57	2.47	2.98	3.49	3.85	4.19[b]
MT123	0.0	2.5	9.6	12.4	13.9	10.2	4.7	0.8	0.0
MT116	0.1	2.8	16.9	24.6	30.8	27.2	16.0	2.0	0.0
MT133	0.0	8.0	35.2	48.3	52.1	53.2	28.9	4.8	0.0
MT133	0.1	6.4	35.3	46.3	53.7	55.2	32.9	4.9	0.0
MT133	0.0	8.9	36.8	49.6	50.9	55.3	29.1	4.1	0.0
MT190	0.0	1.3	10.2	12.2	14.4	17.8	10.7	2.9	0.0
MT228	0.0	0.6	5.2	5.9	6.3	4.3	2.6	0.6	0.0
MT264	0.1	0.8	2.3	2.7	3.1	2.7	1.4	0.2	0.0

(a) Elevation of lower tie plate.
(b) Elevation of upper tie plate.

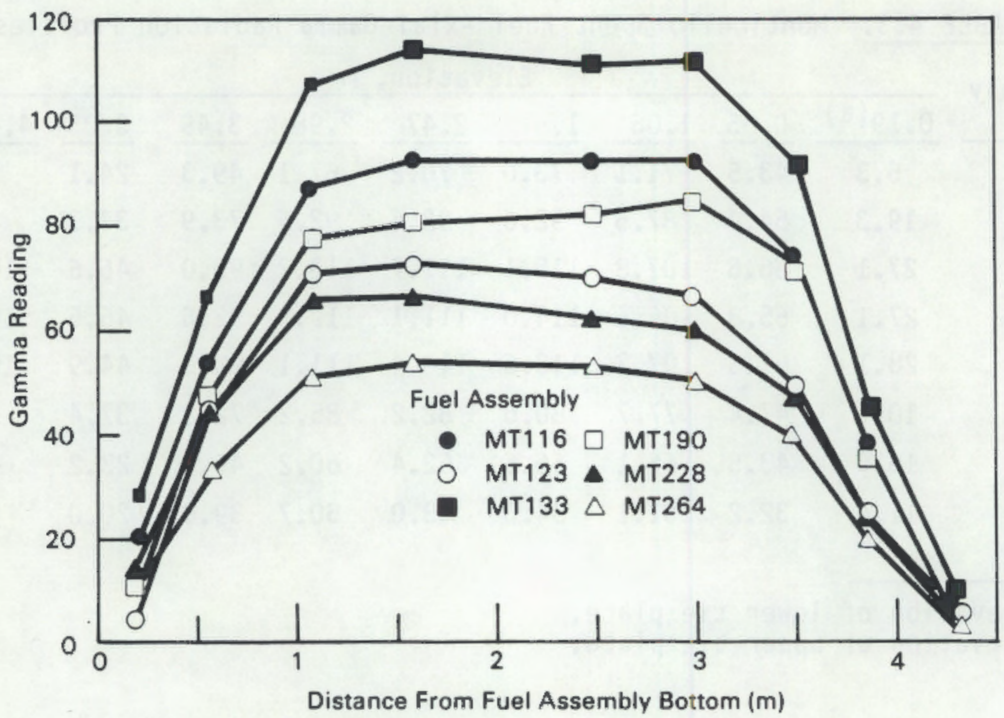

FIGURE 4.5. Monticello Spent Fuel Axial Gamma Profiles

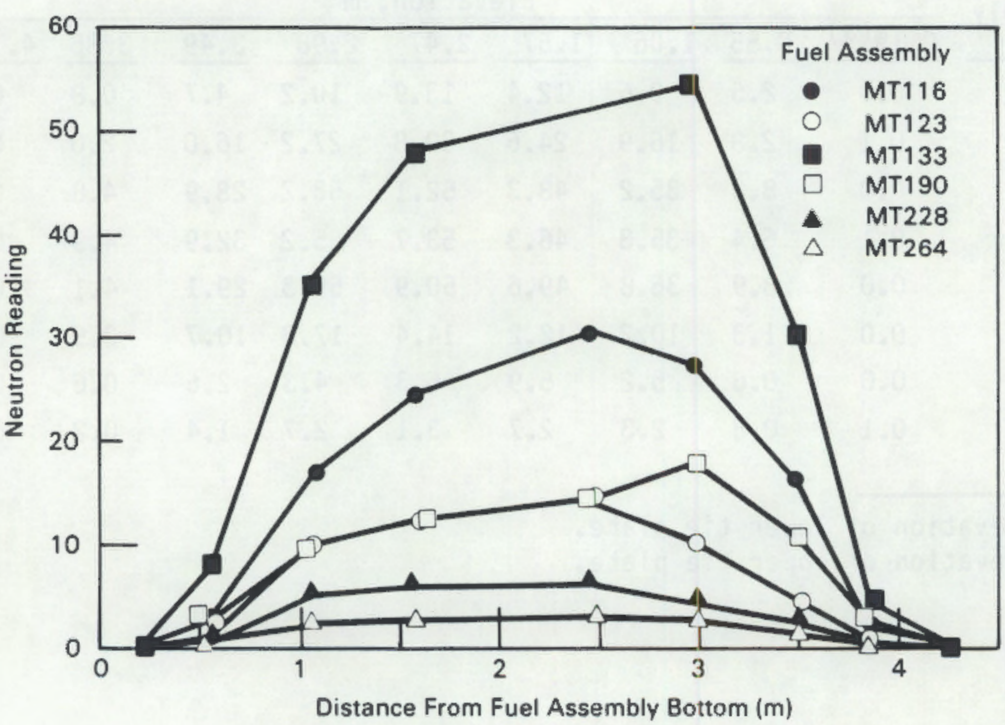

FIGURE 4.6. Monticello Spent Fuel Axial Neutron Profiles

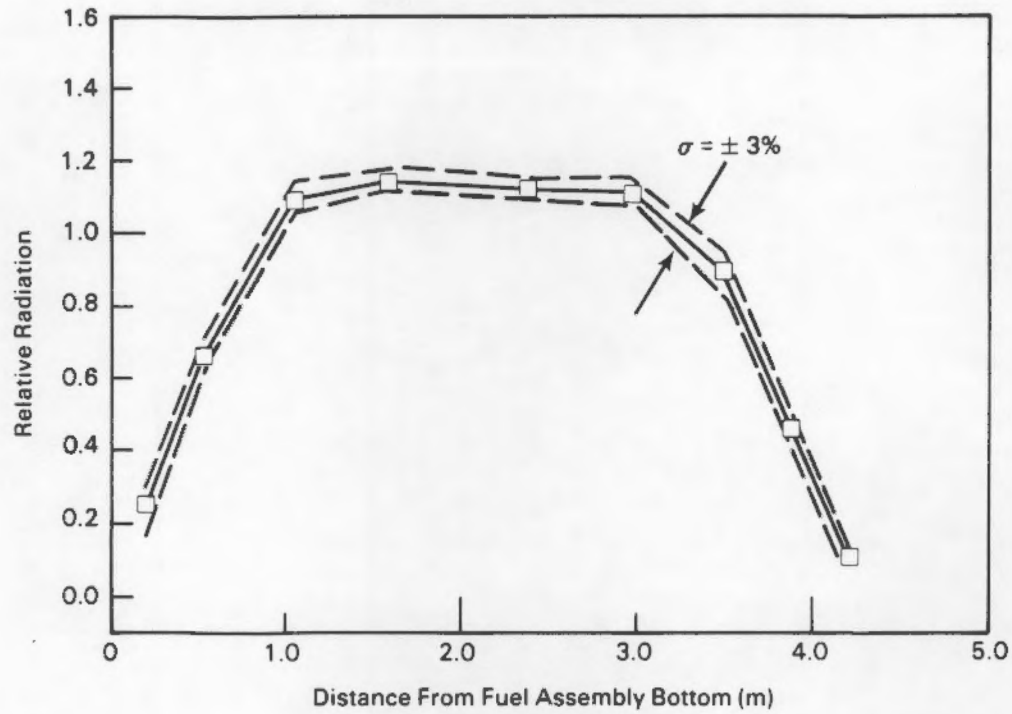

FIGURE 4.7. Monticello Spent Fuel Normalized Gamma Axial Radiation Profile

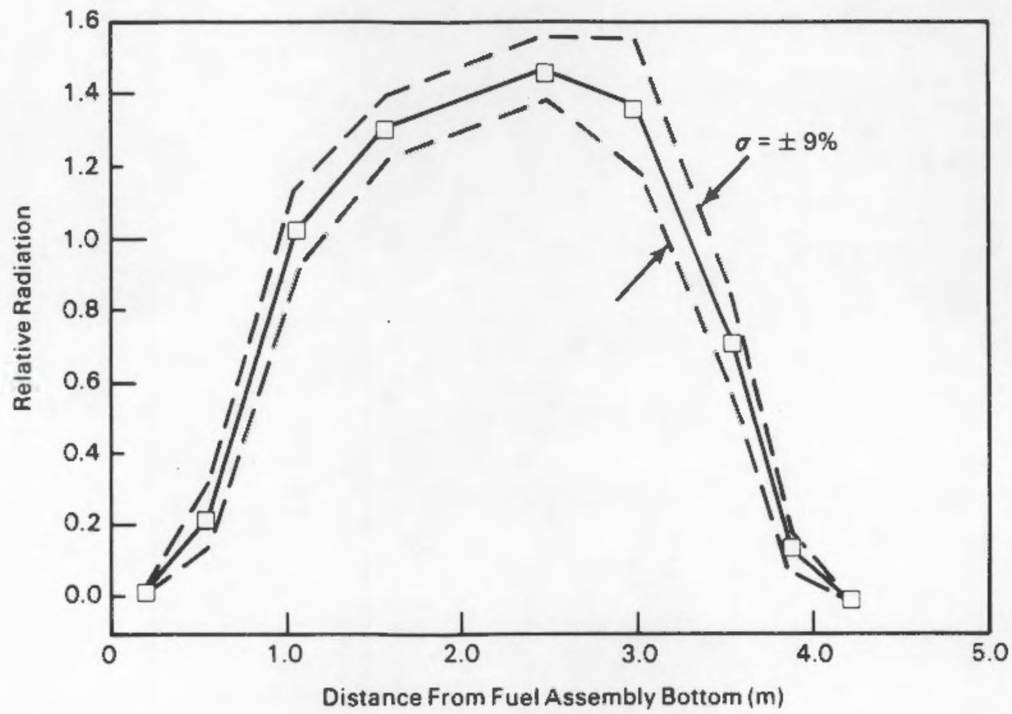

FIGURE 4.8. Monticello Spent Fuel Normalized Neutron Axial Radiation Profile

5.0 DECAY HEAT ANALYSIS AND DATA COMPARISONS

The pre-calorimetry decay heat analysis is discussed in this section. First, the ORIGEN2 computer code is described. Next, the input to the code is identified. Last, predictions of decay heat and axial radiation profiles are compared to actual measurements.

5.1 ORIGEN2 COMPUTER CODE

A standard version of the ORIGEN2 code (Croff 1980a,b) was used to predict the decay heat rates of six Monticello BWR spent fuel assemblies. The ORIGEN2 code is widely used in the nuclear industry to predict decay heat rates of spent fuel assemblies. It is a general purpose burnup and decay code that features extensive data libraries containing information on over 1200 nuclides. The code can be used to perform transmutation calculations in steps of constant power or constant neutron flux level. The resulting nuclide concentrations can be decayed with user-specified time intervals. Output options are available for decay heat rate as well as spent fuel compositions and radioactivity.

Before the ORIGEN2 code was used to predict the decay heat rates of the selected fuel assemblies, code results from a standard problem were compared with results from another standard version of the code run on another computer. The comparison was performed to ensure that the predictions are what would be expected from the code as it would be received from the Oak Ridge National Laboratory (ORNL) Radiation Shielding Information Center.

