NUREG/CR-4485
PNL-5606

The Impact of Fuel Claddding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants

Case Study: PWR During Routine Operations

Prepared by M. P. Moeller, G. F. Martin, D. L. Haggard

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

NUREG/CR-4485
PNL-5606

The Impact of Fuel Cladding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants

Case Study: PWR During Routine Operations

Manuscript Completed: November 1985
Date Published: January 1986

Prepared by
M. P. Moeller, G. F. Martin, D. L. Haggard

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Safety Review and Oversight
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN B2987

ABSTRACT

The purpose of this report is to present data in support of evaluating the impact of fuel cladding failure events on occupational radiation exposure.To determine quantitatively whether fuel cladding failure contributes significantly to occupational radiation exposure, radiation exposure measurements were taken at comparable locations in two mirror-image pressurized-water reactors (PWRs) and their common auxiliary building. One reactor, Unit B, was experiencing degraded fuel characterized as 0.125 percent fuel pin-hole leakers and was operating at approximately 55 percent of the reactor's licensed maximum core power, while the other reactor, Unit A, was operating under normal conditions with less than 0.01 percent fuel pin-hole leakers at 100 percent of the reactor's licensed maximum core power. Measurements consisted of gamma spectral analyses, radiation exposure rates and airborne radionuclide concentrations. In addition, data from primary coolant sample results for the previous 20 months on both reactor coolant systems were analyzed. The results of the measurements and coolant sample analyses suggest that a 3560-megawatt-thermal (1100 MWe) PWR operating at full power with 0.125 percent failed fuel can experience an increase of 540 percent in radiation exposure rates as compared to a PWR operating with normal fuel. In specific plant areas, the degraded fuel may elevate radiation exposure rates even more.

CONTENTS

FIGURES

TABLES

EXECUTIVE SUMMARY

To date, probabilistic risk assessments of the consequences of nuclear power plant operations have been oriented toward severe core damage and other major accidents. However, a significant number of existing and potential generic issues, including several with regulatory and safety goal implications, are concerned with incidents of lesser magnitude such as fuel cladding failure. Although these incidents can result in fuel damage, increased primary coolant activity and increased radiation levels within a plant, they do not generally result in any radiation exposure to the general public.

The NRC's policy is to consider radiation protection when making regulatory decisions pertaining to nuclear power plants. This consideration includes onsite occupational radiation exposures as well as potential offsite exposures to the general public. To adequately evaluate this issue, the impact of less serious reactor incidents on plant workers must be considered. Since, in many cases, radiation exposures may represent the dominant risk to plant workers, the techniques developed in this evaluation may be useful in assessing or estimating changes in occupational radiation exposures attributable to reactor incidents that are less serious than major accidents.

The purpose of this report is to present data in support of evaluating the radiological impact of fuel cladding failure incidents on workers at commercial U.S. nuclear power plants. To accomplish this goal, this report presents a case study designed to quantify the impact of increased fuel cladding failures on a plant's primary coolant activity and the associated radiation exposure rates. Radiation measurements were taken at comparable locations in two mirror-image pressurized-water reactors (PWRs) and their common auxiliary building. One reactor, Unit B, was experiencing degraded fuel characterized as 62 pin-hole leakers and was operating at approximately 55 percent of the reactor's licensed maximum core power. The other reactor, Unit A, was operating under normal fuel conditions at 100 percent of the reactor's licensed maximum core power. The 62 pin-hole leakers translate to 0.125 percent failed fuel, which is more than ten times the normal fuel operating conditions but is within the plant's technical specifications. Measurements consisted of gamma spectral analyses, radiation exposure rates and airborne radionuclide analyses. In addition, data based on primary coolant sample results for the previous 20 months on both reactor coolant systems were evaluated.

Modification of the regulations pertaining to fuel cladding failures and other incidents that are less serious than major accidents would have both positive and negative consequences. Using probabilistic risk assessment techniques to perform a value impact analysis is appropriate to weigh the opposing factors influencing the decision-making process. Data in this report are, therefore, presented in a format that would support the eventual use of probabilistic risk assessment techniques to help resolve this regulatory issue. A value impact analysis is not included here, however, because of the limited number of reactors on which measurements were made; additional radiation measurement data are required to make a value impact analysis meaningful for the majority of the commercial U.S. nuclear power plants.

Since the primary interest of this study was to identify contributions to occupational radiation exposures caused by increased reactor coolant activity resulting from degraded fuel, the selection of measurement locations was based on the need for routine worker access. To facilitate a comparison of data for the two units, every effort was made to select measurement sites common to both units. Because containment entries in Unit B had been curtailed due to elevated radiation levels and airborne contamination, no measurements were made in either of the containment structures. Instead, the measurement efforts were focused on the piping penetration building located between containment and the common auxiliary building and on the areas of the auxiliary building housing the radwaste, chemical and volume control systems.

The radiation measurements involved four discrete pieces of information. First, an intrinsic germanium detector was used to distinguish the energy of the gamma photons through spectral analyses. Second, at the same time and location of the gamma measurements, exposure rates were measured with a portable ionization chamber. Third, in measurement locations where airborne radionuclide concentrations were suspected, air samples were collected with a high-efficiency filter and silver-zeolite cartridge for analyses of particulates and iodine, respectively. Finally, data from the scheduled coolant sample analyses for the time period corresponding to when the measurements were performed were used to verify the percentage of failed fuel. Other information collected includes the reactor's age, power level, fuel burnup time and general operating history for each reactor. As a result, it was possible to determine the percentage of the dose rate attributable to fuel cladding failures as opposed to corrosion product buildup for both the failed fuel and baseline conditions. The air sample measurements were used to analyze the potential contribution to occupational exposures via inhalation. Consequently, the study was designed to provide the necessary information to determine whether fuel cladding failures contribute significantly to the occupational radiation exposure of workers.

Analyses of the gamma spectra began with the use of a computer code to perform a Gaussian fit and then identify significant photon peaks by energy. Peaks not identified by the code were identified manually. The identified peaks were classified according to whether they resulted from natural background, annihilation radiation, fission products or activation products. A computer code was developed and used to estimate the radiation exposure rate attributable to each identified radionuclide. From this information, the contributions to exposure rates due to fission products and activation products were determined for each gamma spectral measurement. To facilitate comparisons between Unit A and Unit B measurements, the Unit B measurements have been normalized to account for differences in the ages and operating power levels of the two reactors.

Based on the data collected and analyzed, the following conclusions can be drawn. The methodology and procedures presented are adequate to quantitatively evaluate the impact of fuel cladding failures on occupational radiation exposure rates at a nuclear power plant. The data suggest that a 3560-megawatt-thermal (1100 MWe) PWR operating at full power with 0.125 percent failed fuel can experience an increase of 540 percent in radiation exposure rates, as contrasted to a similar unit operating under normal fuel conditions.

In specific plant areas, the degraded fuel may elevate radiation exposure rates even more. These estimates assume that releases of fission products into the reactor coolant increase linearly with power level. Because the percentage of failed fuel was 1000 percent greater in Unit B than in Unit A, the 540-percent increase in radiation exposure rates suggests that fission product leakage rates into the reactor coolant and associated occupational radiation exposure rates would not increase linearly with the magnitude of fuel cladding failures. This is probably due to variations in the efficiency of the reactor cleanup systems for different levels of failed fuel and to the baseline of radiation exposure rates from activation products. Although occupational exposures will probably increase as a result of the increased radiation exposure rates, the magnitude of such an increase will depend on the amount of time workers spend in the affected areas. The primary impact on occupational radiation exposures will probably occur when workers perform routine maintenance and outage tasks on affected reactor and coolant-system components. The data suggest that the degraded fuel does not significantly affect the concentrations of airborne radionuclides in working areas inside the plant. These data may be misleading because plant personnel stated that the degraded fuel frequently results in periodic releases of radionuclides into work areas inside the plant, especially in the radwaste area. These releases included isotopes of xenon and iodine. Consequently, the degraded fuel could result in an increase in occupational radiation exposure from increased airborne radionuclide concentrations.

INTRODUCTION

Ever since it was established, the U.S. Nuclear Regulatory Commission (NRC) has supported programs designed to evaluate occupational radiation exposures and the relative significance of safety issues at commercial U.S. nuclear power plants. To ensure the continued safe operation of existing nuclear power plants, more attention is being given to refining the use of value impact analyses in implementing NRC policy on safety goals and in other regulatory applications especially amenable to probabilistic risk assessment techniques. These other applications could include regulating generic safety issues, formulating new NRC requirements, assessing and evaluating new designs, and setting priorities for nuclear power plant research and inspection.

To date, probabilistic risk assessments concerning the consequences of nuclear power plant operations have been oriented toward severe core damage and other major accidents. However, a significant number of existing and potential generic issues, including several with regulatory and safety goal implications, involve incidents of lesser magnitude, such as fuel cladding failure. Although, these incidents can result in fuel damage, increased primary coolant activity, and increased radiation exposure rates within a plant, they do not generally result in any radiation exposure to the general public. Before regulatory decisions pertaining to nuclear power plants are made, NRC considers both onsite and offsite radiation protection. Therefore, to adequately evaluate this issue, the impact of less serious incidents on plant workers must be considered. Since, in many cases, radiation exposures may represent the dominant risk to plant workers, this evaluation should assess or estimate any changes in occupational radiation exposures attributable to these incidents.

The purpose of this report is to present data on the radiological impact of fuel-cladding failures on workers at commercial U.S. nuclear power plants. To accomplish this goal, this report presents a case study designed to quantify the impact of increased fuel cladding failures on a plant's primary coolant activity and the associated radiation exposure rates. Radiation measurements were taken at comparable locations in two mirror-image pressurized-water reactors (PWRs) and at locations in their common auxiliary building. One reactor, Unit B, was experiencing degraded fuel characterized as 62 pin-hole leakers and was operating at approximately 55 percent of the reactor's licensed maximum core power. The other reactor, Unit A, was operating under normal fuel conditions at 100 percent of the reactor's licensed maximum core power. The 62 pin-hole leakers translate to 0.125 percent failed fuel, which is more than ten times the normal fuel operating conditions but is within the plant's technical specifications. Measurements consisted of gamma spectral analyses, radiation exposure rates and airborne radionuclide analyses. In addition, data based on primary coolant sample analyses for the previous 20 months on both reactor coolant systems were evaluated.

Modification of regulations pertaining to fuel cladding failures and similar incidents would have both positive and negative consequences. The attributable benefits of relaxing the regulations include the economic advantages of lengthening a plant's operating time through additional fuel exposure and the subsequent reduction in the frequency of refueling outages. Because the refueling outages result in significant radiation exposures to plant workers, reducing

their frequency could potentially decrease the plant's collective occupational radiation exposures. A negative aspect of relaxing the regulations would be the increased plant radiation levels, which could increase collective occupational radiation exposures. Therefore, quantifying the magnitude of this increase for all types of nuclear power plants is essential to the value impact analyses supporting the regulatory decision. Data presented in this study provide a means of characterizing the magnitude of this increase for the PWRs evaluated. Before a regulatory decision, a comparison should be made between the magnitude of the potential for reduced collective occupational doses because of fewer outages and the magnitude and consequences of the anticipated increased plant radiation exposure rates during routine operation.

