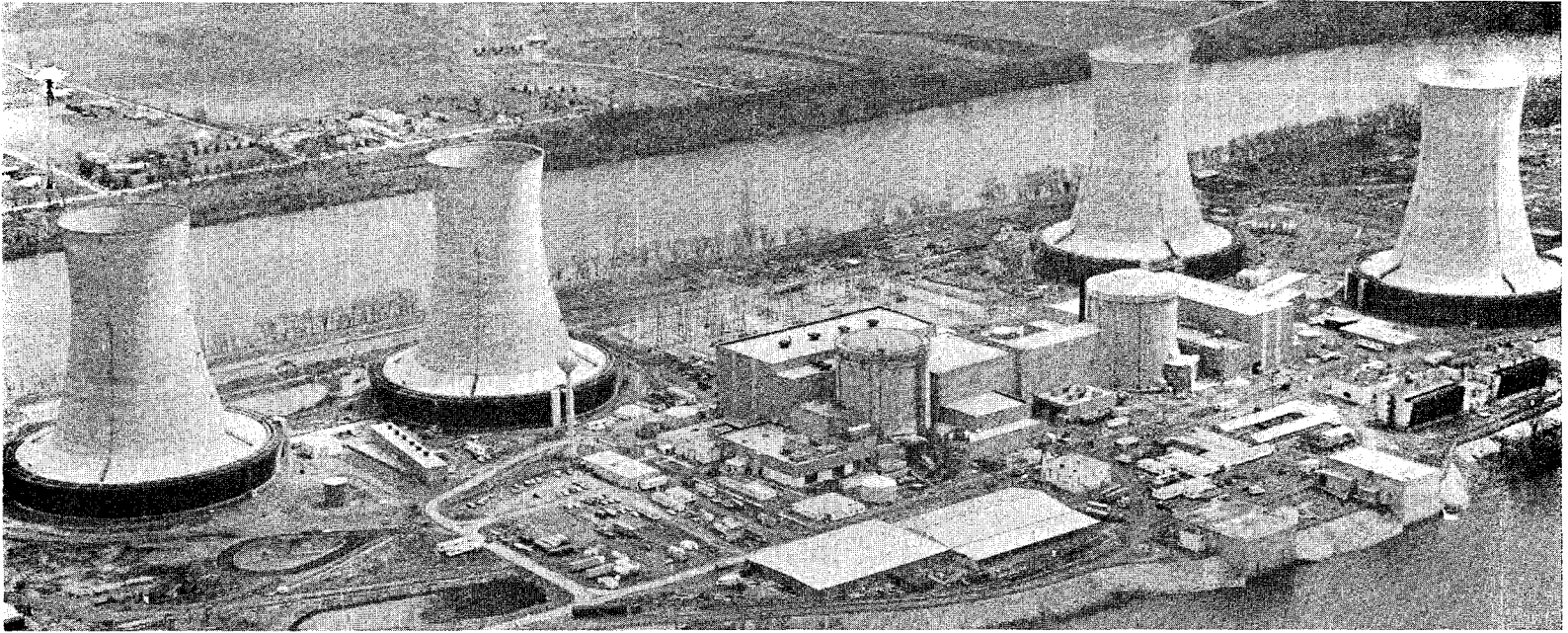


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TMI-2 Reactor Building Source Term Measurements: Surfaces and Basement Water and Sediment

**Charles V. McIsaac
David G. Keefer**

October 1984

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Three Mile Island Operations Office
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**TMI-2 REACTOR BUILDING SOURCE TERM
MEASUREMENTS: SURFACES AND BASEMENT
WATER AND SEDIMENT**

**Charles V. McIsaac
David G. Keefer**

Published October 1984

**EG&G Idaho, Inc.
Idaho Falls, Idaho 83415**

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ABSTRACT

Presented in this report are the results of radiochemical and elemental analyses performed on samples collected from the Three Mile Island Unit 2 Reactor Building from August 1979 to December 1983. The quantities of fission products and core materials that were measured on the external surfaces in the Reactor Building or in the water and sediment in its basement are summarized. Recent analysis results for access panels removed from the air cooling assembly and for liquid and particulate samples collected from the Reactor Building sump and reactor coolant drain tank are included in the report. Measurements show that 59% of the ^3H , 2.7% of the ^{90}Sr , 15% of the ^{129}I , 20% of the ^{131}I , and 42% of the ^{137}Cs originally in the core at the time of the accident could be accounted for outside the core in the Reactor Building. With the exceptions of ^{90}Sr and ^{144}Ce , the vast majority of each radionuclide released was found dispersed in the water and sediment in the basement.

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TMI-2 REACTOR BUILDING SOURCE TERM MEASUREMENTS:
SURFACES AND BASEMENT WATER AND SEDIMENT

INTRODUCTION

During the first three days following the shutdown of the Three Mile Island Unit 2 (TMI-2) reactor in March 1979, an estimated 1.0×10^6 L of contaminated primary coolant escaped from the reactor coolant system (RCS) through the power-operated relief valve (PORV) on the pressurizer. This lost coolant flowed through interconnecting piping to the reactor coolant drain tank (RCDT), located in the southwest quadrant of the Reactor Building basement. The rupture disk on the RCDT burst within minutes of the initial opening of the PORV, allowing primary coolant to pass from the tank through a vent line to the Reactor Building basement. After about 1 h, 40 min. of continuous leakage through the PORV, enough coolant escaped to uncover the core. During the 2 h the core is estimated to have been uncovered, significant amounts of hydrogen gas and fission products were released to the RCS as a result of the oxidation and embrittlement of the zircaloy fuel cladding. The hydrogen then escaped through the PORV to the Reactor Building, where there occurred a hydrogen burn. The resulting pressure spike actuated the post-loss-of-coolant accident Reactor Building spray system. This system, located in the Reactor Building dome, sprayed 6.4×10^4 L of water into the Reactor Building atmosphere. During this time, the Reactor Building air cooling assembly on the 305-ft elevation floor was circulating moisture-laden air from its immediate environment to the Reactor Building dome.

The Reactor Building atmosphere likely remained saturated with water for months following the accident, resulting in extensive pooling of contaminated water on upper-level horizontal surfaces. During these months the volume of contaminated water in the basement continued to increase because the RCS and the river water cooling system, which was supplying water to the Reactor Building air cooling assembly, were leaking. The water in the basement would eventually reach a depth of 2.59 m by September 1981.

Shortly after the accident, four organizations interested in both plant recovery and accident data acquisition formally agreed to cooperate in obtaining and distributing information in these areas. The organizations, commonly referred to as the GEND Group--GPU Nuclear, Electric Power Research Institute, Nuclear Regulatory Commission, and Department of Energy (DOE)--are currently involved in reactor recovery and data acquisition activities. To ensure that an adequate level of effort was maintained to acquire new data throughout the recovery process, DOE created a Data Acquisition Program when it established in 1979 the DOE Technical Information and Examination Program office at TMI-2. One objective of the Data Acquisition Program has been to determine the quantity of fission products and core materials that were released to the Reactor Building environment. Toward this end, DOE has funded the collection and analysis of many of the samples that have been removed from the Reactor Building.

The first sampling of the water and sediment in the Reactor Building basement took place in August 1979 when the volume of water in the basement was about 2×10^6 L. Since that time, liquid and sediment samples have been collected from the basement at six other locations, the most current samples having been collected in August 1983. In addition, similar samples were collected from the RCDT in December 1983. Several regions of the basement, including the inside of the RCDT, were also visually examined in 1983 using a closed circuit television system.

The first systematic samplings of Reactor Building surfaces took place in December 1981 and March 1982, before and after the Reactor Building gross decontamination experiment. One hundred eighty samples were collected from Reactor Building structural surfaces such as floors and D-ring walls. Using a different sampling technique, additional surface samples were collected in September 1983 to determine the depth activity penetrated into the painted structural concrete. Seventeen core samples were removed from Reactor Building concrete floors and D-ring walls.

In October 1981, in situ gamma spectral measurements were made of the total activities deposited on the surfaces of three of the five cooling

coils that are installed in the Reactor Building air cooling assembly. To augment these surface activity data, a number of scrapes and smears were collected in 1983 from various external and internal surfaces of the assembly. In addition, the five cooling coil access panels were removed from the assembly for subsequent laboratory analysis.

Many of the analyses results for the surface and basement water and sediment samples previously described have been reported elsewhere. The results of the radiochemical and gamma-ray spectrometry analyses performed on the surface samples collected during 1981 and 1982 were reported in GEND-037, published in October 1983.¹ Analyses results for the concrete core samples collected in September 1983 were published in June 1984 as GEND-INF-052.² The results of radiochemical and elemental analyses performed on the samples of water and sediment collected from the Reactor Building basement, with the exception of results for the Reactor Building sump and RCDT samples, were previously reported in GEND-INF-11, Vol. I, II, and III, which were published in July 1981,³ October 1982,⁴ and June 1983,⁵ respectively.

The purpose of this report is to summarize the quantities of fission products and core material elements that were measured either on the external surfaces in the Reactor Building or in the water and sediment in its basement. Recent surface analysis results for the five cooling coil access panels that were removed from the Reactor Building air cooling assembly are included in the summary. The analysis results for samples that were collected from the Reactor Building sump and RCDT in 1983 are also included. The radiochemical and elemental analyses results for these latter samples are compiled with those for the water and sediment samples that were collected from the basement during the period August 1979 through January 1983. In most cases, the data cited for the basement water and sediment samples were obtained from the original letter data reports that were issued by the laboratories that performed the analyses.

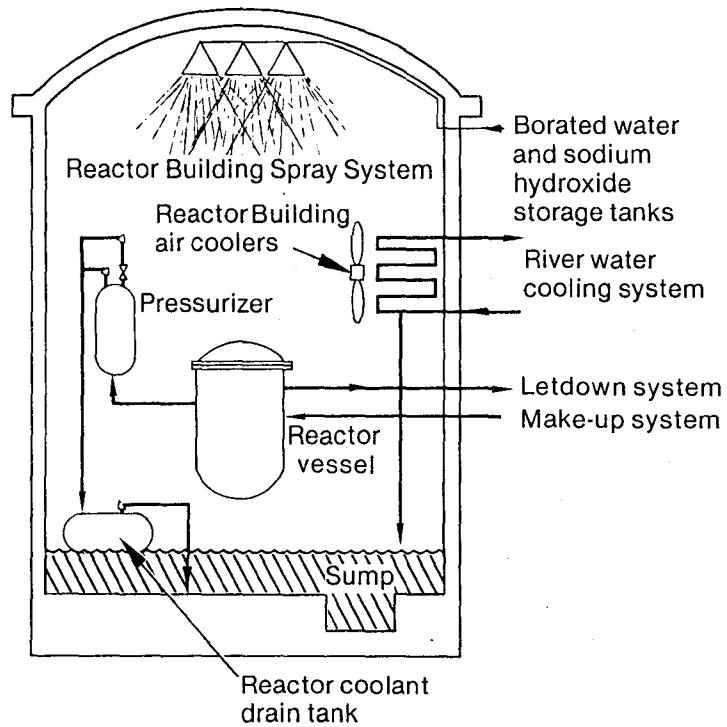
FLOODING OF THE REACTOR BUILDING BASEMENT

The three floors in the Reactor Building are designated by their elevations, in feet, above sea level. Thus the basement floor, which is at an elevation of 282 ft, 6 in., is commonly known as the 282-ft floor. The basement is the area between the 282-ft and 305-ft floors. A number of cubicles located in the basement shield components such as the reactor coolant drain tank (RCDT), RCDT pump, leakage coolers, leakage transfer pumps, letdown coolers, and sump. All of these components, except the sump, are at or above the level of the basement floor. The volume of the RCDT, 2.74×10^4 L, is about three times the volume of the Reactor Building sump.

When the first batch of water was pumped out of the basement on September 23, 1981 for processing through the Submerged Demineralizer System (SDS) and EPICOR II ion-exchange resins, the water level in the basement had reached about 2.6 m (8 ft, 6 in.). This water is attributed to three major sources: the reactor coolant system (RCS), the Reactor Building spray system, and the river water cooling system. These three sources of water that flooded the basement are shown in Figure 1.

Accident Water

The TMI-2 plant had been operating about three months when the series of events that led to the accident began at 0400 on March 28, 1979. At about that time, the pumps that normally supplied secondary water to the steam generators shut down because of an unusual electrical malfunction. The auxiliary feedwater pumps automatically turned on when the main pumps tripped, but because the block valves downstream of the auxiliary pumps were closed, the flow of secondary water to the steam generators stopped. When the flow of feedwater was interrupted, the level of secondary water in the steam generators dropped rapidly, thereby reducing their capacity to remove heat from the RCS. As a result, the pressure in the RCS increased dramatically and the reactor automatically shut down.



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Figure 1. Reactor Building basement water sources include the reactor coolant system, the spray system, and the air coolers.

The pressure in the RCS quickly increased beyond the high-level set point of the power-operated relief valve (PORV) that is located on the RCS pressurizer. When this relief valve opened, reactor coolant began to escape from the RCS. The lost coolant flowed through a 36-cm-diameter pipe to the RCDT located in the basement some 18 m below the level of the PORV. As a consequence of rapid pressurization, the rupture disk on this tank soon burst and coolant escaped to the basement floor through a 46-cm-diameter pipe that encloses the rupture disk. The rupture line rises vertically from the top of the tank, turns 90 degrees, and terminates in a penetration in the west wall of the RCDT cubicle. Coolant continued to escape to the Reactor Building basement via this pathway until 0620 when the PORV block valve was closed. Additional coolant, in the form of steam and water, and hydrogen gas escaped through the PORV from 0713 to 1700 hours, when the block valve was intermittently opened to regulate RCS pressure. An estimated 1×10^6 L of reactor coolant was released to the basement via this same pathway during the first three days following the onset of the accident.⁶

In addition to the 1×10^6 L of RCS water released during the accident, an average of 29.5 L/h flowed through the PORV block valve for more than two years following the accident. This leakage contributed 6.74×10^5 L of RCS water to the basement water volume (see Reference 6). Thus the total volume of RCS water that escaped to the basement was approximately 1.67×10^6 L, which is about 69% of the total volume of water released to the basement as of September 23, 1981.

As a result of the hydrogen burn pressure spike that occurred at 1350 hours on the day of the accident, the Reactor Building spray system activated and remained on for 5 min., 40 s. During that time, the system discharged an estimated 6.43×10^4 L of chemically treated water, containing boron and sodium hydroxide, into the Reactor Building atmosphere (see Reference 6). Sodium hydroxide is added to the water to remove halogens like I and Br. Upper level radiation monitors registered decreases following the spray, indicating that the sprayed water effectively removed at least some of the airborne contaminants. Most of this water probably eventually drained to the basement. The volume of

water discharged by the spray system represents about 3% of the total basement water volume as of September 23, 1981.

Further increases in the basement water level after the accident are attributed to leakage from the river water cooling system of the Reactor Building air cooling assembly. The leakage is suspected to have been from a relief valve on the assembly cooling coils. Based on back projections of water level and reconstruction of events associated with water inventory, an estimated 6.81×10^5 L of river water was released to the basement from this source before it was secured (see Reference 6). The river water from this source represents about 28% of the maximum basement water inventory prior to the start of SDS processing in September 1981.

Decontamination Water

After some of the initial accident water was removed from the basement and processed through the SDS and EPICOR II systems, it was staged for use in decontaminating the Reactor Building. This water began to be recycled to the Reactor Building in March 1982 when upper level floors and walls, cable trays, and major pieces of equipment were sprayed with high- and low-pressure processed water. Most of this decontamination water drained to the basement, carrying with it the fission products that were washed from the structural and equipment surfaces.

The depth of the water in the Reactor Building basement from May 1979 through December 1983 is shown graphically in Figure 2. Prior to the start of SDS processing on September 23, 1981, the water level had been increasing at a fairly constant rate due to leakage from the RCS and the river water cooling system. However, by the time the gross decontamination experiment commenced six months later in March 1982, about 2.3×10^6 L of contaminated water had been pumped from the basement and processed through the SDS. This initial processing, which was done in 16 separate batches, reduced the water depth to about 17 cm (7 in.). The gross decontamination experiment and subsequent decontamination operations periodically increased the water depth. As of mid-April 1983, an estimated 1.4×10^6 L of processed water had been used for decontamination purposes and had returned to the basement.

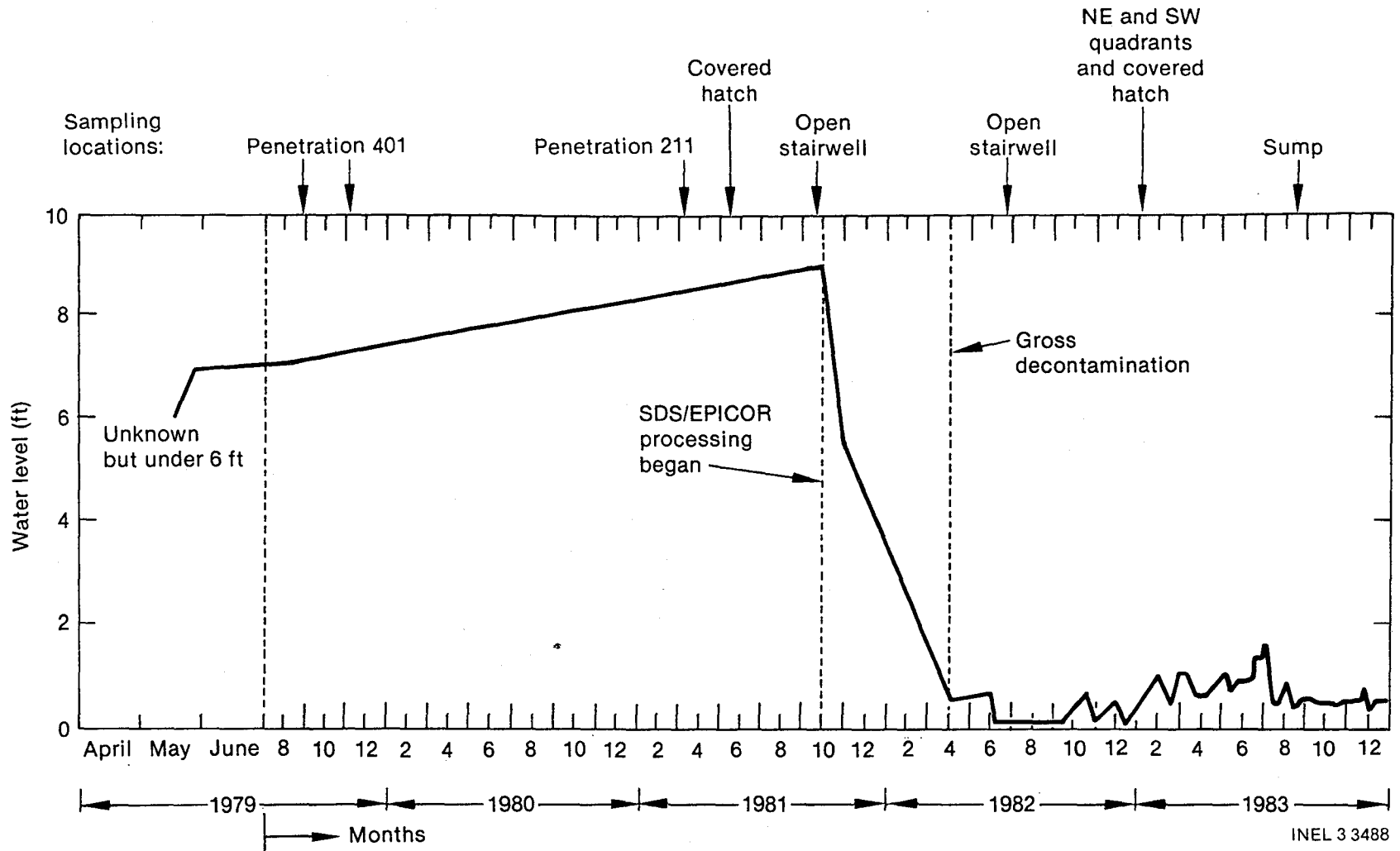


Figure 2. Reactor Building water level during the period the basement was sampled.

REACTOR BUILDING BASEMENT SAMPLES

The samples of liquid and sediment that have been collected from the Reactor Building basement since August 1979 are summarized in Table 1. To date, 24 samples have been acquired using a variety of sampling techniques. Samples obtained prior to the start of Submerged Demineralizer System (SDS) processing were collected to determine the unperturbed inventories of fission products and core materials in the water and sediment in the basement. Because SDS processing removed the majority of the cesium and strontium activity from the water that was used for Reactor Building decontamination, samplings performed since March 1982 focused on characterizing the sediment present on the basement floor. The locations where liquid and sediment samples have been collected to date are shown in Figure 3, which is the floor plan of the Reactor Building basement.

On August 28, 1979, a flexible tygon tube with a weighted end was inserted through Reactor Building penetration 401, and three 30-mL samples of basement water were pumped into separate sample containers. The first sample was extracted from near the surface of the water, the second at a depth nearly midway into the water, and the third from just above the basement floor. Penetration 401 is directly south of where water entered the basement through the reactor coolant drain tank (RCDT) rupture line. The same penetration and sampling method were used on November 15, 1979 to collect a 1-L sample from near the basement floor. The next sampling occurred on March 19, 1981, when three 1-L samples of water were pumped from about 4 ft below the surface of the water in the vicinity of penetration 211, which is on the opposite side of the building with respect to the RCDT. The majority of the basement water samples collected at later dates were obtained using manually and electrically operated vacuum-actuated samplers developed by EG&G Idaho, Inc.

Two versions of the manually operated sampler were used at various times to collect samples of water and sediment from the basement. One model of the sampler was equipped with a single, evacuated glass serum vial positioned at the sampler's lower extremity, while the second was equipped with eight such vials positioned along its length to allow the simultaneous

TABLE 1. REACTOR BUILDING BASEMENT SAMPLES

| Sample Collection Date | Location ^a | Specific Location | Number of Samples Obtained | Quantity (mL) | Physical Character | Sampling Technique Used | Laboratory ^b |
|------------------------|-----------------------|---|----------------------------|---------------|--------------------|--|-------------------------|
| 8/28/79 | 1 | Penetration 401, 292-ft elevation | 3 | 30 ea. | Water, slurry | Liquid suction through tygon tubing | ORNL |
| 11/15/79 | 1 | Penetration 401 | 1 | 1050 | Slurry | Liquid suction through tygon tubing | ORNL |
| 3/19/81 | 2 | Penetration 211 | 3 | 1000 ea. | Water | Liquid suction through tygon tubing | ORNL |
| 5/14/81 | 3 | Covered equipment hatch, 305-ft elevation | 8 | ~100 ea. | Water, slurry | Vacuum actuated, plunger-operated sampler | INEL |
| 9/24/81 | 4 | Open stairwell, 305-ft elevation | 1 | ~100 | Slurry | Vacuum actuated, plunger-operated sampler | INEL |
| 6/23/82 | 5 | Bottom of open stairwell | 1 | ~45 | Slurry | Manual scoop | ORNL, WHEDL |
| 1/11/83 | 3 | Covered equipment hatch, 305-ft elevation | 1 | 45 | Slurry | Remotely operated, solenoid-actuated sampler | PNL/TMI |
| 1/11/83 | 6 | Northeast quadrant, penetration 238 | 1 | 55 | Slurry | Remotely operated, solenoid-actuated sampler | PNL/TMI |

TABLE 1. (continued)

| Sample Collection Date | Location ^a | Specific Location | Number of Samples Obtained | Quantity (mL) | Physical Character | Sampling Technique Used | Laboratory ^b |
|------------------------|-----------------------|-------------------------------------|----------------------------|---------------|--------------------|--|-------------------------|
| 1/11/83 | 7 | Southwest quadrant, penetration 225 | 1 | 55 | Slurry | Remotely operated, solenoid-actuated sampler | PNL/TMI |
| 8/22/83 | 8 | Sump pump discharge line | 2 | 300 ea. | Slurry | On-line sampling bomb | INEL, WHEDL |
| 12/5,12/83 | 9 | Reactor coolant drain tank | 2 | ~120 ea. | Water, slurry | Vacuum actuated, plunger-operated sampler | INEL, WHEDL |

a. Location number refers to Figure 3.

b. ORNL: Oak Ridge National Laboratory; INEL: Idaho National Engineering Laboratory; WHEDL: Westinghouse Hanford Engineering Development Laboratory; PNL/TMI: Pacific Northwest Laboratory Mobile Facility at TMI.