ORIGEN2 results are based on a large library of one energy group cross sections of the nuclides. These cross sections are the result of extensive calculations starting with a numerical description of the cross section of each isotope as a function of neutron energy. The basic cross sections are averaged over the energy range of 0 eV to 17 MeV using a computed neutron energy spectrum. The calculation of the neutron spectrum is done with a composition appropriate to a specific reactor core design and operating condition. The user is provided with various data libraries, each representing a reactor type, core loading, and operating condition. There is one BWR cross section library for a ^{235}U enriched core; this actinide library has the numerical designation 252. Associated with it are activation product library 251 and fission product library 253, which were generated using the same neutron spectrum as was used to generate library 252.

A special concern in making decay heat rate predictions with ORIGEN2 for BWR fuel, as opposed to PWR fuel, is the effect of steam voids on the neutron spectrum. The BWR core operating environment contains appreciable steam voids. The ratio of plutonium to uranium fissions and the actinide composition at a given burnup are influenced by differences in the neutron spectrum. Assembly decay heat rates are determined by different fission product yields for uranium and plutonium and by the mix of actinide isotopes in the spent fuel. A series of calculations was performed to evaluate the sensitivity of decay heat rates to variations of core steam void fractions. A version of the LEOPARD code (Barry 1963) was used to calculate the effect of unit cell steam voiding on the one group spectrum-averaged cross sections of the isotope responsible for most of the decay heat. The change in the spectrum-averaged cross section at a given void fraction relative to the ORIGEN2 library default void fraction of 31.6% was determined for a range of void fractions. These relative change factors were used to alter the cross section of ORIGEN2 library 252 via code input for a series of ORIGEN2 cases, each representing a specific core steam void fraction in the range of 0% to 90%. As a result of these sensitivity calculations, it was found that core void variations of 0% to 90% can cause the decay heat rate to vary by 11% to 30%, depending on the time out of reactor.

The spectrum used in computing libraries 251, 252, and 253 was calculated assuming a BWR-6 assembly (8x8 rod array) at 31.6% core average steam void fraction (Croff et al. 1978). The Monticello fuel assemblies used in this study are of the earlier 7x7 design. The void fraction that has the same hydrogen-to-uranium (H/U) ratio as the 31.6% used in calculating the ORIGEN2 library is 36% for the 7x7 rod design. The H/U ratio is a reasonable basis to use to determine the equivalent void fraction. It is a good measure of the hardness of the neutron spectrum because the relative moderation and absorption rates are determined by the H/U ratio. The Monticello BWR assemblies had operating void fractions of 39% to 40% void, which is close to the 36% equivalent void fraction of the library. Therefore, no corrections for void fraction were made.

5.2 ORIGEN2 INPUT SPECIFICATIONS

Summaries of Monticello fuel assembly design and burnup data used as input to ORIGEN2 were presented in Tables 3.1 and 3.2 in Section 3.0. Neither table gives the assemblies' structural material content. These materials contribute to the decay heat from neutron activation. Because the fraction of decay heat from neutron activation of assembly structural materials is less than 5% of the total decay heat, generic values were used. These values are given in the ORNL document (Croff et al. 1978) that describes the makeup of the ORIGEN2 BWR library. The two elements contributing the largest share of activation heating are cobalt and gadolinium. A value of 1573 ppm was assumed for the gadolinium concentration in the uranium fuel. The cobalt content of the 304 stainless steel was assumed to be 800 ppm. The Zircaloy-2 cladding was assumed to contain 10 ppm cobalt, and the uranium oxide was assumed to contain 1 ppm cobalt.

All assemblies were initially enriched to 2.25 wt% ^{235}U averaged over all rods in each assembly. Sensitivity studies were conducted (McKinnon et al. 1985) using ORIGEN2 with different enrichments to ensure that calculated decay heat rates based on single average assembly enrichments closely approximated average decay heat rates from ORIGEN2 based on individual rod enrichments in the assemblies.

The Monticello reactor power history for the first four operating cycles is shown in Figure 5.1. The specific reactor powers (tabulated values are in Appendix B) are based on a design core power of 18.2 MW/MTU when the reactor is operated at its full thermal power of 1670 MWt.

Power histories for the assemblies were determined from burnup histories shown in Table 3.2, Section 3.0, and from the reactor power history shown in Figure 5.1. Assembly power histories within a reactor operating cycle were calculated by multiplying the ratio of incremental burnup for the cycle (Table 3.2) to the core average incremental burnup for that cycle by the core

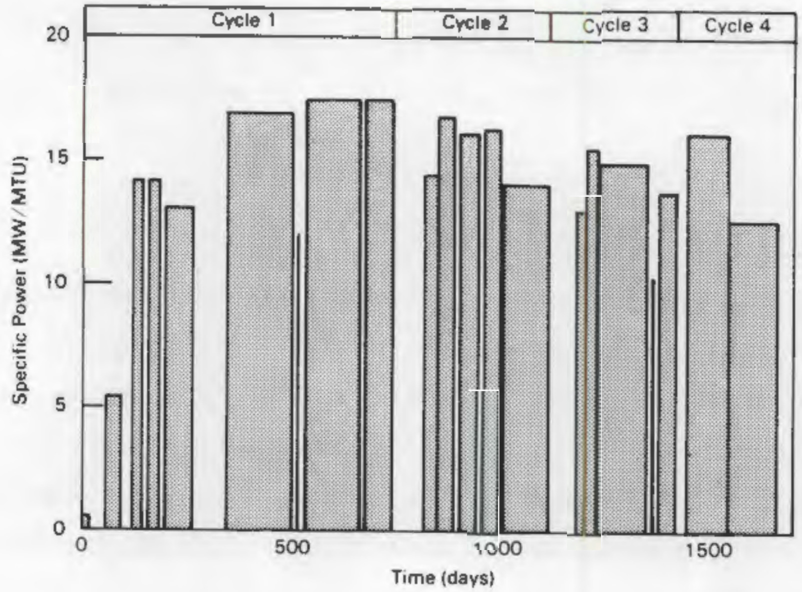

FIGURE 5.1. Monticello Reactor Operating History (Cycles 1 through 4)

averaged power history (Figure 5.1). A resulting typical assembly-specific power history used as input to ORIGEN2 is shown in Figure 5.2 for assembly MT133. The complete input file for the ORIGEN2 prediction of the decay heat of MT133 is presented in Appendix B.

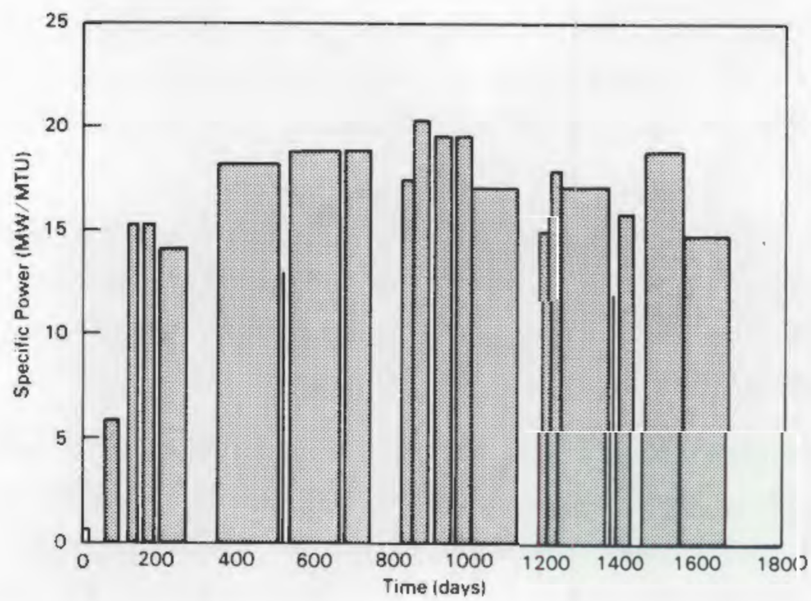

FIGURE 5.2. Monticello Assembly MT133 Specific Power History

5.3 ORIGEN2 PREDICTIONS COMPARED TO DATA

Predictions of both the magnitude of assembly decay heat rates and axial decay heat profiles are compared to experimental data in the following subsections. Predictions and decay heat data previously obtained (McKinnon et al. 1985, 1986) for the Dresden and Cooper BWR spent fuel assemblies are included for completeness.

5.3.1 Decay Heat Rates

Measured values of decay heat generation rates and ORIGEN2 pre-calorimetry predictions are shown in Table 5.1 for every run made with the six Monticello spent fuel assemblies. Table 5.1 contains predictions made using final burnup values based on the Cycle Summary reports and Form 30 reports. Plots of the information contained in Table 5.1 are shown in Figures 5.3, 5.4, and 5.5. The scatter in the data shown in Figure 5.3 prompted an investigation into the reason for the large differences among repeat measurements on the same fuel assembly; see the measured decay heat values in Table 5.1 for assemblies MT123, MT133, and MT 228. The investigation did not identify the reason for the large differences among repeat measurements but did suggest a small change in the operation of the calorimeter that resulted in better measurement repeatability. The new method of operation, described in Section 4.1, prompted a recalibration of the calorimeter using the new (1985) method of operation.

All the data taken using the 1985 method of operation is shown in Figure 5.4. The correlation between the predictions and data is much more apparent in Figure 5.4 than it is in Figure 5.3. Figure 5.4 also shows that there would be much less scatter from the best fit straight line for predictions based on the Cycle Summary final burnups than for predictions based on Form 30 final burnups. The EPRI calorimetry study (McKinnon et al. 1985) and the calorimetry done in conjunction with BWR cask performance testing (McKinnon et al. 1986) also showed that the predictions based on Cycle Summary total burnups have a tighter fit to the measured data than do the Form 30 total burnup-based predictions.

TABLE 5.1. Monticello Spent Fuel Assembly Calorimetry Results

Assembly ID	1984 Method: Measurement Date	1984 Method: Measured Decay Heat, W	1984 Method: Predicted Decay Heat, Form 30, W	1984 Method: Predicted Decay Heat, Cycle Summary, W	1985 Method: Measurement Date	1985 Method: Measured Decay Heat, W	1985 Method: Predicted Decay Heat, Form 30, W	1985 Method: Predicted Decay Heat, Cycle Summary, W
MT116	06/10/85	114.9	105.3	102.1	08/27/85	119.6	104.5	101.5
MT123	06/05/85	66.8	73.2	79.5	08/27/85	97.2	72.7	79.0
	06/08/85	95.3	73.2	79.5				
	06/11/85	65.9	73.1	79.4				
MT133	05/29/85	152.6	124.3	125.4	06/13/85	146.0	124.2	125.3
	06/06/85	129.0	124.3	125.4	06/13/85	145.4	124.2	125.3
	06/09/85	154.8	124.3	125.4	08/20/85	146.0	123.3	124.4
	06/12/85	106.7	124.2	125.4	08/21/85	146.8	123.3	124.4
					08/29/85	149.9	123.1	124.3
					08/30/85	144.7	123.1	124.2
					08/31/85	147.0	123.1	124.2
					09/01/85	147.8	123.1	124.2
MT190	06/08/85	99.2	88.7	89.7	08/28/85	107.6	88.0	89.0
MT228	05/30/85	101.0	70.7	73.3	08/20/85	90.3	70.2	72.7
	06/07/85	71.2	70.7	73.2				
	06/11/85	76.4	70.6	73.2				
MT264	06/05/85	46.1	52.6	57.9	08/28/85	76.2	52.2	57.5