Using probabilistic risk assessment techniques to perform a value impact analysis is appropriate to weigh the opposing factors influencing the decision - making process. Data in this report are presented in a format that would support the eventual use of probabilistic risk assessment techniques to help resolve this regulatory issue. A value impact analysis is not included here, however, because of the limited number of plants on which measurements were made and because additional radiation measurement data are required to make a value impact analysis meaningful for the majority of the commercial U.S. nuclear power plants.

Other sections of this report describe specific aspects of this project. These include the methodology and theory of the radiation measurement techniques used, a description of the location and equipment setup for each measurement, a presentation and analysis of the results, and a discussion of suggested conclusions to be drawn from the data.

Included in the appendices are a review of the radionuclides of interest and a presentation of the gamma spectral analyses, the source code list for the computer calculations, the dose rate data, and the reactor plant data.

METHODOLOGY

The collection of radiation measurement data to determine the contribution of fuel cladding failures to occupational radiation exposures is the primary objective of this project. To meet this objective, a study design is presented that eliminates many confounding variables. Key aspects of the study design include 1) the selection of the nuclear power plant site where the radiation measurements were to be collected, 2) the locations where the measurements were to be made, 3) the types of measurements to be performed, and 4) the supporting data to be requested from the plant.

In determining what type or characteristics of a plant would be advantageous, incidents potentially affecting onsite or offsite radiation levels, but not considered major reactor accidents, were reviewed. A potentially measurable incident that has an impact on occupational radiation exposure is fuel cladding failure. Such failures can affect occupational radiation exposures because they result in elevated primary coolant radionuclide concentrations and increased plant radiation levels. For comparison, measurements must be made at a plant experiencing fuel cladding failure and at a plant that does not have degraded fuel. The latter measurements are essential in order to establish a baseline for quantifying fluctuations in the measured parameters.

Collecting meaningful data in a cost-effective manner was a criterion for selecting the nuclear power plant site. A cost-effective approach is to take the measurements at a two-unit nuclear power plant where one reactor is experiencing fuel cladding failure and an identical reactor is not. This would make it possible for all measurements to be made during the same site visit. This approach was used, and measurements were taken in two mirror-image PWRs that share a common auxiliary building at one plant site.

Locations within the power plant units where radiation measurements were made were chosen so that variations in the source terms and associated occupational radiation exposures could be quantified. With respect to the source term, measurements were made throughout the chemical and volume control system. This system includes most major components outside of containment. With respect to occupational radiation exposures, measurements were made in areas of the plant where workers would be expected to be exposed during routine plant activities. Since routine plant activities at the site did not include containment entries, measurements were limited to areas outside the containment structure. Because the measurements were taken in two mirror-image units, it was possible to select common measurement locations in both units. This permitted direct comparison of the resulting data.

The radiation measurements involved four discrete pieces of information. First, an intrinsic germanium detector was used to distinguish the energy of the gamma photons through spectral analyses. Second, at the same time and location of the gamma measurements, exposure rates were measured with a portable ionization chamber. Third, in measured locations where radionuclide concentrations were suspected, air samples were collected with a high-efficiency filter and silver-zeolite cartridge for analyses of particulates and iodine, respectively. Finally, data from the scheduled coolant sample analyses for the time period corresponding to the measurements were used to verify the percentage of failed fuel. As a result, it was possible to determine the percentage of the dose rate attributable to the fuel cladding failures as opposed to

corrosion product buildup for both the failed fuel and the baseline conditions. The air sample measurements were used to analyze the potential contribution to occupational exposures via inhalation. Consequently, the study was designed to provide the necessary information to determine whether fuel cladding failures contribute significantly to the occupational radiation exposure of workers.

Supporting data requested from plant personnel included primarily coolant sample analyses. To eliminate confounding variables such as plant age, power level and fuel burnup time, data obtained through coolant sample analyses for the two units were reviewed. Evaluation of these data was designed to provide trends that should facilitate the elimination of some potentially confounding variables. Other information provided by plant personnel included the units' age, power level, fuel burnup time and general operating history for each unit. In addition, plant personnel characterized the degraded fuel as equivalent to 62 fuel-pin leakers.

MEASUREMENT PROCEDURES

Three different types of measurements were taken. These included gamma spectral data, radiation exposure rates, and airborne radionuclide concentrations. The following section discusses the equipment and procedures used for each type of measurement. An equipment list for each type of measurement performed is presented at the end of this section.

Gamma spectral measurements were made to identify the contributing radionuclides. The system included an intrinsic germanium (Ge) detector, preamplifier, high-voltage power source, amplifier, and multichannel analyzer. Data on the collected spectra were stored on magnetic tape cassettes for transfer to a computer for analyses. The multichannel analyzer was calibrated to 1 keV per channel over 4096 channels. The equipment block diagram is presented in Figure 1. Equipment calibration was verified before measurements were taken in Unit A and Unit B.

Raw spectral data were collected as pulse-height distributions. These pulse-height distributions were corrected for detector efficiency and scattering losses during analysis, and an energy calibration was performed in order to associate a photon energy with the pulse heights. A computer code was used to determine which spectral peaks were statistically significant and to identify the radionuclides. In addition to identifying radionuclides, the number of photons associated with each peak was recorded.

A detector can be calibrated for efficiency as a function of photon energy. The efficiency curve for an intrinsic Ge detector, similar to that used for the photon spectroscopy, is presented in Figure 2. The low-energy decline in efficiency is due to the thickness of the detector window, while the high-energy decline is due to the increased penetrating power of higher energy photons. The efficiency was measured using radionuclide mixtures with known relative photon intensities.

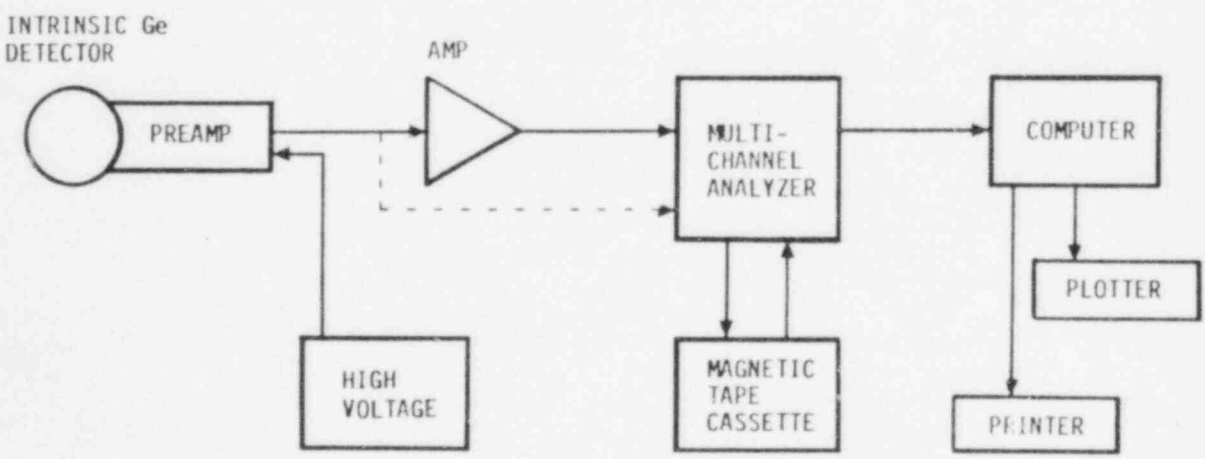

FIGURE 1. Equipment Arrangement for the Gamma Spectrometer

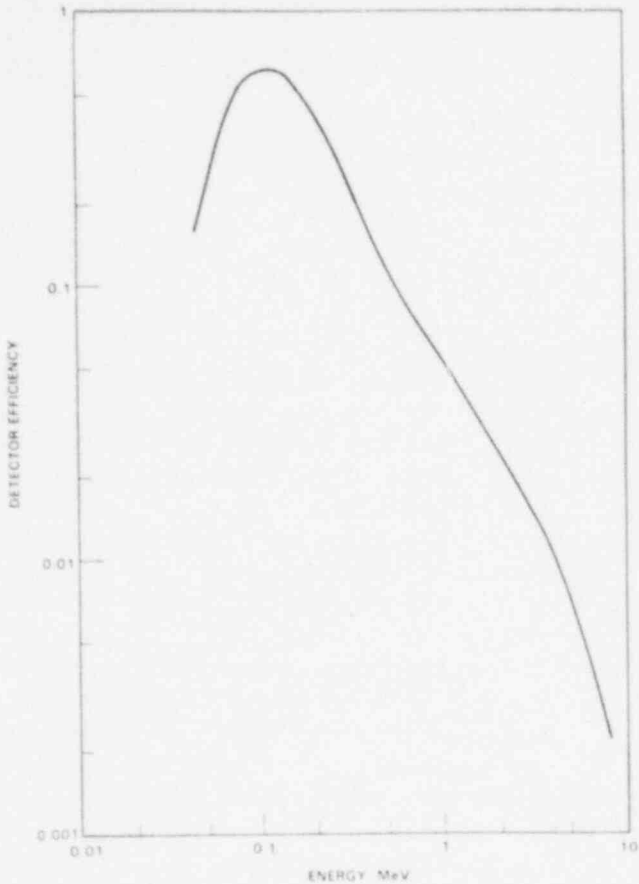

FIGURE 2. Efficiency Curve for a Ge Detector

For photon spectroscopy measurements at locations where the radiation exposure rate was in excess of 10 milliroentgens per hour (mR/hr), a lead collimator was used. The collimator consisted of a cylindrical, 2-inch-thick annulus with an inner diameter of 3 inches and an outer diameter of 7 inches. A 4-inch square, 2-inch-thick faceplate was fitted in front of the annulus. The faceplate had a 0.25-inch-diameter hole in its center. The collimator was positioned such that the detector was inside the annulus and behind the faceplate. The detector did not support any of the collimator's weight.

In addition to reducing the incident radiation flux at the detector, the collimator also affected the recorded spectra because of photon attenuation and scattering. The effect of these energy-dependent phenomena on detector response was determined experimentally and factored into the data analyses. Measurement spectra recorded using the collimator also exhibited lead x-ray peaks.

At each measurement location, a survey instrument was used to determine the dose rate at the detector position. In each case, the survey reading was taken with the instrument held in the same physical position as the detector during the measurements. The instrument used was a portable ionization chamber selected from instruments routinely in use at the plant. Calibration of the instrument was verified. This type of instrument is typically used to measure beta, gamma, and x-ray radiation, with four linear ranges of operation from 5 to 5,000 mR/hr. The detector chamber is air filled, vented to the atmosphere, and fitted with a mylar window and a sliding aluminum beta shield. For this study, all readings were made with the beta shield closed. Energy response curves provided by the manufacturer for this instrument indicate that for energies over 100 keV the instrument response is essentially independent of energy.

At measurement locations where it was suspected that airborne concentrations of radionuclides might exist, air samples were taken. Air samples were drawn using a high-volume portable air sampler fitted with a combination filter and cartridge holder. The sampler and holder were selected from instruments routinely used for plant surveys. The air sampler provided a continuously adjustable flow rate from 0 to 8 cubic feet per minute (cfm) and the holder assembly accommodated a 2-inch-diameter piece of filter paper and a 2.25-inch-diameter cartridge. The particulate filter paper used was designed to provide a high collection efficiency for approximately 0.3-micrometer or greater diameter particles at the flow rates created by the sampler. The cartridges used were silver-impregnated zeolite cartridges that were 37 percent silver by weight. They have a 100 percent collection efficiency for iodine. The samples were drawn using the plant's collection procedure, which requires a 4-cfm flow rate for 10 minutes, thus yielding a sample volume of 40 cubic feet. All filters and cartridges were counted using the plant's lithium-drifted germanium spectroscopy system and analyzed using energy peak identification software.