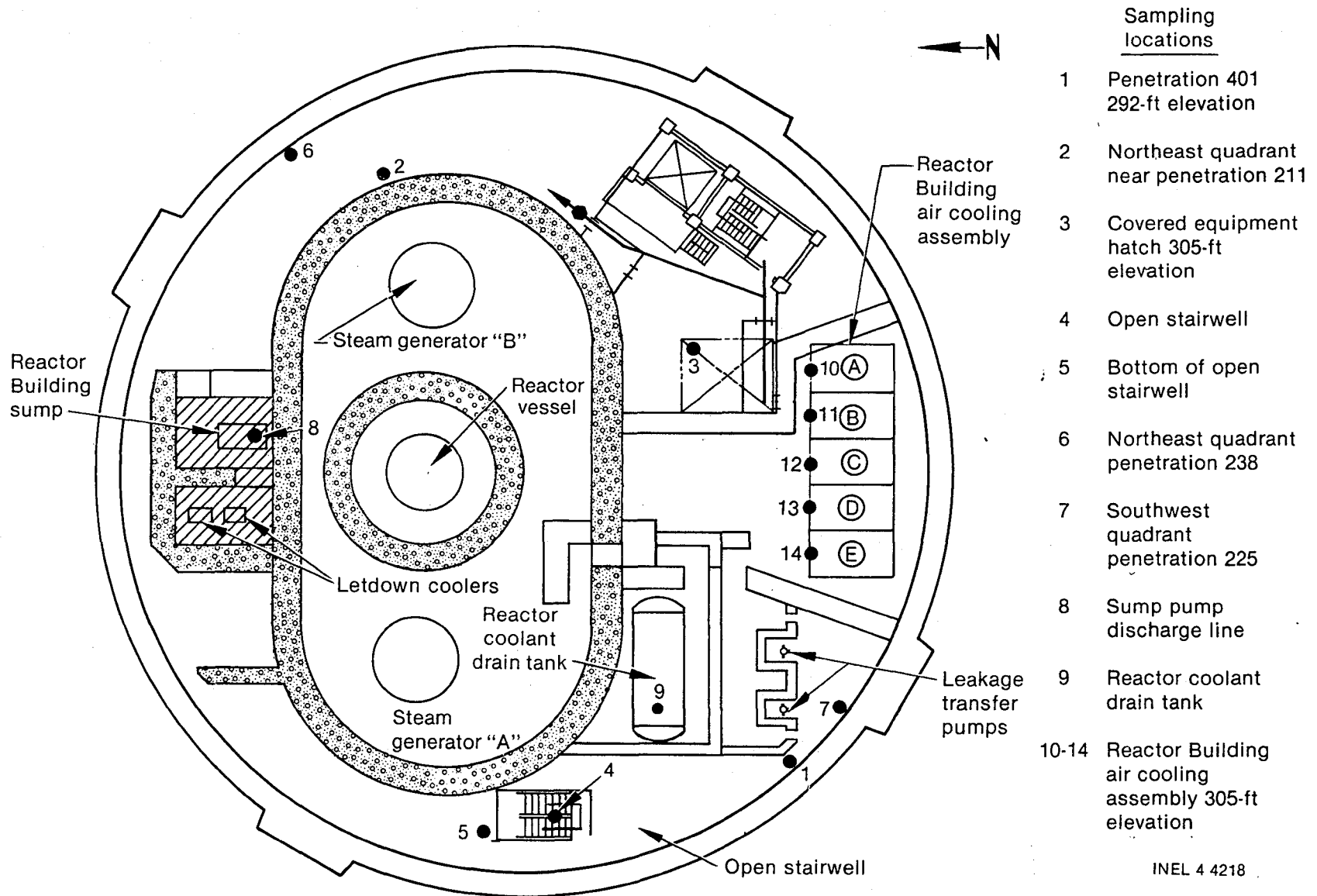


Figure 3. Reactor Building basement samples were obtained from nine locations, and an access panel was removed from each air cooling assembly unit.

collection of four pairs of samples at varying heights above the basement floor. In each case, the sampler was manually lowered into the water using extension poles until the bottom of the sampler rested on the basement floor. A downward push on the extension pole caused hypodermic needles to puncture the septums on the evacuated glass vials, so that water would be drawn into the bottles. When the pole was released, the bottles retracted and the self-sealing septums prevented any loss of sample material. The multilevel sampler was used on May 14, 1981 to collect a total of eight samples from three depths in the water and from the basement floor directly beneath the covered hatch on the 305-ft elevation. The single-level version of the sampler was used on September 24, 1981 to obtain a sample of water and sediment from the floor near the center of the open stairwell. The open stairwell, like penetration 401, is in the vicinity of the RCDT. This single-level sampler was also used on December 5 and 12, 1983 to collect samples of liquid and particulate matter from the bottom inside surface of the RCDT, directly beneath the rupture disk and vertical section of the rupture line.

The three samplings of basement sediment performed on January 11, 1983 utilized the electrically operated, vacuum-actuated sampler developed by EG&G Idaho. This sampler is operated by electrically opening a solenoid valve using a 24-V power source; the sample is drawn into a 50-cc stainless steel chamber by vacuum. Using this sampler, samples of sediment and water were collected from the northeast, southeast, and southwest quadrants of the building. The sample of sediment that was obtained from the floor near the bottom of the open stairwell on June 23, 1982 was collected manually using a metal scoop.

The Reactor Building sump was sampled on August 22, 1983 using a sampling point on the sump pump discharge line. Two samples were collected in 300-cc stainless steel sample containers. The first sample was obtained when the sampling line was first opened for purging, and the second was collected after the line had been purged and the sump water had been recirculated for about 20 min. The latter sample is considered representative of the water and solids present in the sump.

REACTOR BUILDING SURFACE SAMPLES

Structural Surfaces

To measure the effectiveness of the Reactor Building gross decontamination experiment performed in March 1982, 85 surface samples were collected from Reactor Building structural surfaces in December 1981, and an additional 95 surface samples were obtained from the same surfaces in late-March 1982 following the completion of the decontamination experiment. The samples were collected using a milling tool that was designed to allow sampling over a range of depths. Paint shavings and concrete dust or metal shavings generated during milling were, in each case, swept from the surface being sampled and collected on a sample collection filter. Detailed descriptions of the sampler and the sampling procedure used may be found in Reference 1.

Before and after the decontamination experiment, a number of floor locations were sampled several times at different depths to determine if the amount of activity collected increased with sampling depth. However, because surface activity was found to be nonuniform over even small areas, the results regarding activity penetration were ambiguous. These data made it apparent that a different sampling technique was required to determine if activity penetration had occurred.

The alternate sampling technique eventually chosen was core boring. From September 16 to September 26, 1983, 17 high-quality concrete cores were extracted from the floors and D-ring walls in the Reactor Building. The cores typically were 4.5 cm in diameter and 5.5 cm long. A detailed description of the core boring tool and the sampling procedure used may be found in Reference 7. The results of the individual autoradiography, gamma spectroscopy, and radiochemical analyses that were performed on the core samples are presented in Reference 2.

Air Cooling Assembly Surfaces

To better understand the fission product transport mechanisms that were in effect in the Reactor Building atmosphere, surface samples were

obtained from the five Reactor Building air coolers. A variety of samples were collected and measurements made, including particulate scrapes and smears from various locations, metal coupons removed from the access panels, and in situ gamma scans of the cooling coils and drip pans.

The Reactor Building air cooling assembly consists of five air cooling units connected to a common system of duct work for air distribution (see Figure 4). Each cooling unit consists of finned, water type cooling coils and an axial flow fan. All five units are assembled in a common metal housing with suitable divider plates and backdraft dampers.

The air cooling assembly housing is primarily mild carbon steel with a 10-mil nuclear grade epoxy base coating of Phenoline 368. The assembly measures approximately 45 ft by 13 ft and is 32 ft high. The total surface area of the assembly exposed to the Reactor Building atmosphere is approximately $9.7 \text{ E}+6 \text{ cm}^2$ ($10,400 \text{ ft}^2$).⁸ Associated ventilation duct work has an exposed surface area of approximately $2.3 \text{ E}+7 \text{ cm}^2$ ($24,860 \text{ ft}^2$). The duct work is mostly mild carbon steel with a 6-mil coat of Phenoline 368.

Each of the five cooling units consists of coil assemblies that are 18 by 12 rows of copper nickel tubes, each having an outside diameter of 5/8 in. The tubes are 54 in. long, bisecting six plate fins per inch. The plate fins are composed of 16-gauge steel and have dimensions of approximately 20 in. by 27 in. There are six coil assemblies per air cooler unit. Airflow is downward through the coil sections. The calculated surface area of the air cooler fin plates, support plates, and coolant tubes is $1.35 \text{ E}+7 \text{ cm}^2$ ($14,500 \text{ ft}^2$) per air cooler unit (see Appendix A).

The air cooler drip pans are located at the base of the common plenum. The drip pans provide a catch basin for condensation generated on the cooler coils during loss-of-coolant accident conditions. The surface area of the air cooler drip pans is approximately $8.2 \text{ E}+4 \text{ cm}^2$ (88 ft^2) per air cooler. Table 2 summarizes the surface areas of components associated with the Reactor Building air coolers.

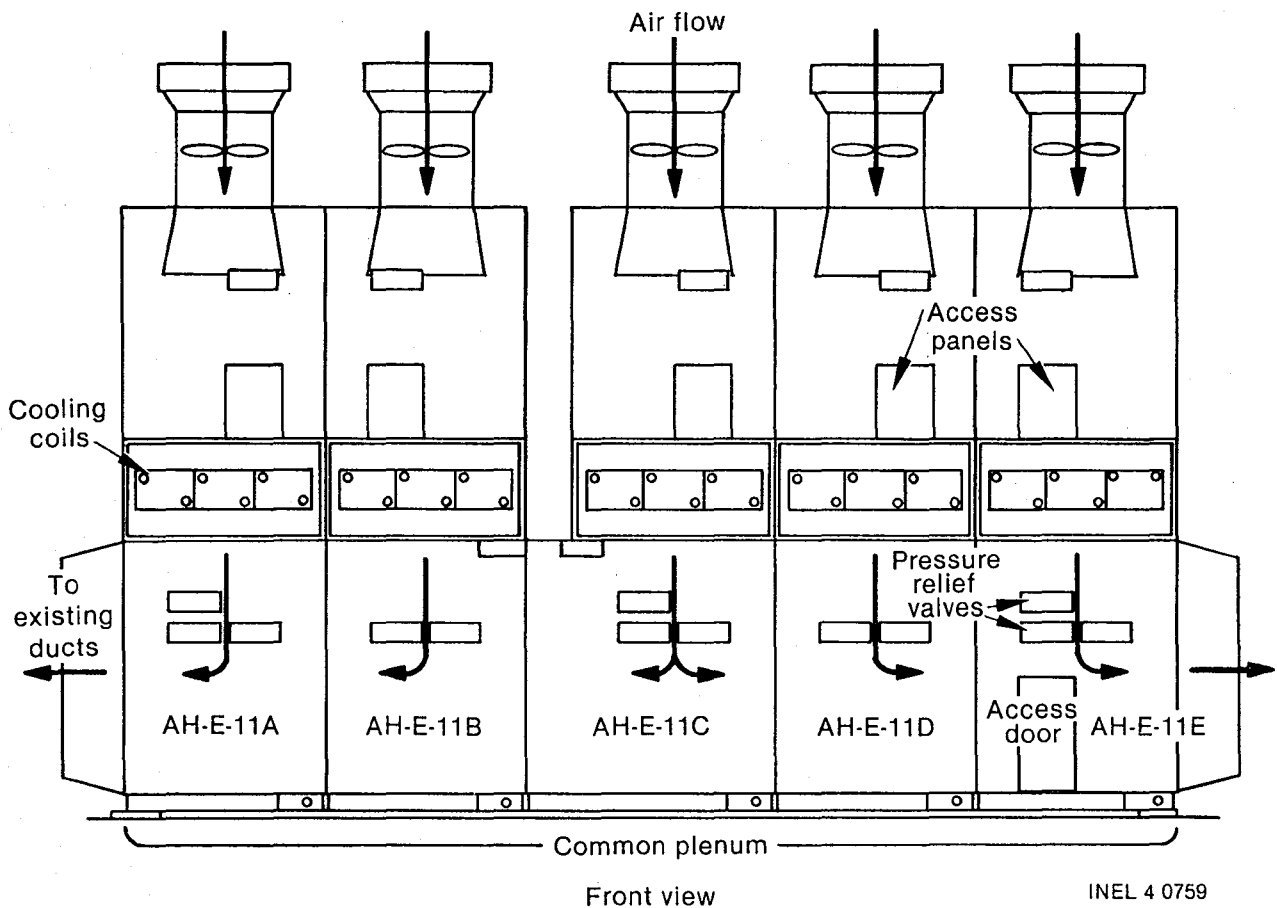
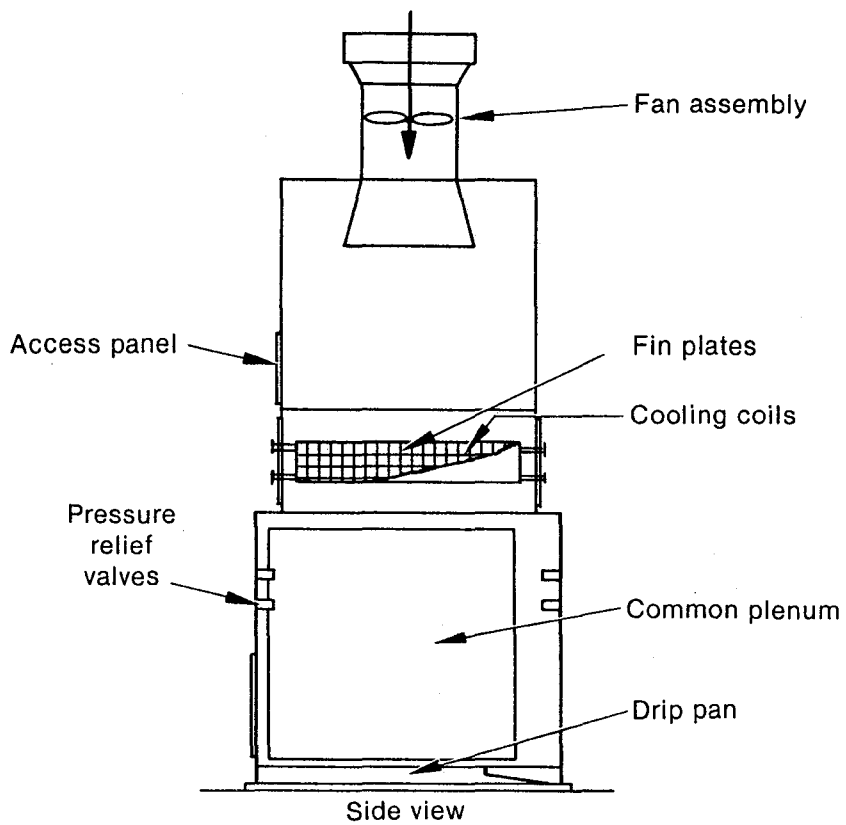


Figure 4. Reactor Building air cooling assembly.

TABLE 2. SURFACE AREAS OF REACTOR BUILDING AIR COOLER COMPONENTS

| Unit | Surface Area per Air Cooler | | Total Surface Area for Five Air Coolers | |
|---------------|--------------------------------|--------------------|--|--------------------|
| | (ft ²) | (cm ²) | (ft ²) | (cm ²) |
| Housing | -- | -- | 10,400 | 9.7 E+6 |
| Vent ducts | -- | -- | 24,860 | 2.3 E+7 |
| Cooling coils | 14,500 | 1.35 E+7 | 72,500 | 6.75 E+7 |
| Drip pans | 88 | 8.2 E+4 | 440 | 4.10 E+5 |
| Total | -- | -- | 107,760 | 1.00 E+8 |

In October 1981, Science Applications, Inc. personnel performed in situ gamma scans of the 11C, 11D, and 11E cooling coils and drip pans using a mobile germanium spectrometer. The intent of these measurements was to compare the surface activities collected on coolers that had been running since the accident (represented by 11C and 11E) with the surface activities measured on a cooler that was known to be off since June 1979 (represented by 11D). Results of the in situ scans were reported in November 1981.⁹

In April 1983, a scrape sample was obtained from the internal surface of the louvered inlet to the 11A air cooler and analyzed onsite by GPU Nuclear. At about this same time, the panels that allow access to the Reactor Building air coolers were removed for surface analysis. A total of five panels were removed, one from each Reactor Building cooler. The dimensions of the panels are 30 in. by 40 in., and they are located at the 322-ft elevation just above the cooling coils. Smear surveys were performed on the internal surfaces of the 11A, 11B, and 11C access panels prior to shipment. GPU Nuclear and Battelle Pacific Northwest Laboratory (PNL) analyzed the smear samples onsite. In addition, smear and scrape samples were obtained from the 11D and 11E cooler access panels. Initially, a smear sample was taken over an area of approximately 100 cm². Following the collection of the smear sample, a scrape sample was removed from the same area. The smear and scrape samples were analyzed onsite by PNL. The five Reactor Building access panels were eventually shipped to the Idaho National Engineering Laboratory (INEL), where they were subjected to a series of analyses. The results of those analyses are presented in a later section of this report.

DISCUSSION OF ANALYSES RESULTS

Basement Samples

Table 3 presents a summary of the volumes and masses of the Reactor Building basement samples that have been analyzed to date. Four of the eight samples that were collected May 14, 1981 were archived and as yet have not been analyzed. The sample of sediment from near the bottom of the open stairwell on June 23, 1982 was split into two portions that were sent to different laboratories for analysis. The volumes of all of the samples analyzed ranged from 18 to 1050 mL, and the masses of the dried solids that were filtered from these samples using 0.45- μ m filters ranged from less than 1 mg to about 719 mg.

The dry masses of solids in the samples collected from the basement floor on August 28 and November 15, 1979 are presented in Table 3 as estimated values since no direct measurements of their masses were made. Both samples were centrifuged, and in each case only the volume of the centrifuged solids was measured. Consequently, the radiochemical and elemental analyses results for these two solids samples were originally reported as microcuries and micrograms per milliliter of either the total volume of the sample or the volume of centrifuged solids.^{10,11} These same results were later reported on a per-gram basis by assigning a dry density of 1 g/mL to the centrifuged solids (see References 5 and 12). However, measurements made on the two samples collected from the basement floor on May 14, 1981 and June 23, 1982 indicate that the actual dry density of the centrifuged solids was substantially less than 1 g/mL. Solids from both the May and June samples were centrifuged and subsequently dried and weighed. Expressed as dry grams per mL of centrifuged wet solids, the densities measured were, respectively, 0.049 and 0.078 g/mL. The average of these two values, 0.0635 g/mL, was multiplied times the measured volumes of the centrifuged solids of the August and November 1979 samples, which were 1.20 and 5.25 mL, respectively, to obtain the estimates of their masses that are given in Table 3.

The masses of the solids present in the three samples collected in January 1983 were determined, in each case, by passing the sample through

TABLE 3. VOLUMES AND MASSES OF BASEMENT SAMPLES THAT HAVE BEEN ANALYZED

| Sampling Date (m/d/y) | Sample Volume (mL) | Mass of Filtered Solids (mg) | Sampling Date (m/d/y) | Sample Volume (mL) | Mass of Filtered Solids (mg) |
|--------------------------|-----------------------|---------------------------------|--------------------------|-----------------------|---------------------------------|
| 8/28/79 | 30 | --a | 5/14/81 | 110 | 108 |
| 8/28/79 | 30 | --a | 9/24/81 | 120 | 24.6 |
| 8/28/79 | 30 | 76.2 ^b | 6/23/82 | 18 | 389.4 |
| 11/15/79 | 1050 | 333 ^b | 6/23/82 | 27.5 | 718.6 |
| 3/19/81 | 1000 | --a | 1/11/83 | 45 | 2 |
| 3/19/81 | 1000 | --a | 1/11/83 | 55 | <1 |
| 3/19/81 | 1000 | --a | 1/11/83 | 55 | 497 |
| 5/14/81 | 85 | --a | 8/22/83 | 200 | 71.8 |
| 5/14/81 | 105 | --a | 12/05/83 | 120 | --a |
| 5/14/81 | 110 | --a | 12/12/83 | 128 | 9.1 |

a. Not measured. Insufficient solids.

b. Mass was estimated based on volume of centrifuged solids.

two 0.45- μm filters that were sandwiched together. When the filtration was completed, the backup filter was weighed separately to determine the mass of the water absorbed by the leading filter. The mass of the dry filter, which was measured before filtration commenced, and the mass of the absorbed water were then subtracted from the mass of the leading filter following filtration to determine the mass of the particulates retained by the filter. Unfortunately, one of these steps, the subtraction of the mass of absorbed water, was omitted when the analyses results for these three samples were reported by the laboratory that performed the analyses. Consequently, the radiochemical and elemental analyses results previously reported for these January 1983 samples are incorrect (see References 5 and 12). The masses of the solids presented in Table 3--2, <1, and 497 mg--have been corrected for absorbed water and should be considered to supersede the masses previously reported, which were 80, 45, and 583 mg, respectively.

Following their segregation, the liquid and solids portions of the basement samples were separately subjected to various radiochemical, gamma-ray spectroscopy, and elemental analyses. The specific analyses methods that were used on individual samples as well as estimates of the uncertainties in the results may be found in References 3 through 6, 10, and 12 through 17.

Radionuclide and Elemental Concentrations

Because they have not been previously published, the radiochemical and gamma-ray spectroscopy analyses results for the samples collected from the Reactor Building sump and the reactor coolant drain tank (RCDT) are presented separately in Table 4. The quoted uncertainties are for counting statistics only and are given at the two-sigma level. Absolute uncertainties are probably not better than $\pm 5\%$. The activities given in Table 4 are decay-corrected to April 24, 1984, the approximate date the samples were analyzed.

The radiochemical, gamma-ray spectroscopy, and elemental analyses results for the samples that were collected from the Reactor Building basement August 1979 through January 1983 are summarized in Tables 5 and 6,

TABLE 4. RADIOCHEMICAL ANALYSIS RESULTS FOR THE REACTOR BUILDING SUMP AND REACTOR COOLANT DRAIN TANK SAMPLES
(Decay-corrected to April 24, 1984)

| Nuclide | Sump Sample | | Reactor Coolant Drain Tank Sample | |
|-------------------|--------------------------------|---------------------------------|-----------------------------------|---------------------------------|
| | Solids ($\mu\text{Ci/g}$) | Liquid ($\mu\text{Ci/mL}$) | Solids ($\mu\text{Ci/g}$) | Liquid ($\mu\text{Ci/mL}$) |
| ^3H | --a | --a | --b | $3.43 \pm 0.07 \text{ E-2}$ |
| ^{54}Mn | $1.0 \pm 0.2 \text{ E-1}$ | --b | --b | --b |
| ^{60}Co | $2.27 \pm 0.08 \text{ E+0}$ | --b | $3.04 \pm 0.05 \text{ E+1}$ | $1.0 \pm 0.1 \text{ E-3}$ |
| ^{90}Sr | $1.49 \pm 0.06 \text{ E+2}$ | $6.0 \pm 0.3 \text{ E+0}$ | $1.39 \pm 0.07 \text{ E+4}$ | $2.49 \pm 0.04 \text{ E+0}$ |
| ^{106}Ru | --b | --b | $6.1 \pm 0.3 \text{ E+1}$ | --b |
| ^{125}Sb | $1.4 \pm 0.1 \text{ E+0}$ | --b | $1.57 \pm 0.08 \text{ E+1}$ | $5.8 \pm 1.0 \text{ E-3}$ |
| ^{129}I | $<2.5 \text{ E-5}$ | --b | $5.2 \pm 0.4 \text{ E-8}$ | $3.9 \pm 0.2 \text{ E-7}$ |
| ^{134}Cs | $2.83 \pm 0.07 \text{ E+0}$ | $4.92 \pm 0.04 \text{ E+0}$ | $5.7 \pm 0.4 \text{ E+0}$ | $6.84 \pm 0.05 \text{ E-2}$ |
| ^{137}Cs | $5.29 \pm 0.03 \text{ E+1}$ | $9.40 \pm 0.05 \text{ E+1}$ | $9.7 \pm 0.2 \text{ E+1}$ | $1.22 \pm 0.01 \text{ E+0}$ |
| ^{144}Ce | $7.5 \pm 2.3 \text{ E+0}$ | --b | $9.6 \pm 0.2 \text{ E+1}$ | --b |
| | (mg/g) | (mg/mL) | (mg/g) | (mg/mL) |
| ^{235}U | $5.3 \pm 0.4 \text{ E-3}$ | $<3 \text{ E-6}$ | $9.0 \pm 0.5 \text{ E-2}$ | $<3 \text{ E-6}$ |

a. Not measured.

b. Not detected.