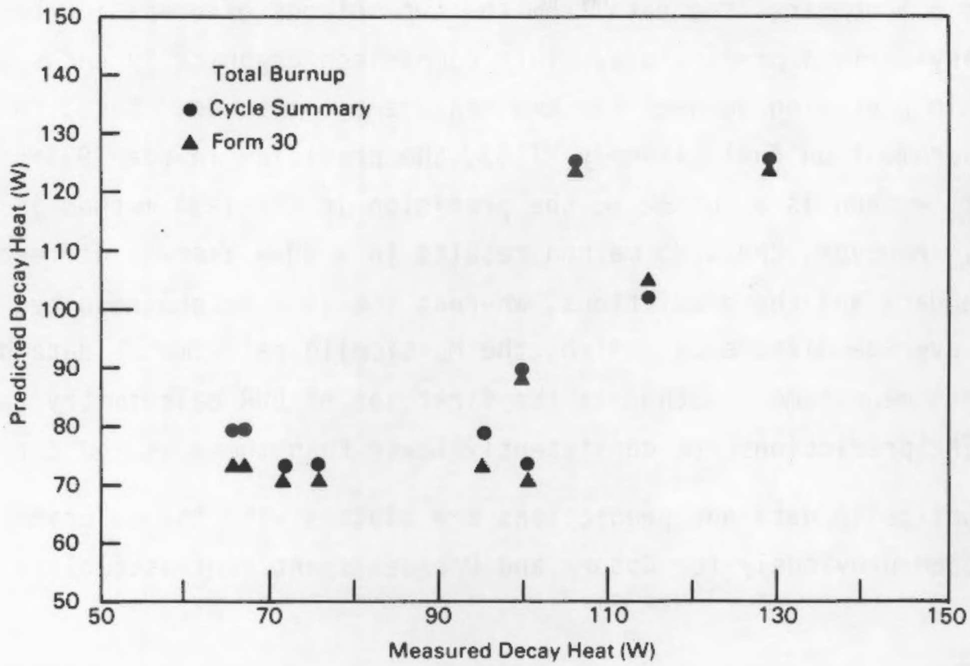

FIGURE 5.3. Predicted and Measured Monticello Spent Fuel Decay Heat Rates Obtained Using the 1984 Measurement Method

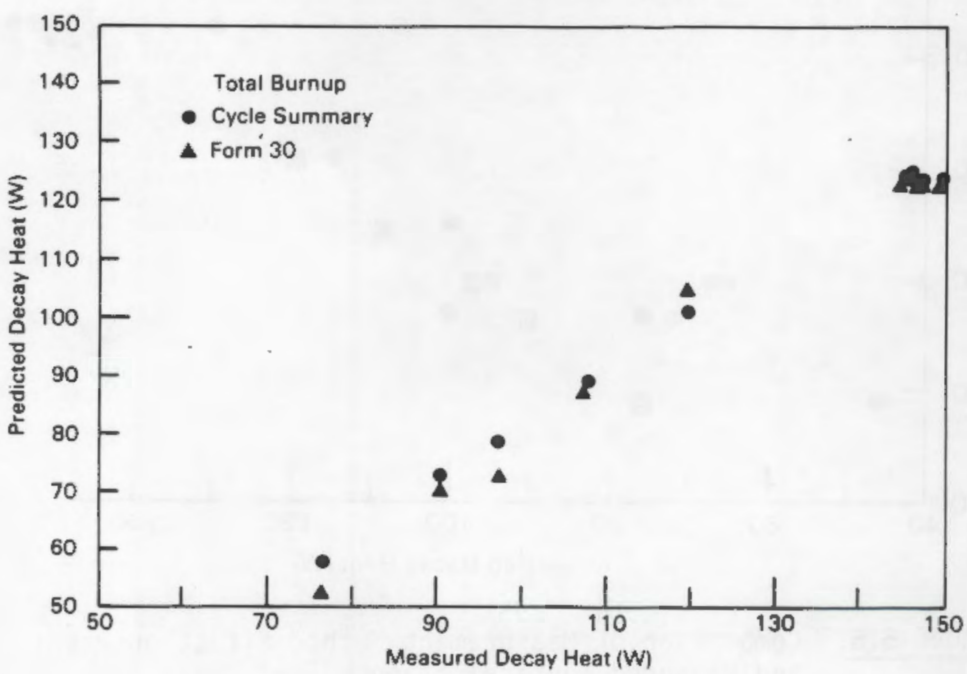

FIGURE 5.4. Predicted and Measured Monticello Spent Fuel Decay Heat Rates Obtained Using the 1985 Measurement Method

Figure 5.5 compares the data from the two methods of operation based on Cycle Summary-derived predictions. This comparison graphically shows the difference in precision between the two measurement methods. Based on the repeat measurement on fuel assembly MT133, the precision in the 1985 measurements method is about ±2 W; the precision in the 1984 method is greater than ±15 W. However, the 1985 method results in a 20-W average difference between the data and the predictions, whereas the 1984 method resulted in less than a 4-W average difference. Also, the Monticello calorimetry data obtained from the 1985 measurement method is the first set of BWR calorimetry data in which ORIGEN2 predictions are consistently lower than the measured data.

The Monticello data and predictions are plotted with the calorimetry data collected previously for Cooper and Dresden spent fuel assemblies in

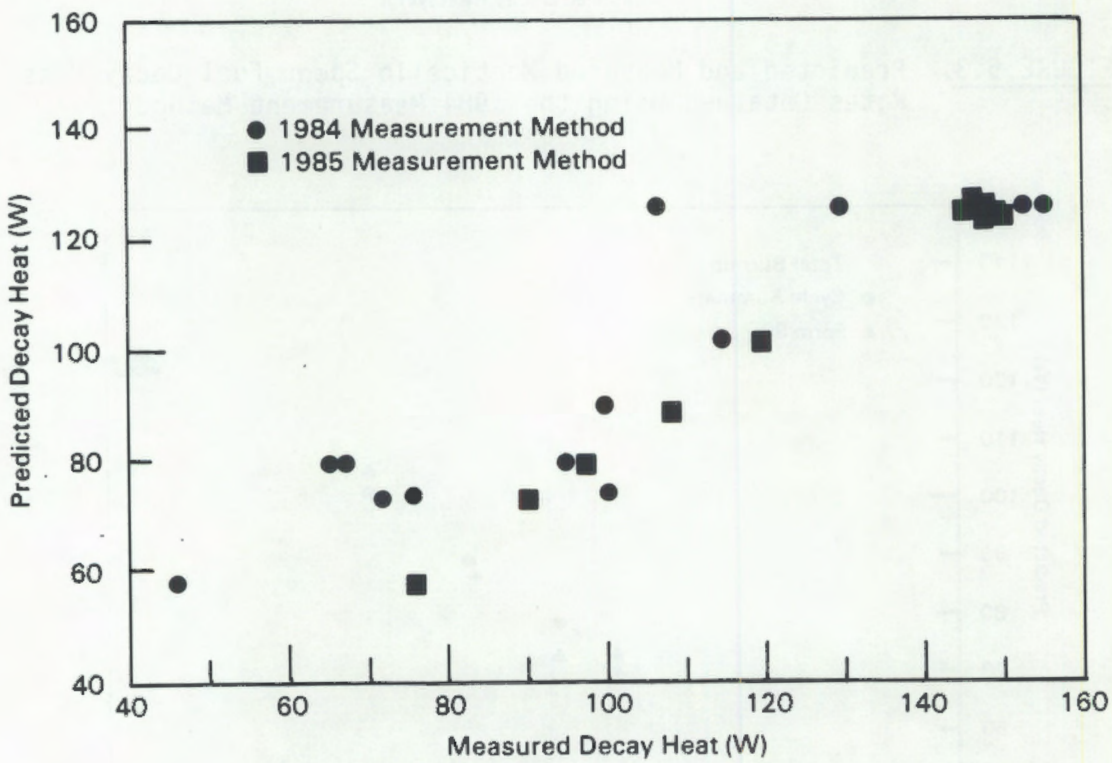

FIGURE 5.5. Comparison of Measurement Method Effect on Predicted and Measured Monticello Spent Fuel Decay Heat Rate

Figure 5.6. Linear regression analysis was used to fit a straight line to the data shown in Figure 5.6. The equation of this line is

$$\text{Measured decay heat} = \text{Predicted decay heat} - 6$$

The standard deviation of the data about the line is ±14.3 W. The -6 intercept is a result of the Monticello data obtained from the 1985 measurement method.

It seems strange that the curve representing the relationship between the Monticello predictions and data from the 1985 method would parallel the curve for the other data but be displaced from it, unless there is something in the prediction or measurement process that can introduce a constant that is independent of decay heat magnitude.

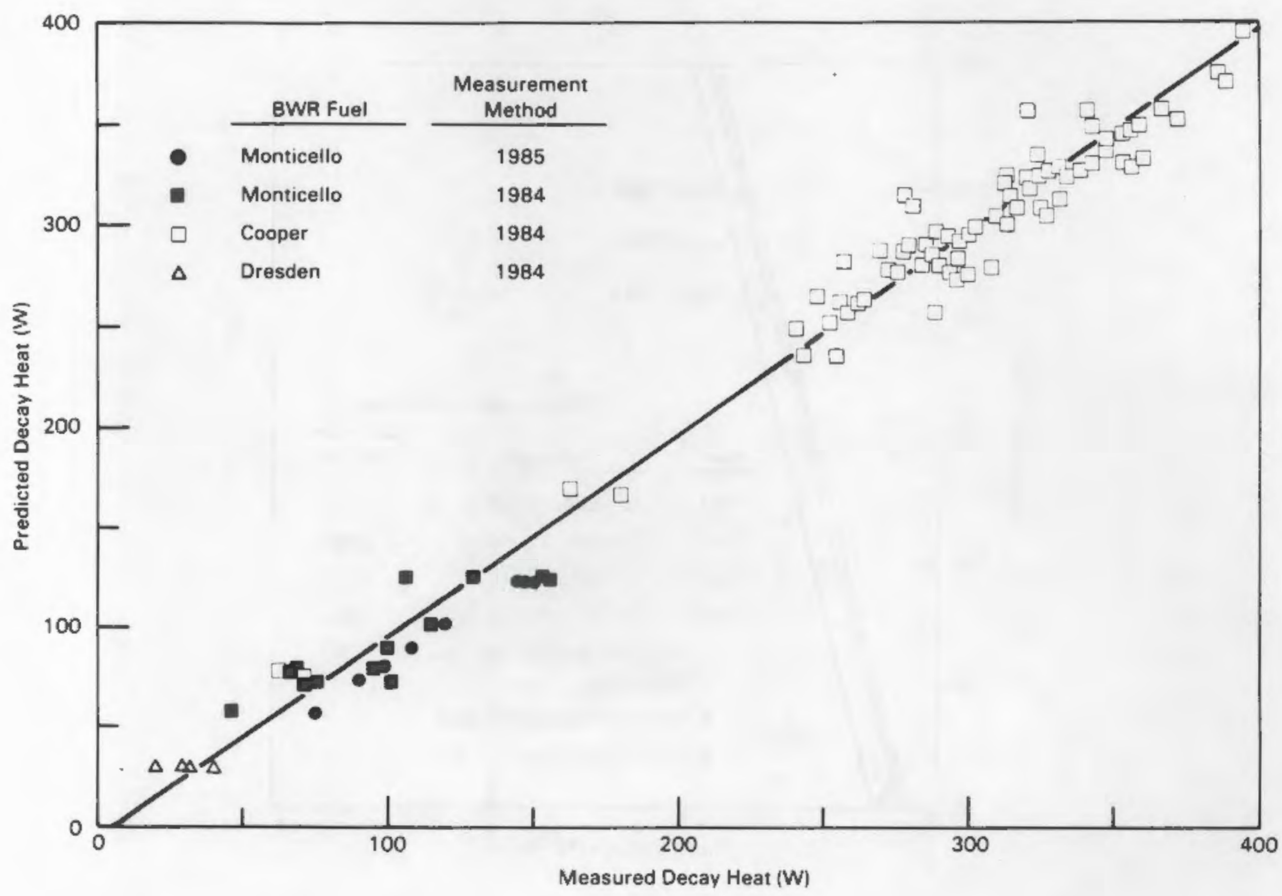

FIGURE 5.6. Predicted and Measured Decay Heat Rates for Dresden, Cooper, and Monticello Spent Fuel Assemblies

This opportunity does exist in the measurement process where temperatures are recorded over a 5-hour period. The temperature-time curve is determined by using regression analysis to fit a curve to the data (Figure 4.2, Section 4.1). The slope of this curve at time = zero is then entered into a calorimetry calibration curve of the form $y = mx + b$, where x is the slope at time zero, y is the decay heat, and m and b are constants determined from a best fit to the calibration data. If the constant b were in error for some reason, then measured values would be displaced from their true values by a constant amount.

