Residual Radionuclide Distribution and Inventory at Dresden Nuclear Power Station Unit No. 1

April 1984

Prepared for
the U.S. Nuclear Regulatory Commission
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

TOPICAL REPORT

RESIDUAL RADIONUCLIDE DISTRIBUTION AND
INVENTORY AT THE DRESDEN UNIT ONE NUCLEAR
GENERATING PLANT

K. H. Abel
D. E. Robertson
E. A. Lepel
M. W. Leale

April, 1984

Prepared for
the U.S. Nuclear Regulatory Commission
under Contract DE-AC06-76RLO 1830
NRC FIN B2299

Pacific Northwest Laboratory
Richland, Washington 99352

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
under Contract DE-AC06-76RLO 1830

TABLE OF CONTENTS

EXECUTIVE SUMMARY	iii
LIST OF FIGURES	vii
LIST OF TABLES	viii
1.0 INTRODUCTION	1
2.0 SAMPLING AND MEASUREMENTS PROGRAM	2
2.1 PIPING AND HARDWARE	3
2.2 CONCRETE CORES	6
2.3 SOIL SAMPLING	7
2.4 SAMPLE HANDLING AND ANALYSES	7
3.0 RESIDUAL RADIONUCLIDE CONCENTRATIONS	9
3.1 REACTOR SYSTEMS CORROSION FILMS	9
3.2 WASTE DISPOSAL CONSIDERATIONS	16
3.3 CONCRETE	18
3.4 SOILS	22
4.0 ESTIMATED RESIDUAL RADIONUCLIDE INVENTORIES	22
4.1 PIPING AND HARDWARE	24
4.2 CONCRETE	32
4.3 SOILS	32
5.0 CONCLUSIONS	33
REFERENCES	36
APPENDIX A - DESCRIPTION OF DRESDEN NUCLEAR PLANT, UNIT ONE	
APPENDIX B - DRESDEN UNIT ONE OPERATING HISTORY	
APPENDIX C - SAMPLE INVENTORY FOR DRESDEN NUCLEAR POWER STATION UNIT ONE	

EXECUTIVE SUMMARY

The residual inventory and distribution of long-lived radionuclides associated with Dresden Unit One have been estimated as part of a generic assessment of residual radioactivity in light water nuclear power stations. The programmatic goal is to provide the NRC with a data base for use in establishing strategies and policies for the decommissioning of nuclear power plants after power generation ceases.

The Dresden Nuclear Station is located near Morris, Illinois, approximately 100 km southwest of Chicago, Illinois. The plant is a dual cycle 700 MW_t Boiling Water Reactor (BWR) which operated commercially from July, 1960 to October, 1978.

In August, 1982, Pacific Northwest Laboratory (PNL) conducted a sampling and measurement program at Dresden One. Samples of corrosion films associated with piping and plate surfaces, concrete, and soils were collected and measured to determine the quantities and composition of the residual radionuclides in and around the nuclear unit (excluding neutron-activated parts of the pressure vessel, the biological shield, residues and resins in tanks and sumps, and the spent fuel stored on site). The samples were analyzed for the following wide spectrum of long-lived gamma, beta, x-ray, and alpha-emitting radionuclides ⁵⁴Mn, ⁵⁵Fe, ⁵⁹Ni, ⁶⁰Co, ⁶³Ni, ⁶⁵Zn, ⁹⁰Sr, ⁹⁴Nb, ⁹⁹Tc, ¹⁰³Ru, ¹⁰⁶Ru, ^{108m}Ag, ^{110m}Ag, ¹²⁵Sb, ¹²⁶Sn, ¹³⁴Cs, ¹³⁷Cs, ¹⁴⁴Ce, ¹⁵²Eu, ¹⁵⁴Eu, ¹⁵⁵Eu, ^{166m}Ho, ²³⁷Np, ²³⁸Pu, ²³⁹⁻²⁴⁰Pu, ²⁴¹Am, ²⁴²Cm and ²⁴⁴Cm.

The most abundant radionuclides in the corrosion films on piping and hardware were ⁶⁰Co ($t_{1/2}$ = 5.27 yrs) and ⁵⁵Fe ($t_{1/2}$ = 2.7 yrs), which accounted for approximately 64 and 24 percent, respectively, of the total residual radionuclide inventory present at the time of our sampling.

The highest radionuclide concentrations were found associated with the reactor feedwater piping, followed by the fuel pool system, the steam system and turbine, and the fuel transfer areas. However, in none of the samples were the radionuclide concentrations above the limits allowable for Class A wastes as defined in 10 CFR 61(9). Thus, based upon our limited sampling, all contaminated piping and hardware systems (excluding the pressure vessel internals) could readily be disposed of as the least restrictive class of low level

radioactive waste in commercial shallow land burial sites when the plant is eventually decommissioned.

Radionuclide contamination of concrete surfaces was found to reside mainly in the top centimeter of concrete, often being attached primarily to the paint surface. Cesium-137 and ^{60}Co were the most abundant radionuclides associated with the concrete.

Soil samples collected adjacent to the containment sphere equipment access penetrations and in the radwaste treatment area contained low levels of ^{60}Co and ^{137}Cs contamination.

Inventories of the detectable radionuclides were determined for the piping and the various major systems in the nuclear plant. The total residual radionuclide inventory is listed below in Table S.1 (excluding the neutron-activated portions of the pressure vessel, biological shield, concrete surfaces, sludges and resins in tanks and sumps, and spent fuel stored on site). The inventory has been constructed for the dates of our sampling (August, 1982), and extrapolated back to shutdown in October, 1978.

Cobalt-60 and ^{55}Fe would continue to be the most abundant radionuclides for the next 10-20 years in the corrosion residues. Thereafter, ^{63}Ni will become the most abundant radionuclide. The long-lived radionuclide ^{59}Ni will only begin to be of significance after a period of 400 to 800 years after shutdown.

The very low concentrations of ^{94}Nb , ^{99}Tc , and ^{129}I observed in corrosion films throughout the nuclear plant systems indicate that they will never become an important part of the radionuclide inventory for these systems.

The transuranic nuclides found in reactor piping systems at Dresden One accounted for approximately 0.1% of the residual long-lived radionuclide inventory at shutdown. Transuranic isotopes measured in the corrosion films included ^{237}Np , ^{238}Pu , $^{239-240}\text{Pu}$, ^{241}Am , ^{242}Cm , and ^{244}Cm . The most abundant transuranic radionuclides at the time of our sampling were ^{244}Cm , ^{238}Pu and ^{241}Am , which were present in approximately two times greater concentrations than the $^{239-240}\text{Pu}$.

The residual radionuclide inventory in the Dresden One Nuclear Generating Plant is about 28% of the estimated inventory for a 1155 MWe reference BWR used in the conceptual study for decommissioning a BWR power station(2).

TABLE S.1. Total Residual Radionuclide Inventory
at Dresden One^(a)

<u>Radionuclide (b)</u>	<u>Half-Life (years)</u>	<u>Inventory (Curies)</u>	
		August, 1982	Oct. 31, 1978 (Shutdown)
⁶⁰ Co	5.27	660	1,080
⁵⁵ Fe	2.7	250	650
⁶³ Ni	100	110	110
⁶⁵ Zn	0.668	9.3	450
⁵⁹ Ni	75,000	2.1	2.1
⁵⁴ Mn	0.855	1.0	20
¹⁴⁴ Ce	0.779	0.98	30
¹³⁷ Cs	30.2	0.96	1.0
²⁴⁴ Cm	18	0.069	0.08
¹³⁴ Cs	2.06	0.060	0.21
²³⁸ Pu	87.7	0.059	0.061
²⁴¹ Am	432	0.046	0.046
²³⁹⁻²⁴⁰ Pu	24,100	0.030	0.030
²⁴² Cm	0.446	0.0085	2.9
Total		1,030	2,350

(a) Excluding neutron-activated pressure vessel and internals, biological shield, concrete surfaces, residues, sludges and resins in tanks and sumps, and spent fuel.

(b) Other long-lived radionuclides specifically listed in 10 CFR 61⁽⁹⁾, e.g., ⁹⁴Nb, ⁹⁹Tc, and ¹²⁹I, were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

The Dresden electrical output rating (210 MWe) is only 18% of that for the reference BWR in the conceptual study. Thus, the Dresden One radionuclide

inventory does not exactly scale with the electrical output rating and inventory for the reference BWR. The relatively higher Dresden inventory may be due to the dual cycle design of the Dresden steam supply system, and the original use of admiralty brass heat exchangers in the feedwater system which contributed to elevated trace metal corrosion products in the primary coolant. These trace metals subsequently became neutron-activated and deposited throughout the various systems of the plant.

The radionuclide mixture present in the long-lived residue at Dresden One is significantly different from the single sludge sample used for the previous conceptual study for decommissioning a BWR. Although the ^{60}Co contribution at Dresden One is similar to that for the conceptual study, the ^{54}Mn is much lower (a factor of 40) and the ^{65}Zn is much higher (a factor of 30) at Dresden One. Also, the ^{55}Fe inventory at Dresden One is higher than would be expected from the reference BWR decommissioning assessment.

LIST OF FIGURES

2.1	Scraping the Fuel Pool Wall	4
2.2	Out of Service Fuel Pool Recirculation Pump	4
2.3	Reactor Steam Vent Line	5
2.4	Low Pressure Inlet to Turbine	5
2.5	Concrete Coring Assembly	8
2.6	Hand-held Sediment Coring Apparatus	8
2.7	Inlet Cooling Canal Sediment Core	8
3.1	Original Materials Used in Dresden One Nuclear Steam System	13
A.1	Schematic Diagram of Steam Generation System	A-2
A.2	Reactor and Recirculation Loop Isometric	A-3
A.3	Primary Drum Steam-Water Mixture Separator	A-7
A.4	Secondary Steam Generator	A-10
A.5	Unloading Heat Exchanger System	A-14
A.6	Fuel Handling System	A-18

LIST OF TABLES

S.1	Total Residual Radionuclide Inventory at Dresden One	v
3.1	Residual Radionuclide Concentrations in Corrosion Films from Dresden One - August 1982	10
3.2	Residual Radionuclide Concentrations in Corrosion Films from Dresden One, August 1982	11
3.3	Residual Radionuclide Concentrations in Corrosion Films from Dresden One, August 1982	12
3.4	Maximum Allowable Radionuclide Concentrations for Class A Segregated Waste	16
3.5	Radionuclide Concentrations* (pCi/cm ²) in Dresden Unit One Concrete Core Segments	18
3.6	Residual Radionuclide Concentrations in Dresden One Concrete Cores	19
3.7	Radionuclide Concentrations (pCi/g) in Selected Onsite Soils and Sediments from Dresden Nuclear Station (August 1982)	21
4.1	Total Surface Area Estimates for Dresden One Operating System Components	22
4.2	Samples Used for Radionuclide Inventory Construction	24
4.3	Residual Radionuclide Concentrations Utilized for Radionuclide Inventory Estimate at Dresden One, August 1982	25
4.4	Residual Radionuclide Inventories in Various Operating Systems of Dresden One, August 1982	27
4.5	Total Residual Radionuclide Inventory at Dresden One - August, 1982 and Shutdown - October, 1978	28
4.6	Comparison of Inventory Estimates in the Dresden One Primary System at Shutdown (1978)	29
A.1	Rated Load Design Conditions	A-8

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has been charged with the responsibility of developing a general decommissioning policy for commercial nuclear facilities in the United States, including nuclear power plants. Since the nuclear industry has matured to the point where some of the early nuclear power plants have reached retirement status, it is imperative that the NRC develop detailed information to provide guidance for the decommissioning of these plants.

Several studies have conceptually assessed the technology, safety, and costs associated with various alternatives for decommissioning nuclear power plants.(1-5) One of the key elements of such assessments is a characterization of the radionuclide inventory within a nuclear power plant. This information is essential for understanding the radiological problems which will be encountered during decommissioning. However, empirical data relating to the composition, distribution, and quantity of residual radionuclides within nuclear power plants have been extremely limited, and calculated or estimated quantities have mainly been used in previous assessments.

To provide the NRC with an actual data base of residual radionuclide measurements within nuclear power plants, Pacific Northwest Laboratory (PNL) has conducted a comprehensive sampling and analyses program at a number of nuclear power plants. The objective of these studies is to provide information on the range of compositions, quantities, and locations of radionuclide residues likely to be encountered in retired reactor power stations (exclusive of the reactor pressure vessel) and in the immediate station environs.

This is the fourth in a series of topical reports detailing the results of this program and addresses the residual radionuclide distribution and inventory at Dresden Unit One. Previous reports have described the residual radionuclides at the partially decommissioned Pathfinder Nuclear Plant in Sioux Falls, South Dakota⁽⁶⁾, at the shut-down Humboldt Bay Power Plant near Eureka, California⁽⁷⁾, and at the Monticello Nuclear Generating Plant near Monticello, Minnesota⁽⁸⁾. Topical reports on three more nuclear power plants are in progress, and a final, generic report assessing the residual radionuclide distribution and inventory within and around commercial nuclear power plants will follow and summarize overall findings. A total of four boiling water reactors and three pressurized water reactors has been sampled.

The on-site sampling and measurements program was conducted at Dresden Unit One in August, 1982 to determine the residual radionuclide concentrations, distribution, and inventory at the plant. The program emphasized the characterization of long-lived radionuclides which had been transported from the pressure vessel and deposited in the various systems and components of the plant. Samples of soil and cooling canal sediments from locations in the restricted area of the plant were also analyzed for residual radioactive material. The results of the sampling and measurement program for the Dresden Unit One Generating Plant are summarized in this report.

Dresden Unit One is a dual cycle boiling water reactor (BWR) rated at approximately 210 MWe ($\sqrt{700}$ MW_t) designed by General Electric Company and operated by Commonwealth Edison Company. The nuclear plant began operation in 1959, and was placed into cold shutdown on October 31, 1978. Appendix A contains a more detailed description of the dual cycle steam system and other operational systems. Appendix B contains a short summary of the operating history of Unit One, and Appendix C contains a listing of samples obtained during the PNL field sampling effort at the site.

2.0 SAMPLING AND MEASUREMENTS PROGRAM

The sampling and field measurements at Dresden Unit One were conducted during August, 1982 by a five-man team including a health physics specialist.

On-site measurements of radionuclide composition and concentrations were made using a portable intrinsic germanium gamma-ray spectrometer to serve as a guide during the sampling program and to establish an inventory for shipping purposes from the Dresden site to PNL. A beta-gamma detector was also utilized for monitoring smearable contamination levels on equipment and samples.

The intrinsic germanium detector (a 30% efficient Princeton Gamma-Tech coaxial diode with a resolution of 1.94 keV FWHM) was housed in lead shielding. The detector output was coupled to a Canberra Series 80 multichannel analyzer and minicomputer. A magnetic tape unit and hard copy printer were also interfaced to the multichannel analyzer for data storage purposes and/or on-site printout of the spectral data.

Corrosion film samples obtained onsite were returned to our Richland, Washington laboratory, dissolved in hot concentrated HCl and aliquoted for detailed analyses. Selected samples were sent to the E.A.L. Corporation, Richmond, California for radiochemical analyses. Concrete cores, 10 cm diameter by 6 to 10 cm deep, were also collected for radionuclide analyses from various contaminated locations within the nuclear station in order to assess the radionuclide concentrations associated with the concrete. A sample inventory for the Dresden site is contained in Appendix C.

2.1 PIPING AND HARDWARE

Dresden Unit One is presently in cold shutdown status and its future uncertain. Preparations for a chemical decontamination of the primary loop are currently underway and future power generation is conceivable. Therefore, it was important that our sampling not destroy the integrity in any operational system. The majority of our samples were thus limited to surficial corrosion films scraped from strategic locations which were accessible.

High dose rates associated with a number of primary loop systems (e.g., steam generators, primary steam drum) also limited our sampling possibilities.

Corrosion films on accessible systems were sampled by scraping the surface of interest using a razor blade scraper or paint scraper and the loosened material collected into a polyethylene bag taped to the plate or pipe. For calculation purposes, it is estimated that 100% of the corrosion film deposited on the surface was removed by this scraping technique.