TABLE 5. SUMMARY OF ANALYSES RESULTS FOR LIQUID FRACTIONS OF REACTOR BUILDING BASEMENT SAMPLES
(Activities Decay-corrected to Sample Collection Dates)

| Sampling location: | Penetration 401 | | Covered Hatch | Open Stairwell | Open Stairwell | | Covered Hatch | Penetration 238 | Penetration 225 | Sump | Reactor Coolant Drain Tank |
|----------------------------------|-------------------|-------------------|-------------------|-------------------|-------------------|----------------------|----------------------|----------------------|---------------------------------|--|----------------------------|
| Analysis laboratory: | ORNL ^a | ORNL ^b | INEL ^c | INEL ^c | ORNL ^d | WHEDL ^{e,f} | PNL/TMI ^g | PNL/TMI ^g | PNL/TMI ^g | INEL ^h and WHEDL ⁱ | |
| Sample collection date: | 8/28/79 | 11/15/79 | 5/14/81 | 9/24/81 | 6/23/82 | 6/23/82 | 1/11/83 | 1/11/83 | 1/11/83 | 8/22/83 | 12/12/83 |
| Basement water volume (mL): | 1.95 E+9 | 2.01 E+9 | 2.36 E+9 | 2.38 E+9 | 9.99 E+7 | 9.99 E+7 | 1.83 E+8 | 1.83 E+8 | 1.83 E+8(1.03 E+7) ^j | (2.74 E+7) ^k | |
| Nuclide concentrations (μCi/mL): | | | | | | | | | | | |
| ³ H | 1.03 E+0 | 1.04 E+0 | 6.03 E-1 | 5.87 E-1 | --1 | --1 | --1 | --1 | --1 | --1 | 3.5 E-2 |
| ⁹⁰ Sr | 2.81 E+0 | 2.3 E+0 | 5.2 E+0 | 4.8 E+0 | 6.98 E+0 | 5.8 E+0 | 2.49 E+0 | 2.36 E+0 | 3.45 E+0 | 6.1 E+0 | 2.51 E+0 |
| ¹⁰⁶ Ru | 7.0 E-3 | 3.1 E-3 | --m | --m | <3.6 E-2 | 2.6 E-3 | <1.4 E-2 | <3.7 E-2 | <9.4 E-3 | --m | <2.7 E-2 |
| ¹²⁵ Sb | 1.5 E-2 | 2.3 E-2 | 3.0 E-2 | <2 E-2 | <3 E-3 | 2.6 E-2 | <7 E-3 | <2 E-2 | <5 E-3 | --m | 6.4 E-3 |
| ¹²⁹ I | 1.38 E-5 | --1 | 4.3 E-6 | 2 E-5 | <2 E-4 | --1 | 1.08 E-6 | 1.11 E-6 | 4.09 E-7 | --m | 3.9 E-7 |
| ¹³⁴ Cs | 3.99 E+1 | 3.21 E+1 | 1.92 E+1 | 1.62 E+1 | 1.40 E+1 | 1.44 E+1 | 9.8 E-1 | 6.7 E-1 | 6.6 E-1 | 6.16 E+0 | 7.73 E-2 |
| ¹³⁷ Cs | 1.76 E+2 | 1.62 E+2 | 1.43 E+2 | 1.37 E+2 | 1.51 E+2 | 1.59 E+2 | 1.20 E+1 | 8.22 E+0 | 8.26 E+0 | 9.55 E+1 | 1.23 E+0 |
| ¹⁴⁴ Ce | 6.3 E-3 | 8.7 E-4 | --m | <3.5 E-2 | --m | --m | <4.2 E-3 | <1.2 E-2 | <3.0 E-3 | --m | <1.5 E-2 |
| Fuel concentrations (μg/mL): | | | | | | | | | | | |
| U | 1.6 E-2 | --1 | --1 | <3 E-2 | 1.6 E-2 | 5 E-3 | --1 | 5.7 E-2 | --1 | 5.7 E-2 | <1.3 E-1 |
| Pu | 1.8 E-5 | --1 | 2.2 E-4 | --1 | 5.6 E-5 | <1 E-4 | --1 | 5.3 E-8 | --1 | 5.3 E-8 | --1 |
| Element concentrations (μg/mL): | | | | | | | | | | | |
| Li | 1.53 E+0 | --1 | 1.8 E+0 | 1.8 E+0 | --1 | 1.0 E+1 | --1 | --1 | --1 | 2.35 E+0 | --1 |
| B | 2.02 E+3 | 2.0 E+3 | 2.12 E+3 | 2.30 E+3 | ~3 E+3 | 8.0 E+3 | --1 | --1 | --1 | 7.97 E+3 | --1 |
| Na | 1.16 E+3 | 1.2 E+3 | 1.18 E+3 | 1.24 E+3 | ~3 E+3 | 6.0 E+3 | --1 | --1 | --1 | 2.56 E+3 | --1 |
| Mg | <2 E+0 | --1 | 5.4 E+0 | 7.3 E+0 | 5 E+0 | 8.0 E+1 | --1 | --1 | --1 | 6.80 E+0 | --1 |
| Al | 3 E+0 | --1 | 1.4 E+0 | 1.8 E+0 | 3 E+0 | 1.7 E+0 | --1 | --1 | --1 | 3.16 E+0 | --1 |
| Si | --1 | --1 | 4.8 E+0 | 6.8 E+0 | 2.0 E+1 | 9.0 E+1 | --1 | --1 | --1 | 8.22 E+0 | --1 |
| K | 4 E+0 | --1 | 1.6 E+1 | 2.0 E+1 | 7.0 E+1 | 2.0 E+2 | --1 | --1 | --1 | 2.50 E+1 | --1 |
| Ca | 9.3 E+0 | --1 | 3.6 E+1 | 4.1 E+1 | 3.0 E+1 | 2.0 E+1 | --1 | --1 | --1 | 2.66 E+1 | --1 |
| Cr | 7 E-1 | --1 | 2 E-1 | 3 E-1 | 3 E+0 | 2 E+0 | --1 | --1 | --1 | 2.50 E-2 | --1 |
| Mn | <1 E-1 | --1 | <2 E-1 | <2 E-1 | 2 E-1 | --m | --1 | --1 | --1 | 1.42 E+0 | --1 |
| Fe | 1.2 E+0 | --1 | 9 E-1 | 7 E-1 | 6 E-1 | 3 E+0 | --1 | --1 | --1 | 1.36 E+0 | --1 |
| Co | <1 E-1 | --1 | <5 E+0 | <5 E+0 | 1 E-1 | --m | --1 | --1 | --1 | 2.90 E-2 | --1 |
| Ni | <1 E+0 | --1 | <1 E+0 | 1.1 E+0 | 5 E-1 | --m | --1 | --1 | --1 | 6.63 E-1 | --1 |

TABLE 5. (continued)

| Sampling location: | Penetration 401 | | Covered Hatch | | Open Stairwell | | Open Stairwell | | Covered Hatch | Penetration 238 | Penetration 225 | Sump | Reactor Coolant Drain Tank | | |
|---|-------------------|-------------------|-------------------|----|-------------------|-----|-------------------|----------------------|----------------------|----------------------|----------------------|--|----------------------------|----------|-----|
| Analysis laboratory: | ORNL ^a | ORNL ^b | INEL ^c | | INEL ^c | | ORNL ^d | WHEDL ^{e,f} | PNL/TMI ^g | PNL/TMI ^g | PNL/TMI ^g | INEL ^h and WHEDL ⁱ | | | |
| Element concentrations (µg/mL) (continued): | | | | | | | | | | | | | | | |
| Cu | <3 | E+0 | --1 | <1 | E+0 | <1 | E+0 | 5 | E+0 | 6 | E-1 | --1 | --1 | 9.59 E+0 | --1 |
| Zn | 4.7 | E-1 | --1 | <5 | E+0 | <5 | E+0 | --1 | --m | --1 | --1 | --1 | 5.57 E-1 | --1 | |
| Zr | --1 | --1 | --1 | <1 | E+0 | 1.4 | E+0 | <4 | E-1 | --m | --1 | --1 | <5 | E-2 | |
| Ag | <3 | E-1 | --1 | <1 | E+0 | <1 | E+0 | 1 | E-1 | 4 | E-2 | --1 | --1 | 1.30 E-1 | |
| Cd | <3 | E-1 | --1 | <2 | E+0 | <2 | E+0 | 2 | E+0 | 1.7 | E-1 | --1 | --1 | 2.08 E-1 | |
| In | <1 | E-1 | --1 | <5 | E+0 | <5 | E+0 | 1 | E+0 | 8 | E-2 | --1 | --1 | <5 | |
| Sn | --1 | --1 | --1 | <5 | E+0 | <5 | E+0 | <2 | E+0 | 2 | E-2 | --1 | --1 | 3.46 E-1 | |
| Gd | --1 | --1 | --1 | <1 | E+0 | <1 | E+0 | --1 | --m | --1 | --1 | --1 | --1 | <1 | |

- a. W. D. Shults, letter from Oak Ridge National Laboratory to J. A. Daniel, GPU Service Corp., September 14, 1979.
- b. D. O. Campbell, et al., Evaluation of the Submerged Demineralizer System (SDS) Flowsheet for Decontamination of High-Activity-Level Water at the Three Mile Island Unit 2 Nuclear Power Station, ORNL/TM-7448, July 1980.
- c. T. E. Cox, et al., Reactor Building Basement Radionuclide Distribution Studies, GEND-INF-011, Vol. II, October 1982.
- d. E. H. Gischel, letter from GPU Nuclear to Distribution, February 8, 1983.
- e. J. J. McCown, letter from Westinghouse Hanford Engineering Development Laboratory to J. T. Horan, EG&G Idaho, Inc., September 10, 1982.
- f. J. J. McCown, letter from Westinghouse Hanford Engineering Development Laboratory to J. T. Horan, EG&G Idaho, Inc., November 8, 1982.
- g. C. W. Thomas, private communication, Pacific Northwest Laboratory/TMI Mobile Laboratory, June 1983.
- h. D. W. Akers, letter from EG&G Idaho, Inc., to C. V. McIsaac, EG&G Idaho, Inc., June 7, 1984.
- i. J. J. McCown, letter from Westinghouse Hanford Engineering Development Laboratory to J. T. Horan, EG&G Idaho, Inc., April 2, 1984.
- j. Volume of the Reactor Building sump.
- k. Volume of the reactor coolant drain tank.
- l. Not measured.
- m. Not detected.

TABLE 6. SUMMARY OF ANALYSES RESULTS FOR SOLIDS FRACTIONS OF REACTOR BUILDING BASEMENT SAMPLES
(Activities Decay-corrected to Sample Collection Dates)

| Sampling location: | Penetration 401 | | Covered Hatch | Open Stairwell | Open Stairwell | | Covered Hatch | Penetration 238 | Penetration 225 | Sump | Reactor Coolant Drain Tank |
|---------------------------------|-------------------|-------------------|-------------------|-------------------|-------------------|----------------------|----------------------|----------------------|----------------------|--|----------------------------|
| Analysis laboratory: | ORNL ^a | ORNL ^b | INEL ^c | INEL ^c | ORNL ^d | WHEDL ^{e,f} | PNL/TMI ^g | PNL/TMI ^g | PNL/TMI ^g | INEL ^h and WHEDL ⁱ | |
| Sample collection date: | 8/28/79 | 11/15/79 | 5/14/81 | 9/24/81 | 6/23/82 | 6/23/82 | 1/11/83 | 1/11/83 | 1/11/83 | 8/22/83 | 12/12/83 |
| Mass of basement solids (g): | 3.76 E+5 | 3.76 E+5 | 3.76 E+5 | 3.76 E+5 | 3.76 E+5 | 3.76 E+5 | 3.76 E+5 | 3.76 E+5 | 3.76 E+5 | (3.70 E+3) ^j | (2.6 E+4) ^k |
| Nuclide concentrations (μCi/g): | | | | | | | | | | | |
| 60Co | 5.2 E-1 | 1.3 E+0 | 1.21 E+1 | 2 E+1 | 9.6 E+0 | 1.17 E+1 | <8.0 E-1 | >3.3 E+1 | 1.47 E+0 | 2.48 E+0 | 3.19 E+1 |
| 90Sr | 8.2 E+1 | 6.1 E+2 | 8 E+2 | 2.2 E+3 | 2.38 E+3 | 4.9 E+3 | 1.2 E+3 | >5.7 E+2 | 8.38 E+1 | 1.51 E+2 | 1.40 E+4 |
| 106Ru | 1.7 E+1 | 1.8 E+1 | 1.04 E+2 | 5.8 E+1 | -- ^l | 4.1 E+1 | <1.2 E+1 | >5.9 E+1 | 2.72 E+0 | -- ^l | 7.8 E+1 |
| 125Sb | 1.1 E+1 | 2.8 E+1 | 4.87 E+2 | 1.2 E+1 | 1.33 E+2 | 1.42 E+2 | 1.7 E+1 | >3.6 E+2 | 1.13 E+1 | 1.7 E+0 | 1.72 E+1 |
| 129I | 4.8 E-3 | -- ^m | 1.1 E-1 | 2.7 E-3 | 1.8 E-1 | -- ^m | 1.6 E-2 | -- ^m | 1.97 E-3 | <2.5 E-5 | 5.2 E-8 |
| 134Cs | 5.7 E+0 | 1.5 E+1 | 1.07 E+2 | 3.9 E+1 | 7.43 E+1 | 1.84 E+2 | 3.2 E+2 | >4.2 E+1 | 2.30 E+0 | 3.55 E+0 | 6.4 E+0 |
| 137Cs | 2.5 E+1 | 7.6 E+1 | 8.08 E+2 | 3.24 E+2 | 8.02 E+2 | 2.04 E+3 | 3.6 E+3 | >5.4 E+2 | 2.99 E+1 | 5.37 E+1 | 9.8 E+1 |
| 144Ce | 1.2 E+1 | 3.9 E+1 | 6.6 E+1 | 9.4 E+1 | -- ^l | 5.2 E+1 | 2.6 E+1 | >1.3 E+1 | 2.03 E+0 | 1.4 E+1 | 1.3 E+2 |
| Fuel concentrations: | | | | | | | | | | | |
| U (mg/g) | 4.3 E-2 | 3.2 E-1 | 4.0 E+0 | 3.9 E-1 | 2.97 E+0 | 3.9 E+0 | 2.2 E+0 | -- ^m | 3.0 E+0 | 1.8 E-1 | 3.7 E+0 |
| 235Pu (μg/g) | 6.3 E-2 | -- ^m | 2.9 E+0 | -- ^m | 4.41 E+0 | 6.1 E+0 | 3.8 E+0 | >4.5 E-1 | 5.2 E-1 | 1.3 E+0 | -- ^m |
| 235U (at.%) | 2.35 E+0 | -- ^m | 2.6 E+0 | <4 E+0 | 2.37 E+0 | 2.4 E+0 | 2.74 E+0 | -- ^m | -- ^m | -- ^m | -- ^m |
| Element concentrations (μg/g): | | | | | | | | | | | |
| Li | <1.2 E+2 | -- ^l | -- ^l | -- ^l | -- ^l | 5 E+0 | -- ^m | -- ^m | -- ^m | 1.09 E+1 | 1.58 E+2 |
| B | 1.2 E+3 | -- ^m | 4 E+4 | 1.4 E+5 | ~3 E+3 | 1 E+4 | -- ^m | -- ^m | -- ^m | 3.22 E+4 | 9.01 E+4 |
| Na | <4 E+2 | -- ^l | 1.9 E+3 | -- ^m | >3 E+3 | 3 E+2 | -- ^m | -- ^m | -- ^m | 2.95 E+2 | 1.34 E+4 |
| Mg | 2.7 E+3 | 2.3 E+3 | 2 E+3 | 4 E+3 | >2 E+3 | 5 E+3 | -- ^m | -- ^m | -- ^m | 9.92 E+1 | 8.48 E+3 |
| Al | 3.2 E+3 | 2.3 E+4 | 1.1 E+4 | 5 E+4 | >5 E+3 | 1.2 E+4 | -- ^m | -- ^m | -- ^m | 1.11 E+4 | 1.88 E+4 |
| Si | -- ^m | 1.1 E+4 | 7 E+4 | 3 E+4 | >2 E+4 | 7 E+3 | -- ^m | -- ^m | -- ^m | 4.86 E+2 | 2.60 E+3 |
| K | 4.0 E+2 | 2.3 E+2 | 1.7 E+3 | -- ^m | 8 E+2 | 5 E+1 | -- ^m | -- ^m | -- ^m | 1.00 E+4 | 1.06 E+5 |
| Ca | 7.8 E+2 | 7.1 E+3 | 2 E+4 | 4 E+4 | 2 E+3 | 3 E+3 | -- ^m | -- ^m | -- ^m | 2.30 E+2 | 8.17 E+4 |
| Cr | 7.8 E+2 | 4.0 E+2 | 1 E+4 | 3 E+4 | 4 E+2 | 1 E+2 | <6.7 E+2 | <1.5 E+2 | <4.4 E+2 | 7.58 E+1 | 2.72 E+3 |
| Mn | 4.0 E+2 | 1.5 E+3 | 2.4 E+3 | 2 E+4 | 1 E+3 | 5 E+1 | 1.6 E+3 | <1 E+2 | 1.4 E+3 | 6.25 E+1 | 6.40 E+3 |
| Fe | 4.0 E+3 | 1.4 E+4 | 1.5 E+5 | 1.2 E+5 | >8 E+3 | 3 E+3 | 5.4 E+3 | 3.9 E+3 | 4.8 E+3 | 9.88 E+3 | 1.14 E+5 |
| Co | <4 E+1 | 5.5 E+1 | -- ^l | -- ^l | 1 E+1 | 2 E+1 | -- ^m | -- ^m | -- ^m | 6.42 E+0 | 2.62 E+2 |
| Ni | 4.0 E+3 | 4.0 E+4 | 3 E+4 | 2.5 E+4 | >1.2 E+3 | 8 E+3 | <2 E+2 | <4 E+1 | 2.2 E+3 | 1.51 E+2 | 1.11 E+5 |

TABLE 6. (continued)

| Sampling location: Analysis laboratory: | Penetration 401 | | Covered Hatch | Open Stairwell | Open Stairwell | | Covered Hatch | Penetration 238 | Penetration 225 | Sump | Reactor Coolant Drain Tank |
|--|-------------------|-------------------|-------------------|-------------------|-------------------|----------------------|----------------------|----------------------|----------------------|--|----------------------------------|
| | ORNL ^a | ORNL ^b | INEL ^c | INEL ^c | ORNL ^d | WHEDL ^{e,f} | PNL/TMI ^g | PNL/TMI ^g | PNL/TMI ^g | INEL ^h and WHEDL ⁱ | |
| Element concentrations ($\mu\text{g/g}$) (continued): | | | | | | | | | | | |
| Cu | 2.2 E+4 | 1.2 E+5 | 1.0 E+5 | 3 E+3 | >4 E+4 | 4.9 E+4 | 1.7 E+3 | 3.1 E+2 | 1.3 E+4 | 1.42 E+2 | 3.22 E+5 |
| Zn | 7.8 E+2 | 6.3 E+3 | 1.8 E+4 | --m | >2 E+3 | 2 E+2 | 3.5 E+2 | <2 E+3 | <6 E+2 | 1.21 E+2 | 2.47 E+4 |
| Zr | --l | 1.1 E+2 | --l | --l | >2 E+3 | 2 E+2 | 1.2 E+2 | 6.5 E+1 | 5.6 E+2 | 2.03 E+2 | 1.89 E+3 |
| Ag | 3.2 E+3 | 8.6 E+2 | 1.6 E+4 | --m | >5 E+3 | 2.5 E+4 | --m | --m | --m | 2.94 E+2 | 6.29 E+3 |
| Cd | <2 E+2 | 8.6 E+2 | 6.4 E+3 | --m | >5 E+3 | 5 E+3 | --m | --m | --m | 1.47 E+2 | 1.19 E+4 |
| In | 1.2 E+2 | 4.7 E+2 | 1 E+4 | --m | 1.5 E+3 | 3 E+3 | --m | --m | --m | <7 E+1 | 6.20 E+3 |
| Sn | --l | 6.3 E+2 | --l | --l | 1 E+3 | 1.4 E+3 | --m | --m | --m | <4 E+1 | 4.96 E+3 |
| Gd | --l | --l | --l | --l | --l | --l | --m | --m | --m | <1 E+1 | 1.95 E+2 |
| Total weight percent: | 4.6 | 23.1 | 49.3 | 46.2 | 10.6 | 13.6 | 1.0 | 0.7 | 2.6 | 6.6 | 93.8 |

a. W. D. Shults, letter from Oak Ridge National Laboratory to J. A. Daniel, GPU Service Corp., September 14, 1979.

b. D. O. Campbell, et al., Evaluation of the Submerged Demineralizer System (SDS) Flowsheet for Decontamination of High-Activity-Level Water at the Three Mile Island Unit 2 Nuclear Power Station, ORNL/TM-7448, July 1980.

c. T. E. Cox, et al., Reactor Building Basement Radionuclide Distribution Studies, GEND-INF-011, Vol. II, October 1982.

d. E. H. Gischel, letter from GPU Nuclear to Distribution, February 8, 1983.

e. J. J. McCown, letter from Westinghouse Hanford Engineering Development Laboratory to J. T. Horan, EG&G Idaho, Inc., September 10, 1982.

f. J. J. McCown, letter from Westinghouse Hanford Engineering Development Laboratory to J. T. Horan, EG&G Idaho, Inc., November 8, 1982.

g. C. W. Thomas, private communication, Pacific Northwest Laboratory/TMI Mobile Laboratory, June 1983.

h. D. W. Akers, letter from EG&G Idaho, Inc., to C. V. McIsaac, EG&G Idaho, Inc., June 7, 1984.

i. J. J. McCown, letter from Westinghouse Hanford Engineering Development Laboratory to J. T. Horan, EG&G Idaho, Inc., April 2, 1984.

j. Measured mass of solids in solution in the Reactor Building sump following sump water recirculation for 20 min.

k. Estimated mass of solids in the reactor coolant drain tank.

l. Not detected.

m. Not measured.

along with similar results for the sump and RCDT samples. The results for the liquid portions of the samples are given in Table 5, and those for the solids portions are given in Table 6. Samples are identified in those tables by sampling location, sample collection date, and by the laboratory that performed the analyses. Laboratories that performed the analyses were the Oak Ridge National Laboratory (ORNL), the INEL, the Westinghouse Hanford Engineering Development Laboratory (WHEDL), and the PNL mobile emergency response laboratory that is stationed at TMI-2. Also listed for each sample are the estimated volume of water and estimated mass of sediment in the basement on the date the sampling occurred. The radionuclide concentrations presented in Tables 5 and 6 are decay-corrected to the sample collection dates.

The concentrations presented in Table 5 for the liquid portions of the samples collected August 28, 1979 and May 14, 1981 are averages of the concentrations measured in the several samples collected on each date. As was mentioned previously, three samples were obtained on August 28 and eight samples were collected on May 14 at varying heights above the basement floor. On both sampling dates, the concentrations of soluble fission products and stable elements (e.g. ^3H , ^{90}Sr , ^{137}Cs , ^{131}I , B, and Na) were found to be essentially constant over the range of depths sampled. In other words, no significant stratification of soluble species was observed.

The analyses results for the three water samples collected on March 19, 1981 from beneath core flood tank A are not presented in Table 5 because only a few of the results had been uncovered at the time of this writing. The concentrations of ^{134}Cs , ^{137}Cs , and ^{129}I were reported as 19.2 ± 0.2 , 138 ± 3 , and 5.5×10^{-6} $\mu\text{Ci/mL}$, respectively, decay-corrected to the day the sampling took place.^{18,19} The concentrations of these three radionuclides in the water beneath core flood tank A are essentially identical to their respective concentrations in the samples collected May 14, 1981 from beneath the covered hatch. Since these two sampling locations are on opposite sides of D-ring B, the good agreement suggests that soluble radionuclides were homogeneously dispersed in the water throughout large regions of the basement.

During the two years prior to the start of basement water processing on September 23, 1981, the water level in the basement increased at a nearly constant rate due to leakage from the reactor coolant system (RCS) and the river water cooling system. Excluding the losses from the RCS that occurred during the first few days following the accident, the volumes of water that leaked from these two systems are estimated to be about equal (6.74×10^5 L from the RCS and 6.81×10^5 L from the river water cooling system). The changes in the concentrations of radionuclides and stable elements in the basement water during the sampling period August 28, 1979 through September 24, 1981 reflect not only the cumulative effect of water being added simultaneously from those two systems but also the net effect of ongoing processes such as radioactive decay, precipitation, dissolution, condensation, and volatilization.