Figure 5.7 shows the calorimeter calibration curves that have been determined over the course of the calorimeter's existence (Judson et al. 1982; McKinnon et al. 1985, 1986). It is interesting to note that the slope values for all of the curves vary by less than ±2%, whereas intercept values vary from

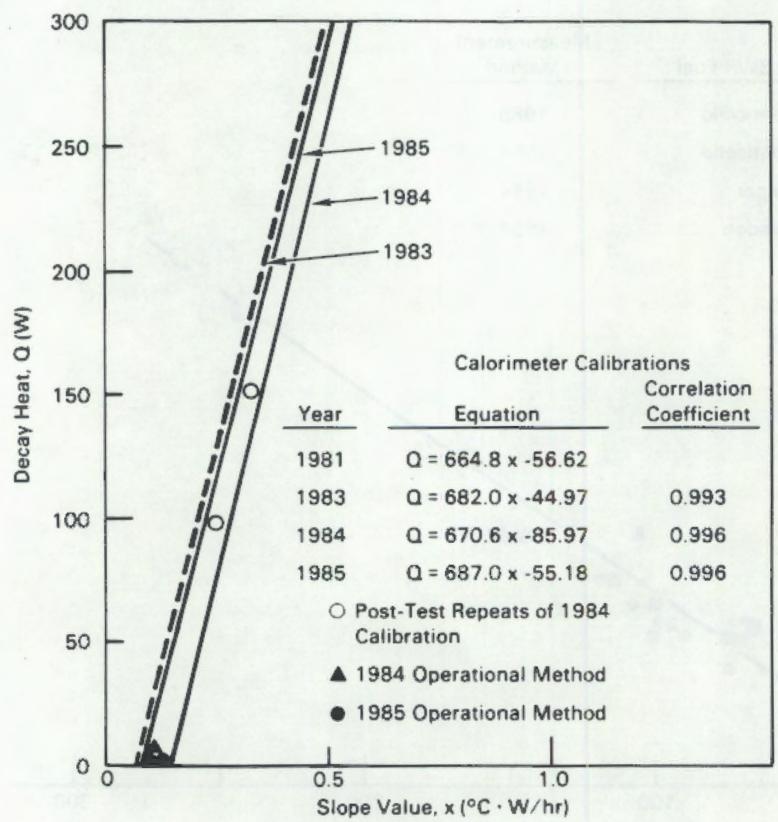

FIGURE 5.7. All Calorimeter Calibration Curves

-44 to -85. The ±2% variation in slope can account for no more than ±5 W difference over the range of the Monticello data and would not give a constant difference between the measurements and the predictions. However, the intercept value can account for over 40 W difference. It can easily account for the constant difference between the measurements and the predictions.

Repeat measurements on Dresden assembly DN212 can also be used to assess the calibration of the calorimeter. The assembly was discharged from the Dresden reactor in February 1972 and had an ORIGEN2-predicted output of 30 W in October 1983. This assembly was measured four times during the period from October 1983 to October 1985. The results of these measurements are shown in Table 5.2. It is interesting to note that the most recent measurement on DN212 shows the same trend as the Monticello assemblies: the ORIGEN2 prediction is lower than the calorimeter measurements. Previous measurements were less than the predictions.

The 1984 and 1985 calibrations of the calorimeter were the first attempts to quantify the performance of the calorimeter with no power to the electrical heater. In the 1985 calibration, the slope of the heat-up curve at zero power was measured to be 0.1637 and 0.1964. In the 1984 calibration, the slope at zero power was measured to be 0.1564. These three points are plotted on Figure 5.3 and are much closer together than their respective calibration curves. The differences in the various calibration curves could be associated with

TABLE 5.2. Dresden Assembly DN212 Repeat Measurements

Measurement Date	Slope	Measured Decay Heat, W[(a)]
10/83	0.1952	29.5
10/03/84	0.2364	0.1
10/18/84	0.2853	22.0
10/85	0.2436	39.2

(a) Based on respective 1983, 1984, and 1985 calibration curves.

power measurements during calibration with the electric heater. Further investigation into this hypothesis was not within the scope of this study. It is apparent from comparison of all ORIGEN2 predictions with BWR calorimetry data that, for assemblies of interest to at-reactor and interim storage facilities (200 W or greater), ORIGEN2 predictions are accurate to within ±10%, an acceptable range.

Based on these results, additional effort should be extended to isolate the source of the calibration offset error. Without further study, the predictions appear to be as good as, if not better than, the measurements. The results also show that the data repeatability is sensitive to the operational method used and that the accuracy of the data is linked to the calibration process.

5.3.2 Axial Decay Heat Profiles

The comparison between the predicted axial decay heat profile and the measured gamma output of the fuel assemblies is shown in Figure 5.8. The pre-calorimetry prediction was made by using the core-averaged axial burnup distribution. The measurement data was obtained with the ION-1. The core-averaged axial gamma ray source is predicted to be very close to the axial burnup distribution because the total gamma ray source strength is proportional to burnup to within 5% for burnups above 5.0 GWd/MTU. The ION-1 distribution is based on an average of the normalization of the measured values shown in Table 3.4. The degree to which the predicted axial gamma profile departs from the measurements depends upon local influences from control rods, the axial burnup distributions of adjacent assemblies, and the local steam void history.

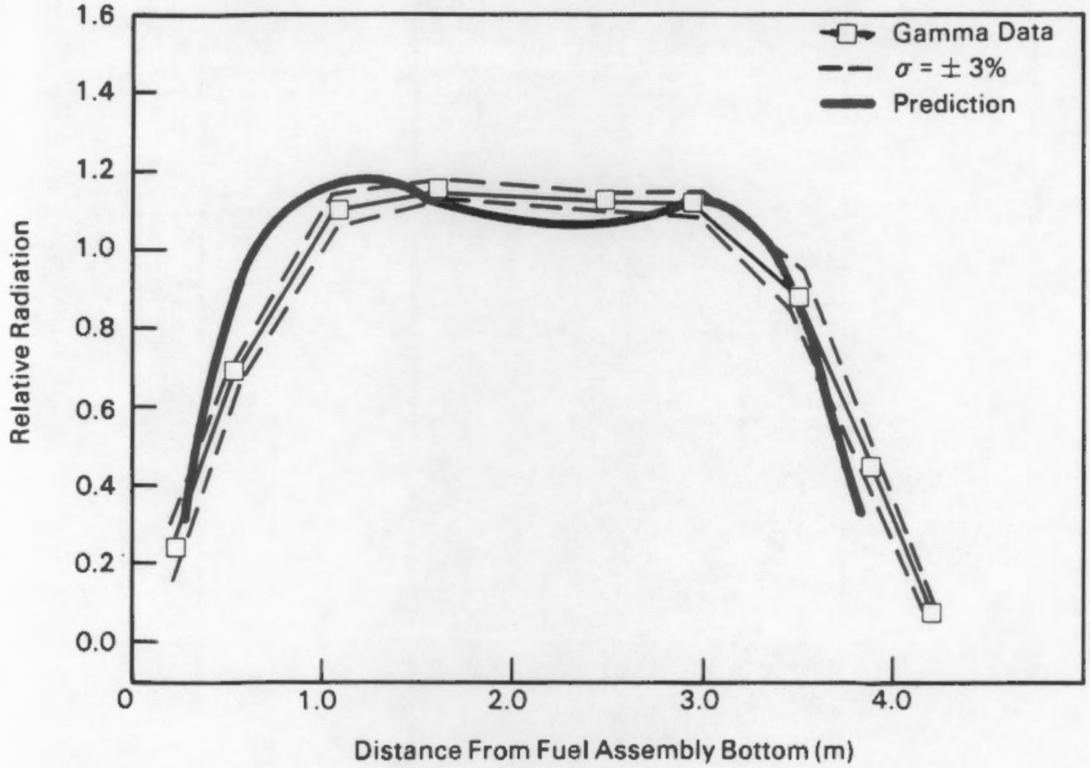

FIGURE 5.8. Monticello Spent Fuel Assembly Measured Average Axial Gamma Profile and Predicted Axial Decay Heat Profile

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APPENDIX A

CALORIMETER DATA AND CORRECTION FACTORS

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CALORIMETER DATA AND CORRECTION FACTORS

The first part of this appendix contains the Monticello calorimetry data arranged in chronological order. This is followed by a summary of Cooper and Dresden calorimetry data taken from McKinnon et al. (1985, 1986). The final part of the appendix gives the methodology used to adjust raw calorimeter data for differences in heat capacity between the calibration and measurement mode of operation, and the method used to compensate for gamma energy losses during the measurement mode of operation.

Bundle ID	MT116	
Date/Time Bundle Loaded	6-10-85	11:25
Date/Time Bundle Removed	6-10-85	17:35

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	74.94	
1	0.49	1.48	74.95	
2	0.98	1.94	74.97	
3	1.43	2.37	74.98	
4	1.86	2.78	75.00	
5	2.25	3.16	75.01	

"b" Coefficient	0.5177857
Index of Determination	0.999954
Slope Watts	106.9233
Corrected Slope Watts (0.9842)	105.2336
Ave. Corrected Radiation Value	645.9 R/hr
Gamma Watts (0.01499)	9.68
Total Thermal Output	114.9

Bundle ID	MT123	
Date/Time Bundle Loaded	6-05-85	10:50
Date/Time Bundle Removed	6-05-85	19:53

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	0.99	73.62	250.0
1	0.39	1.35	73.63	249.81
2	0.74	1.68	73.64	249.74
3	1.09	2.00	73.66	249.36
4	1.40	2.29	73.67	249.36
5	1.70	2.56	73.69	248.91

"b" Coefficient	0.393893
Index of Determination	0.999952
Slope Watts	60.76778
Corrected Slope Watts (0.9842)	59.80765
Ave. Corrected Radiation Value	468.7 R/hr
Gamma Watts (0.01499)	7.0
Total Thermal Output	66.8

Bundle ID	MT123
Date/Time Bundle Loaded	6-08-85 17:56
Date/Time Bundle Removed	6-08-85 00:51

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	74.49	243.02
1	0.49	1.47	74.49	
2	0.90	1.86	74.49	
3	1.28	2.23	74.49	
4	1.64	2.59	74.50	
5	2.00	2.92	74.52	

"b" Coefficient	0.4710358
Index of Determination	0.99961
Slope Watts	89.5069
Corrected Slope Watts (0.9842)	88.09269
Ave. Corrected Radiation Value	481.9 R/hr
Gamma Watts (0.01499)	7.22
Total Thermal Output	95.3

Bundle ID	MT123
Date/Time Bundle Loaded	6-11-85 14:48
Date/Time Bundle Removed	6-11-85 22:38

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	72.63	251.15
1	0.36	1.29	72.64	250.14
2	0.67	1.91	72.37	250.29
3	0.92	2.80	71.66	250.19
4	1.11	3.57	71.17	249.94
5	1.25	4.15	70.78	249.87

"b" Coefficient	0.3910716
Index of Determination	0.999987
Slope Watts	59.71666
Corrected Slope Watts (0.9842)	58.77314
Ave. Corrected Radiation Value	472.2 R/hr
Gamma Watts (0.01499)	7.1
Total Thermal Output	65.9

Bundle ID	MT133
Date/Time Bundle Loaded	5-29-85 06:06
Date/Time Bundle Removed	5-29-85 16:00

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	73.78	--
1	0.61	1.59	73.31	248.7
2	1.16	2.10	73.84	248.4
3	1.65	2.58	73.85	248.2
4	2.11	3.03	73.86	248.0
5	2.55	3.44	73.88	248.1

"b" Coefficient	0.614893
Index of Determination	0.999894
Slope Watts	143.3
Corrected Slope Watts (0.9842)	140.8
Ave. Corrected Radiation Value	790 R/hr
Gamma Watts (0.01499)	11.8
Total Thermal Output	152.6