Samples in the Fuel Storage Area included scrapings of the fuel pool wall and ladder and a scraping from an out-of-service fuel pool recirculating pump (Figures 2.1 and 2.2).

The fuel transfer canal surface and the interior of the reactor steam vent line were also scraped for deposited corrosion film. The reactor steam vent line is shown in Figure 2.3.

Several corrosion film samples were also obtained from the turbine area including the steam relief valve, low pressure inlet, and high pressure outlet. Figure 2.4 shows the low pressure inlet area sampled.

FIGURE 2.1. Scraping the Fuel Pool Wall

FIGURE 2.2. Out of Service Fuel Pool Recirculation Pump

FIGURE 2.3. Reactor Steam Vent Line

FIGURE 2.4. Low Pressure Inlet to Turbine

One piece of hardware, a floor drain from instrument room A, was obtained, since it had been removed from service. A total of thirteen samples were obtained from the Dresden One nuclear systems. A description of these piping and hardware samples is contained in Appendix C.

All corrosion film scrapings were double or triple bagged in plastic, labeled, and finally packaged into metal cans, which were subsequently sealed with tape. The hardware sample was wrapped in rags to cover rough and sharp corners, then double bagged in plastic, labeled, and finally also placed into a metal can.

2.2 CONCRETE CORES

Nine concrete cores (10 cm diameter by approximately 6 to 10 cm deep) were collected from Dresden Unit One. Five cores were taken in the containment sphere, two in the turbine building, and two in the radwaste building.

Containment cores were taken from: (1) a floor area contaminated by a spill during previous decontamination work, (2) secondary Steam Generator Room B, (3) the hallway in front of the Accumulator Room, (4) the sub-pile room, and (5) the unloading heat exchanger room. The two turbine building cores were taken in the Make-up Demineralizer Room and the Condensate Pump Room Pit. The two cores from the radwaste building were from the basement area. The core locations and surficial dose rates for all Dresden concrete cores are listed in Appendix C.

Prior to taking a core, the floor was surveyed with a G-M meter to locate any "hot spots" of contamination. Reinforcing rebar was also located prior to coring with a portable rebar locator to insure that no rebar was encountered during the coring process.

A Korit® concrete coring apparatus utilizing a diamond-tipped core barrel was employed (Figure 2.5). Recirculated water was used as a bit coolant and cutting fluid. After coring, the residual water was wiped up, and the core segment broken free in place, removed, dried, wrapped in rags, and double bagged in polyethylene. The sample was then labeled and placed into a metal can for shipment. The borehole was then refilled with concrete and the new surface finished by troweling.

2.3 SOIL SAMPLING

Surficial soil samples on the plant site were obtained in the north, east, and south compass directions from the containment sphere near equipment entry points into the sphere. Two other soil samples were obtained in the locked radwaste treatment area at locations of known contamination. Finally, surficial sediments were obtained from the inlet and outlet cooling water canal using a hand-held corer. Figures 2.6 and 2.7 illustrate the coring apparatus and the type of core obtained.

Soil samples were obtained with a shovel from a 0.1 m² area. The soil was placed into double polyethylene bags, labeled, and placed into a metal can for shipment. The sediment cores were extruded and packaged into double polyethylene bags after surficial water was carefully decanted. The bags were then labeled and placed into a metal can for shipment to our laboratory.

2.4 SAMPLE HANDLING AND ANALYSES

Upon return of the samples to our Richland laboratory, small pieces of a number of the low activity samples were repackaged into plastic bags, the bags sealed, and the samples were then analyzed directly on Ge(Li) gamma-ray spectrometers. The concrete cores were likewise repackaged and directly analyzed. The concrete cores were later cut into 1-cm slabs with a diamond blade lapidary saw and each slab was analyzed to determine the radionuclide contamination as a function of depth in the concrete.

The corrosion film samples were contacted with hot concentrated hydrochloric acid to dissolve the material for subsequent radiochemical and gamma spectrometric analyses. The instrument room floor drain was likewise contacted with hot, concentrated hydrochloric acid in order to remove the surface associated corrosion film.

The HCl solutions were filtered to remove any insoluble matter, the filter rinsed, and the final solution brought to one liter. Appropriate aliquots were then removed and packaged into standardized geometries for Ge(Li) gamma-ray spectrometric analysis. Radionuclides measured by the direct Ge(Li) analysis and computer-assisted data reduction were ⁵⁴Mn, ⁶⁰Co, ⁶⁵Zn, ⁹⁴Nb, ¹⁰³Ru,

FIGURE 2.5. Concrete Coring Assembly

FIGURE 2.6. Hand-held Coring Apparatus

FIGURE 2.7. Inlet Cooling Canal Sediment Core

^{106}Ru , ^{108}mAg , ^{110}mAg , ^{125}Sb , ^{126}Sn , ^{134}Cs , ^{137}Cs , ^{144}Ce , ^{152}Eu , ^{154}Eu , ^{155}Eu , ^{166}mHo , and ^{241}Am .

The filters containing any insoluble matter remaining from the hydrochloric acid dissolution were also analyzed by Ge(Li) spectrometry. Normally, the concentrations of the insoluble radionuclides associated with the residues were only a few percent of the soluble fraction.

Following the direct gamma-ray spectrometry, aliquots of leachate were taken for radiochemical analyses. Radionuclides measured included ^{55}Fe , ^{59}Ni , ^{63}Ni , ^{90}Sr , ^{99}Tc , ^{129}I , ^{237}Np , ^{238}Pu , $^{239-240}\text{Pu}$, ^{241}Am , ^{242}Cm , and ^{244}Cm . These radiochemical measurements were performed by E.A.L Corp. of Richmond, California.

3.0 RESIDUAL RADIONUCLIDE CONCENTRATIONS

The residual radionuclide concentrations in contaminated piping hardware, concrete and soils are reported in this section.

3.1 REACTOR SYSTEMS CORROSION FILMS

Residual radionuclide concentrations associated with the corrosion films deposited on contaminated surfaces are shown in Tables 3.1 to 3.3.

As illustrated in Tables 3.1, 3.2, and 3.3, the sample scraped from the reactor steam vent line contained the highest residual radionuclide concentrations. This stainless steel line was exposed to a steam-water mixture and should be fairly typical of the stainless piping in the primary system. The most abundant radionuclides are the activated corrosion products ^{60}Co , ^{55}Fe , ^{63}Ni , ^{65}Zn , ^{59}Ni , and ^{54}Mn . The fission products ^{137}Cs and ^{144}Ce are also present at lesser concentrations. Transuranic radionuclide concentrations in August, 1982 ranged from a low of 28 pCi/cm² for ^{242}Cm , to a high of 840 pCi/cm² for ^{244}Cm . However, at the time of shutdown (October, 1978) the ^{242}Cm was by far the most abundant transuranic radionuclide, but had decayed to very low concentrations at the time of sampling. The transuranic concentrations in this sample were among the highest observed at the various reactors examined in the overall field program.

The relatively high concentrations of ^{54}Mn and ^{65}Zn on contaminated piping and hardware approximately five years after shutdown indicate that the deposition

TABLE 3.1. Residual Radionuclide Concentrations in Corrosion Films from Dresden One - August 1982

SAMPLE	CONCENTRATION - pCi/cm ² (pCi/gm)															
	⁵⁴ Mn	⁶⁰ Co	⁶⁵ Zn	⁹⁴ Nb	¹⁰⁶ Ru	^{108m} Ag	^{110m} Ag	¹²⁵ Sb	¹²⁶ Sn	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵² Eu	¹⁵⁴ Eu	¹⁵⁵ Eu	^{166m} Ho
D-SC-10 FUEL POOL WALL	1200 ± 100 (250 ± 30)	176,000 ± 1000 (35,800 ± 100)	1500 ± 300 (310 ± 60)	<40 (<8)	<700 (<100)	<500 (<100)	<700 (<100)	<200 (<40)	<400 (<90)	10,600 ± 100 (2160 ± 20)	86,900 ± 200 (17,660 ± 40)	<300 (<60)	<100 (<30)	<200 (<30)	<100 (<20)	<60 (<10)
D-SC-11 FUEL POOL RECIRC. PUMP	<70	11,200 ± 200	<20	<21	<400	<300	<400	<100	<20	650 ± 50	7300 ± 100	<200	<80	<80	<60	<30
D-SC-12 FUEL POOL LADDER	1700 ± 100 (340 ± 30)	200,000 ± 1000 (41,600 ± 100)	2800 ± 300 (560 ± 60)	<40 (<7)	<600 (<100)	<400 (<80)	<800 (<100)	500 ± 200 (100 ± 40)	<30 (<7)	6000 ± 100 (1210 ± 10)	50,100 ± 200 (10,170 ± 20)	<200 (<50)	<100 (<20)	<100 (<30)	<80 (<20)	<50 (<10)
D-SC-13 INSIDE SPHERE WALL	<0.3	120 ± 1	2.7 ± 0.8	<0.1	<2	<1	<2	<0.4	<0.1	<0.3	3.4 ± 0.2	<0.6	<0.3	<0.4	<0.2	<0.1
D-SC-14 AIR VENT DUCT	<0.2	93 ± 1	1.2 ± 0.4	<0.1	<1	<0.6	<1	<0.2	<0.05	0.43 ± 0.09	5.5 ± 0.1	0.7 ± 0.4	<0.2	0.5 ± 0.2	<0.1	<0.08
D-SC-15 FLOOR DRAIN INSTR. ROOM A	34 ± 11 (15 ± 5)	26,400 ± 40 (11,630 ± 20)	390 ± 30 (170 ± 10)	<3 (<1)	<60 (<30)	<40 (<20)	<60 (<30)	<20 (<7)	<3 (<1)	990 ± 10 (437 ± 3)	7600 ± 30 (3360 ± 30)	<20 (<10)	<10 (<5)	<10 (<6)	<8 (<4)	<5 (<2)
D-SC-16 FUEL CHUTE	3.6 ± 0.9	790 ± 3	11 ± 2	<0.3	<4	<3	<5	<1	<0.2	3.3 ± 0.4	24 ± 1	4 ± 2	<0.8	<1	<0.5	<0.4
D-SC-17 FUEL CANAL	<0.2 (<0.04)	53 ± 1 (10.8 ± 0.2)	1.5 ± 0.5 (0.30 ± 0.10)	<0.1 (<0.01)	<1 (<0.2)	<1 (<0.2)	<1 (<0.2)	<0.4 (<0.08)	<0.08 (<0.02)	5.6 ± 0.2 (1.13 ± 0.04)	39 ± 1 (7.91 ± 0.07)	<0.5 (<0.1)	<0.3 (<0.05)	<0.3 (<0.06)	<0.2 (<0.04)	<0.1 (<0.02)
D-SC-18 REACTOR STEAM VENT LINE	12,000 ± 2000 (800 ± 100)	7.90 × 10 ⁶ ± 0.01 × 10 ⁶ (522,000 ± 1000)	113,000 ± 4000 (7500 ± 300)	<500 (<30)	<8000 (<600)	<5000 (<300)	<9000 (<600)	<2000 (<100)	<400 (<30)	<800 (<50)	5500 ± 1000 (360 ± 60)	12,000 ± 3000 (800 ± 200)	<2000 (<100)	<2000 (<100)	<1000 (<70)	<700 (<50)
D-SC-19 STEAM RELIEF VALVE	<0.6	510 ± 2	7 ± 1	<0.2	<3	<2	<3	<0.7	<0.1	<0.3	<4	2 ± 1	<0.5	<0.7	<0.4	<0.3
D-SC-20 LOW PRESSURE INLET TO TURBINE	<2	2270 ± 10	37 ± 4	<0.5	<8	<5	<8	<4	<0.4	<0.7	3.1 ± 0.9	<3	5 ± 1	<2	<1	<0.7
D-SC-21 HIGH PRESS. OUTLET FROM TURBINE	<0.6	975 ± 2	15 ± 2	<0.2	<3	<2	<3	<0.7	<0.1	<0.3	6.6 ± 0.9	4 ± 1	<0.5	1.7 ± 0.7	<0.4	<0.3
D-SC-22 STEAM LINE PIPING	1 ± 0.5	2750 ± 10	36 ± 3	<0.3	<5	<3	12 ± 6	<1	<0.2	<0.5	3.0 ± 0.6	<2	2.4 ± 0.9	<1	<0.6	<0.4

Table 3.2 Residual Radionuclide Concentrations in Corrosion Films from Dresden One, August 1982

SAMPLE	CONCENTRATION - pCi/cm ² (pCi/gm)					
	⁵⁵ Fe	⁵⁹ Ni	⁶³ Ni	⁹⁰ Sr	⁹⁹ Tc	¹²⁹ I
D-SC-12 FUEL POOL LADDER	164,000 ± 10,000 (33,000 ± 2,000)	560 ± 40 (110 ± 10)	6,600 ± 500 (1,330 ± 90)	80 ± 5	45 ± 3 (9.1 ± 0.6)	<0.4 (<0.08)
D-SC-15 FLOOR DRAIN INSTRUMENT ROOM A	540 ± 30 (0.32 ± 0.02)	85 ± 5 (0.051 ± 0.003)	1,300 ± 80 (0.77 ± 0.05)	6,200 ± 400	0.93 ± 0.06 (0.0006 ± 0.0001)	0.09 ± 0.02 (5 × 10 ⁻⁵ ± 1 × 10 ⁻⁵)
D-SC-16 FUEL TRANSFER CHUTE	440 ± 30	8.9 ± 0.7	3,700 ± 300	0.062 ± 0.004	0.009 ± 0.002	<0.02
D-SC-18 REACTOR STEAM VENT LINE	2.91 × 10 ⁶ (192,000 ± 11,000)	26,000 ± 2,000 (1,700 ± 100)	1.28 × 10 ⁶ ± 0.08 × 10 ⁶ (85,000 ± 5,000)	690 ± 40	4.5 ± 0.3 (0.30 ± 0.02)	<1 (<0.07)
D-SC-20 LOW PRESSURE INLET TO TURBINE	2,200 ± 300	10 ± 1	1,000 ± 100	2.9 ± 0.2	0.05 ± 0.01	<0.01
D-SC-22 STEAM LINE PIPING	2,100 ± 200	10 ± 1	720 ± 90	3.8 ± 0.2	0.05 ± 0.02	0.03 ± 0.01

TABLE 3.3. Residual Radionuclide Concentrations in Corrosion
Films from Dresden One, August 1982

SAMPLE	CONCENTRATIONS - pCi/cm ² (pCi/gm)					
	²³⁷ Np	²³⁸ Pu	^{239,240} Pu	²⁴¹ Am	²⁴² Cm	²⁴⁴ Cm
D-SC-12 FUEL POOL LADDER	0.034 ± 0.004 (0.007 ± 0.001)	61 ± 2 (12.5 ± 0.5)	24 ± 1 (4.8 ± 0.2)	33 ± 2 (6.7 ± 0.3)	7.2 ± 0.7 (1.5 ± 0.1)	56 ± 3 (11.4 ± 0.7)
D-SC-15 FLOOR DRAIN INSTRUMENT ROOM A	<0.003	7.3 ± 0.6	13.0 ± 0.7	20 ± 1	0.28 ± 0.03	1.7 ± 0.1
D-SC-16 FUEL TRANSFER CHUTE	0.0008 ± 0.0004	0.16 ± 0.01	0.067 ± 0.003	0.11 ± 0.01	0.020 ± 0.04	0.20 ± 0.01
D-SC-18 REACTOR STEAM VENT LINE	0.27 ± 0.13 (0.018 ± 0.009)	720 ± 30 (47 ± 2)	360 ± 10 (24 ± 1)	550 ± 30 (36 ± 2)	100 ± 10 (6.6 ± 0.6)	840 ± 50 (55 ± 3)
D-SC-20 LOW PRESSURE INLET TO TURBINE	0.0004 ± 0.0002	0.42 ± 0.02	0.68 ± 0.03	0.98 ± 0.04	0.015 ± 0.004	0.19 ± 0.01
D-SC-22 STEAM LINE PIPING	0.0006 ± 0.0003	0.21 ± 0.01	0.44 ± 0.02	0.70 ± 0.02	0.009 ± 0.003	0.17 ± 0.01

of neutron-activated corrosion products in Dresden One was significant. The presence of large amounts of ^{65}Zn also indicates the original use of admiralty brass heat exchangers in the feedwater and condenser systems. Although these were replaced with stainless steel heat exchangers, a corrosion residue is still present throughout the operating loop. Figure 3.1 shows the materials of construction initially used within Dresden One.