By the time the basement water was first sampled on August 28, 1979, the concentrations of ^3H , ^{134}Cs , and ^{137}Cs in the RCS water that was leaking to the basement had decreased to values substantially lower than their respective concentrations in the basement water. On the other hand, the concentration of ^{90}Sr in the RCS water being added to the basement remained at least a factor of 2 higher than its concentration in the basement water throughout the 1979 to 1981 sampling period. As a result, the effect of RCS leakage during this time was to decrease the concentrations of ^3H , ^{134}Cs , and ^{137}Cs and increase the concentration of ^{90}Sr in the basement water. The concentrations of ^3H and ^{137}Cs in the basement water decreased from initial values of 1 and 176 $\mu\text{Ci/mL}$, respectively, on August 28, 1979, to about 0.6 and 137 $\mu\text{Ci/mL}$ on September 24, 1981. After being corrected for radioactive decay, the concentrations of ^3H and ^{137}Cs decreased by 36 and 18%, respectively. During this same time period, the concentration of ^{90}Sr increased from 2.8 to about 5 $\mu\text{Ci/mL}$.

The concentration of boron in the RCS coolant fluctuated between 2900 and 4300 $\mu\text{g/mL}$ during this same time frame, and its time-weighted average concentration was about 3600 $\mu\text{g/mL}$, which is approximately a factor of 2 higher than its mean concentration in the basement water. The data presented in Table 5 show that the concentrations of B and Na in the

basement water remained essentially constant throughout the 19 months prior to the start of basement water processing through the Submerged Demineralizer System (SDS). Their mean concentrations in the basement water during this period were $2100 \mu\text{g/mL} \pm 7\%$ and $1200 \mu\text{g/mL} \pm 2.5\%$, respectively, where the uncertainties are standard deviations at the one-sigma level. Since the river water that leaked from the air cooling assembly entered the basement on the east side of the building and contained only trace quantities of boron and the RCS coolant entered on the west side outside the RCDT cubicle, the constancy of the concentrations of B and Na at several different sampling locations is additional evidence that soluble elements and compounds were uniformly dispersed in the basement water.

An abridged summary of the concentrations of B, Na, ^3H , ^{90}Sr , ^{131}I , and ^{137}Cs measured in reactor coolant samples collected before and during the 1979 to 1981 basement sampling periods is provided in Table 7. The half-life of ^{131}I is so short (8.04 days) that it was not detected in samples of RCS water collected after July 1979. It was, however, measured in the samples of water and sediment collected from the basement on August 28, 1979. Its mean concentration in the samples of water was $1.23 \pm 0.06 \times 10^{-2} \mu\text{Ci/mL}$, and its concentration in the samples of sediment was $5.3 \pm 1.4 \mu\text{Ci/g}$ as of 0800 August 28, 1979. Thereafter, only the concentration of ^{129}I was measured in RCS and basement water samples.

It is evident from an examination of the data presented in Table 5 that the concentrations of dissolved ^{129}I measured in the samples collected from beneath core flood tank A on March 19, 1981 and from beneath the covered hatch on May 14, 1981 are factors of 3 to 5 lower than the concentrations of dissolved ^{129}I measured in the water samples collected through penetration 401 on August 28, 1979 and from the floor beneath the open stairwell on September 24, 1981. The concentrations of dissolved ^{129}I at these four locations were measured to be 5.5×10^{-6} , 4.3×10^{-6} , 1.38×10^{-5} , and $2 \times 10^{-5} \mu\text{Ci/mL}$, respectively. It should be noted that the concentrations given for the first three locations are, in each case, averages of the concentrations measured in three or more

TABLE 7. CONCENTRATIONS MEASURED IN REACTOR COOLANT SAMPLES^{a,b}
 (Activities Decay-corrected to Sample Collection Dates)

| Sampling Date (m/d/y) | pH | B ($\mu\text{g/mL}$) | Na ($\mu\text{g/mL}$) | ^3H ($\mu\text{Ci/mL}$) | ^{90}Sr ($\mu\text{Ci/mL}$) | ^{131}I ($\mu\text{Ci/mL}$) | ^{137}Cs ($\mu\text{Ci/mL}$) |
|--------------------------|-----|---------------------------|----------------------------|---------------------------------------|---|---|--|
| 3/29/79 | -- | -- | -- | 1.50 | 0.39 | 14,500 | 275 |
| 4/10/79 | 8.0 | 3408 | -- | -- | 13 | 9,500 | 315 |
| 4/17/79 | 8.2 | 3553 | -- | -- | 32.1 | 4,800 | 270 |
| 5/18/79 | 7.6 | 2870 | -- | 0.52 | 44.1 | 210 | 190 |
| 6/19/79 | 7.6 | 3295 | 330 | 0.35 | 16.1 | 7.7 | 110 |
| 7/17/79 | 7.8 | 3370 | 550 | 0.27 | 17.7 | 0.55 | 94 |
| 8/29/79 ^c | 8.0 | 3430 | 750 | 0.26 | 17.2 | ND ^d | 73 |
| 9/17/79 | 8.0 | 3640 | 930 | 0.19 | 25.7 | -- | 74 |
| 10/15/79 | 8.0 | 3800 | 1360 | 0.25 | 27.0 | -- | 69 |
| 11/15/79 ^c | 7.8 | 3850 | 1470 | 0.21 | 24 | -- | 59 |
| 5/19/80 | 7.8 | 4100 | 1500 | 0.09 | 28 | -- | 40 |
| 11/17/80 | 7.6 | 3770 | 910 | 0.08 | 20 | -- | 26 |
| 2/23/81 | 7.6 | 3820 | 850 | 0.063 | 25 | -- | 25 |
| 5/11/81 ^c | 7.6 | 3850 | 1060 | 0.061 | 23 | -- | 22 |
| 6/22/81 | 7.6 | 3908 | 960 | 0.075 | 19 | -- | 19 |
| 7/20/81 | 7.6 | 3780 | 1010 | 0.050 | 13 | -- | 19 |
| 8/24/81 | 7.6 | 3699 | 943 | 0.04 | 19 | -- | 17 |
| 9/21/81 ^c | 7.6 | 3754 | 1163 | 0.04 | 19 | -- | 17 |

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a. Science Applications, Inc., Iodine-131 Behavior During the TMI-2 Accident, NSAC-30, September 1981.

b. TMI-2 Technical Planning Department, GPU Nuclear-Bechtel National, Inc. Reactor Coolant System Sample Results, TPO/TMI-122, Rev. 0, July 1984.

c. Date is on or near a date when samples were collected from the basement water.

d. Not detected.

samples, whereas the concentration cited for the fourth location, the open stairwell, is the result of a measurement made on a single sample collected from the elevation of the floor. If for the moment we ignore the result for the open stairwell, the data indicate that the concentration of dissolved ^{129}I in the basement water decreased from $1.38 \times 10^{-5} \mu\text{Ci/mL}$ in August 1979 to about $5 \times 10^{-6} \mu\text{Ci/mL}$ by March 1981.

The concentrations of ^{129}I measured in RCS samples collected on March 29, 1979 and August 14, 1980 were 7.4×10^{-5} and $7.1 \times 10^{-6} \mu\text{Ci/mL}$, respectively.²⁰ The concentrations of dissolved ^{129}I in the filtrates of these two samples were 5.3×10^{-6} and $3.4 \times 10^{-6} \mu\text{Ci/mL}$, respectively (see Reference 20). The concentration of dissolved ^{129}I was also measured in a sample of water collected from the RCDT in December 1983. Its concentration was measured to be $3.9 \times 10^{-7} \mu\text{Ci/mL}$. This latter concentration is probably representative of the concentration of dissolved ^{129}I in the RCS water from December 1982 to July 1983, the last period during which the RCS leaked water to the RCDT before RCDT sampling in December 1983.

The quoted concentrations of ^{129}I in the RCS water during and following the accident are sufficiently high to account for the concentration of $1.38 \times 10^{-5} \mu\text{Ci/mL}$ that was measured in the samples collected from the basement on August 28, 1979. Using a concentration of $7.4 \times 10^{-5} \mu\text{Ci/mL}$ for the 1×10^6 L of coolant that escaped during the first three days and a concentration of $7.1 \times 10^{-6} \mu\text{Ci/mL}$ for the estimated 1×10^5 L that subsequently escaped to the basement April 1 to August 28, 1979, the upper limit of the concentration of ^{129}I in the basement water on August 28, 1979 was calculated to be $3.8 \times 10^{-5} \mu\text{Ci/mL}$. A lower limit of $3 \times 10^{-6} \mu\text{Ci/mL}$ was calculated using the concentrations of dissolved ^{129}I in the two RCS samples. However, the decrease in concentration from $1.38 \times 10^{-5} \mu\text{Ci/mL}$ on August 28, 1979 to $4.3 \times 10^{-6} \mu\text{Ci/mL}$ on May 14, 1981 cannot be accounted for by dilution alone because the volume of water in the basement increased by only 24% during the period between these dates. Apparently either a significant amount of ^{129}I precipitated from solution during the 21 months between samplings, or the concentration of ^{129}I measured on August 28 was not indicative of its average concentration throughout the basement.

The concentration of ^{129}I in the filtrate of the sample of water and sediment collected from beneath the open stairwell on September 24, 1981 was measured to be $2 \pm 1 \times 10^{-5}$ $\mu\text{Ci/mL}$, a factor of about 4 to 5 higher than the concentrations measured several months earlier in the samples obtained from beneath the covered hatch and penetration 211. The higher concentration below the open stairwell might be explained by the fact that it is relatively near the point where water discharged from the RCDDT while the covered hatch and penetration 211 are located in the southeast and northeast quadrants of the Reactor Building (see Figure 3). An alternate explanation is that the filtrate of the open stairwell sample was contaminated by ^{129}I that originally was associated with the particulates in the sample.

The data presented in Table 6 indicate that significant quantities of ^{129}I did precipitate from the water in the basement. The concentrations of insoluble ^{129}I in the samples of sediment collected August 1979 through September 1981 from the basement floor range from 2.7×10^{-3} to 1.1×10^{-1} $\mu\text{Ci/g}$. The concentration of ^{129}I in the sediment collected from beneath the covered hatch on May 14 is about a factor of 20 higher than its concentration measured in the sediment collected through penetration 401 on August 28, and it is about a factor of 40 higher than its concentration measured in the sediment collected September 24 from beneath the open stairwell. Water draining through the covered hatch might have carried chemical agents and/or elements that enhanced the precipitation of iodine from the water below.

The results of a chemical species analysis that was performed on the three 1-L samples of water that were drawn from the basement through penetration 211 in March 1981 indicate that the iodine in solution was predominantly iodide. The results of that analysis are as follows (see Reference 19).

| <u>Iodine Species</u> | ^{129}I <u>Concentration ($\mu\text{g/mL}$)</u> |
|---------------------------|--|
| Total iodine | 0.031 |
| Iodide | 0.030 |
| Iodate | 0.00056 |
| Elemental | 0.00061 |

If radioiodine was released from the core and entered the water as elemental iodine (I_2), then iodide (I^-) would have immediately accounted for about 50% of the total iodine in solution, according to the reaction



The fraction that was iodide would have increased to about 83% after a few days had elapsed, according to the overall reaction



Iodate (IO_3^-) would not have been present initially, but after a few days, it would have accounted for about 17% of the iodine in solution. The measured concentration of iodate corresponds to 1.8% of the total iodine in solution and is therefore a factor of 10 less than the fraction expected if iodine entered the coolant as elemental iodine. The conclusion that must be drawn is that the fission product iodine entered the water as iodide and not as elemental iodine (see Reference 19).

Following the completion of processing of the basement water through the Submerged Demineralizer System in early June 1982, the concentrations of ^{137}Cs and ^{90}Sr that dissolved in the water remaining in the basement were slightly higher than their concentrations prior to the start of processing. The decontamination operations that were conducted on the upper levels of the building as well as dissolution of activity in the sediment on the floor would have contributed activity to the inventory in solution. By January 1983 the concentrations of ^{137}Cs and ^{90}Sr in solution had been reduced to average values of about 9.5 and 2.8 $\mu\text{Ci/mL}$, respectively, as a result of additional processed water being added to the basement water inventory. Apparently the volume of processed water that mixed with the water in the Reactor Building sump was small, as the concentrations of ^{137}Cs and ^{90}Sr that were measured in the sump water in August 1983 were about 95 and 6 $\mu\text{Ci/mL}$, respectively.

The activity concentrations measured in the water samples that were collected from the RCDT in December 1983 are, in general, substantially

lower than their respective concentrations in the basement water on earlier sampling dates. The concentrations measured in samples collected from the RCDT may be understood in terms of leakage from the RCS. The RCS was first depressurized in July 1982, and it remained depressurized until mid-December 1982. During the time it was depressurized, the RCS did not leak. At the end of December 1982 the RCS pressure was increased to 60 psig, and this pressure was maintained until late July 1983 when it was decreased to atmospheric pressure. During this latter period when the RCS was pressurized, water again began to leak to the RCDT. Initially, the RCS leak rate increased rapidly from 0 to 0.72 L/min. It gradually decreased to 0.63 L/min. by late May 1983 and remained at that value until late July 1983, when the RCS stopped leaking. The total volume of water that leaked from the RCS to the RCDT during this time frame is therefore about 2×10^5 L, or about eight RCDT volumes. Since no RCS leakage occurred from July to December 1982, the activity concentrations in the RCDT liquid were expected to be similar to those in the RCS liquid during the December 1982 to July 1983 time frame.

The analysis of RCDT samples collected in December 1983 revealed that the concentrations of ^3H and ^{90}Sr in the RCDT water were essentially identical to their average concentrations in the RCS water in July 1983, which were 0.034 and 2.6 $\mu\text{Ci/mL}$, respectively.²¹ However, the concentration of ^{137}Cs in the RCDT water was 1.23 $\mu\text{Ci/mL}$, which was about a factor of 5 higher than its average concentration in the RCS water in June and July 1983. The average concentration of ^{137}Cs in the RCS water during the entire December 1982 to July 1983 time frame when leakage occurred was only 0.56 $\mu\text{Ci/mL}$, suggesting that the actual concentration of ^{137}Cs in the RCS water that leaked to the RCDT during this period was higher than 0.56 $\mu\text{Ci/mL}$.

The data presented in Table 6 for the solids fractions of the Reactor Building basement samples indicate that ^{90}Sr is the predominant activity in both the sediment on the basement floor and the particulate matter retained in the RCDT. The average concentration of ^{90}Sr in the sediment covering the basement floor is about 1.4 mCi/g (mCi per dry gram), and its concentration in the solids in the RCDT is a surprising 14 mCi/g. The

nuclide having the second highest concentration in the basement sediment is ^{137}Cs ; its average concentration is about 0.86 mCi/g. The average concentrations of ^{125}Sb , ^{134}Cs , ^{106}Ru , and ^{144}Ce in the basement sediment were all about a factor of 10 to 20 lower than the average concentration of ^{137}Cs .

As was previously mentioned, the sample of sediment collected June 23, 1982 was divided following its collection, and portions were sent to ORNL and WHEDL for analysis. ORNL received about 390 mg of solids, and WHEDL received about 720 mg of solids. A comparison of the analyses results presented in Table 6 indicates that the nuclides and elements present in the original sample were not homogeneously dispersed. The radionuclide concentrations reported by WHEDL are, on the average, a factor of 2 higher than the concentrations reported by ORNL. With the exception of ^{90}Sr , the radionuclide concentrations measured in the two samples of sediment that were collected from the same floor location through penetration 401 on August 28, 1979 and November 15, 1979 generally agree within a factor of 2.5. The concentration of ^{90}Sr , however, was a factor of 7.4 higher in November than it was in August, suggesting that ^{90}Sr was precipitating from the basement water during the several months between samplings.

A number of compounds and elements measured to be present in the water and sediment in the basement can combine with strontium to form slightly soluble or insoluble compounds. An abbreviated list of candidate strontium compounds is provided in Table 8. Significant amounts of boron, carbon, and silicon were detected in many of the sediment samples, and dissolved carbonate, oxalate, and sulfate were measured in several samples of water. Phosphate, on the other hand, was not detected in these same samples that were analyzed using ion chromatography.

The river water that leaked into the basement from the air cooling system is probably the source of the majority of the carbon and a significant fraction of the silicon in the water and sediment. The results of an elemental analysis performed on a sample of Susquehanna River silt collected from a drain in the river water cooling system indicate that Fe, Si, Mn, Pb, Ca, K, S, Al, Ba, Na, and Ti are, in descending order of

TABLE 8. SLIGHTLY SOLUBLE OR INSOLUBLE STRONTIUM COMPOUNDS^a

| Compound Name | Formula | Solubility | |
|----------------------------|---|--------------------------|------------------------|
| | | Cold Water (g/100mL) | Hot Water (g/100mL) |
| Strontium hexaboride | SrB ₆ | Insoluble | Insoluble |
| Strontium carbonate | SrCO ₃ | 0.0011 | 0.065 |
| Strontium hyponitrite | SrN ₂ O ₂ · 5H ₂ O | Very slightly soluble | Slightly soluble |
| Strontium oxalate | SrC ₂ O ₄ · H ₂ O | 0.0051 | 5 |
| Strontium diorthophosphate | SrHPO ₄ | Insoluble | Insoluble |
| Strontium metasilicate | SrSiO ₃ | Insoluble | Insoluble |
| Strontium monosulfide | SrS | Insoluble | Decomposes |
| Strontium sulfite | SrSO ₃ | 0.0033 | -- |

a. R. C. Weast (ed.), Handbook of Chemistry and Physics, 52nd edition, Cleveland: CRC Press, 1971-1972, pp. B164-165.

concentration, its major elements, excluding C and O. The concentrations of these eleven elements in the sample of silt range from greater than 10 to 0.1 weight percent (wt%) (see Reference 13). Smaller amounts of Cr (0.05 wt%), Cu (0.03 wt%), and Zn (0.03 wt%) were also detected (see Reference 13). A second source of many of the elements mentioned is the concrete dust in the Reactor Building. The analyses results for a sample of scarified concrete indicate that it is composed primarily of, in descending order of concentration, Si, Al, Ca, Fe, S, K, and Na. The concentrations of these seven elements in the concrete range from greater than 10 wt% to 0.1 wt% (see Reference 13). Concrete dust on upper level surfaces would have been washed to the basement during the first few months following the accident and during the decontamination flushings of upper level floors and walls. Also, because the concrete walls in the basement, excluding the D-ring walls, are unpainted above a height of about 1.8 m, modest quantities of the seven elements would have been leached from the bare walls.

Most of the other elements that are itemized in Table 6 were transported to the basement in the coolant that escaped from the RCS. Video surveys of the reactor core region, which were first performed in 1983, have revealed that the core was extensively damaged as a result of the accident. Core materials have been measured to be present in sections of the reactor coolant Makeup and Purification System. Because the portions of this system containing core materials were isolated from the reactor early in the accident sequence, there is reason to believe that extensive disruption of the core occurred when the system water velocities were sufficient to distribute core materials throughout the RCS.

The active core region contains fuel, structural materials, neutron absorbers and trace amounts of materials used for instrumentation, neutron sources, and insulation. The form, composition, mass, and elemental composition of material in each category is indicated in Table 9. Only the initial core condition is given in the table; the contributions of fission and corrosion products and coolant crud deposition are neglected. On the basis of total mass inventory, the fuel, structural, and absorber materials constitute about 70, 27, and 3 wt% of the active core, respectively.

TABLE 9. INVENTORY OF MAJOR ELEMENTS IN THE ACTIVE CORE REGION^a

| Category of Material | Form of Material | Compound/Alloy | Mass (kg) | Elemental Composition (kg) | | | | |
|----------------------|----------------------|--|-----------|----------------------------|-------|------|------|------|
| | | | | U | Ag | In | Cd | |
| Fuel | Ceramic pellets | UO ₂ | 93,049 | 81,883 | -- | -- | -- | |
| Poisons | Metal alloy rod | Ag-In-Cd | 2,749 | -- | 2,199 | 412 | 137 | |
| | | | | U | B | Al | Gd | |
| | Ceramic pellets | B ₄ C in Al ₂ O ₃ | 626 | -- | 5.8 | 327 | -- | |
| | Ceramic pellets | Gd ₂ O ₃ -UO ₂ | 132 | 113 | -- | -- | 3.4 | |
| | | | | Zr | Ni | Cr | Fe | Sn |
| 38 Structures | Fuel rod cladding | Zircaloy-4 | 20,158 | 19,754 | -- | 20.2 | 42.3 | 292 |
| | Guide tubes | Zircaloy-4 | 1,220 | 1,196 | -- | 1.2 | 2.6 | 17.7 |
| | Instrument tubes | Zircaloy-4 | 113 | 111 | -- | 0.1 | 0.2 | 1.6 |
| | Poison rod cladding | Zircaloy-4 | 744 | 729 | -- | 0.7 | 1.6 | 10.8 |
| | Spacer sleeves | Zircaloy-4 | 118 | 116 | -- | 0.1 | 0.2 | 1.7 |
| | End plugs | Zircaloy-4 | 676 | 662 | -- | 0.7 | 1.4 | 9.8 |
| | Ceramic spacers | ZrO ₂ | 331 | 244 | -- | -- | -- | -- |
| | Control rod cladding | 304SS | 612 | -- | 61.2 | 116 | 428 | -- |
| | Orifice rod cladding | 304SS | 27 | -- | 2.7 | 5.1 | 18.9 | -- |
| | End plugs | 304SS | 64 | -- | 6.4 | 12.2 | 44.8 | -- |
| | Plenum springs | (Stainless) | 703 | -- | 70.3 | 134 | 492 | -- |
| | Metallic spacers | (Stainless) | 204 | -- | 20.4 | 38.8 | 143 | -- |
| | Spacer grids | Inconel-718 | 1,211 | -- | 606 | 242 | 242 | -- |

a. Nuclear Associates International, TMI-2 Accident Core Heatup Analysis, Part III--"As-Built Design and Material Characteristics of the TMI-2 Core," NSAC-25, June 1981.

The elemental analyses results presented in Table 6 indicate that Cu, Fe, Al, Si, Ni, B, Ag, Ca, and Zn are the major elements, excluding C and O, present in the sediment on the basement floor. The average concentrations of these nine elements range from 3.8 wt% (B) to 0.4 wt% (Zn). The major elements detected in the particulate matter collected from the RCDT are, in descending order of concentration, Cu, Fe, Ni, K, B, Ca, Zn, Al, and Na. The concentrations of these nine elements range from 32 wt% to 1.3 wt%. The concentration of Cu in the sediment on the basement floor ranges from 0.01 wt% in the sump to 12 wt% at penetration 401. The basement floor location having the second highest Cu concentration (10 wt%) is the point below the covered hatch. Since the concentration of Cu in Susquehanna River silt was measured to be only 0.03 wt%, most of the Cu in the basement sediment must have derived from the corrosion of electrical components in the Reactor Building, such as motor windings and buss bars. The high concentration (32 wt%) of Cu in the RCDT solids is surprising because the active core region and the pressurizer contain only trace amounts of Cu, and the walls and plumbing of the RCDT are stainless steel. The source of this Cu is a mystery. Perhaps it was present as a contaminant when the tank was installed.

The concentration of Fe in the sediment on the basement floor ranges from 0.3 wt% beneath the open stairwell to 15 wt% at the sampling location beneath the covered hatch. The higher concentration beneath the covered hatch indicates that the majority of Fe in the sediment derived from concrete dust, corroded steel, and river water rather than from reactor core materials. The element that exhibited the highest average concentration in the sediment was B. Its concentration ranged from 14 wt% beneath the open stairwell to 0.12 wt% beneath penetration 401 (August 28, 1979 sample), and its average concentration in all of the sediment samples was 3.8 wt%. Boron's concentration in the RCDT solids sample was about 9 wt%.

Among the elements that exhibited the highest concentrations in the sediment, Cu and Ag are of special interest because they both can combine with iodine to form insoluble or very slightly soluble compounds. Copper combines with iodide (I^-) to form CuI (or Cu_2I_2), which has a

solubility in cold water of 8 $\mu\text{g/mL}$, and with iodate (IO_3^-) to form $\text{Cu}(\text{OH})\text{IO}_2$ and Cu_2HIO_6 , both of which are insoluble in cold and hot water. Silver combines with iodine to form AgI , which is insoluble, and AgIO_3 , which has a solubility of 30 $\mu\text{g/mL}$ in cold water and 190 $\mu\text{g/mL}$ in hot water. The data presented in Table 6 indicate that there is a correlation between the concentrations of Cu and Ag in the sediment on the basement floor and the concentration of ^{129}I in the sediment. Sampling locations having the highest concentrations of Cu and Ag are also the locations with the highest concentrations of ^{129}I .