Bundle ID	MT133
Date/Time Bundle Loaded	6-06-85 12:46
Date/Time Bundle Removed	6-06-85 23:59

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	73.92	243.93
1	0.55	1.50	73.93	245.72
2	1.03	1.98	73.95	245.64
3	1.51	2.42	73.97	245.53
4	1.95	2.85	73.98	245.56
5	2.35	3.23	73.99	245.36

"b" Coefficient	0.5506787
Index of Determination	0.99993
Slope Watts	119.1774
Corrected Slope Watts (0.9842)	117.2944
Ave. Corrected Radiation Value	778.9 R/hr
Gamma Watts (0.01499)	11.67
Total Thermal Output	129.0

Bundle ID MT133

Date/Time Bundle Loaded 6-09-85 05:07

Date/Time Bundle Removed 6-09-85 11:14

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.06	74.59	246.34
1	0.62	1.67	74.60	246.27
2	1.17	2.20	74.61	246.01
3	1.67	2.68	74.62	245.93
4	2.15	3.15	74.64	246.03
5	2.59	3.57	74.64	246.00

"b" Coefficient	0.6207858
Index of Determination	0.999916
Slope Watts	145.2954
Corrected Slope Watts (0.9842)	142.9998
Ave. Corrected Radiation Value	786.3 R/hr
Gamma Watts (0.01499)	11.8
Total Thermal Output	154.8

Bundle ID MT133

Date/Time Bundle Loaded 6-12-85 01:43

Date/Time Bundle Removed 6-12-85 09:57

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	69.95	250.88
1	0.47	1.43	69.98	250.66
2	0.93	1.84	70.01	250.69
3	1.37	2.23	70.05	250.99
4	1.78	2.61	70.08	250.89
5	2.17	2.96	70.11	250.86

"b" Coefficient	0.4884286
Index of Determination	0.999985
Slope Watts	95.98648
Corrected Slope Watts (0.9842)	94.46989
Ave. Corrected Radiation Value	813.4 r/hr
Gamma Watts (0.01499)	12.2
Total Thermal Output	106.7

Bundle ID	MT190
Date/Time Bundle Loaded	6-08-85 04:15
Date/Time Bundle Removed	6-08-85 11:53

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	74.33	249.25
1	0.48	1.46	74.34	248.68
2	0.91	1.87	74.35	248.02
3	1.31	2.23	74.36	247.28
4	1.68	2.60	74.37	246.89
5	2.04	2.93	74.38	247.04

"b" Coefficient	0.4807143
Index of Determination	0.9999209
Slope Watts	93.11256
Corrected Slope Watts (0.9842)	91.64139
Ave. Corrected Radiation Value	503.3 R/hr
Gamma Watts (0.01499)	7.54
Total Thermal Output	99.2

Bundle ID	MT228
Date/Time Bundle Loaded	5-30-85 04:44
Date/Time Bundle Removed	5-30-85 13:11

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	74.22	--
1	0.49	1.45	74.25	248.82
2	0.90	1.85	74.24	248.89
3	1.29	2.23	74.26	248.71
4	1.63	2.56	74.27	248.61
5	1.95	2.87	74.27	248.50

"b" Coefficient	0.4883215
Index of Determination	0.999861
Slope Watts	95.94659
Corrected Slope Watts (0.9842)	94.43063
Ave. Corrected Radiation Value	445 R/hr
Gamma Watts (0.01499)	6.6
Total Thermal Output	101

Bundle ID	MT228
Date/Time Bundle Loaded	6-07-85 14:56
Date/Time Bundle Removed	6-07-85 23:39

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	74.24	247.39
1	D.39	1.38	74.24	245.60
2	0.77	1.73	74.25	245.60
3	1.12	2.06	74.27	245.50
4	1.45	2.37	74.26	245.39
5	1.75	2.66	74.27	245.34

"b" Coefficient	0.408893
Index of Determination	0.999981
Slope Watts	66.35593
Corrected Slope Watts (0.9842)	65.3075
Ave. Corrected Radiation Value	394.6 R/hr
Gamma Watts (0.01499)	5.9
Total Thermal Output	71.2

Bundle ID	MT228
Date/Time Bundle Loaded	6-11-85 01:16
Date/Time Bundle Removed	6-11-85 09:15

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	75.14	245.54
1	0.38	1.37	75.13	245.70
2	0.74	1.71	75.15	245.59
3	1.06	2.04	75.15	245.57
4	1.34	1.98	75.15	245.72
5	1.54	2.76	75.66	245.83

"b" Coefficient	0.4203572
Index of Determination	0.9996919
Slope Watts	70.62686
Corrected Slope Watts (0.9842)	69.51096
Ave. Corrected Radiation Value	459.6 R/hr
Gamma Watts (0.01499)	6.88
Total Thermal Output	76.4

Bundle ID MT264

Date/Time Bundle Loaded 6-05-85 23:25

Date/Time Bundle Removed 6-05-85 09:05

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	73.75	246.74
1	0.34	1.31	73.76	246.61
2	0.65	1.62	73.77	246.73
3	0.95	1.91	73.78	246.71
4	1.23	2.18	73.79	246.81
5	1.50	2.42	73.80	247.09

"b" Coefficient	0.3411072
Index of Determination	0.999974
Slope Watts	41.10271
Corrected Slope Watts (0.9842)	40.45329
Ave. Corrected Radiation Value	371.6 R/hr
Gamma Watts (0.01499)	5.6
Total Thermal Output	46.1

Bundle ID MT116

Date/Time Bundle Loaded 8-27-85 11:41

Date/Time Bundle Removed 8-27-85 19:25

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	75.54	245.3
1	0.42	1.43	75.54	245.1
2	0.83	1.83	75.55	241.5
3	1.21	2.19	75.56	245.5
4	1.58	2.55	75.57	244.7
5	1.91	2.88	75.57	242.8

"b" Coefficient	0.4376073
Index of Determination	0.999969
Slope Watts	111.9
Corrected Slope Watts (0.9842)	110.1
Ave. Corrected Radiation Value	632 R/hr
Gamma Watts (0.01499)	9.5
Total Thermal Output	119.6

Bundle ID	MT123
Date/Time Bundle Loaded	8-27-85 20:17
Date/Time Bundle Removed	8-28-85 04:00

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	75.90	245.8
1	0.38	1.36	75.90	244.5
2	0.74	1.71	75.91	244.4
3	1.04	2.03	75.90	243.9
4	1.35	2.32	75.90	246.2
5	1.64	2.60	75.90	245.0

"b" Coefficient	0.3840358
Index of Determination	0.999773
Slope Watts	91.5
Corrected Slope Watts (0.9842)	90.0
Ave. Corrected Radiation Value	482 R/hr
Gamma Watts (0.01499)	7.2
Total Thermal Output	97.2

Bundle ID	MT133
Date/Time Bundle Loaded	6-13-85 03:36
Date/Time Bundle Removed	6-13-85 10:52

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.01	71.01	251.71
1	0.58	1.48	71.03	252.44
2	0.96	1.91	71.04	252.55
3	1.40	2.33	71.05	252.56
4	1.82	2.73	71.06	252.30
5	2.22	3.11	71.08	252.28

"b" Coefficient	0.5017858
Index of Determination	0.999979
Slope Watts	136.3
Corrected Slope Watts (0.9842)	134.1
Ave. Corrected Radiation Value	790 R/hr
Gamma Watts (0.01499)	11.8
Total Thermal Output	146.0

Bundle ID MT133

Date/Time Bundle Loaded 6-13-85 11:33

Date/Time Bundle Removed 6-13-85 18:30

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.01	71.82	251.29
1	0.48	1.48	71.81	250.52
2	0.95	1.93	71.83	250.79
3	1.39	2.35	71.84	250.78
4	1.80	2.74	71.85	250.66
5	2.19	3.12	71.86	250.42

"b" Coefficient	0.5001786
Index of Determination	0.999986
Slope Watts	135.7
Corrected Slope Watts (0.9842)	133.6
Ave. Corrected Radiation Value	790 R/hr
Gamma Watts (0.01499)	11.8
Total Thermal Output	145.4

Bundle ID MT133

Date/Time Bundle Loaded 8-20-85 10:32

Date/Time Bundle Removed 8-20-85 19:08

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	73.06	248.0
1	0.50	1.44	73.10	249.1
2	0.96	1.86	73.14	249.2
3	1.39	2.26	73.17	249.5
4	1.81	2.63	73.20	249.5
5	2.20	2.98	73.24	249.6

"b" Coefficient	0.5022501
Index of Determination	0.999968
Slope Watts	136.5
Corrected Slope Watts (0.9842)	134.3
Ave. Corrected Radiation Value	775 R/hr
Gamma Watts (0.01499)	11.6
Total Thermal Output	146.0

Bundle ID	MT133
Date/Time Bundle Loaded	8-21-85 08:32
Date/Time Bundle Removed	8-21-85 16:18

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.02	74.29	248.3
1	0.49	1.49	74.31	248.8
2	0.95	1.92	74.32	249.2
3	1.40	2.34	74.34	248.8
4	1.80	2.74	74.35	248.5
5	2.19	3.09	74.37	248.4

"b" Coefficient	0.5040714
Index of Determination	0.999972
Slope Watts	137.2
Corrected Slope Watts (0.9842)	135.0
Ave. Corrected Radiation Value	786 R/hr
Gamma Watts (0.01499)	11.8
Total Thermal Output	146.8

Bundle ID	MT133
Date/Time Bundle Loaded	8-29-85 15:16
Date/Time Bundle Removed	8-29-85 23:19

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	78.65	243.6
1	0.49	1.50	78.62	244.9
2	0.97	1.96	78.62	245.3
3	1.39	2.41	78.60	245.3
4	1.80	2.81	78.59	245.2
5	2.17	3.19	78.58	245.3

"b" Coefficient	0.5128571
Index of Determination	0.9999509
Slope Watts	140.6
Corrected Slope Watts (0.9842)	138.4
Ave. Corrected Radiation Value	769 R/hr
Gamma Watts (0.01499)	11.5
Total Thermal Output	149.9

Bundle ID	MT133
Date/Time Bundle Loaded	8-30-85 14:34
Date/Time Bundle Removed	8-30-85 23:12

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.01	79.12	247.0
1	0.49	1.49	79.12	245.8
2	0.94	1.95	79.11	245.8
3	1.36	2.39	79.08	246.0
4	1.77	2.78	79.09	245.8
5	2.13	3.16	79.07	245.8

"b" Coefficient	0.4983214
Index of Determination	0.999966
Slope Watts	135.0
Corrected Slope Watts (0.9842)	132.9
Ave. Corrected Radiation Value	780 R/hr
Gamma Watts (0.01499)	11.7
Total Thermal Output	144.7

Bundle ID	MT133
Date/Time Bundle Loaded	8-31-85 02:37
Date/Time Bundle Removed	8-31-85 09:36

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.01	79.27	247.2
1	0.49	1.50	79.26	247.2
2	0.95	1.96	79.25	247.2
3	1.38	2.40	79.25	247.5
4	1.76	2.79	79.24	247.6
5	2.14	3.16	79.24	247.5

"b" Coefficient	0.5045357
Index of Determination	0.999948
Slope Watts	137.4
Corrected Slope Watts (0.9842)	135.2
Ave. Corrected Radiation Value	777 R/hr
Gamma Watts (0.01499)	11.7
Total Thermal Output	147.0

Bundle ID MT133

Date/Time Bundle Loaded 9-1-85 07:30

Date/Time Bundle Removed

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.01	79.77	246.3
1	0.51	1.52	79.77	245.4
2	0.96	1.97	79.76	245.3
3	1.38	2.40	79.75	245.0
4	1.77	2.80	79.75	244.8
5	2.15	3.17	79.75	244.4

"b" Coefficient	0.5066072
Index of Determination	0.999887
Slope Watts	138.2
Corrected Slope Watts (0.9842)	136.0
Ave. Corrected Radiation Value	778 R/hr
Gamma Watts (0.01499)	11.7
Total Thermal Output	147.8