TABLE 1. List of Measurement Equipment

EQUIPMENT DESCRIPTION	Model Number	Serial Number
1. Photon Spectroscopy		
Princeton Gamma-Tech Intrinsic Germanium Detector	IGC 6	1274
Princeton Gamma-Tech Preamplifier	352	4183
Canberra Series 40 Multichannel Analyzer	4203	882641
Canberra Analog to Digital Converter	8075	118132 128169 118154 118150
Canberra Linear Amplifier	2022	683221 883266 883263 583209
Canberra Bin Power Supply	2000	9833875
Canberra Cassette Interface	5421	982899
Realistic (Miniset 9) Cassette Recorder	14812	1831107
2. Dose Rate Measurements		
Eberline Air Ion Chamber	RO-2	2043
3. Air Sampling		
RadeCo Hi-Volume Portable Sampler	H-809V-1	3411
Open-Face Combination Filter and Cartridge Holder	47mm/SAI/809	None

ONSITE MEASUREMENTS

To accomplish the study objectives, the measurement procedures were used to collect data in comparable locations in two reactors at a two-unit PWR plant site. These reactors were chosen as a result of differences in their primary coolant activity. One unit, Unit A, was operating at 100 percent of the reactor's licensed maximum core power under normal fuel conditions. The other unit, Unit B, was operating at 55 percent of the reactor's licensed maximum core power with elevated reactor coolant activity as a result of degraded fuel. The licensee had characterized the degraded fuel as equivalent to 62 fuel pins having pin-hole leaks. This value was determined by analysis of the fuel during the subsequent plant outage. Comparison of the Unit B reactor coolant sample analysis results with the values for 1 percent failed fuel documented in the plant's Final Safety Analysis Report (FSAR) and analysis using a diffusion model supported the 62-pin-hole-leaker value. The value for the percentage of failed fuel (0.125) is calculated by dividing the number of pinhole leakers (62) by the total number of fuel pins (49580) and multiplying by 100 (to obtain a percentage).

The remainder of this section gives the details of the measurement approach and of specific measurements.

MEASUREMENT APPROACH

Since the primary interest of this study was to identify contributions to occupational exposures caused by increased reactor coolant radionuclide concentrations resulting from degraded fuel, measurement locations were selected on the basis of the need for routine worker access. Because containment entries had been curtailed as a result of elevated levels of radiation and airborne contamination in Unit B (which was experiencing degraded fuel), no measurements were made inside the containment structures. Fifteen measurements were taken in seven comparable locations. Four measurements were taken in or near the primary piping penetration room located between containment and the auxiliary building. The remaining measurements were made in the radwaste areas of the auxiliary building. Several measurements were taken of the chemical and volume control system. Table 2 lists the unit (A or B) and the floor level elevation associated with the measurements as well as a brief description of the measurement locations. Figure 3 presents a schematic of the chemical and volume control system for Units A and B, illustrating the relative locations in the flow path where measurements were performed. Figures 4 through 7 are floor plans that identify the measuring points for each of the plant elevations. (The tables and figures are at the end of this section. The bold X's in figures 3 - 7 mark the places where measurements were taken.)

To facilitate a comparison between measurements from the two units, an attempt was made to select areas common to each unit that met the radiation level and routine access criteria. In addition, the measurement locations were selected in areas of the units as similar to each other as possible, and the detector setup was oriented the same for each pair of measurements. Because of design modifications, corresponding areas in the two units were not always the same. In several locations for example, piping layout and valve alignments were different. In these cases, every effort was made to orient

the detector in a similar position to the primary radiation source in the area.

To determine the primary contributors to exposure rates, measurements were taken at strategic locations in the flow path of the chemical and volume control system. For example, the measurements made in the piping penetration area provided a representative distribution of all the radionuclides contained in the reactor coolant system. The measurements taken in the purifying ion-exchanger valve rooms and in the bypass ion-exchanger valve rooms recorded the distribution of nuclides entering and leaving the demineralizers. The measurements made in the charging pump rooms indicated the mixture of radionuclides being returned to the primary reactor coolant system. Routine inspection and maintenance of the boric acid make-up system contribute to occupational radiation exposures. Consequently, measurements were taken in the boric-acid-level control valve rooms to provide an indication of the distribution of radionuclides associated with this system. In addition to the measurements taken on portions of the chemical and volume control system, general area measurements were taken in a well-traversed hallway in the auxiliary building and on a system modification made as a result of the degraded fuel conditions.

Most measurements were taken with the detector supported at a distance of 39 inches from the floor. If the area exposure rate, as measured at the detector with the portable ionization chamber, was greater than 10 mR/hr, then the lead collimator described in the measurement procedures section was used. The collimator allowed a measurement to be taken in a radiation field that would have otherwise overloaded the detector; that is, the electronic dead time would have been too high if the collimator had not been used. In areas where airborne radionuclides were suspected to contribute significantly to dose rates, air samples were taken. Table 2 summarizes the physical conditions for each measurement.

DATA COLLECTION

Measurements 12, 13, 14, and 15 were made in the penetration building on the 30-ft elevation. Measurements 13 and 15 were taken just inside the piping room containing the piping penetrations for the chemical and volume control system. The piping room is approximately 20 feet by 50 feet, one wall of which is the curved wall of the containment structure. The detector was located approximately 5 inches from the floor and oriented towards the wire screen gate into the piping room. No collimation was used on either of these measurements. Measurements 12 and 14 were made similarly just outside the primary piping penetration rooms.

Measurements 1 and 4 were taken on the 24-ft elevation in the middle of a 10-foot-wide hallway that runs the entire length of the auxiliary building. The detector was located near the letdown heat exchanger room and a radioactive pipeway. The detector was supported 39 inches from the floor and oriented parallel to the long axis of the hallway. No collimation was necessary for either of these measurements and no air samples were taken.

Measurements 2 and 5 were made inside the concrete cubicles housing the bypass ion-exchanger valves. These cubicles are 4 feet by 4 feet and are located on the 24-ft elevation. The detector was positioned just inside the entrance and supported 39 inches from the floor. The collimator was used for both measurements and the detector was aimed directly at the middle of the

pipe containing the control valve. For measurements 2 and 5, the detector was located 27 inches and 28 inches from the pipe, respectively. Air samples were collected in both of the valve rooms.

Measurements 3 and 6 were taken in the cubicles housing the boric acid pressure-level control valves. The cubicles measure approximately 4 feet by 4 feet and are on the 24-ft elevation. For both measurements, the detector was located just inside the entrance to the cubicle, supported 39 inches from the floor, and placed 37 inches from the pipe containing the control valve. The collimator was used for measurement 6, but not for measurement 3. No air samples were taken at these locations.

The cubicles containing the purifying ion-exchanger valves are located on the 50-ft elevation and measure approximately 4 feet by 10 feet. Measurements 8 and 9 were taken in these cubicles with the detector placed just inside the entrance, supported 39 inches from the floor and oriented towards the pipe containing the control valve. The detector, without the collimator, was positioned 46.5 and 51.5 inches from the pipe for measurement 8 and 9, respectively. Following the measurement, it was discovered that the measured valve for Unit B was not operating at the time of the measurement. The valve that was operating would have been measured, but it was in a valve room that was too heavily contaminated to permit a measurement. Therefore, the data for measurement 8 may not represent the radionuclide mixture or the radiation exposure rates that would be expected when this valve is operating. Air samples were taken at both locations.

The charging pumps are located on the 9-ft elevation and are housed in concrete rooms measuring approximately 12 feet by 20 feet. Measurements 10 and 11 were taken with the detector positioned just past the wall separating the valve room from the pump room and oriented towards the wall supporting the inlet and outlet piping. For both measurements, the detector was supported 39 inches from the floor and approximately 112 inches from the pipes. No collimation was used for these measurements. Air samples were collected in both locations.

Measurement 7 was taken on the 24-ft elevation of Unit B in the cubicle housing the valves for the pressurizer off-gas line, which runs from the Unit B pressurizer to the volume control tank. The existence of this line is due to a modification caused by the degraded fuel conditions in Unit B. Therefore, there is no corresponding measurement point on Unit A. This measurement was made because it represented an area contributing to occupational dose from the degraded fuel that would not otherwise exist. The cubicle is approximately 11 feet by 6 feet, and the detector was positioned just inside the entrance and just beyond the baffle wall. The detector was supported 39 inches from the floor and oriented toward the off-gas line at a distance of 68 inches. Because of the radiation exposure rate, the collimator was used. No air sample was taken at this location.

TABLE 2. Summary of Onsite Measurements

Measurement Number	Collimator Used	Detector Height (in.)	Distance From Primary Source(in.)	Dose Rate at Detector (mR/hr)	Air Sample Taken	Plant Floor Elevation (ft)	Unit	Area Description
1	N	39	NA	< .1	N	24	A	General area near letdown heat-exchanger room
2	Y	39	28	30	Y	24	A	Bypass ion-exchanger valve room/outlet
3	N	39	37	1.7	N	24	A	Boric acid make-up system room
4	N	39	NA	< .2	N	24	B	General area near letdown heat-exchanger room
5	Y	39	27	26	Y	24	B	Bypass ion-exchanger valve room/outlet
6	Y	39	37	11	N	24	B	Boric acid make-up system room
7	Y	39	68	29	N	24	B	Pressurizer off-gas line (1)
8	N	39	46.5	.5	Y	50	B	Purifying ion-exchanger valve room/inlet (2)
9	N	39	51.5	4	Y	50	A	Purifying ion-exchanger valve room/inlet
10	N	39	112	< .2	Y	9	A	Charging pump room
11	N	39	112	3	Y	9	B	Charging pump room
12	N	39	NA	.2	N	30	B	Outside pipe penetration room
13	N	39	NA	3.5	Y	30	B	Inside pipe penetration room
14	N	39	NA	< .2	N	30	A	Outside pipe penetration room
15	N	39	NA	1.2	Y	30	A	Inside pipe penetration room

(1) No comparable measurement location on Unit A.
(2) Valve not in service at time of measurement.
NA - Not applicable.