The presence of Ag in the sediment on the basement floor and in the debris in the RCDT is also of interest because Ag is the major constituent of the Ag-In-Cd alloy that is the neutron absorber in the majority of the reactor control rods. By weight, the alloy is 80% Ag, 15% In, and 5% Cd. The concentration of Ag in the sediment ranges from 2.5 wt% at the location of the open stairwell to 0.029 wt% in the sump. Its concentration in the RCDT solids is 0.63 wt%. The other elements of the alloy, In and Cd, were also detected in the samples collected from the basement. The concentration of In in the sediment on the basement floor varied between less than 0.007 wt% in the sump to 0.3 wt% at the location beneath the open stairwell. The concentration of Cd in the sediment ranged from a low of 0.015 wt% in the sump to a high of 0.64 wt% beneath the covered hatch. The concentrations of In and Cd in the particulate matter collected from the RCDT were 0.62 and 1.19 wt%, respectively.

The masses of Ag, In, and Cd in the reactor core prior to the accident were 2199, 412, and 137 kg, respectively (see Table 9). On an atom basis, these masses correspond to 6 atoms of Cd and about 18 atoms of In for every 100 atoms of Ag in the core. The relative number of atoms of these three elements in the samples of solids collected from the RCDT and basement floor are given in Table 10.

The fact that the relative abundances of these three elements in the RCDT and basement are different from their known values in the core is not surprising given the low melting temperature of the Ag-In-Cd alloy and the different boiling temperatures of the individual elements. The alloy melts

TABLE 10. RELATIVE NUMBER OF ATOMS

| <u>Location</u> | <u>Ag</u> | <u>In</u> | <u>Cd</u> |
|----------------------------|-----------|-----------|-----------|
| Reactor core | 100 | 18 | 6 |
| Reactor coolant drain tank | 100 | 93 | 182 |
| Basement floor | 100 | 74 | 51 |
| Sump | 100 | <22 | 48 |

at 799°C, and the control rod cladding, which is 304 stainless steel, melts at a temperature between 1399 and 1454°C.²² Pure Cd boils at the comparatively low temperature of 765°C at atmospheric pressure, 34°C below the melting temperature of the alloy. When core temperatures reached the melting temperature of the alloy, some Cd gas would have begun to evolve from the alloy that melted. As core temperatures escalated, Cd vapor would have accumulated in the upper plenums of the control rods. The Cd vapor pressure within the rods would have continued to increase until the cladding began to melt. Calculations that have been performed indicate that the maximum vapor pressure of Cd inside the control rods was not sufficient to cause cladding failure. It has been estimated that Cd vapor, at a maximum, accounted for about 35% of the vapor pressure within the rods and He, which is the gas used to pressurize the rods, accounted for the remaining 65%.²³

Based on TRAC computer code calculations, temperatures of the fuel rods rose to peak values of between 2130 and 2330°C before high pressure injection was resumed.²⁴ When the control rods began to fail, presumably between 1399 and 1454°C, Cd vapor and liquefied alloy would have escaped from the rods. The boiling temperatures of Ag and In at atmospheric pressure are 2212 and 2000°C, respectively, but the high pressure in the RCS would have suppressed the boiling of the two elements when temperatures in the core reached these levels. However, the approximate one-to-one proportionality in the number of atoms of Ag and In measured in the sample of solids removed from the RCDT indicates that Ag and In were not transported there as particles of resolidified alloy. The data indicate that they were transported from the core as condensed vapors. The quantity of Cd in the RCDT solids is disproportionately large compared to the quantities of Ag and In found in the solids because a significantly larger fraction of the Cd in the core was vaporized compared to the fractions of the core inventories of Ag and In that were vaporized.

The specific aerosols that formed when the gases that evolved from these three metals condensed are, as yet, unknown. The peak fuel rod temperature (2330°C) estimated by TRAC is below the boiling points of the major elements in stainless steel and Zircaloy-4. Therefore, it is likely

that elements such as Fe, Cr, Ni, and Zr did not vaporize. Because the boiling point of In (2000°C) is only about 200°C below that of Ag, the two metals would have begun to vaporize at about the same time, probably after much of the Cd in the alloy had boiled away. The near equality of the numbers of atoms of Ag and In in the RCDT solids might imply that Ag-In aerosols were formed when the temperature in the core was at its peak.

A comparison of the concentrations of Ag, In, and Cd in the RCDT solids with their respective concentrations in the sediment collected from the floor near the bottom of the open stairwell indicates that the RCDT did not trap these elements very effectively. The average concentrations of In and Cd in the sample of sediment are only factors of 2.8 and 2.4 lower than their respective concentrations in the RCDT solids. The average concentration of Ag in this same sample of sediment is, surprisingly, a factor of about 2 higher than its concentration in the RCDT solids sample. Other elements that exhibited higher concentrations in the sample of sediment than in the sample of RCDT solids were Mn, Al, Cr, and Si, all of which had sources other than the RCS. The concentrations of the remaining elements that were identified in both samples have higher concentrations in the RCDT solids than in the sediment sample. The ratios of the respective concentrations of these latter elements in these two samples range from 1.7 for Fe and Zr to 260 for K and Co.

Both of the samples described in the previous paragraph, as well as the sample obtained from the Reactor Building sump, were subjected to particle size analyses at WHEDL. The particle size distributions of the solids collected from the RCDT and sump were determined by analyzing 500X photomicrographs of the material filtered from the samples following ultrasonic treatment. The results for these two samples are shown as bar graphs in Figures 5 and 6, respectively. Measurements were made in 2- μ m intervals, and they did not include particles under 1 μ m. An obvious difference between the particle size distributions shown in the figures is the higher population of large particles in the sump. By number, about 12% of the particles collected from the sump are larger than 10 μ m, but only about 6% of the particles collected from the RCDT are larger than 10 μ m. Only about 39% of the sump particles were in the 1- to 3- μ m size range,

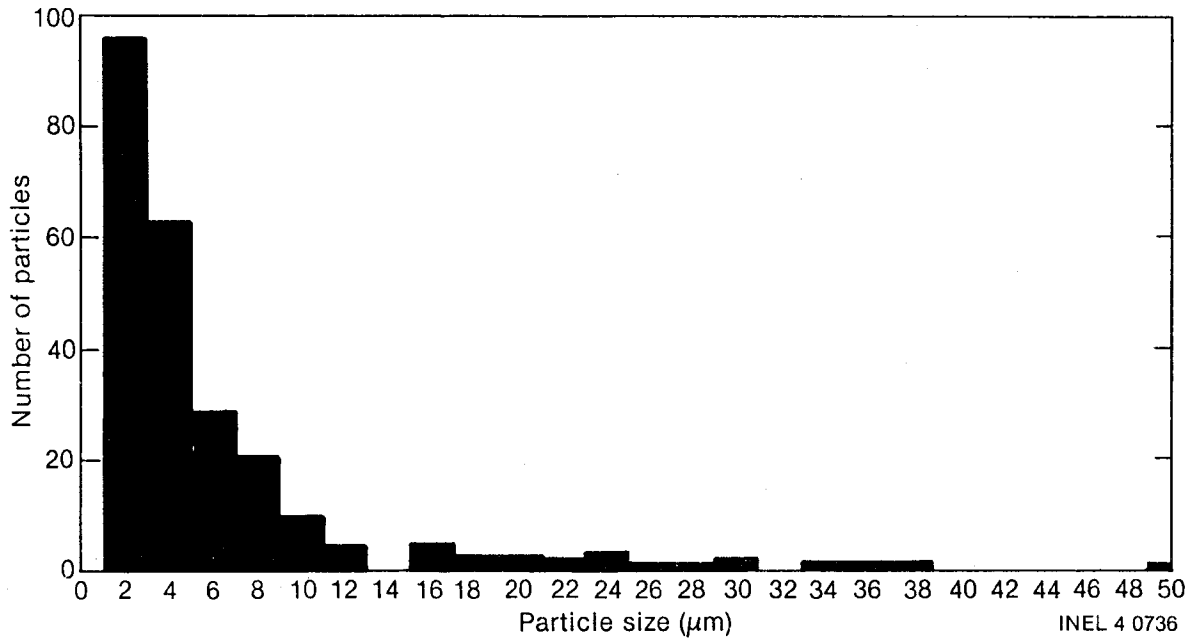


Figure 5. Particle size distribution of sediment collected from the reactor coolant drain tank.

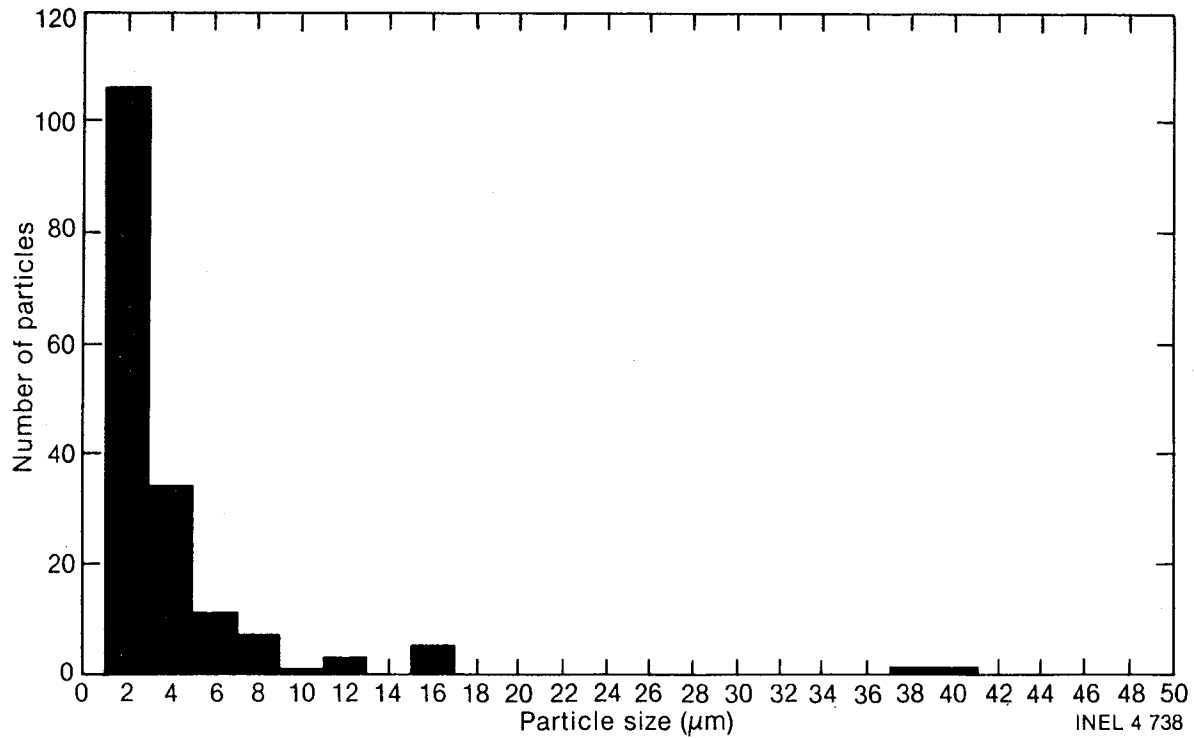


Figure 6. Particle size distribution of sediment collected from the Reactor Building sump.

whereas about 63% of the RCDT particles had sizes in this range. On a population basis, the mean particle sizes measured in the sump and RCDT samples were 6.2 and 3.9 μm , respectively. The particle size distribution analysis of the sediment collected from the floor near the bottom of the open stairwell was performed using a HIAC Particle Size Analyzer. The particle size distribution measured in this sample is very similar to the one determined for the RCDT sample. Like the RCDT sample, the sediment from beneath the open stairwell was found to contain very few particles larger than 10 μm . The mean particle size on a volume basis was measured to be about 20 μm , and on a population basis it was about 4 μm , which is essentially identical to the mean particle size of the RCDT solids. Sixty-five percent of particles in this same sediment sample exhibited sizes between 1.5 and 3 μm , and only 1.5% were found to be larger than 10 μm .

WHEDL also performed microprobe analyses on small portions of the RCDT and sump particulate samples. Tables 11 and 12 summarize the relative x-ray peak height data for the RCDT sample. Table 13 summarizes similar data for the sump sample. The analysis results of the RCDT sample showed the major elements to be Al, Si, Fe, and sometimes Cl and S. One particle showed only the presence of Mo. Minor amounts of Na, Ca, Ti, Ni, and Cu were also detected in the RCDT sample. The analysis of the sump sample showed Si as the major element present, with Fe, S, Cl, and Th also appearing as intermediate elements in some particles. The minor elements detected in the sump sample included Na, Al, P, Ca, Ti, Ni, Cu, and Zn.

The microprobe analyses performed on the RCDT and sump samples did not definitely identify U, although Ce, which is generally associated with U, was detected in each sample using gamma-ray spectroscopy measurement techniques. Neutron activation analyses subsequently performed on the RCDT and sump solids quantified the U in both samples. The concentrations of U measured through neutron activation analyses are 3.7 and 0.18 mg/g, respectively. Since U was not found by the microprobe, it must have been very finely divided and uniformly dispersed over the samples.

The concentrations of U and Pu measured in the solids fractions of the samples collected from the Reactor Building basement are presented in

TABLE 11. RELATIVE ELEMENTAL X-RAY PEAK INTENSITIES OF REACTOR COOLANT DRAIN TANK PARTICLES^a

| Photo Negative Number | Time (s) | Na | Al | Si | P | S | Cl | Ca | Ti | Fe | Ni | Cu | Zn | Comment |
|-----------------------|----------|----|-----|-----|---|-----|----|----|----|-----|----|----|----|----------|
| 124 | 239 | 51 | 55 | 287 | 0 | 528 | 0 | 51 | 29 | 66 | 0 | 11 | 0 | Area |
| 126 | 228 | 15 | 116 | 56 | 0 | 668 | 0 | 44 | 0 | 77 | 0 | 24 | 0 | Area |
| 127 | 4127 | 59 | 67 | 172 | 0 | 498 | 0 | 34 | 21 | 144 | 0 | 14 | 0 | Particle |
| 140 | 37 | 0 | 573 | 59 | 0 | 10 | 0 | 0 | 0 | 353 | 2 | 4 | 0 | Particle |
| 141 | 103 | 14 | 462 | 81 | 0 | 77 | 57 | 12 | 9 | 283 | 0 | 3 | 0 | Particle |

a. Data normalized to 1000 counts and the same sampling time.

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TABLE 12. RELATIVE ELEMENTAL CONCENTRATION OF REACTOR COOLANT DRAIN TANK PARTICLES MEASURED USING ENERGY DISPERSIVE SPECTROSCOPY PHOTOMICROGRAPHS^a

| Photo Negative Number | Time (s) | Na | Al | Si | S | Cl | Ca | Ti | Fe | Cu |
|-----------------------|----------|----|-----|----|-----|-----|----|----|-----|----|
| 114 | 4300 | 2 | 17 | 4 | 18 | >50 | 2 | 1 | 8 | 2 |
| 115 | 216 | 4 | >55 | 10 | 21 | >55 | 0 | 0 | 5 | 2 |
| 116 | 83 | 0 | 4 | 0 | 14 | >50 | 0 | 0 | 0 | 0 |
| 118 | 220 | 6 | 2 | 4 | 34 | 0 | 3 | 0 | 4 | 1 |
| 119 | 32 | 0 | >55 | 20 | 10 | 0 | 0 | 0 | >60 | 1 |
| 123 | 215 | 9 | 2 | 10 | >35 | 10 | 9 | 7 | 3 | 1 |

a. Data normalized due to off scale data.

TABLE 13. RELATIVE ELEMENTAL X-RAY PEAK INTENSITIES OF REACTOR BUILDING SUMP PARTICLES^a

| <u>Photo Negative Number</u> | <u>Time (s)</u> | <u>Na</u> | <u>Al</u> | <u>Si</u> | <u>P</u> | <u>S</u> | <u>Cl</u> | <u>Ca</u> | <u>Ti</u> | <u>Fe</u> | <u>Ni</u> | <u>Cu</u> | <u>Zn</u> |
|--------------------------------------|---------------------|-----------|-----------|-----------|----------|----------|-----------|-----------|-----------|-----------|-----------|-----------|-----------|
| 145 | 1623 | 0 | 50 | 359 | 0 | 186 | 125 | 96 | 0 | 156 | 0 | 28 | 0 |
| 146 | 523 | 0 | 34 | 306 | 25 | 187 | 171 | 81 | 36 | 129 | 0 | 30 | 0 |
| 150 | 1328 | 0 | 18 | 477 | 0 | 192 | 73 | 45 | 0 | 184 | 0 | 10 | 0 |
| 156 | 822 | 0 | 124 | 391 | 13 | 89 | 163 | 56 | 18 | 102 | 15 | 25 | 3 |

a. Data normalized to 1000 counts and the same sampling time.

Table 6. The concentration of U ranges from 4 mg/g at the sampling location beneath the covered hatch to about 0.04 mg/g at the location beneath penetration 401 (August 28, 1979 sample). Expressed as atom percent (at.%), the average (^{235}U) enrichment of the U in the core at the time of the accident was 2.33 at.%. The average enrichment of the U in the three sediment samples that were analyzed by mass spectrometry (i.e., the August 28, 1979 and June 23, 1982 samples) is close to the core average, being 2.37 at.%. The good agreement indicates that the release of U-bearing particles was uniform across the core. The concentration of Pu in the sediment ranges from about 6 $\mu\text{g/g}$ at the sampling location beneath the open stairwell (June 23, 1982 sample) to 0.065 $\mu\text{g/g}$ at the location beneath penetration 401. The value of the ratio of the concentration of U to that of Pu measured in individual sediment samples ranges from about 140 to about 5800. However, four samples exhibited similar U to Pu ratios. The mean value of the ratios for these latter four samples is 640 ± 7 . The theoretical value of the ratio of the mass of U to the mass of Pu in the reactor core during the period the basement water was sampled is about 510, which indicates that the U particles in the samples were slightly depleted of Pu.²⁵ Since the ratios of the concentrations of U to Pu that were measured in basement water samples were generally also higher than the core average ratio, it is reasonable to conclude that the U particles were depleted of Pu before they exited the RCDT.

Core Release Fractions

The total quantities of fission products, fuel, and stable elements that were estimated to be present in the water and sediment in the Reactor Building basement on the dates samples were collected are summarized in this section. The quantities of fission products in the RCS coolant during the period the basement was sampled are also included in the summary.

The inventories of radionuclides and stable elements in the basement water and sediment on each sampling date were calculated by multiplying their measured concentrations times the volume of water or the mass of sediment estimated to be present in the basement. The inventories of a limited number of fission products in the RCS were calculated by

multiplying their measured concentrations times the coolant volume of the RCS when at room temperature (21°C). The RCS volume used was 3.33×10^5 L (8.8×10^4 gal), where 4.16×10^4 L (1.1×10^4 gal) of the total volume is attributed to the pressurizer. The calculated results are expressed as percent of core inventory released. The core inventories of the fission products, fuel, and stable elements that were considered in this summary are presented in Table 14. The activities of the radionuclides and the masses of U and Pu given in the table are decay-corrected to August 28, 1979, the day the basement water was first sampled. The release fractions for other sampling dates were calculated after decay-correcting the activities presented in Table 13 to the sample collection dates.

The volume of water in the basement on any particular sampling date was calculated using the known volumes of the cavities that are below the basement floor and the currently accepted semiempirical correlations between basement water depth and volume. The volumes of the cavities and pipes that are beneath the basement floor are given in Table 15 (see Reference 6).

The calculations that were performed to determine the correlation between water depth and volume took into account the displacement volumes of RCS components, concrete shields, and support walls in the basement. The results relating water depth and volume are given in Table 16 (see Reference 6).

Using these correlations and the volume of the cavities below floor level, the water depth on each sampling date was converted into a volume. The water volumes that were calculated for the dates sampling occurred are provided in Table 17.

The total mass of the sediment distributed over the basement floor during the period the basement was sampled is less precisely known. Personnel who collected the sediment sample on June 23, 1982 from near the bottom of the open stairwell estimated that the sediment was about 1.3 cm (1/2 in.) thick at the location sampled. Video surveys of other regions of

TABLE 14. CORE INVENTORY OF RADIONUCLIDES AND ELEMENTS
(Activities Decay-corrected to August 28, 1979)

| Nuclide | Total | Element | Total | Weight Percent ^c |
|-------------------|-------------------------------|---------|---------------------------|--------------------------------|
| | Activity ^a (Ci) | | Mass ^b (kg) | |
| ³ H | 3.65 E+3 | Cr | 5.71 E+2 | 0.47 |
| ⁹⁰ Sr | 7.68 E+5 | Fe | 1.42 E+3 | 1.15 |
| ¹⁰⁶ Ru | 2.42 E+6 | Ni | 7.67 E+2 | 0.62 |
| ¹²⁵ Sb | 5.00 E+4 | Zr | 2.28 E+4 | 18.59 |
| ¹²⁹ I | 1.95 E-1 | Ag | 2.20 E+3 | 1.79 |
| ¹³¹ I | 1.29 E+2 | In | 4.12 E+2 | 0.34 |
| ¹³⁴ Cs | 1.86 E+5 | Cd | 1.37 E+2 | 0.11 |
| ¹³⁷ Cs | 8.37 E+5 | Sn | 3.34 E+2 | 0.27 |
| ¹⁴⁴ Ce | 1.62 E+7 | Gd | 3.40 E+0 | 0.003 |
| | | U | 8.17 E+4 ^a | 66.81 |
| | | Pu | 1.59 E+2 ^a | 0.13 |

a. J. A. Daniel and E. A. Schlomer, letter from Science Applications Inc. to G. R. Eidam, Bechtel National, Inc., TMI, March 9, 1983.

b. Nuclear Associates International, TMI-2 Accident Core Heatup Analysis, Part III--"As-built Design and Material Characteristics of the TMI-2 Core," NSAC-25, June 1981.

c. Based on a total materials mass of 122,737 kg.

TABLE 15. VOLUME BELOW BASEMENT FLOOR

| Item | Volume | |
|-------------------------|---------|--------|
| | (L) | (gal) |
| Sump | 10,303 | 2,722 |
| Drain pipes | 4,822 | 1,274 |
| Incore instrument chase | ~10,220 | ~2,700 |
| Elevator pit | ~ 2,840 | ~ 750 |
| Total | ~28,185 | ~7,446 |

TABLE 16. VOLUME ABOVE BASEMENT FLOOR

| Depth Interval | | Volume Increment | |
|----------------|------------|------------------|-----------|
| (m) | (ft) | (L/cm) | (gal/in.) |
| 0.00 to 1.07 | 0.0 to 3.5 | 9060 | 6080 |
| 1.07 to 1.98 | 3.5 to 6.5 | 9090 | 6100 |
| 1.98 to 2.59 | 6.5 to 8.5 | 9775 | 6560 |

TABLE 17. DEPTH AND VOLUME OF BASEMENT WATER ON SAMPLING DATES

| Sampling Date (m/d/y) | Depth | | Volume | |
|--------------------------|-------|------|----------|----------|
| | (m) | (ft) | (L) | (gal) |
| 08/28/79 | 2.11 | 6.92 | 1.95 E+6 | 5.16 E+5 |
| 11/15/79 | 2.17 | 7.12 | 2.01 E+6 | 5.31 E+5 |
| 03/19/81 | 2.49 | 8.16 | 2.32 E+6 | 6.13 E+5 |
| 05/14/81 | 2.52 | 8.28 | 2.36 E+6 | 6.23 E+5 |
| 09/19/81 ^a | 2.59 | 8.50 | 2.42 E+6 | 6.40 E+5 |
| 09/24/81 | 2.55 | 8.35 | 2.38 E+6 | 6.28 E+5 |
| 06/23/82 | 0.08 | 0.26 | 9.99 E+4 | 2.64 E+4 |
| 01/11/83 | 0.17 | 0.56 | 1.83 E+5 | 4.83 E+4 |
| 08/22/83 | 0.15 | 0.49 | 1.64 E+5 | 4.32 E+4 |

a. Not a sampling date, but the date on which water was recorded at its maximum height. (Data provided for comparison.)

the basement performed in the fall of 1982 and in 1983 showed that the thickness of the sediment varied from area to area.

The April 1983 video survey of the floor beneath the covered hatch showed that islands of bare floor as large as 2 m in diameter were present. During a separate survey, a floor drain in the northeast quadrant of the basement was examined using a zoom lens. The sediment on the drain plate had holes through it over the holes in the plate. Based on this visual inspection, it was estimated that the sediment on the plate was about 0.6 cm (1/4 in.) thick. A survey of the area around the sump screen outside the sump cubicle showed that the floor in this region was completely covered with sediment. A video inspection of the floor in the vicinity of the RCDT leakage transfer pumps, which are in the southwest quadrant of the basement, showed that sediment uniformly covered the floor in this area. However, as was the case with the survey of the area around the sump screen, no references were seen which could be used to estimate the thickness of the sediment.