Commonwealth Edison is well aware of the corrosion redeposition problem in Dresden One and a decontamination of the primary system of Unit One is currently scheduled for July, 1984. This decontamination was scheduled to reduce dose rates for maintenance work or any retrofitting.

The corrosion film scraped from the turbine area (samples D-Sc-19, 20, and 21) and from the steam line (D-Sc-22) should be representative of surfaces in the steam system, e.g., main steam lines, turbine, steam extraction lines, steam side of the feedwater heaters. The corrosion material (black iron oxide) was typical of turbine and steam line deposits sampled at other sites in our field program.

These four corrosion product samples from the steam side contained much lower residual radionuclide concentrations than observed in the feedwater piping, as exemplified by the reactor steam vent line. The most abundant radionuclides in the materials exposed primarily to steam were ^{55}Fe , ^{60}Co , ^{63}Ni and ^{65}Zn , but at concentrations approximately 1000-fold lower than in the reactor steam vent line which was exposed to primary coolant. Transuranic radionuclides were also detectable in the steam side samples at concentrations approximately 1000 times lower than in the primary coolant side. The most abundant transuranics on the steam side were ^{241}Am , followed by $^{239,240}\text{Pu}$ and ^{238}Pu . The plutonium $^{238}/^{239-240}\text{Pu}$ ratios were also different in the steam side as compared to the feedwater side, 0.52 versus 2.0, respectively.

Three scraping samples (D-Sc-10, 11, 12) were obtained from the fuel pool area. The fuel pool ladder and fuel pool wall surface films contained higher residual radionuclide concentrations than the out-of-service recirculating pump. The most abundant residual radionuclides in these three samples were ^{60}Co , ^{55}Fe , ^{137}Cs and ^{134}Cs . The higher concentrations of cesium isotopes observed in the pool area were expected. Transuranic radionuclide concentrations ranged between those observed in the reactor steam vent line and the

FIGURE 3.1. Materials Used in Dresden One Nuclear Steam System

steam side samples. The most abundant transuranic radionuclides were ^{238}Pu and ^{244}Cm . The 238/239-240 ratio for plutonium of 2.5 was similar to that observed in the reactor vent line sample.

Two other samples (D-Sc-16 and 17) were obtained from surfaces used during fuel transfer activities. The fuel canal surface was scraped and a salt deposit on the outside of the fuel transfer chute was analyzed for residual radionuclides. Both contained relatively low residual radionuclide concentrations. The most abundant radionuclides were ^{60}Co , ^{63}Ni , ^{55}Fe and ^{137}Cs as also observed in the fuel pool samples. The transuranic radionuclide concentrations in the fuel transfer chute scraping sample were the lowest observed in the Dresden One containment building samples. The most abundant transuranics were ^{244}Cm , ^{238}Pu and ^{241}Am . The 238/239-240 ratio for plutonium is 2.5, similar to those in the reactor vent line and the fuel storage pool.

There are two distinct patterns of transuranic deposition based upon the differences in the 238/239-240 plutonium ratios observed in the steam side samples as compared to the samples from the other operating systems in Dresden Unit One. Steam side 238/239-240 plutonium ratios are typically about 0.5 while the plutonium ratios for other systems are typically approximately 2. The absolute concentrations in the steam side samples, it should be remembered, are a factor of approximately 1000 lower than the feedwater samples. The low 238/239-240 ratios present in the steam side samples appear to be a residue of contamination of the steam side during some period early in a fuel cycle where the 238/239-240 ratio is relatively low. The higher ratio observed in the other operating systems are apparently residues representative of contamination during the entire fuel cycle.

The floor drain sample from Instrument Room A (D-Sc-15) had residual radionuclide concentrations greater than observed in the steam side and less than observed in the fuel pool samples. The most abundant radionuclides were ^{60}Co , ^{137}Cs , ^{63}Ni , ^{134}Cs , and ^{55}Fe . The most abundant transuranic radionuclides were ^{241}Am , 239-240Pu, and ^{238}Pu . The 238/239-240 ratio for plutonium of 0.56 was similar to that observed in the steam side samples.

Two scraping samples (D-Sc-13 and 14) were obtained which should reflect general surface contamination from aerosol deposition in the containment area. The sphere wall scraping and the air vent duct scraping both contained very low residual radionuclide concentrations. The only radionuclides observed

were ^{60}Co , ^{137}Cs , ^{65}Zn and ^{134}Cs . For the air vent duct, ^{154}Eu and ^{144}Ce were also detected but with very large analytical errors.

3.2 WASTE DISPOSAL CONSIDERATIONS

For waste disposal purposes, the residual radionuclide concentrations observed on contaminated materials at Dresden One can be compared with maximum allowable concentrations for the least restrictive Class A Segregated Waste as published in 10 CFR 61(9). The standards are specified in Ci/m^3 . These concentrations can be converted to a weight basis for easier comparison with the observed measurements on piping, hardware and concrete, assuming a density of iron or steel of $7.9 \text{ g}/\text{cm}^3$ and a density of concrete of $2.5 \text{ g}/\text{cm}^3$. This conversion is shown in Table 3.4. The first column gives the published maximum allowable concentrations for Class A wastes (converted to mCi/cm^3), and columns two and three are conversions of column one which give the concentrations allowable for various radionuclides in either steel or concrete on a weight basis. The maximum allowable concentrations for Class A wastes are substantially greater than those encountered for all of the Dresden One samples. Thus, if dismantlement and disposal of the nuclear plant systems are the eventual preferred decommissioning options, all contaminated materials (with the possible exception of the pressure vessel) could be disposed of as Class A low level waste in commercial shallow land burial sites.

Niobium-94, a long-lived activation product, has been postulated to be a major dose contributor in reactor pressure vessel internals approximately 100 years after permanent reactor shutdown(1,2). However, ^{94}Nb was never detected in any of the radioactive corrosion films from the Dresden One systems measured in this study. Either the ^{94}Nb is present in the neutron-activated pressure vessel in lower amounts than calculated or it is so insoluble that it is never substantially translocated from the pressure vessel and redeposited on other system surfaces. Niobium-94 concentrations in primary coolant piping, condenser feedwater system piping and hardware, and other major plant systems are not high enough to contribute significant dose rates from these systems.

Iodine-129, ^{90}Sr and ^{99}Tc are long-lived radionuclides of special concern in radioactive waste management as indicated in 10 CFR 61(9). However, these radionuclides are very soluble and do not accumulate to any substantial degree in corrosion product films in contaminated nuclear plant systems.

TABLE 3.4 Maximum Allowable Radionuclide Concentration
for Class A Segregated Waste

Radionuclide	$\mu\text{Ci}/\text{cm}^3$	$\mu\text{Ci}/\text{gm}$ of Iron or Steel (a)	$\mu\text{Ci}/\text{gm}$ Concrete (b)
Any with half life less than 5 years	700	89	280
^3H	40	5.1	16
^{14}C	0.8	0.10	0.32
^{59}Ni	22	2.8	8.8
^{60}Co	700	89	280
^{63}Ni	3.5	0.44	1.4
^{94}Nb	0.02	0.0025	0.0080
^{90}Sr	0.04	0.0051	0.016
^{99}Tc	0.3	0.038	0.12
^{129}I	0.008	0.0010	0.0032
^{135}Cs	84	10	34
^{137}Cs	1.0	0.13	0.40
Enriched U	0.04	0.0051	0.016
Natural or Depleted U	0.05	0.0063	0.020
Alpha Emitting Transuranics	10 nCi/gm	10 nCi/gm	10 nCi/gm
^{241}Pu	350 nCi/gm	350 nCi/gm	350 nCi/gm

(a) This column was derived by dividing the first column by the density of iron or steel ($7.9 \text{ g}/\text{cm}^3$).

(b) This column was derived by dividing the first column by the density of concrete ($2.5 \text{ g}/\text{cm}^3$).

3.3 CONCRETE

The residual radionuclide concentrations observed in the top 2 cm of the Dresden One concrete cores are shown in Table 3.5 and Table 3.6. The most abundant radionuclides (Table 3.5) were ^{60}Co , ^{134}Cs , and ^{137}Cs . Minor amounts of ^{106}Ru , ^{108}mAg , ^{110}mAg , ^{144}Ce , ^{152}Eu , ^{154}Eu and ^{155}Eu were also detected in several core samples. Cobalt-60 was found in highest concentrations in the top centimeter of core DCC-4 obtained from the sub-pile room at the 488' elevation. Cesium-137 concentrations were highest in the sub-pile and secondary steam generator room cores; both cores contained approximately $200\ \mu\text{Ci}/\text{cm}^2$.

The residual radionuclide contamination resided primarily in the top centimeter of the concrete except for the core collected from the area of the chemical decontamination solution spill. Typically, the concentrations in the second centimeter were 1% or less of the top centimeter; however, in the spill area the second centimeter concentrations were 30 and 35% of the top centimeter for ^{137}Cs and ^{60}Co , respectively.

Three of the top 1 cm core intervals (cores DCC-1, DCC-4, and DCC-7) were dissolved in hot hydrochloric acid and aliquoted for radiochemical analyses. The results are shown in Table 3.6. The most abundant radionuclides of the group identified by radiochemical analyses were ^{63}Ni , ^{55}Fe , and ^{59}Ni . The most abundant radionuclides were ^{238}Pu in cores DCC-4 and DCC-7 and ^{241}Am in DCC-1. The plutonium ratios 238/239-240 are similar in cores DCC-4 and DCC-7, but core DCC-1, the Dow Chemical spill, has a significantly lower ratio. The curium radionuclide ratios are all similar for the three cores.

Residual radionuclide concentrations in the first segment (0-1 cm) of all Dresden One cores were above the acceptable surface contamination level ($2300\ \text{pCi}/100\ \text{cm}^2$) specified in NRC Regulatory Guide 1.86(10) for decommissioned facilities. The second segment (1-2 cm) in all cores except the Make-up Demineralizer Room was also above this recommended limit. This would indicate paint stripping the concrete in many contaminated areas at Dresden will not necessarily reduce contamination to acceptable levels. Rather, decommissioning may need to entail removal and disposal of the upper several centimeters of many concrete floors.

TABLE 3.5 RADIONUCLIDE CONCENTRATIONS* (pCi/cm²) IN DRESDEN UNIT NO. 1 CONCRETE CORE SEGMENTS

Location	Depth Interval (cm)	⁵⁴ Mn	⁶⁰ Co	⁶⁵ Zn	⁹⁴ Nb	¹⁰⁶ Ru	^{108m} Ag	^{110m} As	¹²⁵ Sb	¹²⁶ Sn
1 Dow Chemical Spill 529' Elevation	0-1	<50	208,800 ± 100	<200	<20	<300	<300	<300	<80	<20
	1-2	<20	73,300 ± 400	<60	18 ± 7	<100	240 ± 70	<80	<20	<20
2 Secondary Steam Generator Room "B" 529' Elevation	0-1	40 ± 20	33,910 ± 50	<70	<8	<200	<200	<90	<60	<20
	1-2	0.5 ± 0.2	320 ± 1	<0.8	<0.09	4 ± 2	<2	<1	<0.7	<0.2
3 Hallway in Front of Accumulator Room 488' 488' Elevation	0-1	27.8 ± 0.9	7,010 ± 10	<4	<0.4	34 ± 6	<4	<5	1.5 ± 1.2	<0.9
	1-2	<0.03	7.57 ± 0.09	<0.04	<0.02	<0.3	<0.3	<0.2	<0.2	<0.02
4 Sub-pile Room 488' Elevation	0-1	960 ± 70	276,900 ± 100	<900	<30	<500	<400	<500	<100	<40
	1-2	2.5 ± 0.2	1,000 ± 1	<2	<0.2	3 ± 1	<2	<0.7	<0.6	<0.2
5 Make-up Demineralizer 1-2	0-1	<0.2	27.3 ± 0.3	<0.4	<0.05	<1	<2	<0.5	<0.5	<0.1
	1-2	<0.02	1.98 ± 0.05	<0.04	<0.01	<0.1	<0.1	<0.04	<0.04	<0.01
6 Condensate Pump Room Pit 1-2	0-1	54 ± 6	19,050 ± 20	<20	<3	<60	<70	<30	<20	<7
	1-2	<0.06	27.1 ± 0.2	<0.2	<0.03	<0.5	<0.4	<0.3	<0.1	<0.04
7 Unloading Heat Exchanger Room 1-2	0-1	310 ± 20	108,200 ± 500	<80	<8	<130	180 ± 90	350 ± 100	<30	<8
	1.1 ± 0.2	240.7 ± 0.5	<0.7	<0.08	<1	<0.7	<0.9	0.4 ± 0.2	<0.06	
8 Rad Waste Basement 1-2	0-1	29 ± 8	7,560 ± 20	<30	<3	<70	<90	<40	<30	<8
	<0.05	9.2 ± 0.1	<0.2	<0.02	<1	<2	<0.2	0.5 <0.1		
9 Rad Waste Basement Drainage Trough	0-1	63 ± 8	20,750 ± 20	<80	<4	<80	<90	<40	<30	<8
	1-2	0.10 ± 0.04	11.1 ± 0.1	<0.2	<0.02	2 ± 1	<1	<0.2	0.6 ± 0.4	<0.3

* To convert to pCi/g, multiply by 0.45.

Table 3.5 (continued)

TABLE 3.5 RADIONUCLIDE CONCENTRATIONS* (pCi/cm²) IN DRESDEN UNIT NO. 1 CONCRETE CORE SEGMENTS

Location	Internal (cm)	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵² Eu	¹⁵⁴ Eu	¹⁵⁵ Eu	¹⁶⁶ Ho	²²⁸ Ac
1 Dow Chemical Spill 529' Elevation	0-1 1-2	<30 <80	130 ± 30 40 ± 10	<260 <80	<50 <7	<40 <10	<70 50 ± 20	<30 <10	<300 <80
2 Secondary Steam Generator Room "B" 529' Elevation	0-1 1-2	9,870 ± 30 102 ± 0.3	202,200 ± 600 1,644 ± 4	<300 4 ± 2	<14 <0.4	<13 <0.2	<50 <0.4	<11 <0.2	<90 <1
3 Hallway in Front of Accumulator Room 488' Elevation	0-1 1-2	75.4 ± 0.8 0.72 ± 0.04	948 ± 1 18.2 ± 0.1	35 ± 3 <0.2	<1.2 <0.06	6.8 ± 0.9 <0.04	2.4 ± 0.8 <0.05	<0.6 <0.02	<5 1.4 ± 0.2
4 Sub-pile Room 488' Elevation	0-1 1-2	39,970 ± 900 27.6 ± 0.2	197,600 ± 400 1,203 ± 1	800 ± 40 1.3 ± 0.7	<80 13.1 ± 0.4	<60 1.7 ± 0.3	300 ± 100 <0.4	<40 <0.2	<400 <2
5 Make-up Demineralizer	0-1 1-2	8.9 ± 0.1 0.11 ± 0.02	431.1 ± 0.8 1.13 ± 0.03	<1 <0.07	<0.1 <0.06	<0.1 <0.04	<0.3 <0.04	<0.06 <0.02	<0.9 0.8 ± 0.1
6 Condensate Pump Room Pit	0-1 1-2	15,100 ± 200 4.00 ± 0.09	111,800 ± 400 44.6 ± 0.2	180 ± 70 <0.4	<40 0.11 ± 0.07	<4 <0.06	<20 <0.1	<4 <0.05	<30 <0.3
7 Unloading Heat Exchanger Room	0-1 1-2	640 ± 10 0.7 ± 0.1	18,120 ± 20 11.2 ± 0.1	160 ± 100 <0.6	48 ± 12 <0.2	49 ± 13 <0.1	40 ± 30 <0.2	<10 <0.2	<100 <1
8 Rad Waste Basement	0-1 1-2	1,580 ± 10 18.9 ± 0.1	40,780 ± 40 910 ± 1	110 ± 80 <1	<4 0.25 ± 0.08	<5 <0.05	<20 <0.3	<5 <0.03	<40 0.6 ± 0.2
9 Rad Waste Basement Drainage Trough	0-1 1-2	6,870 ± 10 18.2 ± 0.1	90,500 ± 300 862 ± 3	360 ± 80 <0.9	<5 <0.06	12 ± 6 <0.04	<20 <0.2	<6 <0.03	<40 <0.2

* To convert to pCi/g, multiply by 0.45.