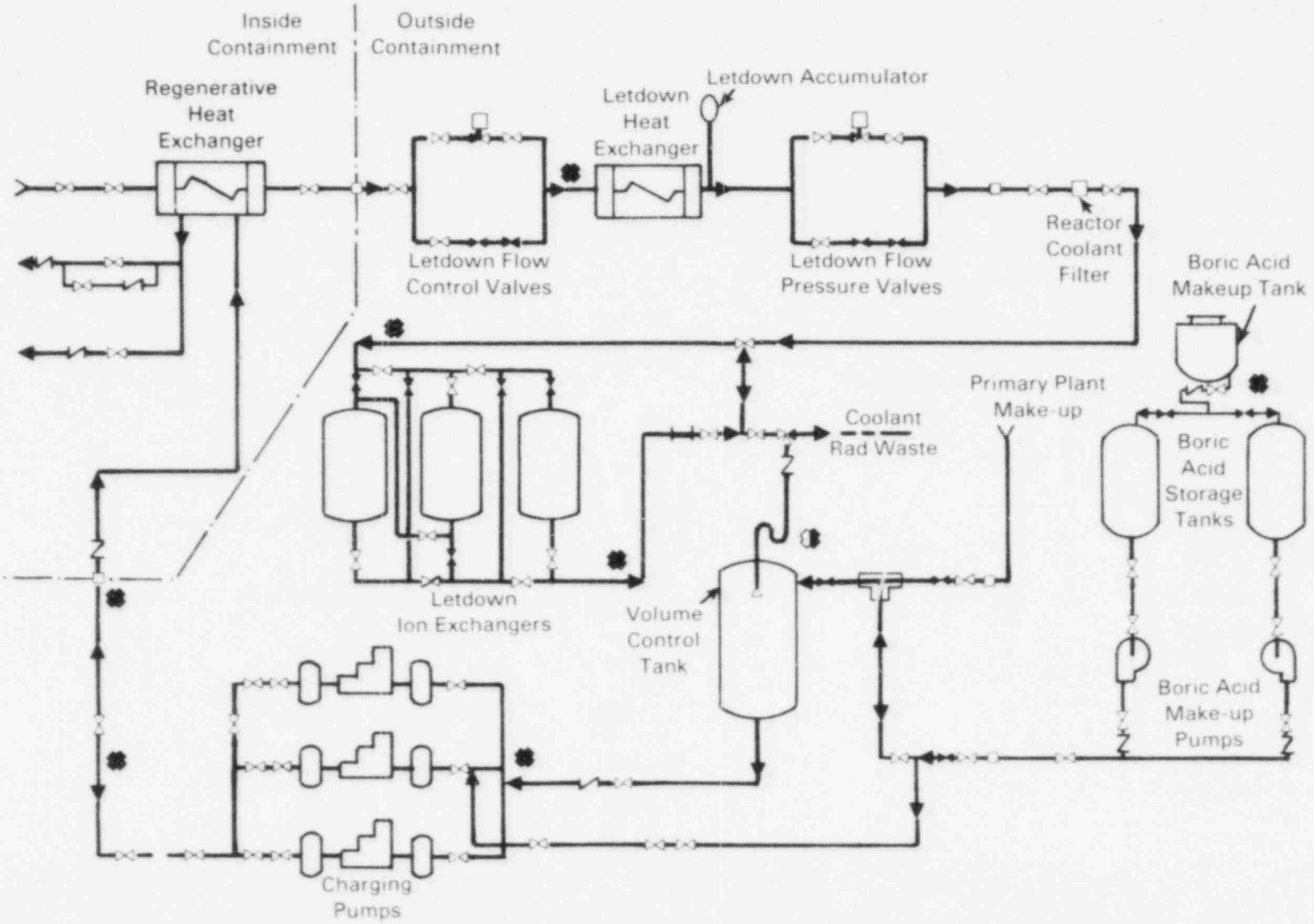

FIGURE 3. Chemical and Volume Control System

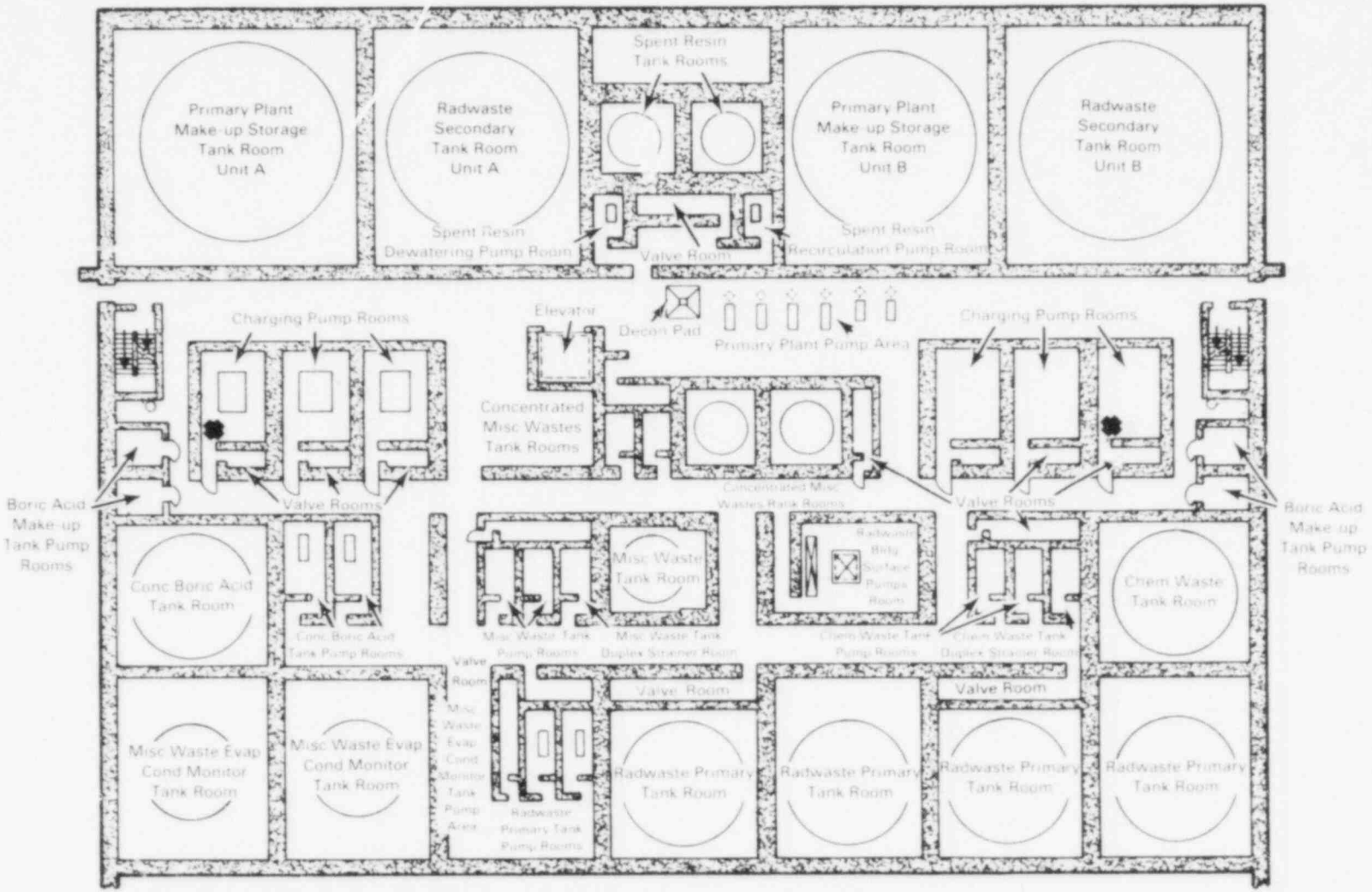

FIGURE 4. Auxiliary Building Floor Plan (3.0-ft elevation)

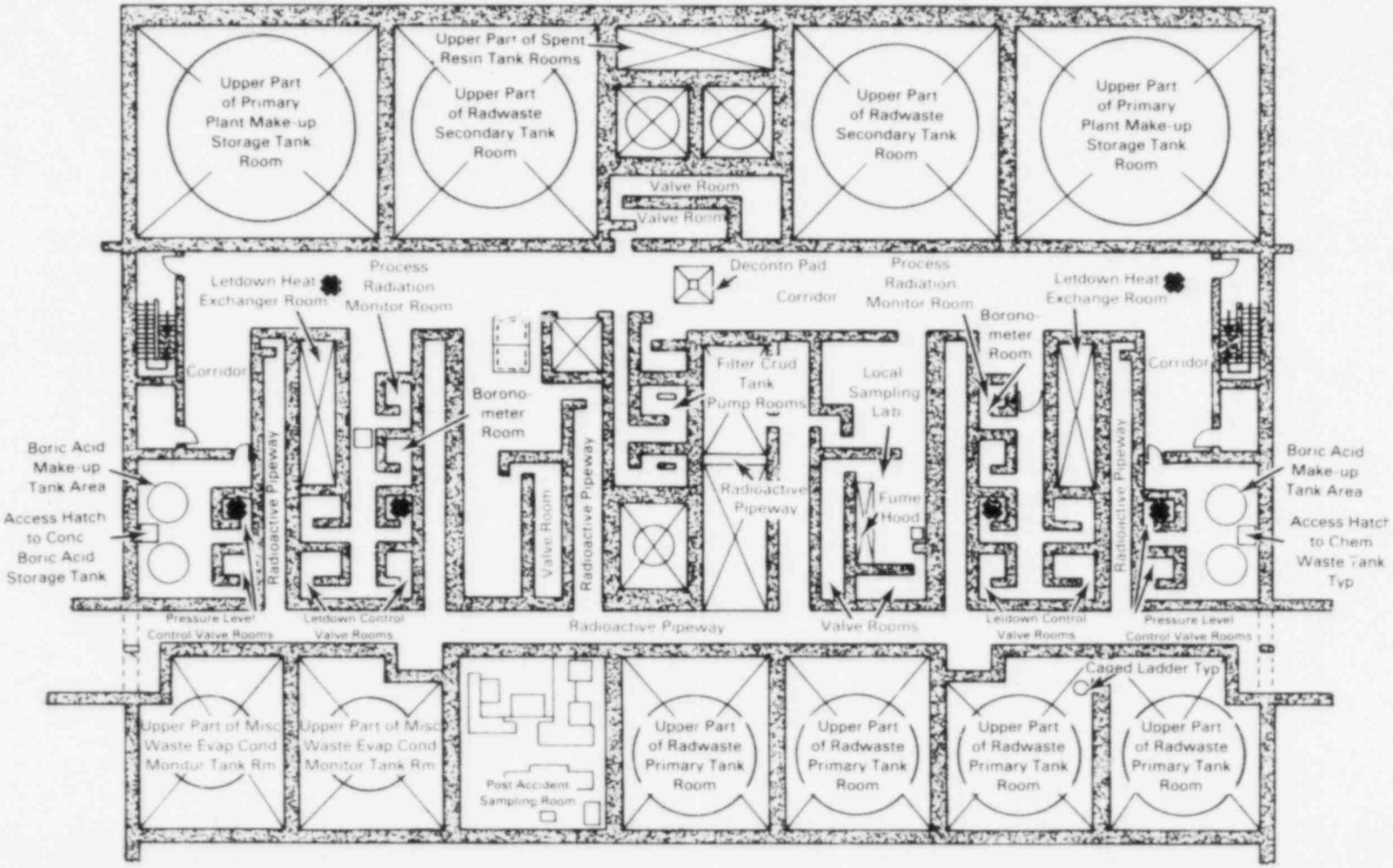

FIGURE 5. Auxiliary Building Floor Plan (24.0-ft elevation)