As was previously mentioned, a sample of sump water was collected on August 22, 1983 at a sampling point on the sump pump discharge line after the sump had been recirculated for about 20 min. The density of solids in this sample was measured to be 0.36 mg/mL. The volume of the sump is $1.03 \times 10^7 \text{ cm}^3$, which implies that 3.7 kg of solids were in solution when the sample was obtained. Since the floor area of the sump is about $5.1 \times 10^4 \text{ cm}^2$, the mean surface density of the solids that settled on the sump floor prior to recirculation must have been at least 73 mg/cm^2 . The surface density is given as a minimum value because it is not known what fraction of the total mass of solids in the sump went into solution during the recirculation of the sump water. The minimum average thickness of the sediment layer in the sump prior to recirculation can be estimated using the results of the volumetric and mass measurements made on the solids of other sediment samples. Two different types of volumetric measurements were made. In two cases, the volume of the solids and water mixture in the bottom of the original sample container was measured after the solids had been allowed to settle out of solution. A measurement of

this type was made on the samples collected August 28, 1979 and November 15, 1979. In four cases, the sample was centrifuged, and the volume of the centrifuged solids was measured. A measurement of this latter type was made on the samples just mentioned, as well as the samples collected May 14, 1981 and June 23, 1982. The centrifuging of the two samples collected in 1979 reduced the volumes of the settled solids by an average factor of 2.25. This factor was used to convert the centrifuged volumes of the solids in the other two samples to settled volumes. The packed (i.e., centrifuged) and settled dry densities of the solids in the four samples were then calculated using the measured and estimated dry masses of the solids. The results are given in Table 18.

When the water in the basement reached its maximum depth in September 1981, the depth of the water above the sump floor was about 4.6 m (15 ft) and the depth above the basement floor was about 2.6 m (8 ft, 6 in.). At that time, the water pressure at the level of the sump floor was about 4.5×10^5 dynes/cm² (6.6 lb/in.²), and the pressure at the level of the basement floor was about 2.5×10^5 dynes/cm² (3.7 lb/in.²). These pressures are similar to the pressures exerted on the solids during centrifugation at 1800 rpm. Therefore, the average packed density is probably more representative of the actual solids density of the sediment in the basement. Dividing the calculated solids surface density in the sump by this volumetric density yields a value of 1.1 cm for the minimum thickness of the sediment layer on the sump floor.

Collectively, the visual inspections of the basement floor indicate that the sediment thickness ranges from 0 to about 1.3 cm (1/2 in.). For the purpose of calculation, it will be assumed that the average thickness of the sediment layer covering the basement floor is 0.635 cm (1/4 in.) and that the solids density of the sediment layer is 63.5 mg/cm³. The surface area of the basement floor is 9.11×10^6 cm².²⁶ Therefore, the estimated volume and mass of sediment on the basement floor are 5.78×10^6 cm³ and 367 kg, respectively. In addition to the sediment distributed over the floor and in the sump, sediment is also expected to have collected in the floor drain pipes, incore instrument chase, and elevator pit. Since these locations have not been sampled, the quantities

TABLE 18. SEDIMENT DENSITIES

| Sampling Date (m/d/y) | Solids Mass (mg) | Solids Volume | | Solids Density | |
|--------------------------|---------------------|------------------------------|-------------------------------|---------------------------------|----------------------------------|
| | | Packed (cm ³) | Settled (cm ³) | Packed (mg/cm ³) | Settled (mg/cm ³) |
| 08/28/79 | 76.2 est. | 1.2 | 3.0 | -- | 25.4 |
| 11/15/79 | 333.0 est. | 5.3 | 10.5 | -- | 31.7 |
| 05/14/81 | 108.0 | 2.2 | 5.0 est. | 49.1 | 21.6 |
| 06/23/82 | 389.4 | 5.0 | 11.3 est. | 77.9 | 34.5 |
| | | | | 63.5 ^a | 28.3 ^a |

a. Average.

of sediment they contain can only be roughly estimated. Expressed as percent of the total sump volume, the sediment layer in the sump occupies about 0.5% of the sump's volume. If a value of 1% is used for the drain pipes and incore instrument chase, the total quantities of sediment they contain are about 3 and 6 kg, respectively. The area of the elevator pit is about $6 \times 10^4 \text{ cm}^2$. If the depth of sediment in the elevator pit is the same as in the sump (i.e., 1.1 cm), then the pit contains about 4 kg of sediment. The total mass of sediment in the basement, including the sump, is therefore about 380 kg.

Like the mass of the sediment on the basement floor, the mass of the debris in the RCDT is not accurately known. Video surveys of the inside of the tank performed in December 1983 showed that the layer of debris was quite thin in the area beneath the rupture disk. Because of limitations inherent in the equipment used to perform the video survey, only the west end of the tank was examined. For the purpose of calculation, the debris on the bottom surface of the tank is assumed to be 0.16 cm (1/16 in.) thick and to uniformly cover one-eighth of the inner surface of the tank from end to end. The area of the deposited debris is then about $5.2 \times 10^4 \text{ cm}^2$, and its total volume is about $8.3 \times 10^3 \text{ cm}^3$. Based on the elemental analysis results for the RCDT solids sample presented in Table 6 (the elements listed collectively account for about 94 wt% of the sample), the density of the debris is about 6.2 g/cm^3 . If we assume that one-half of the total debris volume is water, then the total mass of debris in the RCDT is about 26 kg.

The water volumes given in Table 17 and the sediment and debris masses previously described were used to calculate the total quantities of fission products, fuel, and core materials in the water and solids in the Reactor Building basement and RCDT. The quantities estimated to be in solution in the water in the basement, sump, and RCDT on the dates sampling took place are presented in Table 19. The data indicate that the quantity of radiocesium in solution in the water in the basement was essentially constant from August 1979 to September 1981. Expressed as percent of core inventory, the quantity of ^{137}Cs in solution ranges from 39.1% to 41.9%. These values correspond to a range of between 3.26×10^5 and

TABLE 19. PERCENT OF CORE INVENTORY OF SELECTED RADIONUCLIDES AND ELEMENTS MEASURED TO BE PRESENT IN THE WATER IN THE REACTOR BUILDING BASEMENT, SUMP, AND REACTOR COOLANT DRAIN TANK (RCDT)

| Isotope/ Element | Basement Water | | | | | | | Sump | RCDT |
|---------------------|----------------|----------|----------|----------|----------|----------|----------|----------|----------|
| | 8/28/79 | 11/15/79 | 3/19/81 | 5/14/81 | 9/24/81 | 6/23/82 | 1/11/83 | 8/22/83 | 12/12/83 |
| ³ H | 5.50 E+1 | 5.8 E+1 | --a | 4.29 E+1 | 4.30 E+1 | --a | --a | --a | 3.3 E-2 |
| ⁹⁰ Sr | 7.1 E-1 | 6.1 E-1 | --a | 1.67 E+0 | 1.57 E+0 | 8.9 E-2 | 7.2 E-2 | 9.0 E-3 | 1.0 E-2 |
| ¹⁰⁶ Ru | 5.6 E-4 | 3.0 E-4 | --a | --a | --a | 7.4 E-5 | <2.8 E-3 | --a | <5.8 E-4 |
| ¹²⁵ Sb | 5.9 E-2 | 9.8 E-2 | --a | 2.1 E-1 | <1.6 E-1 | 1.1 E-2 | <2.2 E-2 | --a | 1.0 E-3 |
| ¹²⁹ I | 1.38 E+1 | --a | 6.5 E+0 | 5.2 E+0 | 2.4 E+1 | <1.0 E+1 | 8.1 E-2 | --a | 5.5 E-3 |
| ¹³¹ I | 1.86 E+1 | --a | --a | --a | --a | --a | --a | --a | --a |
| ¹³⁴ Cs | 4.18 E+1 | 3.73 E+1 | 4.01 E+1 | 4.32 E+1 | 4.16 E+1 | 2.0 E+0 | 2.4 E-1 | 1.3 E-1 | 4.8 E-3 |
| ¹³⁷ Cs | 4.10 E+1 | 3.91 E+1 | 3.96 E+1 | 4.19 E+1 | 4.08 E+1 | 2.0 E+0 | 2.2 E-1 | 1.3 E-1 | 4.4 E-3 |
| ¹⁴⁴ Ce | 7.6 E-5 | 1.3 E-5 | --a | --a | <3.3 E-3 | --a | <2.7 E-4 | --a | <1.2 E-4 |
| U | 3.8 E-5 | --a | --a | --a | <8.7 E-5 | 1.3 E-6 | --a | 7.2 E-7 | <4.4 E-6 |
| Pu | 2.2 E-5 | --a | --a | 3.2 E-4 | --a | 3.6 E-6 | --a | 3.4 E-10 | --a |
| Cr | 2.4 E-1 | --a | --a | 8.3 E-2 | 1.3 E-1 | 4.4 E-2 | --a | 4.5 E-5 | --a |
| Fe | 1.6 E-1 | --a | --a | 1.5 E-1 | 1.2 E-1 | 1.3 E-2 | --a | 9.9 E-4 | --a |
| Ni | <2.5 E-1 | --a | --a | <3.1 E-1 | 3.4 E-1 | 6.5 E-3 | --a | 8.9 E-4 | --a |
| Zr | --a | --a | --a | <1.1 E-2 | 1.5 E-2 | 1.8 E-4 | --a | <2.3 E-6 | --a |
| Ag | <2.7 E-2 | --a | --a | <1.1 E-1 | <1.1 E-1 | 3.2 E-4 | --a | 6.1 E-5 | --a |
| Cd | <2.9 E-1 | --a | --a | <3.5 E+0 | <3.5 E+0 | 8.0 E-2 | --a | 1.6 E-3 | --a |
| In | <4.7 E-2 | --a | --a | <2.8 E+0 | <2.9 E+0 | 1.3 E-2 | --a | 1.3 E-3 | --a |
| Sn | --a | --a | --a | <3.5 E+0 | <3.6 E+0 | 5.9 E-4 | --a | 1.1 E-3 | --a |
| Gd | --a | --a | --a | <6.9 E+1 | <7.0 E+1 | --a | --a | 3.0 E-2 | --a |

a. Concentration was not measured.

3.37×10^5 Ci of ^{137}Cs in solution. The data presented in Table 19 indicate that the quantity of ^3H in solution decreased from 58% of the core inventory in November 1979 to 43% of the core inventory by May 1981. After correcting for radioactive decay, this equates to a loss of about 500 Ci of ^3H from the water during the time between samplings. The purges of the Reactor Building atmosphere conducted during the summer of 1980 released only about 1.3 Ci of ^3H , which implies that the quantities of ^3H that were lost as a result of containment purging were insignificant.²⁷ The good agreement among the ^{137}Cs results for samplings performed at locations far removed from one another indicates that soluble species were homogeneously dispersed throughout the basement. Because ^3H is normally very soluble, it is expected that ^3H would also have been uniformly distributed in the basement water. Given these circumstances, the apparent loss of ^3H from the water might be real, but the depletion mechanism is not clearly understood.

The data presented in Table 19 indicate that the quantity of ^{90}Sr in solution increased from about 0.7% (5.5×10^3 Ci) of the core inventory in 1979 to about 1.6% (1.2×10^4 Ci) of the core inventory by 1981. As was previously discussed, the increase in ^{90}Sr activity is attributed to RCS leakage.

The soluble radioiodine activities presented in Table 19 for the August 28, 1979 sampling agree reasonably well. The data indicate that about 14% of the core inventory of ^{129}I and about 19% of the core inventory of ^{131}I were in solution at that time. Two of the three samplings performed in 1981 yielded similar results for the quantity of ^{129}I in solution. The quantities of soluble ^{129}I measured on March 19, 1981, and May 14, 1981, are 6.5 and 5.2% of the core inventory, respectively. The result for the sample collected through the open stairwell on September 24, 1981 is substantially higher, being 24%. For reasons that will be discussed, this result should be rejected. An investigation of the analysis method used on this latter sample determined that the liquid tested through neutron activation analysis was very likely not filtered prior to analysis.²⁸ The volume of the sample that was analyzed for ^{129}I was 0.1 mL. The solids density of the original 120-mL

sample was measured to be about 0.21 mg/mL. Therefore, the 0.1-mL sample might have contained about 0.02 mg of solids. The concentration of ^{129}I in the solids of the September 24, 1981 sample was determined by analyzing a 0.49-mg sample, which corresponds to only about 2% of the total solids in the original sample. Therefore, the result for this sample that was presented in Table 6, which is $2.7 \times 10^{-3} \mu\text{Ci/g}$, is not necessarily representative of the concentration of ^{129}I in the solids in the 0.1-mL sample that was subjected to neutron activation analysis. Its concentration could have been as high as $0.18 \mu\text{Ci/g}$, which is the ^{129}I concentration reported by ORNL for the June 23, 1982 sample that was collected from near the bottom of the open stairwell. If this latter concentration is used for the 0.02 mg of solids that likely contaminated the 0.1-mL liquid sample, the solids would have contributed $3.6 \times 10^{-6} \mu\text{Ci}$ of ^{129}I to the sample. The total ^{129}I activity measured in the 0.1-mL sample was $2.2 \pm 1.1 \times 10^{-6} \mu\text{Ci}$. It is evident that a very modest quantity of solids in the sample could have seriously biased the soluble ^{129}I result, and for this reason the release fraction of 24% presented in Table 19 should be rejected.

The result for the September 24, 1981 sample aside, the results presented in Table 19 indicate that the quantity of ^{129}I in solution decreased from about 14% of the core inventory in August 1979 to about 6% of the core inventory by the spring of 1981. Since only insignificant quantities of ^{129}I were released as a result of Reactor Building purges (see Reference 27), these data imply that about 57% of the radioiodine in solution in August 1979 precipitated from solution and/or plated out on basement structural surfaces. If precipitation was the dominant depletion mechanism, it would have contributed about 8% of the core inventory of ^{129}I (i.e., 0.016 Ci) to the total quantity of ^{129}I measured in the sediment in the basement.

In September 1984, four basement water samples that had been archived since May 1981 were analyzed for ^{129}I .²⁹ One of the four samples was collected directly from the basement floor and contained a sufficient quantity of solids to allow a separate analysis of the solids. The other three archived samples were obtained at three different elevations above

the floor. The concentrations of ^{129}I measured in these latter three samples are $3.27 \pm 0.08 \times 10^{-6} \mu\text{Ci/mL}$, $<1 \times 10^{-6} \mu\text{Ci/mL}$, and $5.30 \pm 0.08 \times 10^{-6} \mu\text{Ci/mL}$. The average of the two concentrations having values above the detection limit is in excellent agreement with the average concentration measured in 1981 in the corresponding liquid samples. The 1981 measurements yielded an average concentration of $4.3 \times 10^{-6} \mu\text{Ci/mL}$ (see Table 5). The concentration of ^{129}I in the filtrate of the archived sample containing the solids was measured to be $5.14 \pm 0.54 \times 10^{-7} \mu\text{Ci/mL}$, and the concentration of ^{129}I in the solids fraction of this sample was measured to be $3.27 \pm 0.03 \times 10^{-1} \mu\text{Ci/g}$. The concentration of ^{129}I in the filtrate of this archived sample is about a factor of 8 lower than the average concentration measured to be in solution in the four samples analyzed in 1981. The concentration of ^{129}I in the solids of the archived sample is about a factor of 3 higher than the concentration reported in 1981 for the solids from the complementary sample. These data certainly are not conclusive proof, but they tend to suggest that ^{129}I precipitated from the water of the archived sample.

The quantities of fission products, fuel, and core material elements measured in the sediment on the basement floor and in the sump are presented in Table 20. The results given in Table 20 are segregated by sampling location and are expressed as percent of core inventory per gram of solids. The overall average concentrations given in the table are arithmetic averages of the concentrations measured at four sampling locations; penetrations 401 and 225 are in the same vicinity in the southwest region of the basement so were considered as a single sampling location. The data for penetration 238 were ignored when the average concentrations were computed because the mass of the sample collected at this location was not determined. The higher concentration of ^{129}I compared to the concentration of ^{131}I probably reflects the effect of precipitation since the ^{131}I result is for a sample collected in August 1979, whereas the ^{129}I result is an average value for all of the sediment samples collected through 1983. The average concentrations of ^{144}Ce and U in the sediment given in Table 20 agree very well, which supports the

TABLE 20. PERCENT OF CORE INVENTORY PER GRAM OF SOLIDS OF SELECTED RADIONUCLIDES AND ELEMENTS MEASURED TO BE PRESENT IN THE SEDIMENT ON THE REACTOR BUILDING BASEMENT FLOOR

| Isotope/ Element | Penetrations | | | | | Average Value ^a |
|---------------------|------------------|-----------------|-----------------|-----------------|-----------|-------------------------------|
| | 401 and 225 | Open Stairwell | Covered Hatch | Penetration 238 | Sump | |
| ³ H | --b | --b | --b | --b | --b | --b |
| ⁹⁰ Sr | 3.41 ± 3.96 E-8 | 4.38 ± 2.12 E-7 | 1.39 ± 0.43 E-7 | >8.05 E-8 | 2.16 E-8 | 1.6 ± 1.9 E-7 |
| ¹⁰⁶ Ru | 9.02 ± 2.23 E-10 | 1.09 ± 0.13 E-8 | 1.39 E-8 | >2.48 E-8 | --b | 8.6 ± 6.8 E-9 |
| ¹²⁵ Sb | 4.45 ± 1.98 E-8 | 3.85 ± 2.99 E-7 | 7.86 ± 10.0 E-7 | >1.67 E-6 | 9.19 E-9 | 3.1 ± 3.6 E-7 |
| ¹²⁹ I | 1.74 ± 1.03 E-6 | 4.68 ± 6.43 E-5 | 3.23 ± 3.41 E-5 | --b | <1.27 E-8 | 2.0 ± 2.3 E-5 |
| ¹³¹ I | 4.11 ± 1.11 E-6 | --b | --b | --b | --b | 4.1 E-6 |
| ¹³⁴ Cs | 5.19 ± 3.04 E-9 | 1.33 ± 1.10 E-7 | 3.18 ± 3.06 E-7 | >7.01 E-8 | 7.30 E-9 | 1.2 ± 1.5 E-7 |
| ¹³⁷ Cs | 5.32 ± 3.32 E-9 | 1.34 ± 1.13 E-7 | 2.82 ± 2.57 E-7 | >6.97 E-8 | 7.03 E-9 | 1.1 ± 1.3 E-7 |
| ¹⁴⁴ Ce | 2.06 ± 1.16 E-10 | 3.81 ± 0.18 E-9 | 2.55 ± 0.96 E-9 | >1.61 E-9 | 2.97 E-9 | 2.4 ± 1.5 E-9 |
| U | 1.37 ± 2.00 E-9 | 2.96 ± 2.23 E-9 | 3.79 ± 1.56 E-9 | --b | 2.20 E-10 | 2.1 ± 1.6 E-9 |
| Pu | 1.83 ± 2.03 E-10 | 3.31 ± 0.75 E-9 | 2.11 ± 0.40 E-9 | >2.83 E-9 | 8.18 E-10 | 1.6 ± 1.4 E-9 |
| Cr | 1.03 ± 0.47 E-7 | 1.78 ± 3.01 E-6 | 1.75 E-6 | <2.63 E-8 | 1.32 E-8 | 9.1 ± 9.9 E-7 |
| Fe | 5.35 ± 3.91 E-7 | 4.33 ± 5.83 E-6 | 5.47 ± 7.20 E-6 | 2.75 E-7 | 7.03 E-7 | 2.8 ± 2.5 E-6 |
| Ni | 2.01 ± 2.78 E-6 | 2.15 ± 1.57 E-6 | 3.91 E-6 | <5.22 E-9 | 1.97 E-8 | 2.0 ± 1.6 E-6 |
| Zr | 1.47 ± 1.40 E-9 | 8.77 ± E-10 | 5.26 E-10 | <2.85 E-10 | 8.92 E-10 | 9.4 ± 3.9 E-10 |
| Ag | 9.23 ± 7.52 E-8 | 1.14 E-6 | 7.27 E-7 | --b | 1.32 E-8 | 4.9 ± 5.4 E-7 |
| Cd | 2.09 E-7 | 1.21 E-6 | 1.55 E-6 | --b | 1.05 E-7 | 7.7 ± 7.2 E-7 |
| In | 2.15 ± 1.81 E-7 | 1.64 ± 0.77 E-6 | 7.30 E-6 | --b | <1.70 E-8 | 2.3 ± 3.4 E-6 |
| Sn | 1.89 E-7 | 3.59 ± 0.85 E-7 | --b | --b | <1.19 E-8 | 1.9 ± 1.7 E-7 |
| Gd | --b | --b | --b | --b | <2.97 E-7 | <3 E-7 |

a. The data for penetration 238 were ignored when average values were computed.

b. Not detected.

hypothesis that signature gamma rays of ^{144}Ce and its daughter ^{144}Pr can be used to estimate the quantities of U in bulk sediment and debris samples.

Best estimates of the total quantities of fission products, fuel, and core materials in the water and sediment in the Reactor Building basement are summarized in Table 21. The quantities given for the sediment were calculated by multiplying the average concentrations in Table 20 by 3.8×10^5 g, the estimated total mass of the sediment in the basement. The average quantities in solution are presented separately for 1979 and 1981. The uncertainties in the quantities presented for the sediment and water are, in each case, simply the standard deviation of the set of measured concentration values. The uncertainties presented in Table 21 do not include the uncertainties in the mass of sediment and volume of water in the basement, nor do they include the uncertainties in the fission product inventories of the core that were used to calculate the release fractions. The best estimates of the total quantities in the basement are subjective summations of the quantities in the sediment and in solution. The results, expressed as percent of core inventory, indicate that 57% of the ^3H , 41% of the ^{137}Cs , 20% of the ^{131}I , 14% of the ^{129}I , and 1.7% of the ^{90}Sr were dispersed in the water and sediment in the Reactor Building basement. The uncertainty in the quantity of ^{129}I in the basement is at a minimum $\pm 9\%$ of the core inventory. This large uncertainty is due primarily to the fact that the concentration of ^{129}I in the samples of sediment varied over a wide range of values.

Similar results for the RCDT are summarized in Table 22. The quantities of ^{106}Ru , ^{144}Ce , U, Ni, Zr, Cd, Sn, and Gd in the RCDT are significant when compared with the quantities of these nuclides and elements measured in the basement water and sediment. For example, the quantity of ^{144}Ce in the RCDT is approximately equal to the total amount of this nuclide in the water and sediment in the basement. The quantities of ^3H , ^{90}Sr , ^{134}Cs , and ^{137}Cs in the coolant in the RCS during the time period when the basement was sampled are presented in Table 23. When the basement water was first sampled on August 28, 1979, the quantities of these nuclides in the RCS coolant, expressed as percent of core inventory,

TABLE 21. BEST ESTIMATES OF TOTAL QUANTITIES OF FISSION PRODUCTS, FUEL, AND CORE MATERIALS IN THE WATER AND SEDIMENT IN THE REACTOR BUILDING BASEMENT

| Isotope/ Element | Percent of Core Inventory | | | |
|---------------------|---|--------------------|-----------------|--------------------------|
| | Sediment ^a (1979 to 1983) | Water ^a | | "Best Estimate" Total |
| | | (1979) | (1981) | |
| ³ H | -- | 5.7 ± 0.2 E+1 | 4.30 ± 0.07 E+1 | 5.7 ± 0.2 E+1 |
| ⁹⁰ Sr | 6.1 ± 7.2 E-2 | 6.6 ± 0.7 E-1 | 1.62 ± 0.07 E+0 | 1.7 ± 0.1 E+0 |
| ¹⁰⁶ Ru | 3.3 ± 2.6 E-3 | 4.3 ± 1.8 E-4 | -- | 4 ± 3 E-3 |
| ¹²⁵ Sb | 1.2 ± 1.4 E-1 | 7.9 ± 2.8 E-2 | 2.1 ± 0.6 E-1 | 3 ± 2 E-1 |
| ¹²⁹ I | 7.6 ± 8.7 E+0 | 1.38 ± 0.04 E+1 | 5.9 ± 0.9 E+0 | 1.4 ± 0.9 E+1 |
| ¹³¹ I | 1.6 ± 0.4 E+0 | 1.86 ± 0.09 E+1 | -- | 2.0 ± 0.1 E+1 |
| ¹³⁴ Cs | 4.6 ± 5.7 E-2 | 3.96 ± 0.32 E+1 | 4.16 ± 0.16 E+1 | 4.2 ± 0.2 E+1 |
| ¹³⁷ Cs | 4.2 ± 4.9 E-2 | 4.01 ± 0.13 E+1 | 4.08 ± 0.12 E+1 | 4.1 ± 0.1 E+1 |
| ¹⁴⁴ Ce | 9.1 ± 5.7 E-4 | 4.5 ± 4.5 E-5 | <3.3 E-3 | 1.0 ± 0.6 E-3 |
| U | 8.0 ± 6.1 E-4 | 3.8 ± 2.6 E-5 | <8.7 E-5 | 8 ± 6 E-4 |
| Pu | 6.1 ± 5.3 E-4 | 1.8 ± 1.3 E-5 | 3.2 ± 1.0 E-4 | 6 ± 5 E-4 |
| Cr | 3.5 ± 3.8 E-1 | 2.4 ± 0.0 E-1 | 1.1 ± 0.3 E-1 | 5 ± 4 E-1 |
| Fe | 1.1 ± 1.0 E+0 | 1.6 ± 0.8 E-1 | 1.4 ± 0.2 E-1 | 1 ± 1 E+0 |
| Ni | 7.6 ± 6.1 E-1 | <2.5 E-1 | 3.4 E-1 | 8 ± 6 E-1 |
| Zr | 3.6 ± 1.5 E-4 | -- | 1.5 E-2 | 4 ± 2 E-4 |
| Ag | 1.9 ± 2.1 E-1 | <2.7 E-2 | <1.1 E-1 | 2 ± 2 E-1 |
| Cd | 2.9 ± 2.7 E-1 | <2.9 E-2 | <3.5 E+0 | 3 ± 3 E-1 |
| In | 8.7 ± 13 E-1 | <4.7 E-2 | <2.9 E+0 | 9 ± 13 E-1 |
| Sn | 7.2 ± 6.5 E-2 | -- | <3.6 E+0 | 7 ± 6 E-2 |
| Gd | <1.1 E-1 | -- | <7.0 E+1 | <1 E-1 |

a. Uncertainty is given at the one-sigma level and is the standard deviation of the set of concentration values.