Table 3.6 Residual Radionuclide Concentrations in Dresden One Concrete Cores

CONCRETE CORE	CONCENTRATIONS - pCi/cm ² *											
	⁵⁵ Fe	⁵⁹ Ni	⁶³ Ni	⁹⁰ Sr	⁹⁹ Tc	¹²⁹ I	²³⁷ Np	²³⁸ Pu	²³⁹ ²⁴⁰ Pu	²⁴¹ Am	²⁴² Cm	²⁴⁴ Cm
DCC-1 CHEMICAL SPILL	18,000 ± 1,000	2,000 ± 100	920,000 ± 80,000	3.0 ± 0.3	117 ± 7	< 0.8	0.036 ± 0.006	15 ± 1	12 ± 1	18 ± 1	0.49 ± 0.15	5.6 ± 0.4
DCC-4 SUB-PILE ROOM	54,000 ± 3,000	2,300 ± 200	770,000 ± 70,000	2,900 ± 200	2.4 ± 0.2	< 0.3	0.026 ± 0.005	48 ± 2	21 ± 1	30 ± 1	5.1 ± 0.5	45 ± 3
DCC-7 UNLOADING HEAT EXCHANGER ROOM	23,000 ± 1,000	670 ± 60	170,000 ± 20,000	17 ± 1	1.3 ± 0.1	< 0.3	0.014 ± 0.004	6.7 ± 0.3	3.6 ± 0.2	6.3 ± 0.3	0.80 ± 0.18	8.2 ± 0.6

*TO CONVERT TO pCi/g MULTIPLY BY 0.45

3.4 SOILS

The surface soils immediately surrounding the reactor sphere sampled near sphere penetrations contained detectable concentrations of reactor-derived radionuclides (Table 3.7). The presence of ^{54}Mn and ^{134}Cs , as well as the elevated concentrations of ^{60}Co and ^{137}Cs (as compared to natural background levels)(11), indicates the reactor as the primary source. The soil sample from the radwaste area was taken from a location of known contamination, and our measurements confirm the presence of elevated concentrations of ^{60}Co , ^{134}Cs , and ^{137}Cs as compared to the natural fallout background.

The Draft Environmental Impact Statement on Decommissioning of Nuclear Facilities(4) suggests residual contamination levels that result in a dose below 10 mrem/year whole-body are acceptable. Several of the soil samples obtained from the Dresden exclusion areas might approach or exceed this limit, notably the radwaste area and the area sampled south of the refuel building.

These concentrations of radionuclides indicate that at the time of decommissioning a careful survey of the soils on site may be necessary and selected small patches of soil may require removal and disposal.

The canal sediments also contained detectable levels of reactor-derived radionuclides. These radionuclide concentrations may not be derived solely from Unit One, however, since there are two other operating units on the site. The presence of traces of ^{54}Mn and ^{134}Cs and the elevated ^{60}Co and ^{137}Cs concentrations indicate a reactor source.

4.0 ESTIMATED RESIDUAL RADIONUCLIDE INVENTORIES

Radionuclide inventories present in the major piping and hardware systems in Dresden One have been estimated from residual concentrations measured in the corrosion films deposited within these systems. These inventories do not include the neutron-activated reactor pressure vessel and internals, the neutron-activated biological shield, the contaminated concrete surfaces, the spent fuel stored on site, or sludges/residues in tanks and sumps. They do, however, include all other major systems which have been exposed to various radioactive fluids.

TABLE 3.7. Radionuclide Concentrations (pCi/g) in Selected Onsite Soils and Sediments from Dresden Nuclear Station (August 1982)

Location	⁴⁰ K	⁵⁴ Mn	⁶⁰ Co	⁶⁵ Zn	¹⁰⁶ Ru	¹²⁵ Sb	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵² Eu	¹⁵⁴ Eu	¹⁵⁵ Eu
Equipment Hatch Area North Side of Sphere	8.3 ± 0.2	0.02 ± 0.01	1.32 ± 0.02	<0.03	<0.07	<0.03	<0.01	0.49 ± 0.01	<0.04	<0.05	<0.04	<0.04
Equipment Storage Doorway East of Sphere	6.9 ± 0.2	0.05 ± 0.02	6.2 ± 0.1	<0.06	<0.1	<0.06	0.06 ± 0.01	1.93 ± 0.02	<0.07	<0.1	<0.08	<0.1
Refuel Building South of Sphere	7.7 ± 0.2	0.23 ± 0.02	13.8 ± 0.1	<0.06	0.2 ± 0.1	<0.06	0.27 ± 0.01	3.36 ± 0.02	0.17 ± 0.07	<0.09	<0.09	<0.03
Rad Waste Area 0-6" In Front of Door	7.4 ± 0.5	0.2 ± 0.1	55.9 ± 0.3	<0.3	<0.8	<0.3	1.71 ± 0.07	96.2 ± 0.3	<0.4	<0.6	<0.4	<0.2
6-12"	8.4 ± 0.5	<0.1	161 ± 1	<0.6	<1	<1	6.3 ± 0.1	260 ± 1	1.5 ± 0.6	<0.9	<0.7	<0.2
Sediments												
Inlet Canal 0-8"	14.4 ± 0.3	2.5 ± 0.9	3.1 ± 0.1	<9	9 ± 4	<0.2	<0.07	1.19 ± 0.02	<7	<0.1	<0.1	<0.1
Outlet Canal 0-8"	10.5 ± 0.3	<0.02	2.79 ± 0.03	<0.05	<1	<0.06	0.02 ± 0.01	2.46 ± 0.02	<0.07	<0.09	<0.07	<0.03

4.1 PIPING AND HARDWARE

The radionuclide concentrations determined in pCi/cm² were converted to the various nuclear plant system inventories (e.g., reactor coolant lines, turbine, condenser, feedwater system, etc.) by multiplying by the total internal surface areas of these systems. The estimated total internal surface areas for the operating systems have been determined through rather extensive examination of schematic drawings, specification documents, and the final safety analysis report for Dresden One.

The estimated surface areas for the major operating systems are shown in Table 4.1. The radwaste system inventory was the most difficult to estimate for Dresden One. The estimated total surface area for this system was based eventually upon that estimated by Oak, et al.⁽²⁾ since the necessary information was not readily available at Dresden. This estimate for the radwaste surface area should be within a factor of two. Detailed system piping and instrument diagrams, blueprints, as-built, and selected isometrics were available for the other operating systems at Dresden One. It is felt that the accuracy associated with surface area estimates for the operating system, except radwaste, is in the range of $\pm 50\%$ or better.

TABLE 4.1. Total Surface Area Estimates for Dresden One Operating System Components

A. Condensate System <u>Component</u>	<u>Surface Area (cm²)</u>
Hotwell	5.72×10^6
Piping	2.72×10^6
Storage Tank	9.15×10^5
Demineralizers	2.95×10^5
Condensate Drip Tank	1.21×10^5
Inter Condensers	1.00×10^6
After Condensers	7.80×10^5
Gland Exhauster	1.86×10^6
Total = 1.34×10^7 cm ²	

TABLE 4.1 Continued

B. Feedwater System		<u>Surface Area (cm²)</u>
	<u>Component</u>	
1	<u>Primary</u>	
	Piping	1.37×10^7
	Steam Drum	8.42×10^6
	Emergency Condenser	1.06×10^6
	Feedwater Heater	3.80×10^7
	Primary Total =	$6.12 \times 10^7 \text{ cm}^2$
2	<u>Secondary</u>	
	Piping	$3.93 \times 10^6 \text{ cm}^2$
	Feedwater Heaters	$2.069 \times 10^7 \text{ cm}^2$
	Steam Generators	$2.533 \times 10^7 \text{ cm}^2$
	Secondary Total =	$5.00 \times 10^7 \text{ cm}^2$
C. Steam Lines		<u>Surface Area (cm²)</u>
	<u>Component</u>	
1	Drum to turbine	$2.29 \times 10^6 \text{ cm}^2$
2	Generator to turbine	$2.45 \times 10^6 \text{ cm}^2$
	Extraction steam lines	4.57×10^6
	Feedwater Heaters 1	$4.015 \times 10^7 \text{ cm}^2$
	2	$2.201 \times 10^7 \text{ cm}^2$
	Extraction steam removal lines	$1.091 \times 10^6 \text{ cm}^2$
	Secondary steam turbine to condenser pipe	$3.902 \times 10^5 \text{ cm}^2$
	Steam separators in extraction system	$8.048 \times 10^5 \text{ cm}^2$
	Total =	$7.38 \times 10^7 \text{ cm}^2$
D. Unloading Heat Exchanger System		<u>Surface Area (cm²)</u>
	<u>Component</u>	
	Piping	$1.327 \times 10^5 \text{ cm}^2$
	Heat exchangers	$9.734 \times 10^5 \text{ cm}^2$
	Total =	$1.11 \times 10^6 \text{ cm}^2$
E. Reactor Cleanup Demineralizer System		<u>Surface Area (cm²)</u>
	<u>Component</u>	
	Piping	$2.817 \times 10^5 \text{ cm}^2$
	Demineralizer	$1.760 \times 10^5 \text{ cm}^2$
	Regen. heat exchangers	$1.506 \times 10^6 \text{ cm}^2$
	Non-Regen. heat exchangers	$1.289 \times 10^6 \text{ cm}^2$
	Total =	$3.25 \times 10^6 \text{ cm}^2$

TABLE 4.1 Continued

F. Rad Waste System	7.8 X 10 ⁶ 6.4 X 10 ⁶
	Total = 1.42 X 10 ⁷
G. Fuel Storage Areas	
<u>Component</u>	<u>Surface Area (cm²)</u>
Spent fuel pool	2.81 X 10 ⁶
Fuel transfer pool	3.97 X 10 ⁶
Fuel transfer tunnel	2.72 X 10 ⁶
Heat exchanger	5.46 X 10 ⁵
	Total = 1.00 X 10 ⁷ cm ²
H. Fuel Transfer Areas	
<u>Component</u>	<u>Surface Area (cm²)</u>
Fuel transfer tube	5.64 X 10 ⁵
Reactor transfer canal	6.99 X 10 ⁶
	Total = 7.55 X 10 ⁶ cm ²

TABLE 4.2. Samples Used for Radionuclide Inventory Construction

System	Sample Used for Inventory Estimate
A. Condensate	D-SC-18
B. Feedwater	D-SC-18
C. Steam	D-SC-20
D. Unloading Heat Exchangers	D-SC-18
E. Reactor Cleanup System	D-SC-18
F. Rad Waste	D-SC-18
G. Fuel Storage	D-SC-12
H. Fuel Transfer	D-SC-16

Table 4.2 lists the samples used for residual radionuclide inventory construction for the various operating components of Dresden One. Table 4.3 contains the radionuclide concentrations in the four samples used for the basis of the inventory construction. Table 4.4 lists the inventories in curies for the residual radionuclides in corrosion films. The largest inventory of residual radionuclides was associated with the reactor feedwater and cleanup systems, including radwaste. Cobalt-60, ^{55}Fe , ^{63}Ni , and ^{65}Zn were the most abundant radionuclides. The next largest inventories were associated with the condensate system followed by the fuel storage and steam system.

Table 4.5 lists the total residual radionuclide inventory in Dresden One. The inventory has been calculated for two dates: our sampling date of August, 1982, and the shutdown date for the Unit of October 31, 1978. This total inventory was formulated by summing the individual operating system inventories in Table 4.4. The inventory in Table 4.5 does not include the neutron-activated pressure vessel and internals, the neutron-activated bioshield, concrete surfaces, residues, sludges, and resins in tanks and sumps, nor the spent fuel stored on site.

The ^{60}Co inventory accounted for 65 percent of the total radionuclide inventory at the time of our sampling and was also the most predominant radionuclide at shutdown, accounting for slightly less than 50 percent of the total inventory. The second most abundant radionuclide present in the Dresden One residual corrosion films was ^{55}Fe . This radionuclide accounted for approximately one-quarter of the total inventory. The third most abundant radionuclide at the time of our sampling was ^{63}Ni . The estimated 110 curies of ^{63}Ni were approximately 10 percent of the total inventory during our sampling period, but only 5 percent of the estimated inventory at shutdown. Because of its relatively long half-life, ^{63}Ni will become an increasingly larger fraction of the total inventory at Dresden One as the shorter-lived radionuclides such as ^{55}Fe , ^{60}Co , ^{65}Zn and ^{137}Cs decay away.

Zinc-65 accounted for less than one percent of the total estimated inventory at the time of our sampling. However, at shutdown it contributed approximately 20 percent of the total estimated inventory. The short half-life of ^{65}Zn , 0.668 yrs, will make this radionuclide insignificant to the overall inventory in several more years.

Table 4.3 Residual Radionuclide Concentrations in Corrosion Films from Dresden One, August 1982

		CONCENTRATIONS - pCi/cm ²															
SAMPLE DESCRIPTION	SAMPLE #	⁵⁴ Mn	⁶⁰ Co	⁶⁵ Zn	⁹⁴ Nb	¹⁰⁶ Ru	^{108m} Ag	^{110m} Ag	¹²⁵ Sb	¹²⁶ Sn	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵² Eu	¹⁵⁴ Eu	¹⁵⁵ Eu	^{186m} Ho
FUEL POOL LADDER	D-SC-12	1,700	200,000	2,800	<40	<600	<400	<600	<500	<30	6,000	50,100	<200	<100	<100	<80	<50
FUEL TRANSFER CHUTE	D-SC-16	3.7	790	11	<0.3	<4	<3	<5	<1	<0.2	3.3	24	4	<0.8	<1	<0.5	<0.4
REACTOR STEAM VENT LINE	D-SC-18	12,000	7.9 × 10 ⁶	113,000	<500	<8,000	<5,000	<9,000	<2,000	<400	<800	5,500	12,000	<2,000	<2,000	<1,000	<700
LOW PRESSURE INLET TO TURBINE	D-SC-20	<2	2,270	37	<0.5	<8	<5	<8	<4	<0.4	<0.7	3.1	<3	5	<2	<1	<0.7

SAMPLE DESCRIPTION	SAMPLE #	⁵⁵ Fe	⁵⁹ Ni	⁶³ Ni	⁹⁰ Sr	⁹⁹ Tc	¹²⁹ I	²³⁸ Pu	²³⁹⁻²⁴⁰ Pu	²⁴¹ Am	²⁴² Cm	²⁴⁴ Cm
FUEL POOL LADDER	D-SC-12	164,000	560	6,600	80	45	<0.4	61	24	33	7.2	56
FUEL TRANSFER CHUTE	D-SC-16	440	8.9	3,700	0.062	0.009	<0.02	0.16	0.067	0.11	0.020	0.20
REACTOR STEAM VENT LINE	D-SC-18	2.9 × 10 ⁶	26,000	1.3 × 10 ⁶	690	4.5	<1	720	360	550	100	840
LOW PRESSURE INLET TO TURBINE	D-SC-20	2,200	10	1,000	2.9	0.05	<0.01	0.42	0.68	0.98	0.015	0.19

Table 4.4 Residual Radionuclide Inventories in Various Operating Systems of Dresden One, August 1982