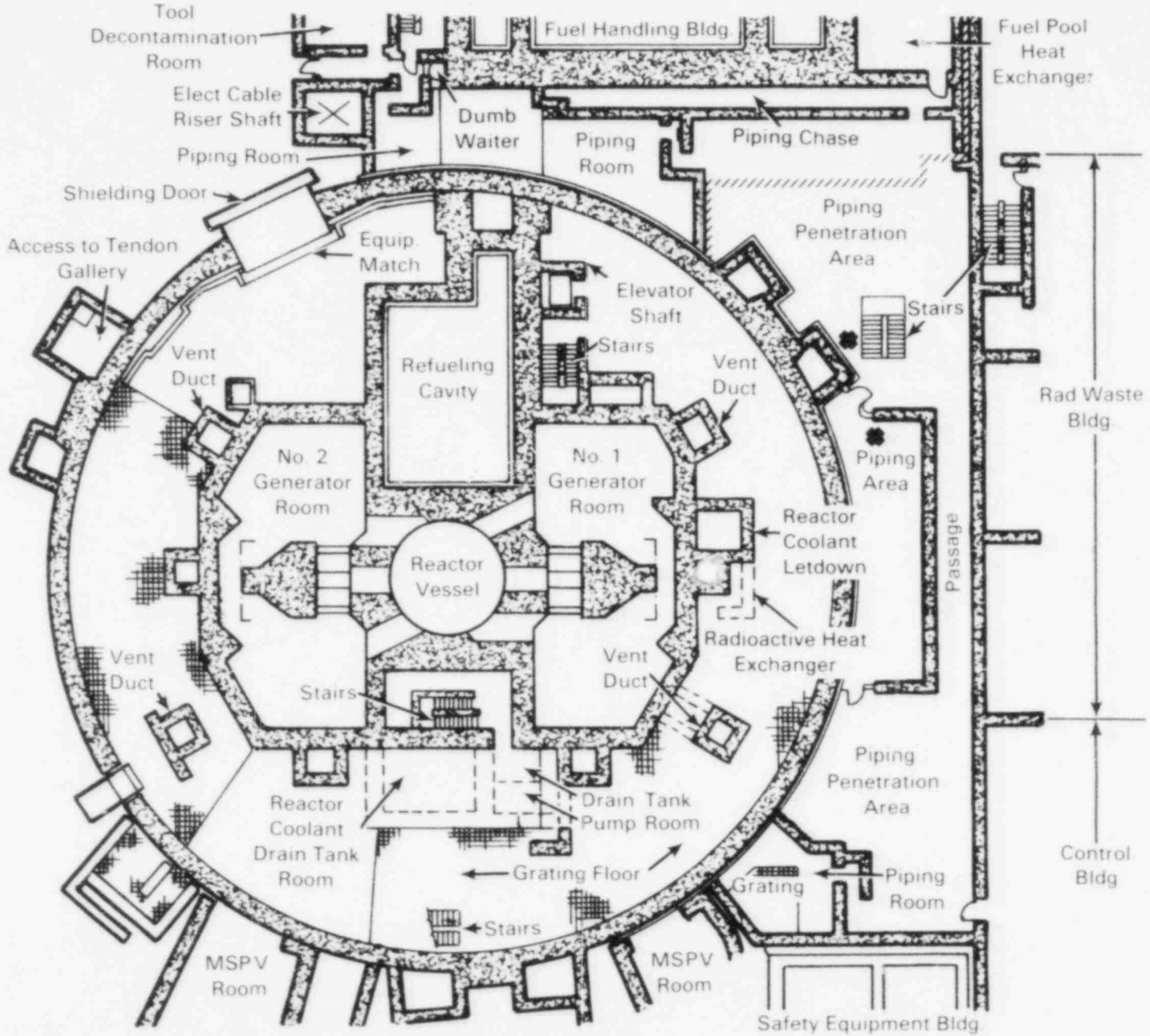

FIGURE 6. Containment Structure and Penetration Building Floor Plan (30.0-ft elevation)

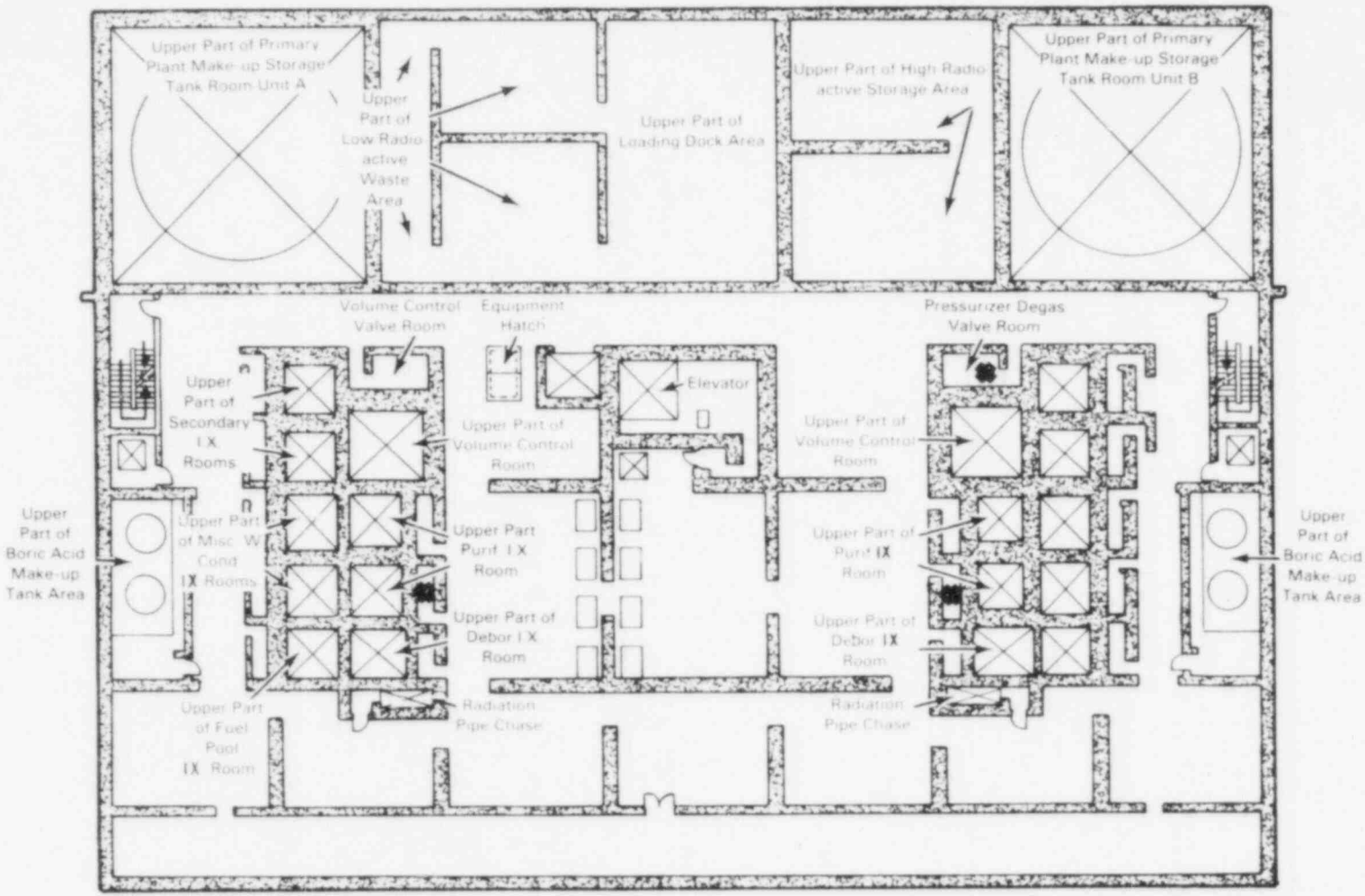

FIGURE 7. Auxiliary Building Floor Plan (50.0-ft elevation)

ANALYSIS AND DISCUSSION

The methodology and procedures presented in this report were designed to facilitate quantitative evaluation of the impact of fuel cladding failures on occupational radiation exposures. Data collection included gamma photon spectroscopy, radiation exposure rate measurements and air samples. In addition, reactor coolant sample analyses for the two units were reviewed to provide an indication of the change in fission product activity with reactor power level.

Because of the similarity between comparable measurements in Units A and B, the gamma photon spectroscopy data were evaluated to determine attributable exposure rates (quantitative analyses) as opposed to merely identifying radionuclides (qualitative analyses). Analysis of the gamma spectra began with the identification of all significant photon peaks using a Gaussian fit model. The radionuclide contributing to the peak was then identified by its characteristic energy. These spectra are presented in Appendix B. Identified peaks were classified according to whether they resulted from natural background, annihilation radiation, fission products or activation products.

The analyses progressed beyond the qualitative stage with the development and application of a computer program that estimated the radiation exposure rate attributable to each radionuclide. The program estimates a radiation exposure rate based on several factors. First, the program determines the observed activity of each radionuclide using the total number of counts in the associated peak and the counting time. Second, the program accounts for the relative efficiency of the intrinsic germanium detector as a function of photon energy. Third, the program calculates a radiation exposure rate based on an energy-dependent gamma dose conversion factor. Finally, the radiation exposure rate estimates for radionuclides observed in each measurement were modified with a geometric correction factor. The value for the geometric correction factor was determined from the physical characteristics of the measurement, including whether or not the collimator was used, and from the actual radiation exposure rate measurement taken with the portable ionization chamber. The source code listing for this computer program is presented in Appendix C.

With an estimate of the radiation exposure rate for each radionuclide peak calculated, the radiation exposure rate attributable to activation products and fission products can be determined. The results of the computer program estimates are presented in Appendix D. To facilitate comparison between Unit A and Unit B measurements, the Unit B measurements have been normalized. The photon peaks of activation products in both measurements give a discrete ratio of the crud buildup in Unit A versus Unit B. Because Unit A has been operating significantly longer, the Unit B attributable activation product exposure rate was normalized by increasing the estimate according to the observed ratio. This procedure normalized the attributable activation product exposure rate according to the operating age of Unit A.

The normalization of the attributable fission product dose rates was based on the data on reactor coolant analyses presented in Appendix E. To normalize the Unit B fission product measurement data from the 55 percent power level in Unit B to the 100 percent power level in Unit A, variations in the gross activity of the reactor coolant in Unit B were analyzed as a function of the reactor power level. The Unit B reactor coolant data obtained for the previous 20

months had a significant number of data points only at power levels of 0, 55 and 100 percent of the reactor's licensed maximum core power. These data are presented in Figure 8. All data represent the range of gross activity in the reactor coolant after the installation of an off-gas line that continuously vents the steam space in the pressurizer. It is significant to note that the gross activity values would be 30 to 40 percent greater without this off-gas line. The measurement data were not adjusted to reflect this off-gas line modification.

As illustrated in Figure 8, the increase in the gross activity of the coolant is linear with reactor power within one standard deviation of the mean. The mean values for gross activity as measured with reactor power at 0, 55 and 100 percent of the reactor's licensed maximum core power are 0.04, 7.94 and 18.7 microcuries per cubic centimeter (μCi/cc), respectively. The calculated standard deviation for these data at 0, 55 and 100 percent of the reactor's licensed maximum core power are 0.03, 1.21 and 3.43 μCi/cc, respectively. While the data satisfy the statistical measures for a linear relationship, the data appear to suggest that the increase may be more closely correlated to a linear-quadratic ($F(P)=a_0 + a_1P + a_2P^2$) than to a linear ($F(P)=a_0 + a_1P$) relationship. However, for this report it has been assumed that the gross activity of the coolant, and therefore the fission product activity, is linear with reactor power. Studies by plant personnel indicate dose rates vary linearly with power. If a linear-quadratic relationship does exist, then the impact of the fission product activity to occupational radiation exposure would be even more significant.