Bundle ID MT190

Date/Time Bundle Loaded 8-28-85 08:56

Date/Time Bundle Removed 8-28-85 19:58

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	76.43	244.0
1	0.30	1.39	76.41	245.2
2	0.77	1.77	76.42	245.9
3	1.13	2.11	76.42	245.8
4	1.47	2.44	76.43	245.6
5	1.77	2.74	76.43	245.5

"b" Coefficient	0.4098572
Index of Determination	0.99711
Slope Watts	101.2
Corrected Slope Watts (0.9842)	99.6
Ave. Corrected Radiation Value	532 R/hr
Gamma Watts (0.01499)	8.0
Total Thermal Output	107.6

Bundle ID MT228

Date/Time Bundle Loaded 8-20-85 22:35

Date/Time Bundle Removed 8-21-85 07:00

t	Delta T_c	Oelta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	73.58	248.5
1	0.36	1.33	73.59	248.7
2	0.70	1.63	73.62	248.9
3	1.02	1.92	73.64	248.9
4	1.31	2.20	73.66	248.9
5	1.60	2.44	73.69	248.8

"b" Coefficient	0.3682501
Index of Determination	0.999961
Slope Watts	85.4
Corrected Slope Watts (0.9842)	84.1
Ave. Corrected Radiation Value	421 R/hr
Gamma Watts (0.01499)	6.3
Total Thermal Output	90.3

Bundle ID MT264

Date/Time Bundle Loaded 8-28-85 21:59

Date/Time Bundle Removed 8-29-85 05:59

t	Delta T_c	Delta T_v	T_{pit}	Pump Power, W
0	0.00	1.00	76.92	244.7
1	0.33	1.31	76.91	244.3
2	0.62	1.61	76.90	244.3
3	0.91	1.88	76.89	244.4
4	1.17	2.16	76.89	244.6
5	1.41	2.41	76.88	244.6

"b" Coefficient	0.3326072
Index of Determination	0.999942
Slope Watts	71.8
Corrected Slope Watts (0.9842)	70.7
Ave. Corrected Radiation Value	373 R/hr
Gamma Watts (0.01499)	5.6
Total Thermal Output	76.2

TABLE A.1. Summary of Cooper and Dresden BWR Calorimetry Data

Assembly ID	Measurement Date	Measured Decay Heat, W	Form 30 Based Pred., W	Cycle Summary Based Pred., W	Form 30 Total Burnup, GWd/MTU	Cycle Summary Total Burnup, GWd/MTU	Measured Minus Form 30 Based Pred., W	Measured Minus Cycle Summary Based Pred., W
DN212	03-Oct-84	31.2	29.7	88.0	5.280	5.280	1.5	-56.8
DN212	18-Oct-84	19.5	29.7	29.7	5.280	5.280	-10.2	-10.2
CZ102	25-Sep-84	62.3	83.5	76.8	11.667	10.733	-21.2	-14.5
CZ102	14-Dec-84	70.4	82.2	75.6	11.667	10.733	-11.8	-5.2
CZ147	04-Nov-84	276.7	296.4	276.9	26.709	24.952	-19.7	-0.2
CZ148	23-Oct-84	273.5	293.6	277.5	26.310	24.868	-20.1	-4.0
CZ182	27-Sep-84	342.6	364.9	354.9	26.824	26.091	-22.3	-12.3
CZ195	30-Oct-84	255.5	289.6	262.3	26.392	23.906	-34.1	-6.8
CZ205	24-Sep-84	324.0	329.5	335.3	25.344	25.793	-5.5	-11.3
CZ205	04-Oct-84	361.5	326.6	332.4	25.344	25.793	34.9	29.1
CZ205	08-Oct-84	343.5	325.1	330.9	25.344	25.793	18.4	12.6
CZ205	09-Oct-84	352.9	324.8	330.6	25.344	25.793	28.1	22.9
CZ205	23-Oct-84	331.8	320.6	326.3	25.344	25.793	11.2	5.5
CZ205	24-Oct-84	338.7	320.3	326.0	25.344	25.793	18.4	12.7
CZ205	29-Oct-84	327.5	319.1	324.8	25.344	25.793	8.4	2.7
CZ205	02-Nov-84	313.1	318.5	324.1	25.344	25.793	-5.4	-11.0
CZ205	05-Nov-84	311.4	317.0	322.6	25.344	25.793	-5.6	-11.2
CZ205	06-Dec-84	314.0	308.2	313.7	25.344	25.793	5.8	0.3
CZ205	12-Dec-84	331.2	306.5	311.9	25.344	25.793	24.7	19.3
CZ205	22-Dec-84	317.1	303.6	309.0	25.344	25.793	13.5	8.1
CZ205	14-May-85	289.7	276.0	280.9	25.344	25.792	13.7	8.8
CZ205	28-May-85	308.0	273.6	278.4	25.344	25.792	34.4	29.6
CZ209	28-Oct-84	279.5	295.1	291.3	25.383	25.056	-15.6	-11.8
CZ211	20-Oct-84	296.0	302.7	283.2	26.679	24.958	-6.7	12.8
CZ211	20-May-85	240.3	266.6	249.4	26.679	24.958	-26.3	-9.1
CZ222	04-Nov-84	355.7	346.2	345.8	26.692	26.665	9.5	9.9
CZ225	02-Oct-84	327.3	321.4	322.8	25.796	25.905	5.9	4.5
CZ239	30-Oct-84	366.5	358.8	357.3	27.246	27.130	7.7	9.2
CZ246	02-Nov-84	320.9	364.4	356.4	27.363	26.760	-43.5	-35.5
CZ246	05-Nov-84	341.7	363.4	355.4	27.363	26.760	-21.7	-13.7
CZ259	28-Oct-84	247.6	293.3	265.2	26.466	23.930	-45.7	-17.6
CZ259	20-Dec-84	288.5	284.0	256.8	26.466	23.930	4.5	31.7
CZ259	14-May-85	254.1	260.0	235.1	26.466	23.930	-5.9	19.0
CZ264	23-Oct-84	263.8	292.6	262.5	26.496	23.767	-28.8	1.3
CZ277	28-Oct-84	262.7	290.9	262.5	26.478	23.891	-28.2	0.2
CZ277	26-May-85	243.0	261.2	235.7	26.478	23.891	-18.2	7.3
CZ286	06-Dec-84	278.4	326.8	313.8	27.141	26.065	-48.4	-35.4
CZ286	29-May-85	284.2	290.8	279.3	27.141	26.065	-6.6	4.9
CZ296	03-Nov-84	256.7	297.0	281.1	26.388	24.973	-40.3	-24.4
CZ296	21-May-85	251.9	266.0	251.7	26.388	24.973	-14.1	0.2
CZ302	24-Oct-84	285.6	290.8	289.0	26.594	26.432	-5.2	-3.4
CZ308	01-Nov-84	269.7	298.7	287.2	25.815	24.817	-29.0	-17.5
CZ311	27-Oct-84	356.9	340.1	328.5	27.392	26.455	16.8	28.4
CZ315	08-Dec-84	328.0	317.2	303.1	26.881	25.685	10.8	24.9

TABLE A.1. (contd)

Assembly ID	Measurement Date	Measured Decay Heat, W	Form 30 Based Pred., W	Cycle Summary Based Pred., W	Form 30 Total Burnup, GWd/MTU	Cycle Summary Total Burnup, GWd/MTU	Measured Minus Form 30 Based Pred., W	Measured Minus Cycle Summary Based Pred., W
CZ318	07-Dec-84	277.6	297.9	287.0	26.568	25.600	-20.3	-9.4
CZ331	24-Sep-84	162.8	161.8	169.4	21.332	22.336	1.0	-6.6
CZ331	21-Dec-84	180.1	158.3	165.8	21.332	22.336	21.8	14.3
CZ337	01-Nov-84	347.7	346.4	346.0	26.720	26.691	1.3	1.7
CZ337	24-May-85	300.4	295.5	295.2	26.720	26.691	4.9	5.2
CZ342	07-Dec-84	280.1	320.1	307.7	27.066	26.018	-40.0	-27.6
CZ342	26-May-85	300.0	286.1	275.0	27.066	26.018	13.9	25.0
CZ346	27-Oct-84	388.7	376.5	369.9	28.048	27.559	12.2	18.8
CZ348	31-Oct-84	342.8	355.5	348.1	27.481	26.910	-12.7	-5.3
CZ351	10-Dec-84	313.8	297.1	300.8	25.753	26.074	16.7	13.0
CZ355	28-Oct-84	290.5	293.0	285.9	25.419	24.803	-2.5	4.6
CZ357	08-Dec-84	320.3	326.3	318.9	27.140	26.528	-6.0	1.4
CZ369	25-Oct-84	347.6	348.3	336.6	26.576	25.680	-0.7	11.0
CZ370	28-Sep-84	288.1	292.4	292.7	26.342	26.367	-4.3	-4.6
CZ372	27-Sep-84	288.8	286.3	296.3	25.848	26.748	2.5	-8.5
CZ379	04-Nov-84	287.4	296.9	291.3	25.925	25.438	-9.5	-3.9
CZ398	27-Oct-84	372.0	361.0	351.9	27.478	26.789	11.0	20.1
CZ415	26-Sep-84	289.3	286.7	296.6	25.863	26.752	2.6	-7.3
CZ416	31-Oct-84	319.8	339.1	322.0	27.461	26.077	-19.3	-2.2
CZ429	26-Oct-84	385.6	375.4	374.7	27.641	27.586	10.2	10.9
CZ430	31-Oct-84	353.3	344.2	344.5	26.825	26.848	9.1	8.8
CZ433	26-Sep-84	287.4	281.6	285.6	25.977	26.350	5.8	1.8
CZ433	21-May-85	256.7	252.7	256.3	25.977	26.350	4.0	0.4
CZ460	09-Dec-84	313.5	308.9	305.5	26.512	26.222	4.6	8.0
CZ466	28-Sep-84	302.1	309.4	299.9	26.077	25.280	-7.3	2.2
CZ468	11-Dec-84	325.3	317.9	308.1	26.757	25.932	7.4	17.2
CZ472	26-Sep-84	325.0	321.2	323.8	25.957	26.171	3.8	1.2
CZ473	10-Dec-84	293.2	297.6	285.1	26.519	25.409	-4.4	8.1
CZ498	25-Oct-84	359.4	345.0	347.3	26.482	26.660	14.4	12.1
CZ508	09-Dec-84	310.0	309.6	304.0	26.357	25.882	0.4	6.0
CZ515	25-Sep-84	294.0	288.4	277.9	27.737	26.728	5.6	16.1
CZ515	26-Oct-84	296.0	282.7	272.4	27.737	26.728	13.3	23.6
CZ526	01-Oct-84	395.4	384.2	396.9	27.596	28.511	11.2	-1.5
CZ526	22-May-85	321.8	323.1	322.1	27.596	27.511	-1.3	-0.3
CZ528	25-Oct-84	297.6	282.5	293.8	25.715	26.748	15.1	3.8
CZ531	30-Oct-84	347.2	343.2	343.0	26.699	26.686	4.0	4.2
CZ536	27-Sep-84	295.2	296.1	294.8	26.589	26.473	-0.9	0.4
CZ542	08-Dec-84	311.9	312.4	303.9	26.691	25.969	-0.5	8.0
CZ545	11-Dec-84	295.2	300.6	287.8	26.668	25.535	-5.4	7.4

DETERMINATION OF HEAT CAPACITY CORRECTION FACTOR

I. Calibration Runs

A. Volume of water in calorimeter = 133.4 gal measured heater attached.
Weight of water = (144.5)(8.3 lb/gal) = 1107.2 lb

B. Weight of pipe holding calibration heater = 151 lb

C. Heat capacity of system

$$
\begin{aligned}
(1107.2\ \text{lb})(1.0\ \text{Btu/lb°F}) &= 1107.2\ \text{Btu/°F} \\
(151\ \text{lb})(0.11\ \text{Btu/lb°F}) &= \underline{\ \ 16.6\ \text{Btu/°F}} \\
&\ 1123.8\ \text{Btu/°F}
\end{aligned}
$$

II. BWR - Dresden Fuel Element

A. Volume of water in calorimeter
BWR fuel assembly displaces 8.08 gal
(8.08 gal)(8.3 lb/gal) = 67.1 lb
Thus, 1107.2
- 67.1
1040.1 lb in calorimeter

B. From published information
UO_2 = 398.8 lb (197 kg U/bundle)
Weight of hardware = 216.8 lb

C. Heat capacity of system
During fuel runs, the calibration heater was removed; therefore, the volume of water would be increased slightly.
151 lb as steel = 18.2 lb as water (equal volumes)
1040.1 lb
+ 18.2 lb
1058.3 lb

$$
\begin{aligned}
(1058.3\ \text{lb})(1.0\ \text{Btu/°F}) &= 1058.3\ \text{Btu/°F} \\
(398.8\ \text{lb})(0.06\ \text{Btu/°F}) &= 23.9\ \text{Btu/°F} \\
(216.8\ \text{lb})(0.11\ \text{Btu/°F}) &= \underline{\ \ 23.8\ \text{Btu/°F}} \\
&\ 1106.0\ \text{Btu/°F}
\end{aligned}
$$

$$\frac{1123.8 - 1106.0}{1123.8} = 0.0158$$

Correction factor = 1 - 0.0158 = 0.9842

The correction factor is used to reduce the measured thermal output ("slope" watts) to correct for bias between heat capacity of system during calibration and actual fuel assembly runs.