	INVENTORY IN Curies																
	AREA (cm ²)	⁵⁴ Mn	⁶⁰ Co	⁶⁵ Zn	⁹⁴ Nb	¹⁰⁶ Ru	^{108m} Ag	^{110m} Ag	¹²⁵ Sb	¹²⁶ Sn	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵² Eu	¹⁵⁴ Eu	¹⁵⁵ Eu	^{166m} Ho
(18) CONDENSATE	1.34 × 10 ⁷	1.6 × 10 ⁻¹	110	1.5	7 × 10 ⁻³	1 × 10 ⁻¹	7 × 10 ⁻²	1 × 10 ⁻¹	3 × 10 ⁻²	5 × 10 ⁻³	1 × 10 ⁻²	7.4 × 10 ⁻²	1.6 × 10 ⁻¹	3 × 10 ⁻²	3 × 10 ⁻²	1 × 10 ⁻²	9 × 10 ⁻³
(18) FEED WATER	5.00 × 10 ⁷	6.0 × 10 ⁻¹	400	5.7	3 × 10 ⁻²	4 × 10 ⁻¹	3 × 10 ⁻¹	5 × 10 ⁻¹	1 × 10 ⁻¹	2 × 10 ⁻²	4 × 10 ⁻²	2.8 × 10 ⁻¹	6.0 × 10 ⁻¹	1 × 10 ⁻¹	1 × 10 ⁻¹	5 × 10 ⁻²	4 × 10 ⁻²
(20) STEAM	7.38 × 10 ⁷	1 × 10 ⁻⁴	1.7 × 10 ⁻¹	2.7 × 10 ⁻³	4 × 10 ⁻⁵	6 × 10 ⁻⁴	4 × 10 ⁻⁴	6 × 10 ⁻⁴	3 × 10 ⁻⁴	3 × 10 ⁻⁵	5 × 10 ⁻⁵	2.3 × 10 ⁻⁴	2 × 10 ⁻⁴	3.7 × 10 ⁻⁴	1 × 10 ⁻⁴	7 × 10 ⁻⁵	5 × 10 ⁻⁵
(18) UNLOADING HT EXCH	1.11 × 10 ⁶	1.3 × 10 ⁻²	8.8	1.3 × 10 ⁻¹	6 × 10 ⁻⁴	9 × 10 ⁻³	6 × 10 ⁻³	1 × 10 ⁻²	2 × 10 ⁻³	4 × 10 ⁻⁴	9 × 10 ⁻⁴	6.1 × 10 ⁻³	1.3 × 10 ⁻²	2 × 10 ⁻⁴	2 × 10 ⁻⁴	1 × 10 ⁻⁴	8 × 10 ⁻⁵
(18) REACTOR CLEANUP	3.25 × 10 ⁶	3.9 × 10 ⁻²	26	3.7 × 10 ⁻¹	2 × 10 ⁻³	3 × 10 ⁻²	2 × 10 ⁻²	3 × 10 ⁻²	7 × 10 ⁻³	1 × 10 ⁻³	3 × 10 ⁻³	1.8 × 10 ⁻²	3.8 × 10 ⁻²	7 × 10 ⁻³	3 × 10 ⁻³	3 × 10 ⁻³	2 × 10 ⁻³
(18) RAD WASTE	1.42 × 10 ⁷	1.7 × 10 ⁻¹	110	1.6	7 × 10 ⁻³	1 × 10 ⁻¹	7 × 10 ⁻²	1 × 10 ⁻¹	3 × 10 ⁻²	6 × 10 ⁻³	1 × 10 ⁻²	7.8 × 10 ⁻²	1.7 × 10 ⁻¹	3 × 10 ⁻²	3 × 10 ⁻²	1 × 10 ⁻²	1 × 10 ⁻³
(12) FUEL STORAGE	1.00 × 10 ⁷	1.7 × 10 ⁻²	2.0	2.8 × 10 ⁻²	4 × 10 ⁻⁴	6 × 10 ⁻³	4 × 10 ⁻³	6 × 10 ⁻³	5.0 × 10 ⁻³	3 × 10 ⁻⁴	6.0 × 10 ⁻²	5.0 × 10 ⁻¹	2 × 10 ⁻³	1 × 10 ⁻³	1 × 10 ⁻³	8 × 10 ⁻⁴	5 × 10 ⁻⁴
(16) FUEL TRANSFER	7.55 × 10 ⁶	2.8 × 10 ⁻⁵	6.0 × 10 ⁻³	8.3 × 10 ⁻⁵	22 × 10 ⁻⁶	3 × 10 ⁻⁶	2 × 10 ⁻⁵	4 × 10 ⁻⁵	8 × 10 ⁻⁶	2 × 10 ⁻⁶	2.5 × 10 ⁻⁵	1.8 × 10 ⁻⁴	3.0 × 10 ⁻⁵	6 × 10 ⁻⁶	8 × 10 ⁻⁶	4 × 10 ⁻⁶	3 × 10 ⁻⁶
TOTAL		1.0	660	9.3					5.0 × 10 ⁻³		0.06	0.96	0.98	3.7 × 10 ⁻⁴			

	⁵⁵ Fe	⁵⁹ Ni	⁶³ Ni	⁹⁰ Sr	⁹⁹ Tc	¹²⁹ I	²³⁸ Pu	^{239 240} Pu	²⁴¹ Am	²⁴² Cm	²⁴⁴ Cm
(18) CONDENSATE	40	3.5 × 10 ⁻¹	17	8.9 × 10 ⁻³	6.0 × 10 ⁻⁵	1 × 10 ⁻⁵	9.6 × 10 ⁻³	4.8 × 10 ⁻³	7.4 × 10 ⁻³	1.4 × 10 ⁻³	1.1 × 10 ⁻²
(18) FEED WATER	150	1.3	65	3.5 × 10 ⁻²	2.3 × 10 ⁻⁴	5 × 10 ⁻⁵	3.6 × 10 ⁻²	1.8 × 10 ⁻²	2.8 × 10 ⁻²	5.1 × 10 ⁻³	4.2 × 10 ⁻²
(20) STEAM	1.6 × 10 ⁻¹	7.4 × 10 ⁻⁴	7.4 × 10 ⁻²	2.1 × 10 ⁻⁴	3.7 × 10 ⁻⁶	7 × 10 ⁻⁷	3.1 × 10 ⁻⁵	5.0 × 10 ⁻⁵	7.2 × 10 ⁻⁵	1.1 × 10 ⁻⁶	1.4 × 10 ⁻⁵
(18) UNLOADING HT EXCH	3.3	2.9 × 10 ⁻²	1.4	7.4 × 10 ⁻⁴	5.0 × 10 ⁻⁶	1 × 10 ⁻⁶	8.0 × 10 ⁻⁴	4.0 × 10 ⁻⁴	6.1 × 10 ⁻⁴	1.1 × 10 ⁻⁴	9.3 × 10 ⁻⁴
(18) REACTOR CLEANUP	9.7	8.5 × 10 ⁻²	4.2	2.2 × 10 ⁻³	1.5 × 10 ⁻⁵	3 × 10 ⁻⁶	2.3 × 10 ⁻³	1.2 × 10 ⁻³	1.8 × 10 ⁻³	3.3 × 10 ⁻⁴	2.7 × 10 ⁻³
(18) RAD WASTE	42	3.7 × 10 ⁻¹	18	9.9 × 10 ⁻³	6.4 × 10 ⁻⁵	1 × 10 ⁻⁵	1.0 × 10 ⁻²	5.1 × 10 ⁻³	7.8 × 10 ⁻³	1.5 × 10 ⁻³	1.2 × 10 ⁻²
(12) FUEL STORAGE	16	5.6 × 10 ⁻³	6.6 × 10 ⁻²	8.0 × 10 ⁻⁴	4.5 × 10 ⁻⁴	4 × 10 ⁻⁶	6.1 × 10 ⁻⁴	2.4 × 10 ⁻⁴	3.3 × 10 ⁻⁴	7.3 × 10 ⁻⁵	5.6 × 10 ⁻⁴
(16) FUEL TRANSFER	3.3 × 10 ⁻³	6.7 × 10 ⁻⁵	2.8 × 10 ⁻²	4.7 × 10 ⁻⁷	6.8 × 10 ⁻⁸	2 × 10 ⁻⁷	1.2 × 10 ⁻⁶	5.1 × 10 ⁻⁷	8.3 × 10 ⁻⁷	1.4 × 10 ⁻⁷	1.5 × 10 ⁻⁶
TOTAL	250	2.1	110	0.078	8.3 × 10 ⁻⁴		5.9 × 10 ⁻²	3.0 × 10 ⁻²	4.6 × 10 ⁻²	8.5 × 10 ⁻³	6.92 × 10 ⁻²

TABLE 4.5. Total Residual Radionuclide Inventory at Dresden One
- August, 1982(a) and at Shutdown - October, 1978

<u>Radionuclide (b)</u>	<u>Half-Life (years)</u>	<u>Inventory (curies)</u>	
		August, 1982	Oct. 31, 1978 (Shutdown)
^{60}Co	5.27	660	1,080
^{55}Fe	2.7	250	650
^{63}Ni	100	110	110
^{65}Zn	0.668	9.3	450
^{59}Ni	75,000	2.1	2.1
^{54}Mn	0.855	1.0	20
^{144}Ce	0.779	0.98	30
^{137}Cs	30.2	0.96	1.0
^{244}Cm	18	0.069	0.08
^{134}Cs	2.06	0.060	0.21
^{238}Pu	87.7	0.059	0.061
^{241}Am	432	0.046	0.046
$^{239-240}\text{Pu}$	24,100	0.030	0.030
^{242}Cm	0.446	0.0085	2.9
Total		1,030	2,350

- (a) Excluding neutron-activated pressure vessel and internals, biological shield, concrete surfaces, residues, sludges and resins in tanks and sumps, and spent fuel.
- (b) Other long-lived radionuclides specifically listed in 10 CFR 61,(9) e.g., ^{94}Nb , ^{99}Tc , and ^{129}I , were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

The total transuranic radionuclides contributed approximately 0.1 percent of the total estimated long-lived inventory at shutdown and only

0.02 percent at the time of our sampling. This decrease in contribution was due primarily to the decay of the relatively short-lived ^{242}Cm ($t_{1/2} = 0.446$ yr). The longer half-lives of the remaining transuranics would make them somewhat more significant to the overall inventory as the shorter-lived activation products (^{60}Co , ^{55}Fe , ^{65}Zn) continue to decrease more rapidly.

Commonwealth Edison has performed another inventory estimation for the primary loop components⁽¹²⁾. The primary loop inventory was estimated for two dates, 1972 and 1979. The earlier estimate was 3000 ± 1000 curies and the more recent inventory estimate was 660 curies, of which 82% was ^{60}Co . The primary loop inventory estimates for 1979 from the Commonwealth Edison report and the primary loop inventory from this work are shown in Table 4.6.

The two inventory estimates for ^{54}Mn , ^{60}Co , and ^{144}Ce are very similar. The estimates for ^{55}Fe , ^{63}Ni and the transuranics are not in such good agreement. This might be anticipated since these radionuclides require extensive radiochemical separation prior to measurement, thus introducing greater potential for analytical error.

TABLE 4.6. Comparison of Inventory Estimates in the Dresden One Primary System at Shutdown (1978)

Nuclide	Half-Life (years)	PNL Estimate	Commonwealth Edison Estimate ⁽¹²⁾
^{54}Mn	0.885	17	15
^{55}Fe	2.7	460	91
^{60}Co	5.27	740	570
^{59}Ni	75,000	1.4	---
^{63}Ni	100	73	26
^{65}Zn	0.668	390	---
^{125}Sb	2.77	<0.3	<0.06
^{137}Cs	30.2	0.33	---
^{144}Ce	0.779	23	29
^{238}Pu	87.7	0.04	0.3
$^{239-240}\text{Pu}$	24,100	0.02	0.2
^{241}Am	432	0.03	0.1
^{242}Cm	0.446	2.8	1.4
^{244}Cm	18.1	0.05	0.1
Total Curies		1700	733

Cobalt-60 is the most abundant radionuclide present, the major contributor to dose for many years, and thus the most important radionuclide in the table for decommissioning considerations. Consequently the good agreement between the two inventory estimates for this radionuclide is significant.

Even though there is agreement between inventory estimates for several radionuclides, the total inventory estimates differ by over a factor of 2. This points up the fact that total inventories can be misleading unless similar radionuclides are considered and all significant radionuclides are included. The major difference in the two inventory totals are due (1) to the exclusion of ^{65}Zn from the Commonwealth Edison inventory, and (2) a large potential underestimation of the ^{55}Fe inventory in the Commonwealth Edison estimate. The short half-life of ^{65}Zn makes it somewhat insignificant in terms of the long-term residual inventory, and the significance of the underestimation of ^{55}Fe can be minimized since it is a very minor contributor to external dose and also has a relatively short half-life of 2.7 years.

4.2 CONCRETE

No inventory estimate for residual radionuclides associated with concrete surfaces was attempted since such an inventory would be tenuous due to the extreme variability of radionuclide distributions throughout the plant. The total quantities of radionuclides associated with contaminated concrete would be a small fraction in comparison to the inventories internally deposited in piping and equipment.

4.3 SOILS

Low but detectable concentrations of reactor-derived residual radionuclides were detected around the Unit One sphere equipment penetrations and in the radwaste treatment area. The extent of this low level contamination could not be assessed during this cursory sampling. Therefore, an inventory estimate has not been attempted. The sampling conducted indicated a more comprehensive examination will be required

to determine the degree and extent of any remedial action for exclusion area soils around Unit One.

5.0 CONCLUSIONS

- The total estimated inventory of long-lived residual radionuclides translocated from the pressure vessel and redistributed throughout Dresden One at shutdown on Oct. 31, 1978 was 2350 curies, of which approximately 1100 curies, or slightly less than 50 percent, were due to ^{60}Co . Approximately 28 percent of the estimated inventory at shutdown, or 650 curies, was due to ^{55}Fe . Zinc-65 was the third most abundant radionuclide and contributed slightly less than 20 percent of the total inventory at shutdown. Concentrations of ^{94}Nb , ^{99}Tc , ^{129}I were insignificant contributors to the radionuclide inventory.
- At the time of our sampling in August, 1982, the residual radionuclide inventory had decreased to an estimated 1030 curies. The most abundant radionuclide was ^{60}Co , which contributed approximately 65 percent of the total inventory, or 660 curies. Iron-55 was still the second most abundant radionuclide, contributing an estimated 250 curies, or approximately 24 percent of the total inventory. Nickel-63 had become the third most abundant radionuclide by August, 1982, and contributed 110 curies, or 11 percent of the total inventory. Zinc-65, though still detectable, only contributed 9.3 curies, or slightly less than 1 percent of the residual radionuclide inventory at the time of our sampling.
- The transuranic radionuclides contributed 0.1 percent of the residual inventory at shutdown and 0.02 percent at the time of our sampling. The decrease between 1978 and 1982 was due to the decay of ^{242}Cm . With further time and radionuclide decay, the longer-lived transuranics (^{238}Pu , $^{239-240}\text{Pu}$ and ^{241}Am) will contribute a continually increasing fraction of the total residual inventory.
- The inventory calculated at shutdown (1978) estimated from this work agrees reasonably well with a Commonwealth Edison estimate in 1979(12). The Commonwealth Edison estimate was constructed preparatory to planned decontamination of the Dresden One primary loop. The total residual radionuclide in the primary loop was estimated at 660 curies in 1979. Approximately

500 curies of ^{60}Co and 70 curies of ^{55}Fe were estimated to be present in the primary loop by Commonwealth Edison. At shutdown these would have been 570 and 90 curies for ^{60}Co and ^{55}Fe , respectively. Using the inventory estimate associated with the feedwater, steam, reactor cleanup and unloading heat exchangers in the present study as comparable to the primary system from the Commonwealth Edison estimate would lead to an inventory estimate of 740 and 460 curies of ^{60}Co and ^{55}Fe , respectively, at shutdown in the primary loop. The agreement between the ^{60}Co estimates is quite good; however the agreement between the ^{55}Fe estimates is not as good. Cobalt-60, however, is a much more significant radionuclide because it is the major contributor to external dose and has a longer half-life.