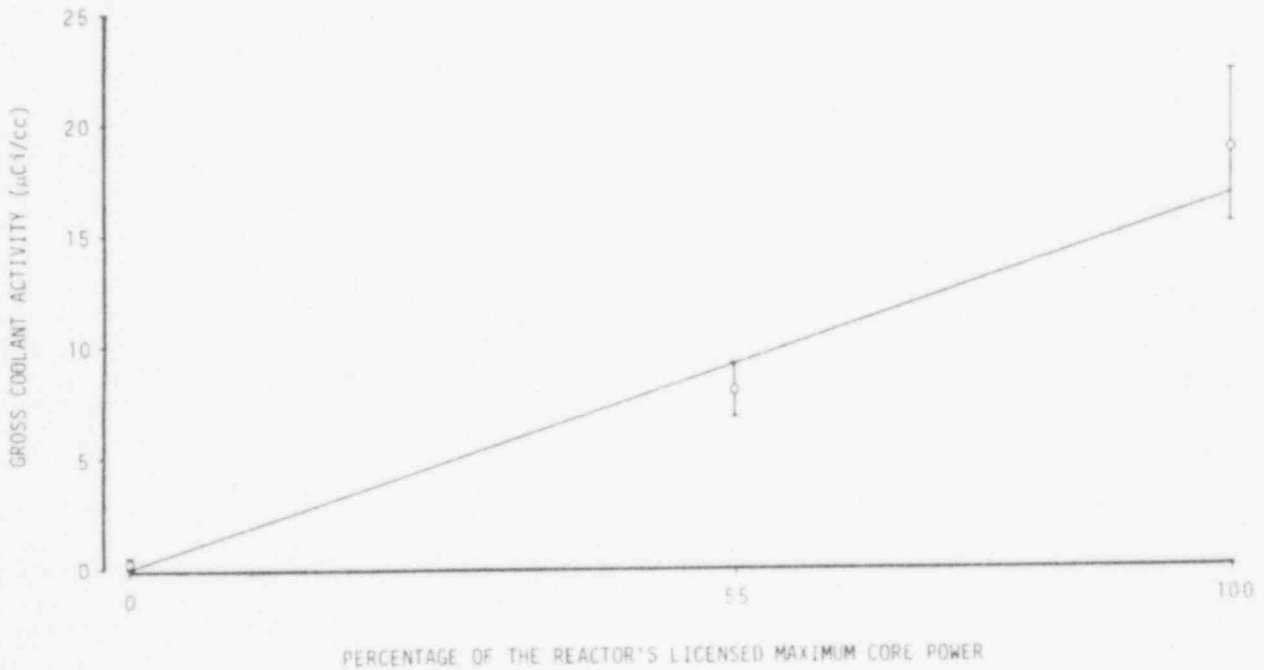

FIGURE 8. Unit B Reactor Coolant Activity vs Reactor Power Level

During the time period of the measurements, Unit B was operating at 55 percent power. Therefore, assuming a linear fission product activity increase with reactor power level, the normalization of the attributable fission product dose rate is straightforward. The Unit B values are multiplied by the inverse of 0.55.

Before the results of this study are presented, it is important to review the applicability of these data to other nuclear power plants. There are several features of the reactors used as part of this study that are noteworthy in relating these findings to what might be expected at other reactors. Because the measurements were taken at PWRs, it is obviously difficult to apply these data to boiling-water reactors, which have a significantly different plant system configuration and layout. When relating these study results to other PWRs, consider that the measurements were taken at relatively new plants of 3560-megawatt-thermal (1100 MWe) power rating. Older and smaller plants would have different power ratings, construction materials, and reactor cleanup and purification systems that would directly affect the buildup of activation products and radiation levels.

Another feature affecting the applicability of these data to other PWRs is a modification to Unit B. The pressurizer steam space sample line on Unit B was modified to provide for continuous venting of the steam space to enhance removal of fission product gases from the reactor coolant system. This modification was necessary because of the high coolant activity levels being experienced from the degraded fuel in this unit. There was no comparable line or equipment installed on Unit A. The impact of this modification has been to reduce the equilibrium coolant activity levels by an estimated 30 to 40 percent. However, it has also resulted in the high specific activity condensate (1-2 mCi/cc) being carried through piping from the steam space vent to areas of the plant that require routine worker access. One such area is the volume control tank valve room where measurements were taken. When considering how the data from this study could be applied to another PWR that does not have the pressurizer steam space vent (off-gas) line, it is necessary to consider what plant locations, systems and components would have elevated radiation levels as a result of the higher coolant activity and radiation levels from the steam space effluent.

The results of the measurements are presented according to the reactor coolant water flow pathway. Systems on this pathway include purification ion exchangers and charging pumps. A schematic of the reactor coolant water flow pathway is presented in Figure 9. The results are presented following the sequence of the coolant flow as it exists from the containment structure, then passes through the system components, and then re-enters containment. In evaluating these data, two points are noteworthy. First, the air samples did not reveal any measurable radioactivity except in one case in which the values were insignificant; the radionuclides were ^{131}I and ^{133}I. The location of measurable radioactivity was inside the primary penetration room in Unit B. Second, while the gamma spectral data do not contain an energy band large enough to observe ^{16}N, additional measurements were taken over an energy range where, if ^{16}N had been present, it would have been observed; although some counts were recorded, no significant ^{16}N photon energy peaks were observed.

Gamma spectra that are characteristic of the two units are presented in Figures 10 and 11. These two spectra are representative of the differences

observed between the baseline (Unit A) and degraded fuel (Unit B) data. The predominant radionuclides recorded on the Unit A spectra were activation products. These included ^{113}Sn, ^{58}Co, ^{60}Co, ^{59}Fe and ^{24}Na. The predominant radionuclides recorded on the Unit B spectra were fission products. These included ^{133}I, ^{133}Xe, ^{135}Xe, ^{134}Cs, ^{137}Cs, ^{95}Nb, ^{88}Kr, ^{88}Rb, ^{135}I and ^{138}Cs. The difference in the operating age between Unit A and B is evident in the magnitude of the ^{60}Co photon counting peaks. Cobalt-60, an activation product, builds up over time as radioactive crud. The ^{60}Co photon counting peaks were larger when measured in locations in the older unit, Unit A.

The results of the measurements and analysis are presented in Table 3. As previously mentioned, each measurement location is identified and presented according to the coolant flow path.