TABLE 22. PERCENT OF CORE INVENTORY IN THE REACTOR COOLANT DRAIN TANK

| Isotope/ Element | Solids | Water | Total |
|---------------------|---------|----------|---------|
| ³ H | --a | 3.3 E-2 | 3.3 E-2 |
| ⁹⁰ Sr | 5.3 E-2 | 1.0 E-2 | 6.3 E-2 |
| ¹⁰⁶ Ru | 1.6 E-3 | <5.8 E-4 | 1.6 E-3 |
| ¹²⁵ Sb | 2.6 E-3 | 1.0 E-3 | 3.6 E-3 |
| ¹²⁹ I | 6.9 E-7 | 5.5 E-3 | 5.5 E-3 |
| ¹³⁴ Cs | 3.8 E-4 | 4.8 E-3 | 5.2 E-3 |
| ¹³⁷ Cs | 3.4 E-4 | 4.4 E-3 | 4.7 E-3 |
| ¹⁴⁴ Ce | 9.5 E-4 | <1.2 E-4 | 9.5 E-4 |
| U | 1.2 E-4 | <4.4 E-6 | 1.2 E-4 |
| Pu | --a | --a | --a |
| Cr | 1.2 E-2 | --a | 1.2 E-2 |
| Fe | 2.1 E-1 | --a | 2.1 E-1 |
| Ni | 3.8 E-1 | --a | 3.8 E-1 |
| Zr | 2.2 E-4 | --a | 2.2 E-4 |
| Ag | 7.4 E-3 | --a | 7.4 E-3 |
| Cd | 2.3 E-1 | --a | 2.3 E-1 |
| In | 4.0 E-2 | --a | 4.0 E-2 |
| Sn | 3.9 E-2 | --a | 3.9 E-2 |
| Gd | 1.5 E-1 | --a | 1.5 E-1 |

a. Concentration not measured.

TABLE 23. PERCENT OF CORE INVENTORY IN REACTOR COOLANT SYSTEM COOLANT

| Sampling Date (m/d/y) | ³ H (%) | ⁹⁰ Sr (%) | ¹³⁴ Cs (%) | ¹³⁷ Cs (%) |
|-----------------------------|-----------------------|-------------------------|--------------------------|--------------------------|
| 8/29/79 | 2.37 | 0.75 | 2.69 | 2.90 |
| 11/15/79 | 1.94 | 1.05 | 2.12 | 2.36 |
| 5/11/81 | 0.61 | 1.04 | 0.89 | 0.91 |
| 9/21/81 | 0.41 | 0.87 | 0.65 | 0.71 |

were 2.37, 0.75, 2.69, and 2.90%, respectively. Subsequent injections of processed water into the RCS, combined with the removal of existing coolant, reduced the concentrations of ^3H and radiocesium. The dissolution of ^{90}Sr from activity-bearing particulates in the RCS increased the concentration of ^{90}Sr in the coolant. By November 15, 1979, about 1% of the core inventory of ^{90}Sr was dispersed in the RCS coolant.

Surface Samples

Reactor Building Air Coolers

The Reactor Building air cooler access panels arrived at the INEL in August 1983. Coupons were cut from the corners of each panel for surface analysis. A total of 20 coupons were removed from the panels. Each coupon had a total surface area of 9.90 cm^2 per side. Photographs were taken during coupon cutting and removal for determination of sample loss during the cutting process. The access panel coupons were gamma scanned in a reproducible geometry. Then the internal surface was removed, and the coupons were rescanned. Appropriate corrections were made to account for gamma-ray attenuation within the coupons. The internal surface material was then analyzed to determine ^{90}Sr and ^{129}I surface concentrations after separation by a bicarbonate fusion process. Elemental analysis by direct current emission spectroscopy was performed on a portion of the particulate surface material removed from the coupon internal surface. Smear samples and particulate scrape samples retrieved from the access panels and the intake louvers were analyzed onsite by PNL and GPU Nuclear personnel.

Results of the surface activity measurements performed at the INEL are presented in Tables 24 and 25. Table 24 presents the surface activities determined for each individual coupon removed from the access panel corners, and Table 25 presents the average surface activity values for each air cooler access panel. Surface activity values for ^{137}Cs on access panel exterior surfaces range from $0.36\ \mu\text{Ci}/\text{cm}^2$ on coupon C from access panel 11B to $1.7\ \mu\text{Ci}/\text{cm}^2$ on coupon C from access panel 11E. Surface

TABLE 24. REACTOR BUILDING AIR COOLER ACCESS PANEL RADIONUCLIDE SURFACE CONCENTRATIONS
(Data Decay-corrected to March 1, 1984)

| | Exterior Surface | | Interior Surface | | | |
|-------|---|---|---|---|--|--|
| | ^{137}Cs ($\mu\text{Ci}/\text{cm}^2$) | ^{134}Cs ($\mu\text{Ci}/\text{cm}^2$) | ^{137}Cs ($\mu\text{Ci}/\text{cm}^2$) | ^{134}Cs ($\mu\text{Ci}/\text{cm}^2$) | ^{90}Sr ($\mu\text{Ci}/\text{cm}^2$) | ^{129}I ($\mu\text{Ci}/\text{cm}^2$) |
| 11A-A | $5.5 \pm 0.3 \text{ E-1}$ | $3.1 \pm 0.1 \text{ E-2}$ | $6.0 \pm 0.4 \text{ E-1}$ | $3.4 \pm 0.3 \text{ E-2}$ | $1.5 \pm 0.8 \text{ E-2}$ | $5.2 \pm 0.5 \text{ E-6}$ |
| -B | $4.8 \pm 0.3 \text{ E-1}$ | $2.6 \pm 0.1 \text{ E-2}$ | $4.6 \pm 0.4 \text{ E-1}$ | $2.7 \pm 0.2 \text{ E-2}$ | -- | -- |
| -C | $5.8 \pm 0.4 \text{ E-1}$ | $3.2 \pm 0.1 \text{ E-2}$ | $4.7 \pm 0.4 \text{ E-1}$ | $2.6 \pm 0.2 \text{ E-2}$ | -- | -- |
| -D | $4.8 \pm 0.3 \text{ E-1}$ | $2.7 \pm 0.1 \text{ E-2}$ | $1.9 \pm 0.3 \text{ E-1}$ | $1.1 \pm 0.1 \text{ E-2}$ | -- | -- |
| 11B-A | $3.7 \pm 0.1 \text{ E-1}$ | $2.1 \pm 0.1 \text{ E-2}$ | $9.2 \pm 1.6 \text{ E-2}$ | $5.4 \pm 1.1 \text{ E-3}$ | -- | -- |
| -B | $4.0 \pm 0.2 \text{ E-1}$ | $2.2 \pm 0.1 \text{ E-2}$ | $8.9 \pm 2.2 \text{ E-2}$ | $6.0 \pm 1.1 \text{ E-3}$ | -- | -- |
| -C | $3.6 \pm 0.2 \text{ E-1}$ | $2.0 \pm 0.1 \text{ E-2}$ | $1.5 \pm 0.2 \text{ E-1}$ | $8.4 \pm 1.1 \text{ E-3}$ | -- | -- |
| -D | $3.9 \pm 0.2 \text{ E-1}$ | $2.2 \pm 0.1 \text{ E-2}$ | $1.2 \pm 0.2 \text{ E-1}$ | $6.8 \pm 1.1 \text{ E-3}$ | $6.4 \pm 0.2 \text{ E-3}$ | -- |
| 11C-A | $1.6 \pm 0.2 \text{ E+0}$ | $8.6 \pm 0.4 \text{ E-2}$ | $9.4 \pm 1.0 \text{ E-1}$ | $5.0 \pm 0.5 \text{ E-2}$ | -- | -- |
| -B | $9.0 \pm 0.7 \text{ E-1}$ | $4.9 \pm 0.2 \text{ E-2}$ | $8.4 \pm 0.7 \text{ E-1}$ | $4.8 \pm 0.4 \text{ E-2}$ | -- | -- |
| -C | $1.5 \pm 0.2 \text{ E+0}$ | $8.0 \pm 0.3 \text{ E-2}$ | $4.1 \pm 6.8 \text{ E-2}^b$ | | $4.9 \pm 0.4 \text{ E-3}^b$ | |
| -- | $6.9 \pm 0.7 \text{ E-6}$ | | | | | |
| -D | $8.9 \pm 0.7 \text{ E-1}$ | $4.9 \pm 0.2 \text{ E-2}$ | $7.6 \pm 0.7 \text{ E-1}$ | $4.3 \pm 0.4 \text{ E-2}$ | -- | -- |
| 11D-A | $3.9 \pm 0.2 \text{ E-1}$ | $2.1 \pm 0.1 \text{ E-2}$ | $3.3 \pm 0.3 \text{ E-1}$ | $1.9 \pm 0.1 \text{ E-2}$ | $7.6 \pm 0.4 \text{ E-3}$ | $2.7 \pm 0.3 \text{ E-6}$ |
| -B | $3.9 \pm 0.2 \text{ E-1}$ | $2.2 \pm 0.1 \text{ E-2}$ | $2.7 \pm 0.2 \text{ E-1}$ | $1.5 \pm 0.1 \text{ E-2}$ | -- | -- |
| -C | $5.9 \pm 0.4 \text{ E-1}$ | $3.2 \pm 0.2 \text{ E-2}$ | $4.1 \pm 0.4 \text{ E-1}$ | $2.3 \pm 0.2 \text{ E-2}$ | -- | -- |
| -D | $5.4 \pm 0.3 \text{ E-1}$ | $2.9 \pm 0.1 \text{ E-2}$ | $2.1 \pm 0.3 \text{ E-1}$ | $1.2 \pm 0.2 \text{ E-2}$ | -- | -- |
| 11E-A | $1.4 \pm 0.2 \text{ E+0}$ | $7.9 \pm 0.3 \text{ E-2}$ | $2.0 \pm 0.1 \text{ E+0}$ | $1.2 \pm 0.1 \text{ E-1}$ | $4.3 \pm 0.2 \text{ E-2}$ | $7.5 \pm 0.8 \text{ E-6}$ |
| -B | $8.7 \pm 0.7 \text{ E-1}$ | $4.8 \pm 0.2 \text{ E-2}$ | $2.2 \pm 0.1 \text{ E+0}$ | $1.3 \pm 0.1 \text{ E-1}$ | -- | -- |
| -C | $1.7 \pm 0.2 \text{ E+0}$ | $9.4 \pm 0.4 \text{ E-2}$ | $1.4 \pm 0.1 \text{ E+0}$ | $8.2 \pm 0.1 \text{ E-2}$ | -- | -- |
| -D | $1.1 \pm 0.1 \text{ E+0}$ | $5.7 \pm 0.2 \text{ E-2}$ | $2.8 \pm 0.2 \text{ E+0}$ | $1.5 \pm 0.1 \text{ E-1}$ | -- | -- |

a. The 1 sigma errors quoted are due to counting errors only.

b. Data not used for calculating the average concentration due to scraping of the sample surface during the coupon cutting operation.

TABLE 25. REACTOR BUILDING AIR COOLER ACCESS PANEL AVERAGE RADIONUCLIDE SURFACE CONCENTRATIONS IN $\mu\text{Ci}/\text{cm}^2$ ^a

| | Exterior Surface | | Interior Surface | | | |
|------|---------------------------|---------------------------|---------------------------|---------------------------|-----------------------------|-----------------------------|
| | ^{137}Cs | ^{134}Cs | ^{137}Cs | ^{134}Cs | ^{90}Sr | ^{129}I |
| 11A | $5.2 \pm 0.5 \text{ E-1}$ | $2.9 \pm 0.3 \text{ E-2}$ | $4.3 \pm 1.7 \text{ E-1}$ | $2.5 \pm 1.0 \text{ E-2}$ | $1.5 \pm 0.8 \text{ E-2}^b$ | $5.2 \pm 0.5 \text{ E-6}^b$ |
| 11B | $3.8 \pm 0.2 \text{ E-1}$ | $2.1 \pm 0.1 \text{ E-2}$ | $1.1 \pm 0.3 \text{ E-1}$ | $6.7 \pm 1.3 \text{ E-3}$ | $6.4 \pm 0.2 \text{ E-3}^b$ | -- |
| 11C | $1.2 \pm 0.4 \text{ E-0}$ | $6.6 \pm 2.0 \text{ E-2}$ | $8.5 \pm 0.9 \text{ E-1}$ | $4.7 \pm 0.4 \text{ E-2}$ | -- | $6.9 \pm 0.7 \text{ E-6}^b$ |
| 11D | $4.8 \pm 1.0 \text{ E-1}$ | $2.6 \pm 0.5 \text{ E-2}$ | $3.1 \pm 0.9 \text{ E-1}$ | $1.7 \pm 0.5 \text{ E-2}$ | $7.6 \pm 0.4 \text{ E-3}^b$ | $2.7 \pm 0.3 \text{ E-6}^b$ |
| 11E | $1.3 \pm 0.4 \text{ E+0}$ | $7.0 \pm 2.0 \text{ E-2}$ | $2.1 \pm 0.6 \text{ E+0}$ | $1.2 \pm 0.3 \text{ E-1}$ | $4.3 \pm 0.2 \text{ E-2}^b$ | $7.5 \pm 0.8 \text{ E-6}^b$ |
| Mean | $7.8 \pm 4.4 \text{ E-1}$ | $4.2 \pm 2.4 \text{ E-2}$ | $7.6 \pm 8.0 \text{ E-1}$ | $4.3 \pm 4.5 \text{ E-2}$ | $1.8 \pm 1.7 \text{ E-2}$ | $5.6 \pm 2.2 \text{ E-6}$ |

a. All errors quoted represent standard deviation.

b. The 1 sigma errors are due to counting errors only, because just one sample analysis result was available.

activity values for ^{134}Cs range from $0.02 \mu\text{Ci}/\text{cm}^2$ to $0.094 \mu\text{Ci}/\text{cm}^2$ on those same two access panels. Interior surface concentrations of ^{137}Cs range from $0.089 \mu\text{Ci}/\text{cm}^2$ on coupon B from access panel 11B to $2.8 \mu\text{Ci}/\text{cm}^2$ on coupon D from access panel 11E, while corresponding ^{134}Cs surface activities range from $0.006 \mu\text{Ci}/\text{cm}^2$ to $0.15 \mu\text{Ci}/\text{cm}^2$. The average concentrations of ^{134}Cs and ^{137}Cs on exterior surfaces of the access panels are greater than their average values on the interior surfaces of the panels, with the exception of access panel 11E. The average interior surface activity concentrations on this panel were approximately twice the exterior values.

Tables 24 and 25 also present the surface concentrations of ^{90}Sr and ^{129}I measured on the Reactor Building air cooler access panels. Analyses for ^{90}Sr and ^{129}I were performed on only a few of the access panel coupon interior surfaces due to the nature and cost of the analyses. Results for ^{90}Sr surface activities range from $0.0064 \mu\text{Ci}/\text{cm}^2$ on access panel 11B to $0.043 \mu\text{Ci}/\text{cm}^2$ on access panel 11E. The mean value of the surface concentration of ^{90}Sr is $0.018 \pm 0.017 \mu\text{Ci}/\text{cm}^2$. Results for ^{129}I surface activities range from $2.7 \text{ E-}6 \mu\text{Ci}/\text{cm}^2$ on access panel 11D to $7.5 \text{ E-}6 \mu\text{Ci}/\text{cm}^2$ on access panel 11E. The average value of the concentration of ^{129}I on the interior surfaces of the access panels is $5.6 \pm 2.2 \text{ E-}6 \mu\text{Ci}/\text{cm}^2$. Surface concentrations of ^{90}Sr on access panel 11C and of ^{129}I on access panel 11B are not available because these analyses were not performed.

The average interior and exterior surface concentrations of ^{90}Sr , ^{129}I , and ^{137}Cs measured on the five access panels are plotted in Figure 7. As the figure illustrates, increased surface concentrations of ^{134}Cs and ^{137}Cs on the access panels are accompanied by increased surface concentrations of ^{90}Sr and ^{129}I . The figure also illustrates the nonuniformity of surface activity concentrations on adjacent air cooler units. There are several possible explanations for the discontinuity in the surface concentrations measured on adjacent air coolers. First, there is some confusion as to which Reactor Building air coolers operated during the accident. Control room logs indicate that all Reactor Building air coolers were on during the accident. Other references state that only

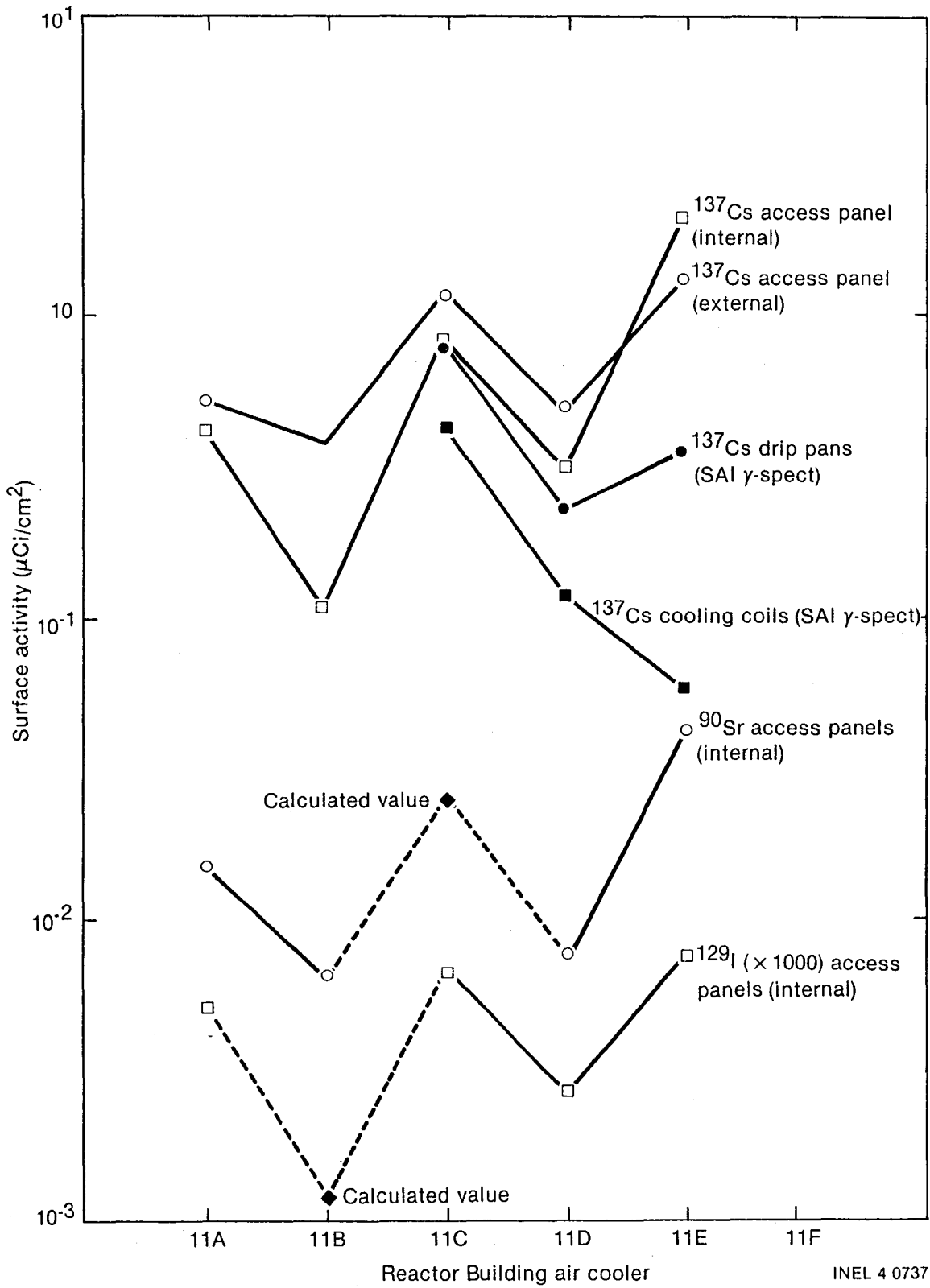


Figure 7. Radionuclide surface concentrations of the Reactor Building air coolers.

three Reactor Building air coolers were running as designed, with two coolers providing redundant backup.³⁰ Which three air coolers were operating is unspecified.

It is likely that volumetric air flow through the coolers varied, and as a consequence different amounts of fission products were deposited on the air cooler internal surfaces. Factors affecting airflow through the individual coolers include backdraft damper positions and backpressure restrictions within the air cooler common plenum and ventilation ducts. Postaccident inspections of the air cooler backdraft dampers indicated that material deposits on switch control rods impeded the operation of two backdraft dampers. The dampers are designed to close by gravity, in the absence of airflow, or by backpressure when the fans are on. Manual operation of the dampers indicated that dampers 11A and 11E were difficult to open and close, while 11B, 11C, and 11D were quite easy to manually operate. Another factor that could have affected the airflow is a backpressure differential in the air cooler common plenum and ventilation ducts. Due to the nature of the air cooling system design (five cooling units connected to one common plenum having two ventilation headers), variable backpressures very likely existed in the common plenum, resulting in unbalanced rates of air flow through the individual coolers. Operations history indicates that air flow through the five cooling units was not balanced.

Still another factor that could have affected the fission product surface concentrations is the variable rate at which moisture condensed on the air cooler surfaces. Different condensing rates within the individual air cooler units would have caused fission products to be washed from the surfaces at different rates. The condensing rates were affected by the rates at which air flowed through the coolers and coolant flowed through the cooling coils. Operations history revealed that although throttle valves are normally used to balance the quantities of coolant flowing through the air cooler coils, they were not used in this manner at TMI-2. The throttle valves were all completely opened during emergency cooling after the H₂ burn. Finally, it is important to note that Reactor Building air coolers 11A, 11B, and 11C were decontaminated with low-pressure

processed water prior to the acquisition of the access panel samples. Coolers 11D and 11E had not been decontaminated when their access panels were acquired. However, cooler 11D was turned off in June 1979 due to a leak in a pressure relief valve. All of these factors would have affected the rates at which fission products were deposited on or washed from internal cooler surfaces.