- The inventory estimates do not take into account the pressure vessel and internals, contaminated concrete, spent fuel, and other materials stored in the spent fuel pool, or sludges which may be present in sumps, storage tanks, fuel pools, or other locations. The inventory estimates include only radionuclides which would be associated with normal corrosion films present in piping and on contaminated structural surfaces.
- The major repositories of residual radionuclides (within the nuclear plant systems) appear to be, in order of significance: the reactor water piping and equipment, the radwaste tanks and equipment, the condenser and condensate system, and the fuel pool and associated systems.
- Residual radionuclide concentrations associated with contaminated piping surfaces and hardware at Dresden One are well below Class A waste limits as defined in 10 CFR 61(9).
- The total measured inventory of residual radionuclides translocated from the pressure vessel and redeposited throughout Dresden One was about one-fourth (28 percent) the inventory predicted for a reference BWR power station by Oak, et al.(2). The Dresden One electrical output rating is approximately 18 percent of that for the reference BWR. Thus, the Dresden inventory estimated in this report does not exactly scale to that of the reference BWR on the basis of power output ratings. The difference may be related to the dual cycle design of the Dresden Unit combined with the initial use of admiralty brass heat exchangers in the feedwater system.

- The greatest contribution to the external whole body gamma dose rate immediately after shutdown was from ^{65}Zn and ^{60}Co . At the time of our sampling, ^{60}Co was the dominant dose contributor and will continue to be for a period of up to about 50 to 60 years following shutdown. After 50 to 60 years, ^{137}Cs would be the dominant external dose contributor although ^{59}Ni and ^{63}Ni would be present in greater abundances. After the ^{137}Cs has decayed, the most abundant long-lived radionuclides will still be ^{63}Ni , and ^{59}Ni , along with ^{241}Am , $^{239-240}\text{Pu}$ and ^{238}Pu .
- A wide variation of radionuclide concentrations was found in concrete cores obtained on the site. As noted at other reactor sites, the vast majority of the radionuclide inventory in the concrete is associated with the upper 1 cm of the cores.
- Low-level soil contamination immediately adjacent to the equipment penetration openings to the Dresden One sphere was detected. Soils in the radwaste treatment area also contained low levels of reactor-derived radionuclides. The cursory examination indicates that further evaluation of exclusion area soils around Dresden One should be conducted at the time of decommissioning.

REFERENCES

1. Smith, R. I., G. J. Konzek and W. E. Kennedy, Jr. 1978. Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. NUREG/CR-0130, Vol. 1 and 2. Prepared for the U. S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.
2. Oak, H. D., G. M. Holter, W. E. Kennedy, Jr., and G. J. Konzek. 1980. Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station. NUREG/CR-0672, Vol. 1 and 2. Prepared for the U. S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.
3. Manion, W. J. and T. S. LaGuardia. 1980. Decommissioning Handbook, DOE/EV/10128-1. Prepared for the U. S. Department of Energy by Nuclear Energy Services, Incorporated, Automation Industries, Danbury, Connecticut.
4. Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities. 1981. NUREG-0586, U. S. Nuclear Regulatory Commission, Office of Standards Development, Washington, D. C.
5. Osterhout, M. M., ed. 1980. Decontamination and Decommissioning of Nuclear Facilities. Plenum Press, New York.
6. Robertson, D. E., C. W. Thomas, K. H. Abel, J. C. Evans, M. W. Leale, E. A. Lepel, and E. M. Woodruff. 1982. Topical Report - Residual Radionuclide Distribution and Inventory at the Pathfinder Generating Plant, PNL-4326. Prepared for the U. S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.
7. Abel, K. H., D. E. Robertson, C. W. Thomas, E. A. Lepel, and W. V. Thomas. 1983. Residual Radionuclide Distribution and Inventory at the Humboldt Bay Nuclear Plant. PNL-4628. Prepared for the U. S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.
8. D. E. Robertson, C. W. Thomas, K. H. Abel, E. A. Lepel, W. V. Thomas, L. C. Carrick and J. C. Evans. Residual Radionuclide Distribution and Inventory at the Monticello Nuclear Generating Plant. PNL-7494. Prepared for the U. S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.
9. USNRC. 1981. Licensing Requirements of Land Disposal of Radioactive Wastes. 10 CFR 61. Federal Register, Vol. 46, No. 142, Friday, July 24, 1981. Proposed Rules.
10. U. S. Nuclear Regulatory Commission. Termination of Operating Licenses for Nuclear Reactors, Regulatory Guide 1.86, June, 1974.
11. Robertson, D. E., C. W. Thomas, R. W. Perkins, V. W. Thomas. 1981. Trans-uranium and Other Long-Lived Radionuclides in the Terrestrial Environs of Nuclear Power Plants. EPRI-EA-2045, Project 1059, Final Report. Prepared by Battelle, Pacific Northwest Laboratories for Electric Power Research Institute, Palo Alto, California.

12. Commonwealth Edison Company, 1980. Final Environmental Statement related to Primary Cooling System Decontamination at Dresden Nuclear Power Station, Unit No. 1. NUREG-0686 U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C.

APPENDIX A

Description of Dresden Nuclear Plant, Unit No. 1

STEAM GENERATION SYSTEM

INTRODUCTION

The Dresden steam generation system is a dual cycle boiling water unit. This is a modified form of the direct boiling water cycle. In a direct boiling water cycle, steam is generated throughout the reactor core and, after adequate separation of the water and steam, is sent directly to the turbine. The dual cycle approach, as utilized at Dresden, removes as much energy as possible by boiling in the upper part of the core without exceeding the limit of reactivity tied up in the voids for good stability.

The application of the dual cycle was designed to improve the power output per unit of core volume and the reactor response to electrical load demands. Both of these improvements resulted from sub-cooling the reactor coolant in the process of generating secondary steam in the secondary steam generators. Thus, with the same steam voids in the core and the same size reactor vessel, this design matched the primary steam production of a direct cycle and, in addition, obtained the secondary steam production. The sub-cooling of the reactor coolant water was essentially a measure of the amount of energy being extracted in the secondary steam generator. Therefore, with the proper sub-cooling vs. reactivity characteristics designed into the reactor core, it was possible to control reactor power levels over a limited range by varying secondary steam flow without moving control rods. This was accomplished by adjusting the position of the turbine secondary admission valves. An increase in sub-cooling produced a rapid increase in reactor heat flux, resulting in a fast response to increase in load demand. The reverse was true on a load drop. Figure A-1 shows a schematic of the steam generation system for the dual cycle system.

The placement of the equipment that made up the steam generation system and its auxiliaries is shown in the Reactor and Recirculation Loop Isometric, Figure A-2.

The primary steam generated in the reactor core by heat transferred directly from the nuclear fuel was carried in a steam-water mixture from the reactor to the primary steam drum where 1,600,000 pounds of steam per hour at rated loads were separated from the mixture and taken off at 965 psia for use in the turbine.

FIGURE A-2. Reactor and Recirculation Loop Isometric

The reactor recirculating water, containing heat also transferred directly from the nuclear fuel in its passage through the core, flowed from the primary steam drum to the suction of the recirculating pumps, and from there was discharged to the tube side of the secondary steam generators. The 25,602,330 pounds per hour of recirculating water continued on from the secondary steam generators, after transmitting a portion of its heat to the production of 1,440,000 pounds per hour of secondary steam, to return to the reactor. Secondary steam, produced at a rated plant load pressure of 475 psia, was supplied to the secondary admission of the turbine.

PRIMARY STEAM

Reactor

The reactor vessel serves as the furnace in which nuclear fuel, through the fission process, produced heat utilized in the direct generation of primary steam. It included components such as the vessel, core, control rods, and miscellaneous instruments and devices required for safe operation.

Vessel

The reactor vessel contains the reactor core and its accessories and provides the containment for the reactor light water moderator and coolant. It was designed for 1250 psig and is normally expected to operate at 1015 psia.

The reactor vessel is approximately 40 feet in height and 12 feet in diameter. The vessel was constructed basically of carbon steel plate internally clad with Type 304 stainless steel. This latter requirement was imposed since the vessel would contact high purity water at pH of 7.0, and it was desirable to keep corrosion products to a minimum in the primary reactor water system.

Access to the reactor vessel internals for refueling, changing control rods, and servicing of instrumentation was through a single hemispherical head closure. This head closure is secured in place during normal reactor operation by 56 head bolts. The major vessel openings, besides the head closure, consist of twelve 16-inch steam-water outlet nozzles located at the upper portion of the cylindrical section, and four 22-inch inlet nozzles entering near the juncture of the cylindrical section and the integral hemispherical bottom head. Each of the latter serve as return inlets for the reactor coolant water coming from the four secondary steam generators. In addition, there are numerous instrument and control rod penetrations.

Core

The reactor core, in which energy is generated by nuclear fission, is contained and supported within the reactor vessel. The basic reactor fuel is uranium dioxide of 2.24 percent ^{235}U enrichment. The uranium dioxide is fabricated as a sintered solid in the form of cylindrical pellets for installation within zircaloy-2 tubes. The core assembly consists of 464 fuel assemblies, each fuel assembly consisting of 36 fuel rods or 35 fuel rods and 1 water rod arranged in a square array inside a zircaloy-2 channel. Design of the core is such that sufficient fuel is contained in the reactor so that partial refueling is necessary no more frequently than once every nine to twelve months. The bottom of the core support plate is at elevation 512 feet, 7 inches.

Risers

The steam-water mixture produced within the core entered the space above and was directed by a turning vane to the twelve reactor vessel outlet nozzles connecting to the 16-inch, Type 304 stainless steel, steam-water risers feeding into the primary steam drum.

Control Rods

The reactor core for full power operation was about 17-20 times larger than the critical size, i.e., it contained much more than the critical amount of fuel. This influenced the number of neutron absorbers or control rods required to effectively maintain full range control for variation in power level and for fast shutdown. To accomplish this, eighty cruciform control rods spaced throughout the reactor core were provided.

As long as the control rods were in the core, the reactor was subcritical, and the neutron density small and constant. By withdrawal of the control rods out of the core, the neutron density, and therefore, the reactor power, increased. The neutron absorbing section of the control rods consists of B_4C powder encapsulated into stainless steel tubes.

The control rods operate external to the fuel channels and use the channel surfaces for guides within the core assembly. Each control rod is activated by a hydraulic drive mechanism through nozzles provided in the bottom of the reactor vessel.

Shield Cooling

During reactor operation, the absorption of gamma and neutron energy from the reactor core becomes a source of heat generated within the biological shield. The biological shield consists of a minimum of three feet of sand retained between a steel cylindrical shell outside the reactor and the 4-foot concrete box-shaped structural wall. To remove this heat, a shield cooling system, consisting of carbon steel cooling coils embedded within the sand fill, was provided. The system used cooling water from the reactor enclosure equipment cooling water system.

PRIMARY STEAM DRUM

General Description

The primary steam drum serves to separate the steam and water mixture and to provide a firm source of feed to the reactor recirculating pumps. The primary steam drum, built by the Foster Wheeler Corporation, was designed for a maximum pressure of 1250 psig. The rated load design conditions are shown in Table A-1.

Steam-Water Mixture Separation

The steam was separated from the steam-water mixture in the following manner (see Figure A-3). The steam water mixture from the reactor flowed up through 12 risers to the drum. Each riser is connected internally to a longitudinal distributing manifold containing guide vanes, which fed the group of separators in the compartment.

In these centrifugal type separators, the mixture was spun around a horizontal center line, centrifugal action driving the water to the outside of the curved section. The water was skimmed off by a knife edge and discharged from a rectangular opening against a baffle which diverts the flow horizontally at the normal water level. Water escaping past the knife edge flowed through the perforations in the splatter plate to the secondary drain. The accumulation of water in the secondary drain formed a trap, the outlet near the normal water level. This trap prevented the discharge of steam through the secondary drain. The steam left the separators horizontally above the water level and discharged into a common space under the driers. The driers consist of multiple sets of "W" type elements arranged to provide for three sharp reversals of direction.

FIGURE A-3. Primary Drum Water Mixture Separator

Any water remaining in the steam leaving the separator contacted the element surface and drained by gravity to a collecting pan with a 1-inch drain discharging well below the water level. The dried steam flowed out through the 4-inch openings at the top to four dry box units, each of which was connected to a 12-inch outlet nozzle.

TABLE A-1

Steam mixture flow to drum - pounds per hour	28,200,000
Steam flow from drum - pounds per hour	1,760,000
Pressure at outlet - psig	1,000
Maximum pressure drop - inlet including entrance and exit losses - psig, design - 5.0 actual	3.72
Steam temperature - F	Saturated
Total solids - ppm	0.5 max.
pH	7.0 to 7.2
Cl ₂ - ppm	0.01 max.
Oxygen - ppm entering drum	3.5
Additives	None
Normal water level	4 inches above C _L
Weight dry--pounds	370,780
Water volume at N.W.L. - Cubic Feet	1,489

The separated water flowed to the bottom of the drum through a series of ten cutouts through the riser manifolds. Each cutout opening had a vortex eliminator mounted directly over it. The vortex eliminator was a set of parallel vertical plates extending up through the water. From the bottom section of the drum, the water entered the downcomer piping through ten connections each equipped with a plate type vortex eliminator and a 1/4-inch mesh, stainless steel screen cover. The latter was provided to prevent any large foreign particles from being carried to the recirculating pumps below.

SECONDARY STEAM GENERATORS

The four secondary steam generators produce 495 psig steam (rated capacity) by utilization of the heat contained in the reactor coolant water.

General Description

The cut-away view of the secondary steam generators is shown in Figure A-4. They are connected into the downcomer loops between the steam drum and the reactor inlet. These steam generators are vertical integral drum shell and tube type units utilizing a U-bend heat exchanger bundle. Each secondary steam generator is located in a shielded compartment at elevation 526 feet in the reactor enclosure with the associated recirculating pump. Each steam generator is designed for operating conditions as shown in Table A-2 below.

	<u>Tube Side</u>	<u>Shell Side</u>
Guaranteed		
Max. continuous flow rate-lbs/hr	water-6,050,000	steam-294,750
Pressure psig	1,050	525
Inlet temperature F	542.8	Saturated
Inlet enthalpy Btu/lb.	540.2	
Outlet temperature F	510	
Outlet enthalpy Btu/lb.	499.8	
Pressure drop psi	15.0	
Feedwater		
Flow - lbs/hr.		298,000
Temperature F		405
Enthalpy Btu/lb.		380.5
Blowdown		
Flow		3,250
Temperature		469
Enthalpy		451.8
Minimum continuous flow percent of max.		5.0
Maximum peak load percent of max.		125
Water Quality		
Total solids ppm	0.5	100
pH	7.0-7.2	8.5-10.5
Additives		None
Cl ₂ ppm	Max. 0.01	Max. 1.0
SiO ₂ ppm		Max. 0.5

Steam Generation Process

The hot primary coolant water enters the water box at the bottom head of the steam generator through a 22-inch inlet pipe, passes through the U-tube

FIGURE A-4. Secondary Steam Generator

bundle giving up heat to the water in the shell side, then flows out through the 18-inch outlet pipe to the reactor where it enters via 27" inlet pipe/nozzle.

The relatively cool feedwater (405°F at rated load) enters the shell just above the transition cone and flows through a circular distributor pipe and down through the annular space between the shell and tube bundle to the upper face of the tube sheet.

The water is then converted to steam as it flows up through the tube bundle. The wet steam enters a circular steam lane and flows through the twenty-eight moisture separators. The moisture separators are arranged in two rings on each side of the steam lane so that steam can be discharged toward the center of the drum, and toward the outside. The steam discharged from the separators flows through twenty-two V-type, inner and outer moisture eliminators, then into demister screens. Continuing, the steam is discharged upward into the topmost shell space from which it is delivered through a 12-inch nozzle to the secondary steam headers. All separators and demisters are drained to points below the water level to prevent moisture re-entrainment.