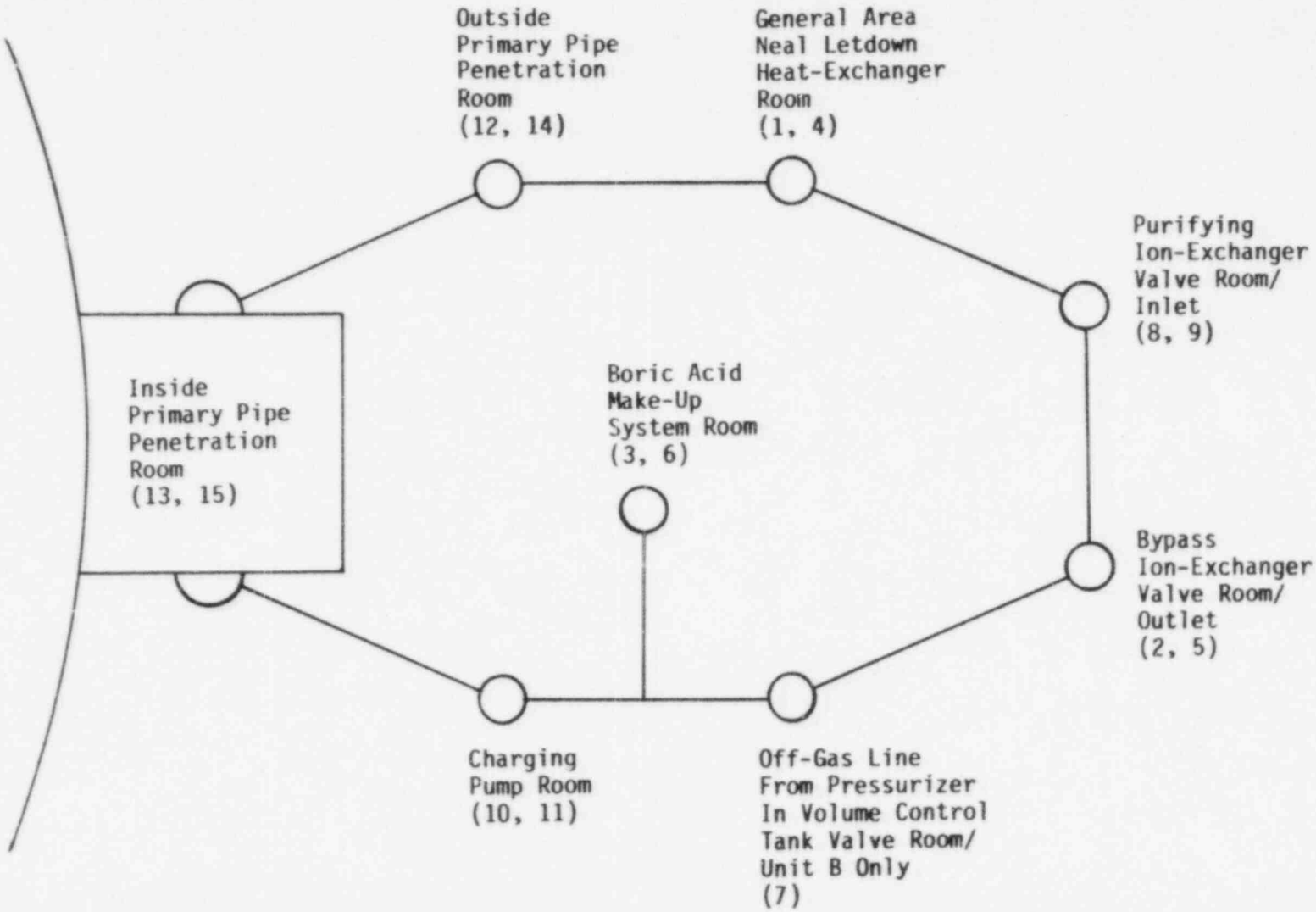

FIGURE 9. Reactor Coolant Water Flow Pathway

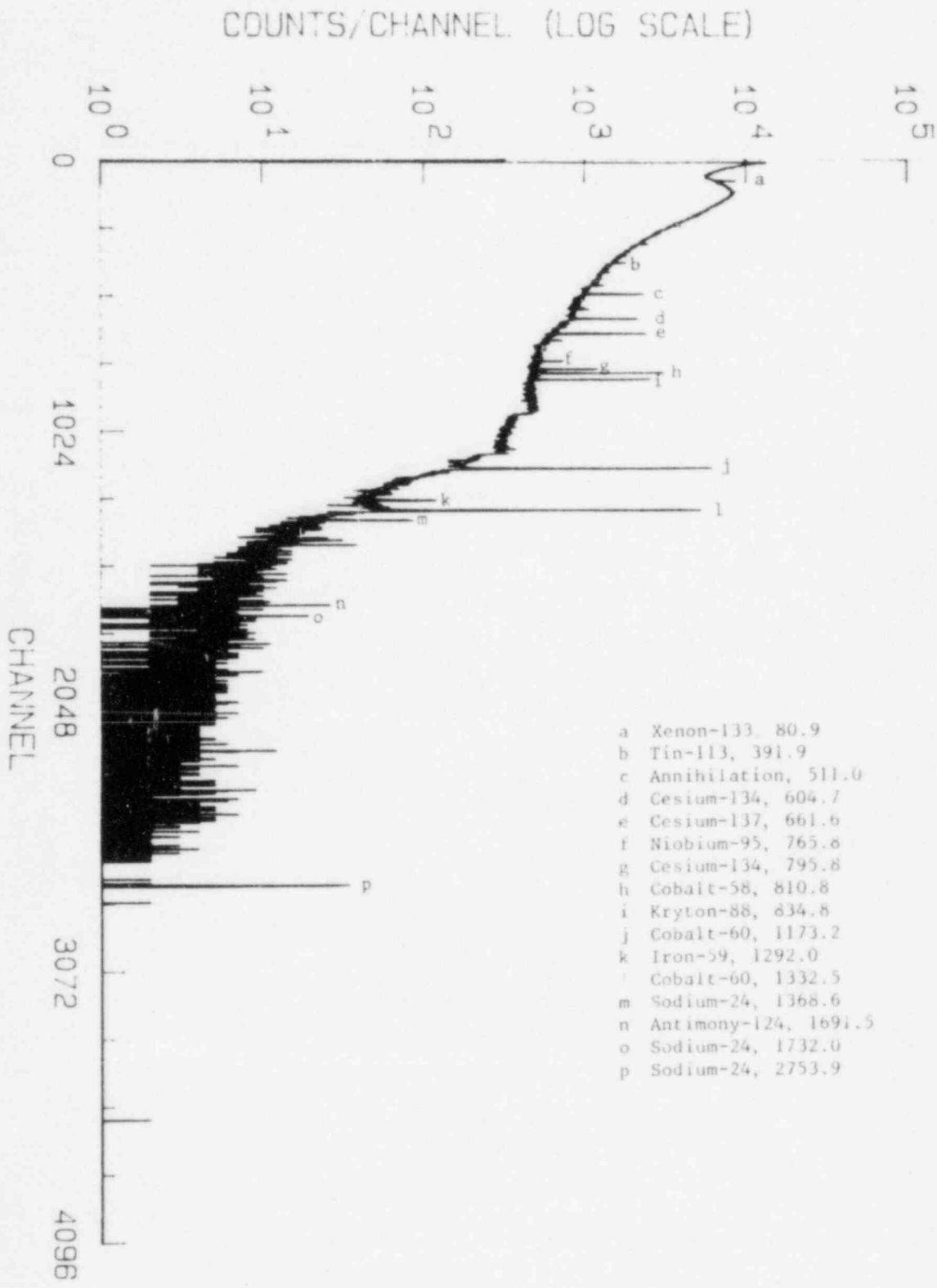

FIGURE 10. Inside Primary Penetration Room - Unit A, Baseline

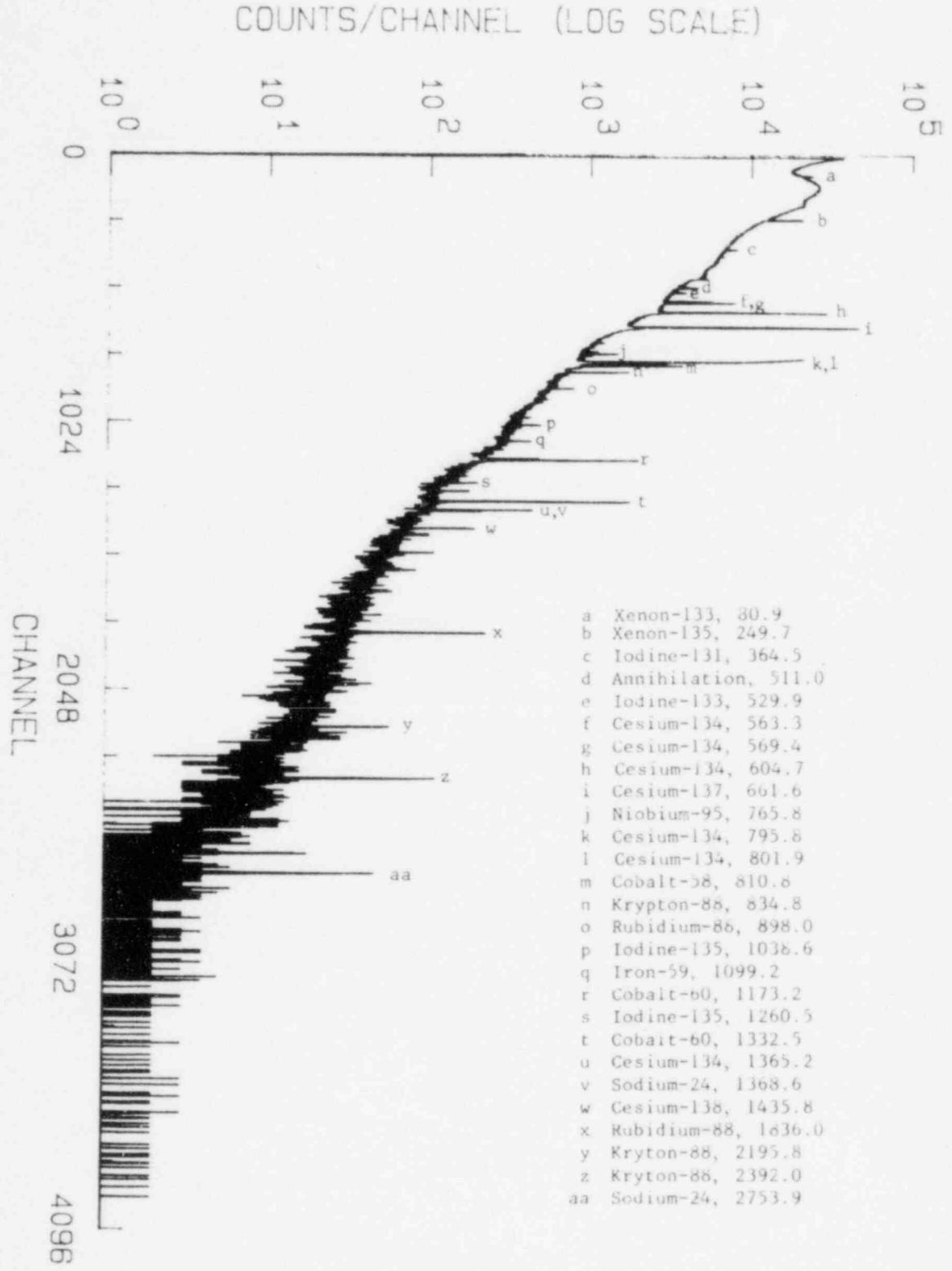

FIGURE 11. Inside Primary Penetration Room - Unit B, Degraded Fuel

TABLE 3. Attributable Radiation Exposure Rates (mR/hr)

Location	Unit A Exposure Rates			Unit B Exposure Rates			Normalized Unit B Exposure Rates		
	Total	Activation Products	Fission Products	Total	Activation Products	Fission Products	Estimated Total	Activation Products	Fission Products
Inside Primary Pipe Penetration Room (13,15)	1.2	0.98	0.20	3.5	0.45	3.01	6.5	1.00	5.47
Outside Primary Pipe Penetration Room (12,14)	<0.2	0.01	0.004	0.2	0.02	0.15	0.3	0.03	0.27
General Area Near Letdown Heat-Exchanger Room (1,4)	<0.2	0.02	0.002	<0.2	0.0	0.001	<0.2(1)	<0.2(1)	.001
Purifying Ion-Exchanger Valve Room/Inlet (8,9)	4	2.11	0.92	0.5	0.28	0.22	NA(2)	NA(2)	NA(2)
Bypass Ion-Exchanger Valve Room/Outlet (2,5)	30	21.9	5.53	26	3.16	21.5	62.6	18.5	39.1
Off-Gas Line From Pressurizer/ Unit B Only (7)	NA(3)	NA(3)	NA(3)	29	2.93	24.9	NA(3)	NA(3)	45.3
Boric Acid Make-Up System Room (3,6)	1.7	1.44	0.20	11	2.49	7.51	18.3	2.87	13.6
Charging Pump Room (10,11)	<0.2	0.09	0.04	3.1	0.11	2.84	5.9	0.65	5.20

(1) No activation products recorded in Unit B. Normalized Unit B activation product contribution assumed equal to Unit A.

(2) Measurements not comparable because only the Unit A ion exchanger contained radioactive material.

(3) No comparable location in Unit A. No data available for normalization of activation product contribution.

INSIDE PRIMARY PIPE PENETRATION ROOM (13, 15)

The data resulting from measurements performed inside the primary pipe penetration rooms are the most representative example of the significant differences between the baseline (Unit A) and the failed fuel (Unit B) exposure rates. In addition, these data permit the most exacting comparison because of the elimination of several potentially confounding variables. First, no collimator was used for either measurement. Second, the measurement positioning and piping layouts were identical. Third, the geometric correction factor was the same for both measurements, which is an indicator of their similarity. Fourth, the two measurement locations were in separate buildings that are a considerable distance apart. Finally, there were four activation product radionuclides observed in both units, which provides significant data for normalizing the Unit B activation product values. The four activation product radionuclides were ^{24}Na, ^{58}Co, ^{59}Fe, and ^{60}Co. The normalization ratio for these measurements was 1 to 2.22 for Unit A to Unit B, which means that Unit A currently has 2.22 times more activation product activity than Unit B.

These data indicate that the failed fuel increased the fission product contribution to occupational radiation exposures from a baseline of approximately 17 percent to 84 percent of the total exposure rate. This represents a factor of 28 increase in the exposure rates from the fission product activity. The total radiation exposure rate increased by approximately 540 percent over the baseline condition. These data are illustrated in Figure 12.

OUTSIDE PRIMARY PIPE PENETRATION ROOM (12, 14)

The data suggest that, in the areas outside the primary pipe penetration rooms, the rate of buildup of activation products is slower than in other areas of the plant, such as the radwaste area. This is based on the exposure rates attributable to the two activation product radionuclides, ^{58}Co and ^{60}Co. The normalization ratio for these measurements was 1 to 1.57 for Unit A to Unit B.

Although the exposure rates are low, the data indicate that the impact of failed fuel results in the fission products providing a contribution to radiation exposure rates in these areas that is greater than 13 times the contribution of activation products. The baseline data describing Unit A indicate that the contribution from fission products to radiation exposure rates is less than half that from activation products. Therefore, the failed fuel increases the fission product contribution to occupational radiation exposures from a baseline of approximately 20 percent to 90 percent of the total exposure rate. This represents an increase in the fission product activity by a factor of 62.