Thus, (measured watts)(0.9842) = corrected watts.

CORRECTION FOR GAMMA HEAT LOST FROM THE CALORIMETER VESSEL

Because the walls of the calorimeter are not thick, some radiation passes through and does not contribute to the temperature rise within the calorimeter. It is essentially "lost" and, if not compensated for, will result in a negative bias in the calorimetric determination of the thermal output of the fuel.

In this analysis the following bases and assumptions apply:

1. Gamma energy absorbed in the inner steel shell of the calorimeter is transferred as heat to the water in the calorimeter.
2. Gamma energy absorbed by the foam insulation is insignificant and is ignored.
3. Gamma energy absorbed in the outer steel shell of the calorimeter is transferred to the basin water, as infinite heat sink.
4. From dose rate measurements made by ion chambers mounted on both sides of the outer shell, the attenuation factor due to the steel shell was determined to be 2.16.

The average energy of the gamma radiation traversing the outer shell may be calculated from the attenuation coefficient using the expression

$$\frac{I}{I_o} = e^{-\mu x}$$

where I is the intensity of the transmitted radiation

I_0 is the intensity of the incident radiation

x is the surface density of the absorber = g/cm^2

μ is the mass attenuation coefficient - cm^2/g.

The wall thickness is 0.953 cm (0.375 in.) and the surface density (0.953 cm) is multiplied by ($7.86\ g/cm^3$) or $7.49\ g/cm^2$.

$$\mu = \frac{1}{x} \ln \frac{I_0}{I} \qquad \text{or} \qquad \mu = \frac{1}{7.49} \ln 2.16 = 0.1028$$

A mass attenuation coefficient of 0.1028 for iron corresponds to a gamma energy of 0.345 MeV (U.S. HEW 1970, p. 138).

At an energy of 0.345 MeV, 1 R/hr is equivalent to an energy fluence rate of $5.25 \times 10^5\ MeV/cm^2$ (U.S. HEW 1970, p. 132).

Because 1 W is equal to 6.24×10^{12} MeV/sec,

$$1\ \text{R/hr} = \frac{5.25 \times 10^5\ \text{Mev}}{\text{cm}^2\ \text{sec}} \times \frac{1\ \text{watt sec}}{6.24 \times 10^{12}\ \text{MeV}} = \frac{8.41 \times 10^{-1}\ \text{watts}}{\text{cm}^2}$$

The calorimeter vessel is 15 ft long and 22 in. in diameter. Its surface area at the inner surface of the outer shell is 3.1416 x 21.25 in. x 180 in. + 2 $(3.1416)(21.25/2)^2$ = 12,795 $in.^2$ = 82,550 cm^2.

For an outer surface dose rate of 1 R/hr the inner surface dose rate is 2.16 R/hr. Hence, the gamma energy lost from the calorimeter for an outer surface dose rate of 1 R/hr is

$$2.16\ \text{R/hr}\ \frac{82550\ \text{cm}^2 \times 8.41 \times 10^{-8}\ \text{watts}}{\text{cm}^2\ \text{R/hr}} = 0.01499\ \text{watt}$$

$$\text{correction factor} = \frac{0.01499\ \text{watt}}{1\ \text{R/hr gamma at outer surface}}$$

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U.S. Department of Health, Education and Welfare. 1970. Radiological Health Handbook. Revised ed. U.S. Government Printing Office, Washington, D.C.

APPENDIX B

ORIGEN2 INPUT

APPENDIX B

ORIGEN2 INPUT

This appendix contains the operating histories used as input to the ORIGEN2 computer code, and the complete ORIGEN2 input for fuel assembly MT133.

TABLE B.1. Monticello Spent Fuel Assembly Operating History

Cycle	Date	Days Since Startup	Core Average Power, MW/MT
1	2-19-71	0	0.0
1		18	0.54
1		57	0
1		95	5.45
1		120	0
1		145	14.17
1		157	0
1		187	14.17
1		195	0
1		266	13.08
1		347	0
1		507	16.90
1		514	0
1		518	11.99
1		530	0
1		664	17.44
1		671	0
1	3-02-73	742	17.44
2	5-19-73	820	0
2		848	14.39
2		852	0
2		892	16.79
2		903	0

TABLE B.1. (contd)

Cycle	Date	Days Since Startup	Core Average Power, MW/MT
2		952	16.13
2		960	0
2		998	16.13
2		1003	0
2		1119	14.06
3	5-19-74	1185	0
3		1203	13.08
3		1207	0
3		1230	15.48
3		1235	0
3		1358	14.93
3		1362	0
3		1365	10.36
3		1371	0
3	1-09-75	1420	13.73
4		1449	0
4		1546	16.13
4		1549	0
4	9-11-75	1665	12.64