REACTOR AUXILIARY EQUIPMENT

INTRODUCTION

The reactor auxiliary equipment chapter contains a description of the equipment directly associated with the reactor and includes the emergency condenser system, the unloading heat exchanger system, the clean-up system, and the poison control system.

EMERGENCY CONDENSER SYSTEM

Function

The emergency condenser system is included as an auxiliary of the primary steam generation system to serve as an alternate heat sink in case the reactor is isolated from the main condenser. It is designed to dissipate only the reactor decay heat in the event the reactor scrams (emergency shutdown) coincident with closure of the primary isolation valves or the turbine stop and bypass valves. With loss of power supply it maintains water in the primary system for at least eight hours.

Initiation of operation of this system is caused by either:

- (1) Manually by operation of switches
- (2) High primary steam pressure (1050 psig)
- (3) Closure of the primary steam isolation valves (in anticipation of high primary steam pressure), or
- (4) Loss of auxiliary power.

SYSTEM DESCRIPTION

The emergency condenser system includes the emergency condenser which consists of a shell and two removable tube and header type bundles, one at each end of the condenser shell, and the double 8-inch steam supply and condensate 6-inch drain piping, both stainless steel, connecting the separate heat exchanger bundles to opposite sides of the primary steam drum. Both steam supply and condensate drain piping are equipped with motor-operated shut-off valves.

Unvalved 3/4-inch stainless steel vent lines connect from the upstream side of the steam valves to the 18-inch primary steam lines. Vent lines, equipped with a double set of remotely operated valves are connected to the downstream side of the steam valves.

Demineralized make-up water from storage tank "A" (contaminated) is supplied to the emergency condenser through a 2-inch line.

The center of the emergency condenser is located at elevation 657 feet, 1 inch; the center line of the primary steam drum is at elevation 634 feet, 2 inches; a difference of about 23 feet. The difference in elevation provides the heat required to return the condensate to the system.

OPERATION OF EMERGENCY CONDENSER SYSTEM

The motor-operated valve in the steam line to the top of each tube bundle is normally open allowing condensed steam to fill the tube bundle and piping. The motor-operated valve in the drain line from each tube bundle is normally closed. When the system is placed in operation, the drain valve is opened, causing the condensate to flow to the lower section of the drum and steam to flow into the tube bundle. The steam is condensed by transferring its latent heat to the water in the shell side which boils away at essentially atmospheric pressure and flows to the atmosphere through the 36-inch emergency condenser

vent penetrating the reactor enclosure. This process continues until the primary steam pressure is reduced below 1050 psig. With the pressure below 1050 psig, either one or both heat exchangers may be taken out of service to reduce the cooling rate; however, if the primary steam pressure should rise again to 1050 psig, the system would automatically be put back into service.

The double-valved vent connections, provided downstream from the motor-operated valves, are used to remove air from the lines. With the emergency condenser in service, a falling off of performance indicates that the tube side is becoming air (non-condensable) bound. Under these circumstances each bundle in turn is isolated by closing the inlet and outlet valves, and the vent valves are opened to permit the escape of the non-condensables. Then the vent valves are closed and the bundle put back into service before repeating the same procedure with the other bundle.

Bypass switches permit the bypassing (safety system sensors) of either tube bundle in case evidence of a leaky tube is indicated by excessive radioactivity in vapor expelled to the atmosphere via the 36-inch vent. These switches also permit venting of the system to be accomplished.

UNLOADING HEAT EXCHANGER SYSTEM

Function

The reactor unloading heat exchanger system is designed to serve two functions; the first, and prime function, is to remove reactor decay heat for sustained periods of time after shutdown, reducing reactor water temperature to 125°F. The second function is for adding heat to the reactor coolant during the initial critical loading experiments. The cooling cycle is used primarily during fuel loading, or reactor maintenance periods.

System Description

A schematic piping and instrument diagram showing the principal equipment of the unloading heat exchanger system and its interconnection through the various reactor system flow paths is shown in Figure A-5 below. The system consists of two separate parallel flow loops. Water is taken from the reactor vessel above the reactor core through 10-inch stainless steel lines and circulated through 6-inch carbon steel piping to the shell side of each heat exchanger by the reactor unloading recirculating pumps.

FIGURE A-5. Unloading Heat Exchanger System

The reactor water is returned to the bottom of the reactor vessel through 6-inch piping that joins with the reactor clean-up demineralizer stainless steel piping to enter the return lines from the secondary steam generators.

Double valving, each with a packing leakoff to the reactor enclosure drain tanks, is provided at the suction to the pumps and at the discharge of the heat exchangers. A double valved 1-inch drain line between each set of shut-off and discharge valves also terminates at the reactor enclosure drain tanks.

Service water is supplied to and discharged from the tube side of each heat exchanger through a 10-inch carbon steel header system. Gate valves are provided in the inlet and globe valves at the outlet of the heat exchangers. Steam from the heating system feeds through a 6-inch line to the 10-inch outlet piping of B unloading heat exchanger through a manually-operated gate valve. Drainage is taken from the inlet line.

CLEAN-UP SYSTEM

Function

The reactor clean-up system performs the function of removing undesirable corrosion-erosion products from the reactor primary coolant system. This system will control the primary coolant purity and, in so doing, will help prevent the buildup of radiation levels throughout the plant.

System Description

The system consists of two separate, duplicate loops, each having a capacity of 100,000 pounds of primary coolant per hour. Each loop contains a recirculating pump, a regenerative heat exchanger, a non-regenerative heat exchanger, demineralizer tank, resin transfer pump, resin storage tank, and associated piping.

The suction to each clean-up recirculating pump is normally through a 4-inch line from the 6-inch pressure equalizing crosstie between the two reactor recirculating pump suction headers. Suction to the clean-up pumps may also be taken from the 4-inch reactor blowdown line.

The primary coolant flows from the clean-up recirculating pump through a 4-inch pipe to the tube side of the regenerative heat exchanger and then to the non-regenerative heat exchanger in series. The primary coolant from the non-regenerative heat exchanger passes through a flow nozzle, then to either the clean-up demineralizer or the radioactive waste system.

After passing through the mixed bed demineralizer, the primary coolant is fed back through the shell side of the regenerative heat exchanger where it is heated.

The water leaving each clean-up loop flows to the reactor through a 4-inch line connecting to the 6-inch discharge line from each unloading system heat exchanger. Each 6-inch line connects to one of the 22-inch reactor recirculating pump discharge lines near the bottom of the reactor vessel.

FUEL AND CONTROL ROD HANDLING SYSTEM

INTRODUCTION

The fuel handling system installed in the Dresden Station and occupying the fuel building and part of the reactor building is designed to accomplish the following:

- (1) Receive and store new fuel.
- (2) Remove and store the reactor vessel head, turning vane assembly and related equipment from inside the reactor vessel.
- (3) Remove spent fuel assemblies from the reactor, transport to the spent fuel storage pool and install with new fuel assemblies.
- (4) Remove orifices from spent fuel, inspect and replace channels on new fuel bundles.
- (5) Store spent fuel.
- (6) Remove channels from spent fuel, inspect and install channels on new fuel bundles.
- (7) Load spent fuel bundles in shipping cask and ship cask.
- (8) Remove spent control rods from reactor and replace with new rods.
- (9) Remove and replace control rod guide tubes in the reactor.
- (10) Remove and repair or replace control rod drive mechanisms in the reactor.
- (11) Remove and store shield blocks.

The fuel handling system essentially consists of a water shielded path for the spent fuel to travel from the reactor vessel to the fuel building, equipment for handling the fuel and various reactor parts, a water shielded storage pool, and shipping and receiving facilities. The overall system is shown in Figure A-6.

To move fuel from or to the reactor vessel, the following operations are performed. The reactor vessel head and the turning vanes are stored in the fuel handling canal. Fuel assemblies are withdrawn from the reactor (or placed in the reactor) using the fuel bundle grapple and hoist.

The fuel assemblies are placed in fuel racks and moved horizontally using the fuel rack carrier through the canal to the south end near the fuel discharge tube. The fuel rack grapple is used to place the fuel racks into the fuel rack baskets.

These baskets are lifted with the 7-1/2-ton auxiliary hook of the reactor service crane, coupled with the fuel rack basket to crane grapple. The basket is lowered through the 42-inch fuel discharge tube into the fuel basket carrier located in the connecting tunnel between the sphere and fuel storage pools.

To store the used fuel in the pools, the fuel basket carrier is moved through the tunnel to the south end of the pools by means of the cable drive assembly. The fuel assemblies are lifted from the fuel rack basket using the fuel bundle grapple and hoist, which is affixed to the storage pit crane, and are then stored in fuel storage racks. Other operations include channel stripping and inspection which may be performed at any time.

STEAM UTILIZATION SYSTEM

INTRODUCTION

The Dresden steam utilization system is quite similar to a conventional plant with differences occasioned by use of primary and secondary steam, a feature of the dual cycle reactor system. Since it is essential to regulate primary steam pressure closely, the primary turbine control valves are positioned by an initial pressure regulator. Under normal load carrying conditions, variation in output is controlled by adjusting the secondary control valves.

FIGURE A-6. Fuel Handling System

Another difference is occasioned by the requirement of returning the highest quality water possible to the reactor. The feedwater supplied by the primary feed pumps through primary heaters to the reactor is demineralized. The feed to the secondary steam generators is pumped by the secondary feed pumps through a separate group of heaters.

TURBINE

GENERAL DESCRIPTION

Unit No. 1 steam turbine, manufactured by the General Electric Company, is a non-reheat, tandem compound, 1800 rpm unit having three elements: high pressure, intermediate pressure, and double flow low pressure sections. See Figures A-1 and A-2.

The unit rated at 192,000 kW is designed for the following steam conditions:

Primary steam pressure	950 psig, 538.4 °F dry saturated
------------------------	----------------------------------

Secondary steam pressure	460 psig, 458.5 °F dry saturated
--------------------------	----------------------------------

The turbine is connected to a hydrogen-cooled, four-pole, sixty cycle, three phase generator by means of a rigid coupling.

The exciters are separate motor driven units. There are two such units each including a main and pilot exciter.

The three turbine elements consist of a dual admission, single flow, high pressure section; a single flow, intermediate pressure section; and a double flow, low pressure section.

The turbine shaft consists of three sections: one for the high pressure, one for the intermediate, and one for the low pressure. The three shaft sections are connected by means of rigid couplings and operate as a single shaft.

The shaft is supported by six bearings. The No. 1 bearing is located in the front standard and the No. 2 bearing is located at the high pressure end of the high pressure turbine. No. 3 and No. 4 bearings are located at either end of the intermediate turbine and No. 5 and No. 6 bearings are located at opposite ends of the low pressure turbine.

The axial position of the rotor is maintained by a thrust bearing which is located in the No. 2 standard just ahead of the No. 3 bearing.

Steam Path

Primary steam from the dual cycle boiling water reactor is brought to the turbine through two 18-inch headers which are tied together in the sphere by a 14-inch line, and they are tied together just ahead of the turbine by means of the by-pass valve chest. Primary steam is admitted to the turbine through two primary stop valves, one in each 18-inch header, and four primary steam control valves mounted on the upper half of the high pressure shell. Flow through the high pressure section is from the intermediate pressure section end toward the front standard.

Secondary steam from the secondary steam generators is brought to the turbine through two 18-inch headers which are tied together ahead of the turbine by a 10-inch line. Secondary steam is admitted to the turbine ahead of the ninth stage through two secondary stop valves, one in each 18-inch header, and four secondary steam control valves. These valves are located in a separate valve chest in front of the turbine below the operating floor level.

The steam exhausting from the high pressure section flows to the intermediate pressure section through two cross-under pipes and from the intermediate pressure section exhausts through a large cross-over equipped with a moisture extractor to the double flow low pressure section. The low pressure section exhausts to the condenser.

Eight extraction points are provided. Five of these are for bleeding steam for feedwater heating and three are for water removal as follows:

11th stage; two 8-inch openings; steam bleed for "E" heaters.

15th stage; one 5-inch opening; water removal.

16th stage; one 14-inch opening; steam bleed for "D" heaters.

19th stage; one 20-inch opening; steam bleed for "C" heaters.

21st stage; one 30-inch opening; steam bleed for "B" heaters.

22nd stage; one 14-inch opening; water removal.

23rd stage; two 30-inch openings; steam bleed for "A" heaters.

24th stage; four 12-inch openings; water removal.

Upstream of the primary stop valves two 18-inch branch lines, connecting each steam lead to a bypass valve chest, are provided in order to allow primary steam to be bypassed to the condenser in case the steam produced by the reactor should exceed the need of the turbine. Eight bypass valves are provided with

the necessary pressure breakdown orifices and two desuperheating water valves which are operated by solenoid air valves controlled by switches on the by-pass valves.

Moisture Removal

Since the turbine operates on dry saturated steam, the design incorporates provisions for removing water from the steam as it flows through the machine.

Moisture removal buckets are used on two stages in the high pressure section, four stages in the intermediate section and four wheels in the low pressure double flow section. The moisture removal buckets have radial grooves in the back side of the vane section on the admission edge. These grooves trap moisture droplets and cause them to be thrown from the bucket tips into pockets formed in the base of the diaphragms from where the accumulated water is drained off.

The other stages, which do not have moisture removal buckets, are provided with pockets between diaphragms for catching the water thrown from the trailing edges of the bucket shroud.

CONDENSING SYSTEM

GENERAL DESCRIPTION OF SYSTEM

Condenser

The main steam condenser is a 120,000 square foot, single pass, divided water box type, manufactured by the Worthington Corp. Its design has special provisions for receiving and condensing large quantities of primary steam dumped to the condenser during a turbine trip-out, large volume of heater drains, and extraction steam and moisture which is bypassed to the condenser during operation with feedwater heaters out of service.

Due to the large volume of non-condensable gases which are entrained in the primary steam, more extensive provisions are made in the condenser for removal of these gases, including a separate reheating steam supply system for the deaerating sections below each tube bank. Large air cooling sections and air removal pipes are also provided at the center of each tube bank.

The single pass, divided water box tube arrangement allows reversal of circulating water flow through each bank of condenser tubes.

Other special provisions include double tube sheets to avoid condensate contamination with river water leaking through the tube rolls, 40,000 gallon hotwell storage of condensate, and a rubber expansion joint connection to the turbine exhaust flange.

Air Removal System

Air removal from the condenser is accomplished through a slotted 16-inch pipe extending the full length of the condenser and located at the top of each of the two tube banks above the air cooling sections. Two connecting 16-inch pipes extend through the south side of the condenser shell above the water boxes and join in a 20-inch pipe leading to the two steam air ejectors and mechanical vacuum pump.

Circulating Water System

A log boom is provided across the entrance to the intake canal at the Kankakee River to prevent floating debris from passing with the circulating water to the intake structure. Bar racks and revolving screens at the intake structure remove the small debris to a size which will not cause excessive condenser tube fouling. Provisions are made for stop log gates between the bar racks and the revolving screens. Water flows to the auxiliary service bay through two 42-inch square openings located in the walls separating this bay and the circulating water pump bays. Each opening is provided with a manually operated vertical slide gate.

Chlorine solution is injected intermittently into the circulating water at the two circulating water pump suctions in the intake structure bays through two 4-inch distributing pipes extending across the entire width of each bay near the bottom, containing 150 1/2-inch holes in two rows 35 degrees apart. Chlorine solution is injected through a similar header at the bottom of the auxiliary service bay. This prevents excessive growth of algae in the service water piping and condenser tubes.

Each circulating pump discharges to a 72-inch pipe which connects to the reversing flow valve manifold from which water flows in either direction through each inlet water box, tube bank, and outlet water box before recombining at the 96-inch line leading to the discharge headworks. A 96-inch square manually operated slide gate is located in the discharge headworks. At this point water enters the discharge canal and flows to the Illinois River. A log boom is located at the end of the discharge canal to prevent access by boats, etc.

A 48-inch equalizing pipe is provided between the two 72-inch condenser inlet pipes approximately 55 feet south of the intake structure. This cross-connection equalizes pressure between the 72-inch lines and allows operation with only one circulating pump. No shut-off valve is provided in this line.