The values of the ratios of ^{90}Sr to ^{137}Cs surface activities on the air cooler access panels are similar to the values of the ratios of their earlier concentrations in the basement water. This suggests that ^{90}Sr and ^{137}Cs were transported to the surfaces of the air cooling assembly from the basement water trapped in water droplets. The values of these ratios are consistent with other ratios of ^{90}Sr to ^{137}Cs activities measured on the vertical metal surfaces at the 305-ft elevation (see Reference 4). The values of the ratios of ^{129}I to ^{137}Cs and ^{129}I to ^{90}Sr concentrations on the access panels are consistent with their measured ratios on other vertical metal surfaces at the 305-ft elevation, but they are unlike their corresponding ratios in earlier RCS water, basement water, or concrete surface samples.

Table 26 compares the average surface activities measured on the access panels with the surface activities measured on other 305-ft elevation vertical metal surfaces. For ^{90}Sr and ^{129}I , average surface activities on the access panels are in good agreement with average values for other vertical metal surfaces at the 305-ft elevation. Average surface concentrations of ^{134}Cs and ^{137}Cs on the access panels appear to be about 2-1/2 times their corresponding concentrations on 305-ft elevation vertical metal surfaces, although the former values are statistically in agreement with the latter values. It is important to note that if we exclude the maximum values observed on the 11C and 11E access panels, the average access panel internal surface concentration of ^{137}Cs becomes $0.28 \pm 0.16 \mu\text{Ci}/\text{cm}^2$, which is in excellent agreement with the corresponding 305-ft-elevation metal surface values of $0.29 \pm 0.10 \mu\text{Ci}/\text{cm}^2$. This suggests that conditions within the 11A, 11B, and 11D air coolers were unlike conditions in the 11C and 11E air coolers. Surface activity

TABLE 26. COMPARISON OF REACTOR BUILDING AIR COOLER RADIONUCLIDE SURFACE CONCENTRATIONS WITH 305-FT-ELEVATION VERTICAL METAL RADIONUCLIDE SURFACE CONCENTRATIONS
(Data Decay-corrected to March 1, 1984)

| Nuclide | Vertical Metal Surfaces ^a ($\mu\text{Ci}/\text{cm}^2$) | Cooler External Surface ($\mu\text{Ci}/\text{cm}^2$) | Cooler Internal Surface ($\mu\text{Ci}/\text{cm}^2$) |
|-------------------|--|---|---|
| ⁹⁰ Sr | $1.6 \pm 0.2 \text{ E-}2$ | -- | $1.8 \pm 1.7 \text{ E-}2$ |
| ¹²⁹ I | $4.2 \pm 0.8 \text{ E-}6$ | -- | $5.6 \pm 2.2 \text{ E-}6$ |
| ¹³⁴ Cs | $1.6 \pm 0.5 \text{ E-}2$ | $4.2 \pm 2.4 \text{ E-}2$ | $4.3 \pm 4.5 \text{ E-}2$ |
| ¹³⁷ Cs | $2.9 \pm 1.0 \text{ E-}1$ | $7.8 \pm 4.4 \text{ E-}1$ | $7.6 \pm 8.0 \text{ E-}1$ |

a. C. V. McIsaac, Surface Activity and Radiation Field Measurements of the TMI-2 Reactor Building Gross Decontamination Experiment, GEND-037, October 1983.

concentrations on the 11C and 11E access panels are as much as 5 times greater than the average 305-ft elevation metal surface values.

Results of the in situ gamma spectral measurements of the Reactor Building cooling coils and drip pans are presented in Table 27. The scans were completed in October 1981 by Science Applications, Inc. (SAI) (see Reference 9), but the results are decay-corrected to March 1, 1984 in the table to allow direct comparison of the results. Surface activity values in $\mu\text{Ci}/\text{cm}^2$ were obtained by dividing the reported total activities on the air cooler coils by the calculated surface area of the air cooler coil (see Appendix A). Calculated surface activity values for ^{137}Cs range from $0.059 \pm 0.029 \mu\text{Ci}/\text{cm}^2$ on air cooler coil 11E to $0.43 \pm 0.07 \mu\text{Ci}/\text{cm}^2$ on air cooler coil 11C. Calculated surface activity values for ^{137}Cs on the 11C and 11D cooling coils are within a factor of 2.5 of the ^{137}Cs surface activities measured on the corresponding access panel internal surfaces, while the ^{137}Cs surface activity calculated for the 11E cooling coils is a factor of 36 lower than its average value on the corresponding access panel internal of the 11E access panel. The lower activity concentrations on the cooling coils, compared to the concentrations on the access panels, are likely due to the cleansing effect of the water vapor that condensed on the coils.

Surface activity concentrations of ^{137}Cs on the air cooler drip pans are also reported in Table 27. These values range from $0.23 \pm 0.12 \mu\text{Ci}/\text{cm}^2$ on drip pan 11D to $0.81 \pm 0.35 \mu\text{Ci}/\text{cm}^2$ on drip pan 11C. The surface activity values for ^{137}Cs on the 11C and 11D air cooler drip pans are in excellent agreement with the concentrations of ^{137}Cs measured on the internal surfaces of the corresponding access panels, while the results for the 11E air cooler drip pan are a factor of 6 lower than the average concentration of ^{137}Cs measured on the internal surface of the corresponding access panel.

Table 28 presents the total quantities of ^{134}Cs , ^{137}Cs , ^{90}Sr , and ^{129}I estimated to be present on the Reactor Building air cooler surfaces. The results are expressed as percent of the original core inventory. Total surface activities were calculated by multiplying the

TABLE 27. REACTOR BUILDING AIR COOLER RADIONUCLIDE SURFACE CONCENTRATIONS^a
(Data Decay-corrected to March 1, 1984)

| Air Cooler Number | ¹³⁷ Cs on Cooling Coils | | ¹³⁷ Cs on Drip Pans | |
|-------------------|------------------------------------|---|--------------------------------|---|
| | (total Ci) | ($\mu\text{Ci}/\text{cm}^2$) ^b | (total Ci) | ($\mu\text{Ci}/\text{cm}^2$) ^c |
| 11C | 5.6 ± 1.0 E+0 | 4.3 ± 0.7 E-1 | 6.6 ± 2.8 E-2 | 8.1 ± 3.5 E-1 |
| 11D | 1.5 ± 0.4 E+0 | 1.2 ± 0.3 E-1 | 1.9 ± 1.0 E-2 | 2.3 ± 1.2 E-1 |
| 11E | 7.6 ± 4.0 E-1 | 5.9 ± 2.9 E-2 | 2.8 ± 2.8 E-2 | 3.5 ± 3.5 E-1 |

a. Quoted uncertainties include statistical uncertainties, estimates of the uncertainties in the detector efficiency calibration, and uncertainties in the data analysis.

b. Assumes surface area of coil $\sim 1.3 \text{ E}+7 \text{ cm}^2$ (from Appendix A).

c. Assumes surface area of drip pans $\sim 8.2 \text{ E}+4 \text{ cm}^2$.

TABLE 28. PERCENT OF TMI-2 FISSION PRODUCT CORE INVENTORY MEASURED ON REACTOR BUILDING AIR COOLER SURFACES
(Activities Decay-corrected to March 1, 1984)

| Nuclide | Total Surface Activity (Ci) | Total Core Inventory ^a (Ci) | Percent of Core Inventory |
|-------------------|-----------------------------|--|---------------------------|
| ¹³⁴ Cs | 3.3 E+0 | 4.1 E+4 | 8.1 E-3 |
| ¹³⁷ Cs | 5.9 E+1 | 7.6 E+5 | 7.7 E-3 |
| ⁹⁰ Sr | 1.4 E+0 | 6.9 E+5 | 2.0 E-4 |
| ¹²⁹ I | 4.3 E-4 | 2.0 E-1 | 2.2 E-1 |

a. Based on a combination of LOR2 and ORIGEN inventories as reported in an SAI letter report from J. A. Daniel and E. A. Schlomer to G. R. Eidam (SAI-83-095), March 9, 1983.

average internal surface activity concentrations measured on the air cooler access panels times the total surface area of the air coolers. The total surface area of the air cooling assembly, $7.7 \text{ E}+7 \text{ cm}^2$, excluding the ventilation duct work, was obtained by adding the surface area of the cooling coils, $\sim 6.8 \text{ E}+7 \text{ cm}^2$, to the surface area of the air cooling assembly housing, $1.0 \text{ E}+7 \text{ cm}^2$. Expressed as percent of core inventory, the quantities of ^{134}Cs and ^{137}Cs on the air cooling assembly are $8.1 \text{ E}-3$ and $7.7 \text{ E}-3\%$, respectively. The quantity of ^{90}Sr estimated to be on the surfaces of the air cooling assembly is $2.0 \text{ E}-4\%$ of the original core inventory. The percent of the core inventory of ^{129}I on the surfaces of the assembly is 0.22% . The value for the ^{129}I core fraction is about 4 times greater than the previously calculated ^{129}I fraction for all Reactor Building surfaces (see Reference 4).

Table 29 presents a compilation of smear and scrape data collected at various locations on the Reactor Building air coolers. The results are presented as total activity per sample; therefore, quantitative comparisons with other data are not possible. Detectible amounts of ^{106}Ru and ^{144}Ce were observed on a few of the 11A intake louver smears, which indicates that fuel debris particulates were probably present on the intake louvers. Because air is deflected 90 degrees or more during its passage through the intake louvers, one would expect that a significant amount of particle impingement occurred inside the louvered inlets. Dose rates on the intake louvers have ranged as high as 1000 mR/h . Coupled with the identification of ^{106}Ru and ^{144}Ce , the survey data indicate that fission product and fuel particulates are deposited on the louvers.

TABLE 29. REACTOR BUILDING AIR COOLER ACCESS PANEL COMPILED SMEAR DATA
(Data Decay-corrected to May 31, 1984)

| Description | ⁹⁰ Sr (total μ/Ci) | ¹³⁴ Cs (total μ/Ci) | ¹³⁷ Cs (total μ/Ci) | ¹⁴⁴ Ce (total μ/Ci) |
|------------------|----------------------------------|-----------------------------------|-----------------------------------|-----------------------------------|
| 11Aa,b | 4.4 ± 0.3 E-1 | 4.2 ± 0.1 E-1 | 5.5 ± 0.1 E+0 | 8.8 ± 0.9 E-3 |
| 11Aa,c | 2.4 ± 0.4 E-1 | 2.2 ± 0.1 E+0 | 3.0 ± 0.1 E+1 | --d |
| 11Aa,e | 6.2 ± 0.3 E+1 | 7.8 ± 0.1 E+0 | 1.1 ± 0.1 E+2 | 4.5 ± 1.9 E-1 |
| 11Aa | 1.7 ± 0.1 E+1 | 3.1 ± 0.1 E+0 | 4.7 ± 0.1 E+1 | --d |
| 11A | 1.7 ± 0.1 E-1 | 1.4 ± 0.1 E-1 | 2.0 ± 0.1 E+0 | --d |
| 11A | 4.0 ± 0.2 E-1 | 1.3 ± 0.1 E-1 | 1.9 ± 0.1 E+0 | 2.2 ± 0.6 E-3 |
| 11B | 2.0 ± 0.1 E-1 | 1.3 ± 0.1 E-2 | 2.0 ± 0.1 E-1 | --d |
| 11C | 9.4 ± 0.9 E-2 | 1.2 ± 0.1 E-1 | 1.7 ± 0.1 E+0 | --d |
| 11C | 2.6 ± 0.3 E-2 | 3.7 ± 0.1 E-2 | 5.3 ± 0.1 E-1 | --d |
| 11C | 2.0 ± 0.3 E-2 | 4.5 ± 0.1 E-2 | 6.9 ± 0.1 E-1 | --d |
| 11C | 1.7 ± 0.3 E-2 | 2.9 ± 0.1 E-2 | 4.4 ± 0.1 E-1 | --d |
| 11C | 1.1 ± 0.1 E-1 | 5.1 ± 0.1 E-2 | 7.7 ± 0.1 E-1 | --d |
| 11D | 3.2 ± 0.3 E-1 | 7.1 ± 0.1 E-1 | 6.8 ± 0.1 E+0 | --d |
| 11D ^c | 6.8 ± 0.1 E-1 | 1.5 ± 0.1 E+0 | 2.2 ± 0.1 E+1 | --d |
| 11E | 2.2 ± 0.2 E-1 | 2.4 ± 0.1 E-1 | 3.6 ± 0.1 E+0 | --d |
| 11E ^c | 1.4 ± 0.2 E-1 | 2.6 ± 0.1 E-1 | 4.0 ± 0.1 E+0 | --d |

a. Sample removed from the intake louvers.

b. ¹⁰⁶Ru was identified at 1.2 ± 0.3 μCi/sample.

c. Scrape sample.

d. Lower limit of detection.

e. ¹⁰⁶Ru and ¹²⁹I were identified at 3.3 ± 0.9(-1) and 5.9 ± 0.8(-5) μCi/sample, respectively.

CONCLUSIONS

Following are the conclusions of the radionuclide measurements of the TMI-2 Reactor Building surfaces and basement water and sediment.

- o The total quantities of fission products, fuel, and core materials present in the Reactor Building, excluding the reactor core region and building atmosphere, are summarized in Table 30. Measurements made from August 1979 to December 1983 have accounted for 59% of the ^3H , 2.7% of the ^{90}Sr , 15% of the ^{129}I , 20% of the ^{131}I , and 42% of the ^{137}Cs originally in the core at the time of the accident. With the exception of ^{90}Sr and ^{144}Ce , the vast majority of the total quantity of each radionuclide released was found dispersed in the water and sediment in the basement.

- o Measurement results show that about 57% of the ^{129}I originally in solution in the basement water in August 1979 was lost from the water by March 1981. The quantity of ^{129}I in solution decreased from about 14 to about 6% of the core inventory during the 19 months between samplings. Based on the measured average concentration, the total amount of ^{129}I in the sediment on the basement floor is 8% of the core inventory, indicating that precipitation was probably the dominant depletion mechanism.

- o Measurement results show that Cu and Ag are among the major elements present in the sediment on the basement floor. The data indicate there is a good correlation between the concentrations of Cu and Ag and concentration of ^{129}I in the sediment. Most of the insoluble iodine in the sediment is probably present in the form of Cu and Ag iodide; iodide was measured to be the predominant species of iodine in solution.

- o The data summarized in Table 30 show that the quantity of ^{129}I on Reactor Building surfaces is relatively small. The quantity measured on structural surfaces is about 0.06% of the original

TABLE 30. TOTAL QUANTITIES OF FISSION PRODUCTS, FUEL, AND CORE MATERIALS
IN THE TMI-2 REACTOR BUILDING

| Isotope/ Element | Percent of Core Inventory | | | | | |
|---------------------|---------------------------|-------------------------------------|------------------------------|-------------------------|-------------------------------------|---------|
| | Basement | Reactor Coolant Drain Tank | Reactor Coolant System | Air Cooling Assembly | Structural Surfaces ^a | Total |
| ³ H | 5.7 E+1 | 3.3 E-2 | 2.2 E+0 | --b | --b | 5.9 E+1 |
| ⁹⁰ Sr | 1.7 E+0 | 6.3 E-2 | 9.6 E-1 | 2.0 E-4 | 2.3 E-3 | 2.7 E+0 |
| ¹⁰⁶ Ru | 4 E-3 | 1.6 E-3 | --b | --b | --b | 5.6 E-3 |
| ¹²⁵ Sb | 3 E-1 | 3.6 E-3 | --b | --b | 1.1 E-2 | 3.1 E-1 |
| ¹²⁹ I | 1.4 E+1 | 5.5 E-3 | 1.2 E+0 ^c | 2.2 E-1 | 6.4 E-2 | 1.5 E+1 |
| ¹³¹ I | 2.0 E+1 | --b | --b | --b | --b | 2.0 E+1 |
| ¹³⁴ Cs | 4.2 E+1 | 5.2 E-3 | 7.7 E-1 | 8.1 E-3 | 3.5 E-2 | 4.3 E+1 |
| ¹³⁷ Cs | 4.1 E+1 | 4.7 E-3 | 8.1 E-1 | 7.7 E-3 | 3.5 E-2 | 4.2 E+1 |
| ¹⁴⁴ Ce | 1.0 E-3 | 9.5 E-4 | --b | --b | --b | 2.0 E-3 |
| U | 8 E-4 | 1.2 E-4 | --b | --b | --b | 9.2 E-4 |
| Pu | 6 E-4 | --b | --b | --b | --b | 6 E-4 |
| Cr | 5 E-1 | 1.2 E-2 | --b | --b | --b | 5.1 E-1 |
| Fe | 1 E+0 | 2.1 E-1 | --b | --b | --b | 1.2 E+0 |
| Ni | 8 E-1 | 3.8 E-1 | --b | --b | --b | 1.2 E+0 |
| Zr | 4 E-4 | 2.2 E-4 | --b | --b | --b | 6.2 E-4 |
| Ag | 2 E-1 | 7.4 E-3 | --b | --b | --b | 2.1 E-1 |
| Cd | 3 E-1 | 2.3 E-1 | --b | --b | --b | 5.3 E-1 |
| In | 9 E-1 | 4.0 E-2 | --b | --b | --b | 9.4 E-1 |
| Sn | 7 E-2 | 3.9 E-2 | --b | --b | --b | 1.1 E-1 |
| Gd | <1 E-1 | 1.5 E-1 | --b | --b | --b | 1.5 E-1 |

a. Total surface activities were reported in Surface Activity and Radiation Field Measurements of the TMI-2 Reactor Building Gross Decontamination Experiments, GEND-037, October 1983, p. 102.

b. Not measured.

c. August 14, 1980 reactor coolant system coolant sample.

core inventory, and the amount estimated to be on the air cooling assembly surfaces is about 0.2% of the original core inventory. Measurements made on samples of Reactor Building air collected beginning 79 h after the onset of the accident have shown the quantities of airborne iodine were negligible, ranging from 0.002 to 0.03% of the core inventory.³¹

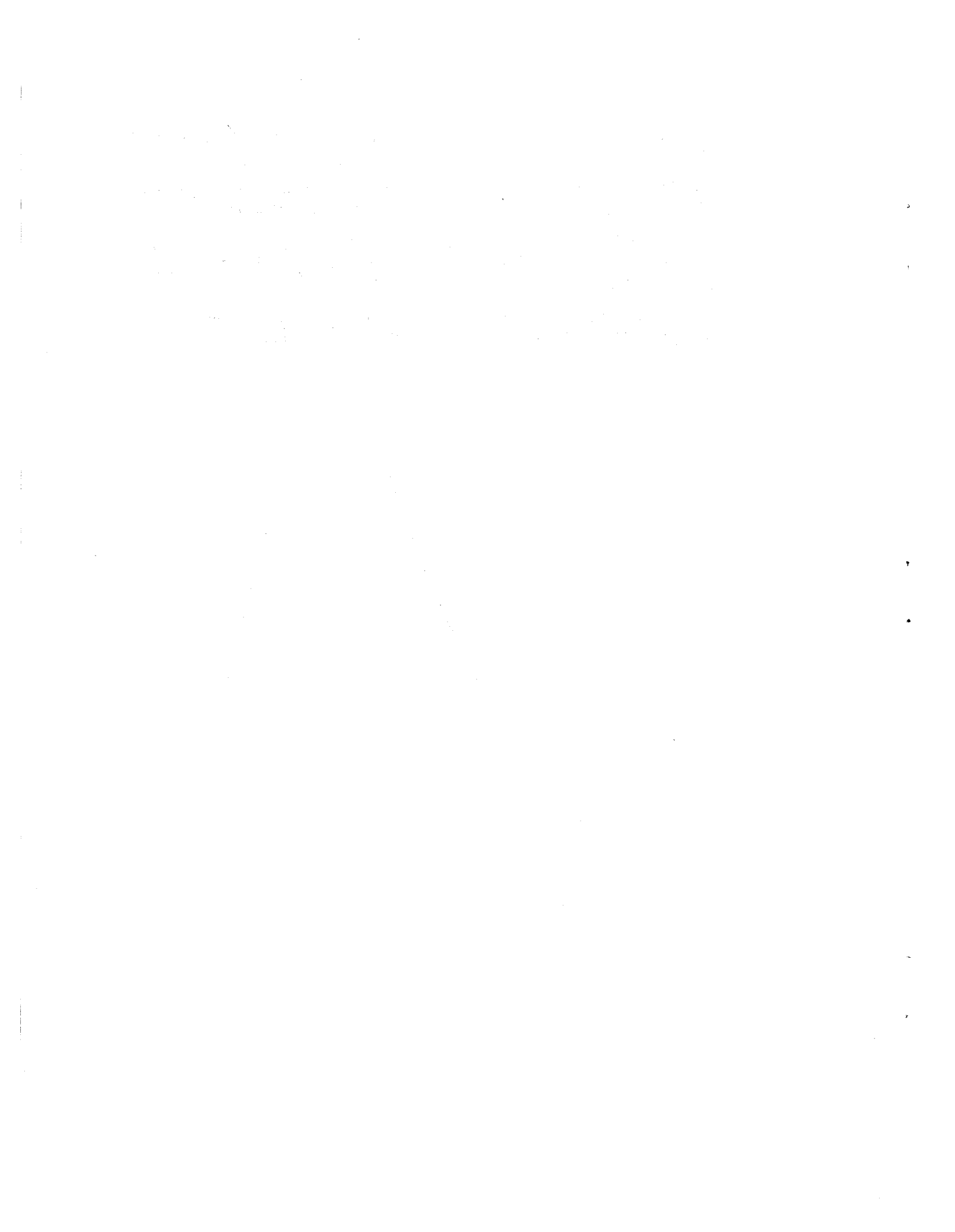
- o The relative quantities of Ag, In, and Cd that were measured in the particulate matter in the reactor coolant drain tank (RCDT) are significantly different than their original ratios in the control rod alloy. This finding indicates that the control rod alloy was not dispersed in the reactor coolant system as fractionated bulk material; rather these metals were transported to the RCDT as condensed vapors.
- o Measurements of the ²³⁵U enrichment of the fuel in the sediment on the basement floor show an average enrichment that is very close to the core average enrichment, which indicates the release of U-bearing particles may have been uniform across the core.
- o The two radionuclides that account for most of the activity in the basement sediment are ⁹⁰Sr and ¹³⁷Cs. Their respective average concentrations in the sediment are about 1.4 and 0.9 mCi per dry gram. Video surveys of the basement combined with calculations indicate that the total mass and packed volume of the sediment are 380 kg and about 6 m³, respectively.

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APPENDIX A
SURFACE AREA CALCULATION OF
TMI-2 REACTOR BUILDING AIR COOLING COILS

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$$\begin{aligned} \text{A. Surface Area coil assembly} &= \text{Surface Area fin plates} \\ &+ \text{Surface Area coolant tubes} \\ &+ \text{Surface Area support plates} \end{aligned}$$

$$\begin{aligned} 1. \text{ Surface Area fin plate} &= \text{Surface Area fins} \\ &- \text{Surface Area tube holes} \\ &= (27 \times 20) - 216 \text{ tube holes} \\ &\quad (\pi \cdot 3.125^2) \\ &= [(540) - (66.27)] \frac{2 \text{ sides}}{\text{fin}} \\ &= 947.5 \text{ in.}^2 \end{aligned}$$

Finned length is 54 in., 6 fins/in. (from vendor reference drawing number 24-00-0103).

$$\text{Surface Area total fin plates} = 947.5 \times 54 \text{ in.} \times \frac{6 \text{ fins}}{\text{in.}}$$

$$= 3.07 \text{ E}+5 \text{ in.}^2$$

$$2. \text{ Surface Area coolant tubes} = [2\pi (.3125)54] 216 \text{ tubes}$$

$$= 2.29 \text{ E}+4 \text{ in.}^2$$

$$3. \text{ Surface Area support plates} = \text{Surface Area side plates} \\ + \text{Surface Area end and center plates}$$

$$= [(29 \times 56 \times 2 \text{ plates}) \\ + (947.5 \times 6 \text{ plates})] \frac{2 \text{ sides}}{\text{plate}}$$

$$= 1.79 \text{ E}+4 \text{ in.}^2$$

$$4. \text{ Surface Area coil assembly} = (3.07 \text{ E}+5) + (2.29 \text{ E}+4) \\ + (1.79 \text{ E}+4)$$

$$= 3.48 \text{ E}+5 \text{ in.}^2/\text{coil assembly} \\ \text{or } 2.24 \text{ E}+6 \text{ cm}^2/\text{coil assembly}$$

There are six coil assemblies per air cooler and five Reactor Building air coolers.

$$B. \text{ Total Surface Area on Reactor Building air cooling coils} \\ = 3.48 \text{ E}+5 \times \frac{6 \text{ coil assembly}}{\text{cooler}} \times 5 \text{ coolers.}$$

$$\text{Total Surface Area on Reactor Building air cooling coils} \\ = 1.04 \text{ E}+7 \text{ in.}^2 \text{ or } 6.75 \text{ E}+7 \text{ cm}^2.$$