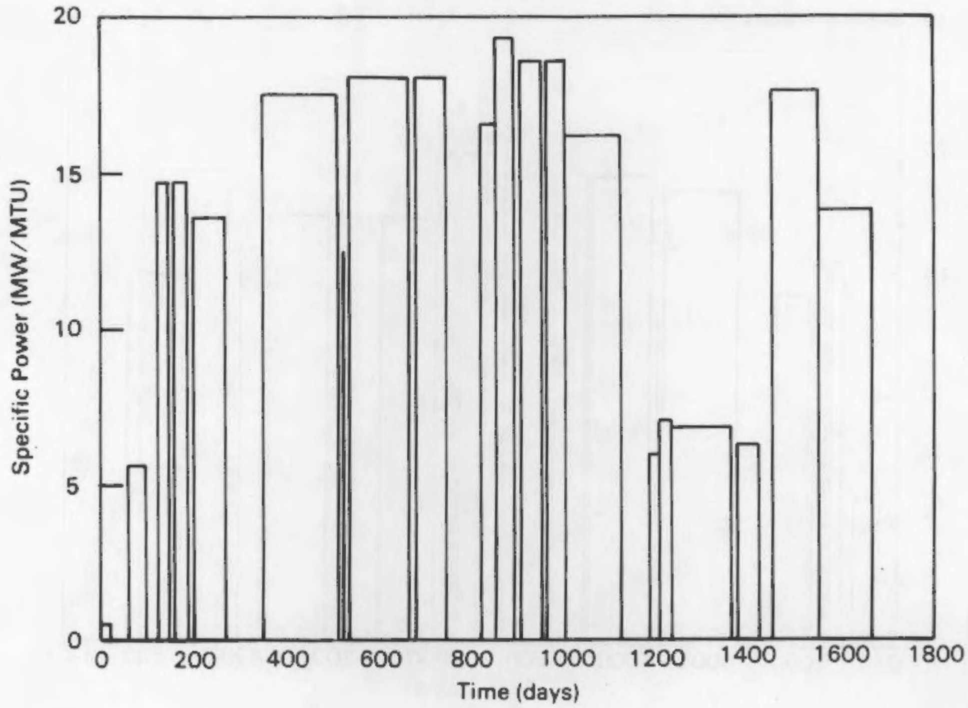

FIGURE B.1. Monticello Assembly MT116 Power History

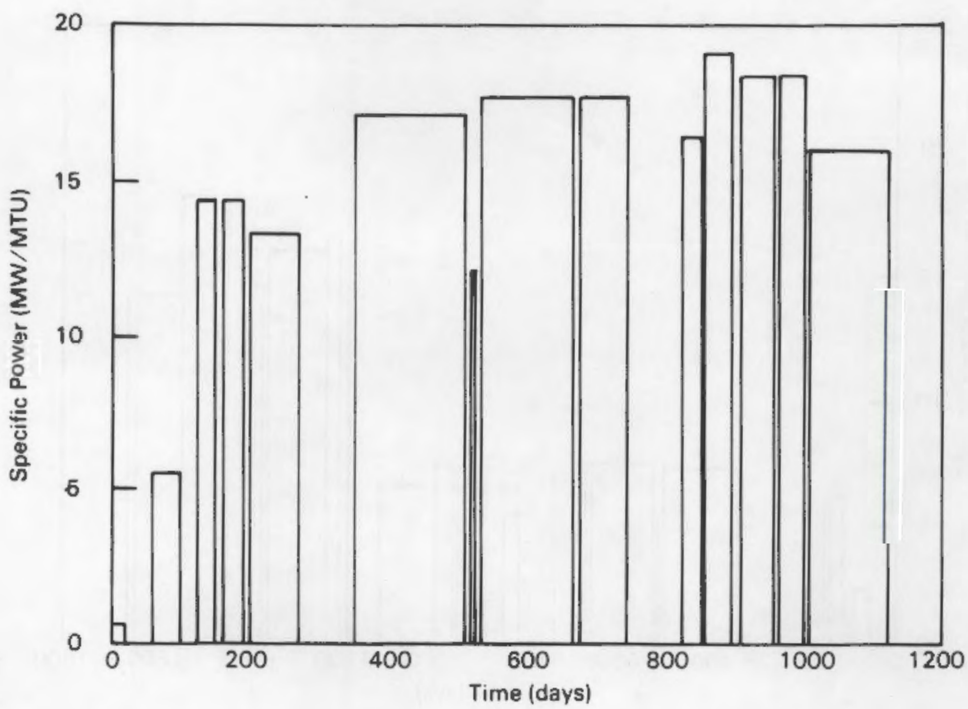

FIGURE B.2. Monticello Assembly MT123 Power History

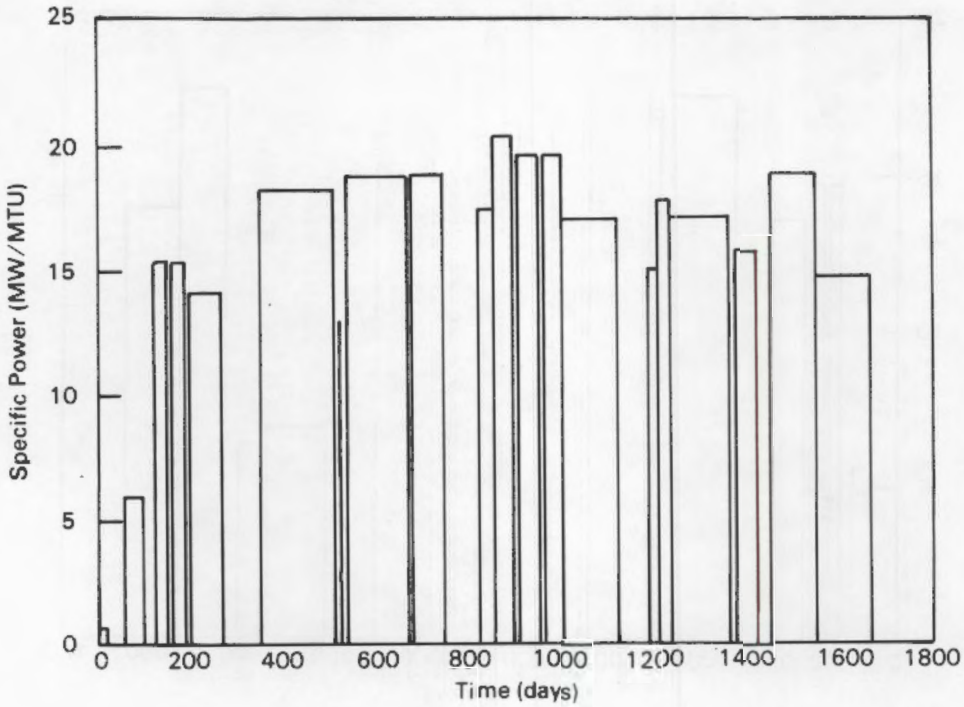

FIGURE B.3. Monticello Assembly MT133 Power History

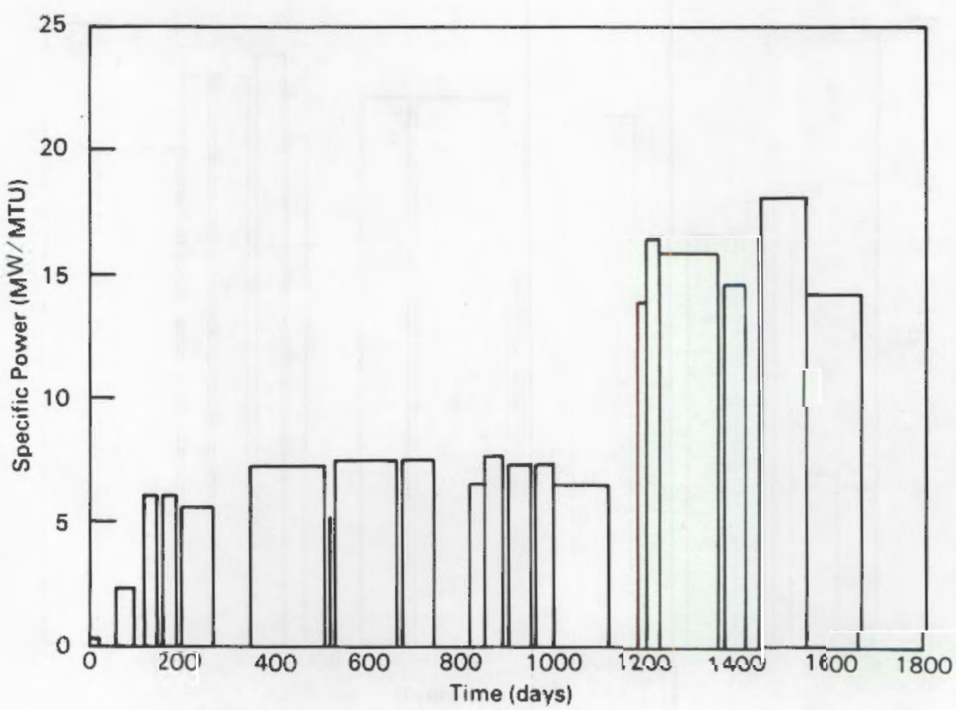

FIGURE B.4. Monticello Assembly MT190 Power History

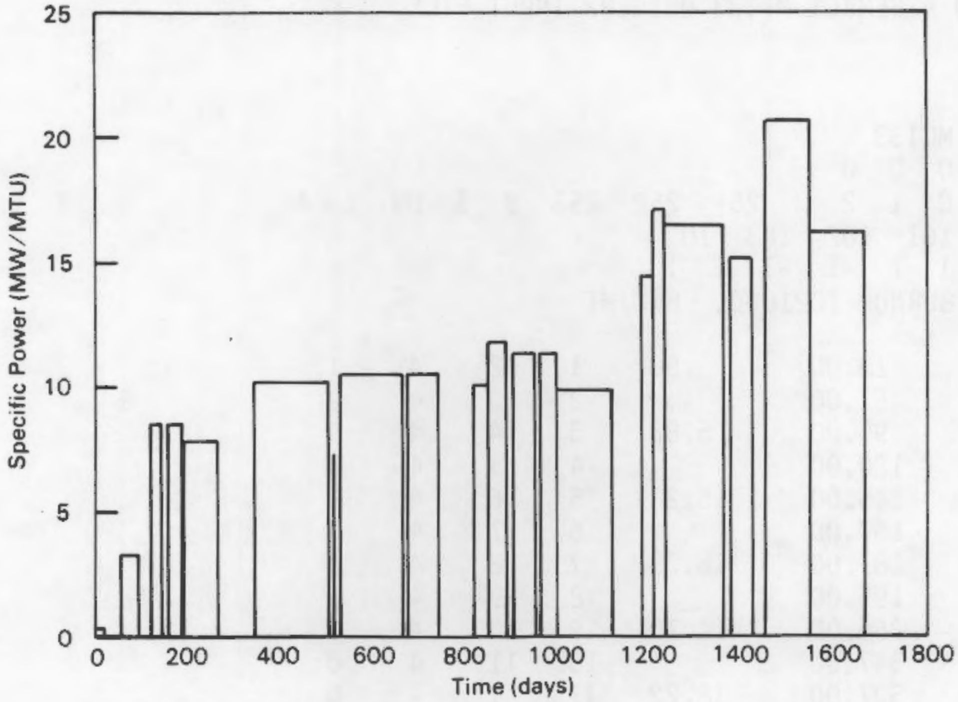

FIGURE B.5. Monticello Assembly MT228 Power History

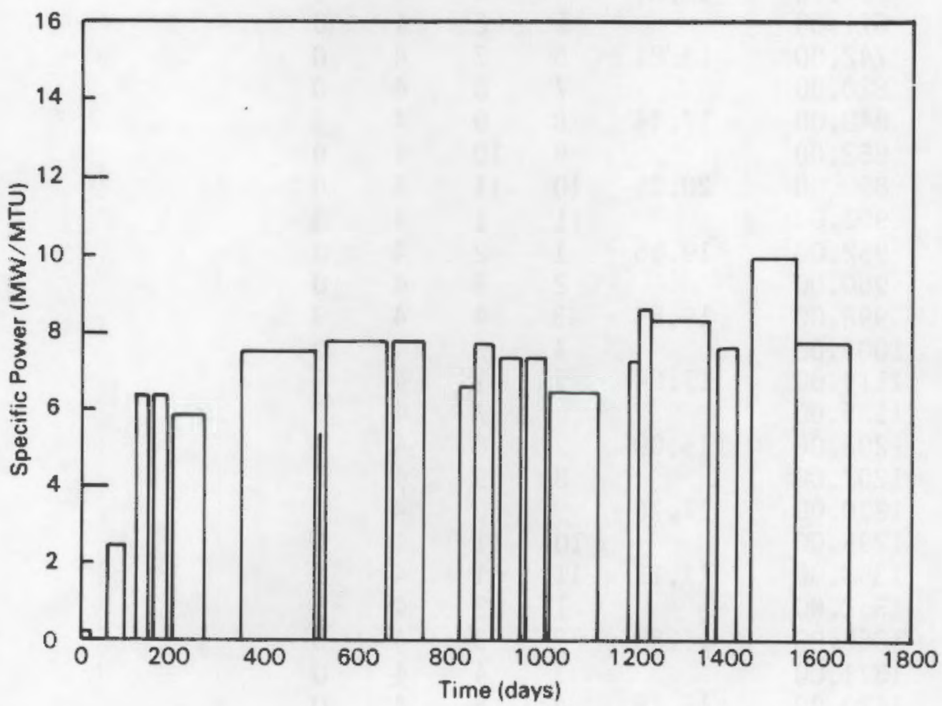

FIGURE B.6. Monticello Assembly MT264 Power History

```
MONTICELLO ASSEMBLY MT133 ORIGEN2 INPUT FILE
-1
-1
-1
TIT       MC133
LIP       0  0  0
LIB       0  1  2  3  251  252  253  9  3  09  1  4
PHO       101  102  103  10
INP       1  1  -1  -1  1  1
RDA       BURNUP TO21000.  MWD/MT
BUP
IRP          18.00       .58    1    2    4    1
DEC          57.00              2    3    4    0
IRP          95.00      5.88    3    4    4    0
DEC         120.00              4    5    4    0
IRP         145.00     15.28    5    6    4    0
DEC         157.00              6    7    4    0
IRP         187.00     15.28    7    8    4    0
DEC         195.00              8    9    4    0
IRP         266.00     14.10    9   10    4    0
DEC         347.00             10   11    4    0
IRP         507.00     18.22   11    1    4    0
DEC         514.00              1    2    4    0
IRP         518.00     12.93    2    3    4    0
DEC         530.00              3    4    4    0
IRP         664.00     18.81    4    5    4    0
DEC         671.00              5    6    4    0
IRP         742.00     18.81    6    7    4    0
DEC         820.00              7    8    4    0
IRP         848.00     17.44    8    9    4    0
DEC         852.00              9   10    4    0
IRP         892.00     20.35   10   11    4    0
DEC         903.00             11    1    4    0
IRP         952.00     19.55    1    2    4    0
DEC         960.00              2    3    4    0
IRP         998.00     19.55    3    4    4    0
DEC        1003.00              4    5    4    0
IRP        1119.00     17.04    5    6    4    0
DEC        1185.00              6    7    4    0
IRP        1203.00     15.00    7    8    4    0
DEC        1207.00              8    9    4    0
IRP        1230.00     17.76    9   10    4    0
DEC        1235.00             10   11    4    0
IRP        1358.00     17.12   11    1    4    0
DEC        1362.00              1    2    4    0
IRP        1365.00     11.88    2    3    4    0
DEC        1371.00              3    4    4    0
IRP        1420.00     15.75    4    5    4    0
DEC        1449.00              5    6    4    0
IRP        1546.00     18.78    6    7    4    0
DEC        1549.00              7    8    4    0
IRP        1665.00     14.72    8    9    4    0
BUP
```

```
MOV       9  1  0  1.0
DEC     3400.   1   2  4  1
DEC     3431.   2   3  4  0
DEC     3459.   3   4  4  0
DEC     3490.   4   5  4  0
DEC     3520.   5   6  4  0
DEC     3551.   6   7  4  0
DEC     3581.   7   8  4  0
DEC     3612.   8   9  4  0
DEC     3763.   9  10  4  0
DEC     3916.  10  11  4  0
HED       1 DISCHARGE
HED       2 JAN 1, 85
HED       3 FEB 1, 85
HED       4 MAR 1, 85
HED       5 APR 1, 85
HED       6 MAY 1, 85
HED       7 JUN 1, 85
HED       8 JUL 1, 85
HED       9 AUG 1, 85
HED      10 DEC31, 85
HED      11 JUN 1, 86

OPTA      8 8 8 8 7 8 8 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTL      8 8 8 8 8 8 8 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTF      8 8 8 8 7 8 8 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OUT     11    1    0   -1
STP       4
4         1000     3.2      50000  0.1          60000 120.9       70000  49.4
4         80000 134685.6    90000 10.7          110000   15.0      120000 2.0
4         130000  37.4      140000 36.6         150000 36.1       160000   9.1
4         170000 5.3        180000 2.0          220000 52.1        230000 7.9
4         240000   959.9    250000   49.9       260000   2022.4   270000 16.5
4         280000 1653.4     290000 7.7          400000 238910.3   420000 65.5
4         470000 0.1        480000 25.1         290000 2.0        500000 3905.4

4         640000    2.5     720000 19.0         740000 6.9        820000 1.0
2         922350 2250.      922380 977500.      0       0.0
0
END
```

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Publishing Coordination (2)