A 20-inch pipe with manually operated shut-off valve located at the discharge headworks connects the circulating water discharge canal with the intake canal to permit some recirculation flow when icing conditions exist.

CONDENSATE SYSTEM

DESCRIPTION OF SYSTEM

The condensate system includes the condensate pumps, steam jet air ejector condensers, gland steam exhauster condensers, condensate demineralizers, chemical feeders, condensate storage tank, and condensate drip tank. The condensate demineralizer, condensate storage tank, and condensate drip tank systems are described in detail later in this chapter.

Condensate Pumps

Condensate leaves the south end of the east and west compartments of the condenser hotwell in two 18-inch carbon steel lines which join to supply a 24-inch header feeding the 20-inch suction lines to the three condensate pumps.

The total flow of condensate from the condenser at rated full load is 2,595,020 pounds per hour. It requires two of the 3100 gpm (1,550,000 pounds per hour) condensate pumps to deliver this flow. The third pump serves as a spare. Each pump discharges into a 10-inch carbon steel pipe which connects into a 14-inch condensate line. Off this section of condensate line is a 6-inch header feeding into two 3-inch lines supplying the primary steam bypass desuperheating stations. The 6-inch header has a 4-inch connection to the waste disposal system used to drain the condensate piping.

The 14-inch condensate line splits into two 14-inch parallel circuits, each of which supplies cooling water for a separate steam jet ejector.

Steam Jet Air Ejector Condensers

Normally the condensate is directed through either one or the other set of steam jet air ejector condensers. The 14-inch inlet line divides into two 12-inch lines which follow parallel paths through the inter- and after-conden-

sers after which 12-inch outlet pipes join to emerge as a 14-inch condensate line.

The steam jet air ejector condenser condenses the steam used in the steam jets. The inter-condenser drains flow through a 2-inch looped pipe to the main condenser and the after-condenser through a 2-inch looped pipe to the condensate drip tank.

Off-gas is vented through a 6-inch line from each after-condenser shell. These lines are equipped with flow transmitters (A, FIT/364 and B, FIT/365), the air signals from which are transmitted to a single flow recorder (FR/351) mounted on panelboard P-1 in the control room. Off of each line is taken a 6-inch vent containing a rupture diaphragm which, when ruptured from excessive air pressure, allows the off-gas to discharge to atmosphere.

Turbine Gland Exhauster Condensers

The 14-inch condensate line from each steam jet air ejector condenser enters the associated turbine gland exhauster condenser. The exhauster blower mounted on this condenser maintains a negative pressure inside the condenser shell drawing steam and inleakage of air from the turbine shaft and valve packing glands. The condensate flowing through the gland steam condenser condenses the steam, and the drains pass through a 2-inch looped pipe to the condensate drip tank.

The 6-inch off-gas line from each exhauster blower connects into a 24-inch carbon steel pipe line approximately 130 feet in length which exhausts into the ventilating stack. The size and length of this line are sufficient to provide a 3-minute holdup of gas which allows any activity to decay to a safe level before discharging to atmosphere.

Condensate Recirculation

The two 14-inch outlet lines from the gland steam exhauster condensers join into a single 14-inch condensate line. An 8-inch condensate recirculating line connects this section back to the condenser. It is required to protect the condensate pumps from overheating during periods of low flow. This minimum amount is in excess of the cooling requirements of the steam jet air ejector and the gland steam exhauster condensers. Therefore, both requirements are satisfied by condensate recirculation when little or no feedwater is required by the reactor or the secondary steam generators.

Condensate Demineralizer

The 1,405,600 pounds per hour of primary condensate feed into a header that serves the three 3-inch influent lines to the condensate demineralizers.

The mixed bed condensate demineralizers remove suspended and dissolved solids to minimize the production of radioactive material in the reactor primary loop.

Feedwater Pump Suctions

The 8-inch effluent lines from the demineralizers join into a single 10-inch line supplying condensate to the heater which feeds the 8-inch suction lines to the three primary feedwater pumps.

The 1,204,160 pounds per hour of secondary condensate feeds into a header which supplies the individual 8-inch suction lines to the three secondary feedwater pumps.

FEEDWATER HEATER SYSTEM

INTRODUCTION

Feedwater heating is accomplished by extracting steam at five stages along the steam path through the turbine, as a means by which the turbine cycle efficiency is improved and moisture removal from the turbine is expedited. Heaters are designated by letters, starting with "A" as the lowest pressure turbine stage at which steam is extracted. Two feedwater heaters are used at each stage since there is a dual feedwater heating system. The dual system consists of two streams designated as the primary (p) and secondary (s); therefore, the heaters are identified as follows: Ap, As, Bp, Bs, Cp, Cs, Dp, Ds, Ep, Es. The primary stream feeds the reactor cycle, while the secondary stream feeds the secondary steam cycle.

Condensate pumped from the hotwell flows through either of two parallel circuits in which are located the steam ejectors and gland steam exhaust condensers. Downstream from the juncture of the parallel circuits the line again divides to supply condensate to the suction of both the primary and the secondary feedwater pumps. Due to the arrangement of having the feed pumps ahead of the heaters, all of the heaters are of high pressure design since high pressure feedwater flows through the tube side of all ten heaters.

Heater drain pumps are incorporated only at "A" stage since all heater drains (condensed extracted steam) are cascaded from the highest pressure heater to the next lower pressure heater, and so on, down to stage "A". At "A" stage the drains are pumped through the "A" drain cooler to the condenser. Drain coolers are used only on the primary heaters at all five stages in order to simplify the system. The top three primary heaters (C, D and E) have integral drain coolers, while heaters "A" and "B" have separate external drain cooler units.

Both the primary and secondary steam is saturated; therefore, moisture must be removed to keep the moisture content of the steam in the turbine from exceeding the desired maximum of 12 percent. This is done in order to reduce blade and casing erosion and is accomplished by moisture removal at three stages (15, 22 and 24) in addition to the five feedwater heater extraction points.

At the three moisture removal stages, one percent of the stage flow is bled off in order to entrain and thus carry a large portion of the moisture present out of the turbine.

To minimize the heater tube erosion problem resulting from the large quantity of entrained moisture in the steam, moisture separators are incorporated in each of the five heater extraction lines. Each separator feeds steam to both the primary and secondary heaters at one stage.

WASTE DISPOSAL FACILITIES

Introduction

The radioactive waste treatment building and storage facilities are enclosed in a fenced area located about 150 feet directly east of the fuel building. This controlled area is 160 feet wide by 200 feet deep. A 16-foot double swing gate provides access to the area at the southwest corner.

Directly inside the west fence at the southwest corner of the area is the dry waste interim storage shelter where dry waste is placed prior to baling and burial in the dry waste storage vault. The dry waste burial vault is located directly east of the storage shelter.

The liquid waste storage tank is located in the northwest corner of the area. Just east of the liquid waste storage tank are the three liquid waste hold-up tanks.

Collection of liquids in the reactor enclosure turbine building and refueling building are routed in piping located in a tunnel extending from the railroad spur track area at the southeast corner of the turbine building east between the north end of the shop-warehouse-fuel building and the reactor enclosure to the north end of the radwaste building.

East of the waste hold-up tanks in the northeast corner of the area is the radioactive waste building which houses the waste testing equipment, control room, and the concentrated waste and permanent resin storage tanks.

Process waste piping has been provided with stub ended tees east of the reactor enclosure opposite a trench extending north. This trench has been excavated and sand filled to expedite a future addition to the turbine generating capacity of the plant.

APPENDIX B

Dresden Unit No. 1 Operating History

DRESDEN UNIT No. 1 OPERATING HISTORY

Unit No. 1 at Dresden produced power commercially from July, 1960, until October 31, 1978.

Initial criticality at Dresden Unit No. 1 was achieved on October 15, 1959, with 28 fuel elements. Testing using larger configurations up to 448 fuel elements continued into March, 1960, and the head was placed upon the reactor vessel March 23, 1960. During April, 1960, testing in the single cycle mode was conducted and the turbine generator rolled on nuclear steam April 15, 1960. Dual cycle testing was conducted in later April, and 315 MW_t (half-rated power) was reached on April 19.

Fully rated capacity 626 MW_t or 192 MW_e was achieved June 29, 1960, and on July 15, 1960, the unit was placed on the Commonwealth Edison electrical system. Full commercial status was started effective August 1, 1960, and the unit was officially dedicated on October 12, 1960, at which time a 40-year license was granted.

The unit had significant problems associated with control rods and undertook a control rod drive repair program from November, 1960, through March, 1961. In April, 1961, criticality testing was conducted with new control blades. On June 2, 1961, turbine generator operation was resumed. The licensed power of the unit was increased to 700 MW_t in September, 1962.

The unit had a history of minor steam leaks and erosion in steam piping in the early and mid-1960's, perhaps related to the use of saturated steam. There were also fuel failures during September-December, 1964, and other periods which, though not leading to off-gas releases above limits, did cause redistribution of radionuclides from fuel to the other primary systems. The rate of fuel failures at Dresden 1 is not typical for the earlier BWR units.

During other outages in the late 1960's, ultrasonic inspections were made on extensive sections of primary piping and welds because of concerns regarding intergranular cracking failures in some of the smaller 304 stainless steel piping.

There was also use of admiralty brass (Cu-Ni) heat exchange surfaces in the several systems at the plant, e.g., the main condenser. Most of these were taken out of service and replaced with stainless. In the 6th partial

refueling, the condenser was retubed from admiralty brass to 304L stainless steel.

The use of Cu-Ni surfaces did lead to translocation and deposition of corrosion production throughout the operating systems at levels above more modern units.

The use of carbon steel in the secondary feedwater system may have also contributed to the elevated corrosion radionuclide levels.

There were 11 fuel cycles from March, 1960, to October 1978. During this operational period, the net electrical generation was 15, 826, 516 MWeHr.

The unit was taken out of service as of October 31, 1978, and license amendments specified by NRC require major modifications primarily related to the IEEE-279 standard, "Criteria for Nuclear Power Plant Protective Systems," issued in 1968 and the addition of a high pressure coolant injection system (ECCS-HPCI).

However, as a result of the accident at TMI, even further modifications would undoubtedly be required for Unit 1 prior to any further power generation.

Commonwealth Edison is currently contemplating the future options for Unit No. 1. A chemical decontamination of the primary loop is scheduled for July, 1984, after which modifications and subsequent power generation or, alternatively, decommissioning, could proceed. A decision regarding future status is anticipated by early 1984.

APPENDIX C

Sample Inventory for Dresden Nuclear Power Station Unit No. 1

SAMPLE INVENTORY FOR DRESDEN NUCLEAR POWER STATION UNIT No. 1

During the period August 2 through August 11, 1982, a sampling program was conducted at Commonwealth Edison's Dresden Nuclear Power Station Unit No. 1.

An extensive sampling program was conducted at this site since the unit has not been operational since October, 1978, and the future status of the unit is uncertain. Commonwealth Edison is considering whether to update to current safety standards or decommission the unit. The staff at the nuclear station was extremely helpful in selecting appropriate sampling locations and hardware for sampling.

Commonwealth Edison is currently planning a decontamination effort in the unit. Therefore, it was necessary to maintain operational system integrity after sampling. Consequently, the sampling hardware was limited to scraping interior surfaces of opened systems.

Samples obtained during the onsite work included five surficial soil core samples from areas associated with Unit No. 1: sediment cores from the intake and discharge canals, nine concrete cores at selected locations through the nuclear unit, and thirteen scraping samples from the various reactor systems.

SAMPLE IDENTIFICATION AND INVENTORY

Samples were designated by a letter code detailing the origin or sample type as follows:

- D CC - Dresden, Concrete Core
- D SC - Dresden, Scraping from Reactor Systems
- D SS - Dresden, Surficial Soil Sample
- D Sed - Dresden, Sediment Core

DRESDEN UNIT No. 1 CONCRETE CORES

Nine concrete cores were taken using a coring system employing an industrial diamond bit and recirculated water as a cooling and lubricating medium. The cores are normally four inches (10 cm) in diameter and range in depth from

5 cm to approximately 20 cm. The table below gives a detailed listing of the coring locations and dose rate data associated with the surface face of each core.

PNL - ID	Dose Rate		Location
	γ	β	
DCC-1	13 mR/hr	12 mRad/hr	Dow Chemical Spill Area - 529' Containment
DCC-2	--	300	Secondary Steam Generator Room "B" - 529' Containment
DCC-3	--	<1	Hall in Front of Accumulator Room - 488' Containment
DCC-4	20	420	Sub-pile Room - 488' Containment
DCC-5	--	<1	Clean Makeup Demineralizer Room - 515' Turbine Bldg.
DCC-6	--	22	Condensate Pump Room Pit Turbine Bldg.
DCC-7	4	230	Unloading Heat Exchanger Room - 517' Containment
DCC-8A	2	32	Rad Waste Bldg. Basement Top of Core - 490'
DCC-8B	--	--	Rad Waste Bldg. Basement Bottom of Core - 490'
DCC-9	--	300	Rad Waste Bldg. Basement North - 490'
			End Drainage Trough

DRESDEN REACTOR SYSTEM SAMPLES

The reactor system samples at Dresden were all scrapings of corrosion films associated with steel piping or plate except for the floor drain from Instrument Room A. These samples were obtained by scraping the interior surface of the various systems as made available by plant operations personnel. The scrapings were taken using a paint scraper and specified surface areas were abraided. The samples are listed below by number; the location and dose rate associated with the sample are also indicated.

PNL - ID	Dose Rate		Location	
	γ	β		
D-SC-10	13 mR/hr	30 mRad/hr	Fuel Storage Pool Wall	517'
D-SC-11	4	9	Fuel Pool Water Circulation Pump	517'
D-SC-12	14	120	Fuel Pool Metal Ladder	517'
D-SC-13	--	--	Inside Sphere Wall	529'
D-SC-14	--	--	Containment Air Vent Inside Sphere	529'
D-SC-15	8	98	Floor Drain - Instrument Room A	502'
D-SC-16	4	12	Fuel Transfer Chute Bottom of Fuel Canal	--
D-SC-17	4	9	Fuel Canal Scrapings	559'
D-SC-18	400	1900	Reactor Steam Vent Line	584'
D-SC-19	--	--	Steam Relief Valve	--
D-SC-20	--	--	Low Pressure Inlet to Turbine	--
D-SC-21	--	--	High Pressure Outlet from Turbine	--
D-SC-22	--	--	Steam Line Piping	--

DRESDEN SOIL AND SEDIMENT SAMPLES

Seven samples (five soils, two sediments) were obtained from areas associated with Unit No. 1 activities. The samples are listed below with depths and sampling locations. The sediment cores were obtained from the canal bank using a plastic corer attached to a hand held long aluminum extension.

PNL - I.D.	Depths (cm)	Location
D-SS-23	0-7.5	North Side of Containment Sphere near Equipment Transfer Hatch
D-SS-24	0-7.5	East Side of Containment Sphere near Equipment Storage Area

D-SS-25	0-15	Rad Waste Area - in front of building doorway
D-SS-25B	15-30	Rad Waste Area - in front of building doorway
D-SS-26	0-7.5	Rad Waste Storage Tank Area
D-SS-27	0-7.5	South Side of Containment Sphere near refueling building
D-Sed-28	0-20	Inlet Canal Sediments
D-Sed-29	0-20	Outlet Canal Sediments

DISTRIBUTIONNo. of
CopiesOFFSITE

10 Dr. Carl Feldman
Nuclear Regulatory Commission
Office of Nuclear Regulatory
Research
Mail Stop 1130SS
Washington, D.C. 20545

12 D. J. Scott
Station Superintendent
Dresden Nuclear Power Station
RR No. 1
Morris, IL 60450

2 U.S. Nuclear Regulatory
Commission
Division of Technical Information
and Document Control
7920 Norfolk Avenue
Bethesda, MD 29914

2 DOE Technical Information

ONSITE

31 Pacific Northwest Laboratory

K. H. Abel (8)
D. E. Robertson (15)
E. A. Lepel
M. W. Leale
N. A. Wogman
Technical Information (3)
Publishing Coordination (2)