GENERAL AREA NEAR LETDOWN HEAT-EXCHANGER ROOM (1, 4)

Although the exposure rates are also low near the letdown heat-exchanger room, the gamma spectroscopy data indicate that fission product activity was dominant in Unit B, while the activation product activity was dominant in Unit A. These data also indicate the lesser extent of ^{60}Co activation product buildup in Unit B versus the older Unit A. These measurements were taken such that the lower energy fission products would be reduced by shielding while the

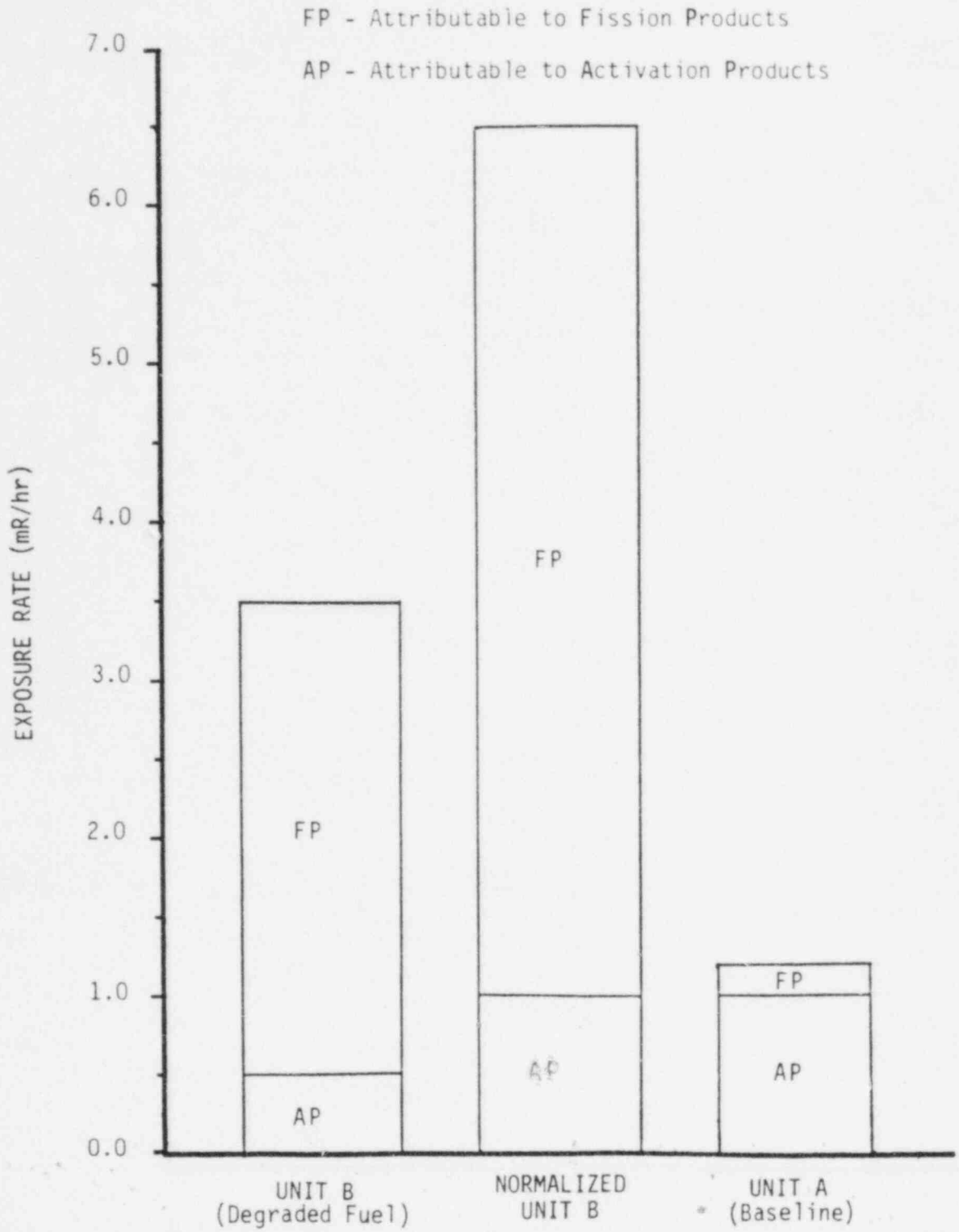

FIGURE 12. Attributable Exposure Inside Unit A and Unit B Primary Penetration Rooms

higher energy ^{60}Co would be observed. The data indicate a similarity in the quality of the measurements because of the excellent agreement on the natural background radionuclide ^{40}K. In addition, other natural decay schemes were evident through the presence of ^{212}Pb, ^{212}Bi, and ^{208}Tl.

PURIFYING ION-EXCHANGER VALVE ROOM/INLET (8, 9)

Data collected in the purifying ion-exchanger valve rooms were complicated by several confounding factors that limit the data's usefulness. The ion-exchanger resin where the measurement was taken in Unit A contained radioactive materials. In Unit B, however, the ion-exchanger resin was being bypassed in the area of the measurement, and therefore, the area did not contain radioactive materials. Measurement of the active resin in Unit B was prohibited by significant contamination levels, which would have contaminated the intrinsic germanium detector.

Data on the ratio of activation product activity between Unit A and Unit B are useful. There were three activation product radionuclides present in both measurements. These were ^{24}Na, ^{58}Co and ^{60}Co. The normalization ratio for this area was 1 to 5.03 for Unit A to Unit B. This ratio is consistent with other normalization values determined in the radwaste area.

BYPASS ION-EXCHANGER VALVE ROOM/OUTLET (2, 5)

For the bypass ion-exchanger valve room, the normalization of the activation product contribution in Unit B was based on the presence of ^{24}Na and ^{60}Co in both measurements. The normalization ratio for this area was 1 to 5.86 for Unit A to Unit B. As in all cases, the fission product normalization was based on the power level. The normalization estimate assumed a linear increase in fission product activity as a function of reactor power level.

The data collected in this area indicate the failed fuel results in a significant increase in the fission product contribution to radiation exposure rates. These data suggest that the failed fuel increases the fission product contribution to occupational radiation exposures from a baseline of approximately 18 percent to 62 percent of the total exposure rate. This represents an increase in the fission product activity by a factor of 7.

OFF-GAS LINE FROM PRESSURIZER LOCATED IN THE VOLUME CONTROL TANK VALVE ROOM - UNIT B ONLY (7)

One consequence of the fuel cladding failure in Unit B was the installation of an off-gas line from the pressurizer. The gamma spectroscopy data for this line were collected in the volume control tank valve room. There was no comparable equipment installed in Unit A. Given the operating conditions in Unit B, the fission product contribution in this area was 86 percent of the total exposure rate. The necessity of the off-gas line indicates the significance of the degraded fuel, while the total exposure rate of 29 mR/hr indicates the potential significance of the fuel cladding failure to occupational radiation exposures.

BORIC ACID MAKE-UP SYSTEM (3, 6)

The data collected on the boric acid make-up system indicated 1) the activation products did not build up as rapidly in this system as they did in other areas of the plant and 2) failed fuel significantly impacts occupational exposures in this area of the plant. The relatively constant and low level of activation products observed in the boric acid make-up systems of Units A and B, which are of different ages, indicate that the activation products are not collecting as rapidly in this system compared with other plant systems. This may be because the boric acid make-up system, which serves as an injection system into the loop, is not on the main water flow pathway. The normalization value for the activation product contribution was assumed to be 1.0, because it is based on two radionuclides, ^{58}Co and ^{60}Co, which result in a ratio of 1.15 to 1 for this area for Unit A to Unit B.

The data collected in this area indicated that the elevated levels of failed fuel significantly increase the fission product contribution to radiation exposure rates. These data suggest that the failed fuel increases the fission products contribution to occupational radiation exposures from a baseline of approximately 12 percent to 74 percent of the total exposure rate. This represents an increase in the fission product activity by a factor of 69.

CHARGING PUMP ROOM (10, 11)

The charging pump room in Unit B was heavily contaminated. As a result, the geometric correction factor was estimated to be significantly different between the two units, even though the measurement procedure and location were similar. Regardless of the geometric correction factor variation, it was evident that the radiological source term was predominantly from the fission product contribution in Unit B. Normalization of the activation product contribution in Unit B was difficult because the only activation product observed in Unit B was an isotope of manganese (^{56}Mn) that was not observed in Unit A. An estimate for the normalization ratio could be determined from the upstream measurements at the bypass in the exchanger room. That normalization ratio was 1 to 5.86 for Unit A to Unit B. Applying this ratio to the charging pump room measurements is conservative for evaluating the impact of failed fuel on exposure rates.

Applying the normalization ratio to the data resulted in an activation product contribution of 0.65 mR/hr in Unit B. Therefore, the failed fuel appears to significantly impact radiation exposure rates in this area. The data and estimates suggest that the failed fuel increased the fission product contribution to occupational radiation exposures from a baseline of approximately 30 percent to 88 percent of the total exposure rate. This represents an increase in the fission product activity by a factor of 130.

DATA SUMMARY

These data are summarized in Table 4.

TABLE 4. Summary of Measurement Analysis

Location	Unit A Fission Product (FP) Exposure Rates (ER) ÷ Unit A Total ER	Normalized Unit B FP ER ÷ Normalized Unit B Total ER	Normalized Unit B FP ER ÷ Unit A FP ER	Normalized Unit B Total ER ÷ Unit A Total ER
Inside primary penetration room (13, 15)	0.17	0.84	28	5.4
Outside primary penetration room (12, 14)	0.20(1)	0.90	62	14.9(1)
Bypass ion-exchanger valve room (2, 5)	0.18	0.62	7	2.1
Boric acid make-up system (3, 6)	0.12	0.74	69	10.8
Charging pump room (10,11)	0.31(1)	0.88	130	44.0

(1) Value based on estimated total exposure rate in Unit A (listed in Appendix D), which was <0.2 mR/hr.

CONCLUSIONS

Based on the data collected for this study, the following conclusions can be drawn:

- The methodology and procedures presented are adequate to quantitatively evaluate the impact of incidents such as fuel cladding failures on occupational radiation exposures for the plant conditions measured.

- The data suggest that a 3560-megawatt-thermal (1100 MWe) PWR operating at full power with approximately 0.125 percent failed fuel (62 failed pins) can expect an increase of 540 percent in radiation exposure rates in areas of routine exposure over normal fuel operating conditions. In specific plant areas, the degraded fuel may elevate radiation exposure rates even higher. These estimates assume that fission product activity increases linearly with power level.

- Because the percentage of failed fuel was 1000 percent greater in Unit B than in Unit A, the 540 percent increase in radiation exposure rates suggests that occupational radiation exposure rates do not increase linearly with the magnitude of fuel cladding failures. This is probably due to variations in the efficiency of the reactor cleanup systems for different levels of failed fuel and to part of the radiation being from activation products.

- Although collective occupational doses will undoubtedly increase as a result of the increased radiation exposure rates, the magnitude of such an increase will depend on the amount of time workers spend in the affected areas. For example, in areas of a plant where radiation exposure rates are very low, a 540 percent increase in radiation exposure rates does not necessarily result in a significant increase in collective occupational doses. The primary impact on collective occupational doses may occur when workers perform routine maintenance and outage tasks on affected reactor and coolant system components located in areas with higher exposure rates.

- The data suggest that the degraded fuel does not significantly affect the magnitude of airborne activity in work areas inside the plant. This could be misleading because plant personnel said that the degraded fuel resulted in periodic releases of radioactive material into work areas inside the plant, especially in the radwaste area. These releases include isotopes of xenon and iodine. Consequently, the degraded fuel could result in an increase in occupational radiation exposures from increased airborne radionuclide concentrations.

- Additional measurements are needed at a plant that has experienced fuel cladding failure to comprehensively evaluate the impact of degraded fuel on occupational radiation exposures during an outage.

- The data analyses are applicable only to the fuel conditions and to the PWR plant described in this report. Assessing the impacts of fuel cladding failure on types of plants, such as boiling-water reactors, will require a set of detailed onsite measurements similar to those presented in this report.

BIBLIOGRAPHY

Atomic Industrial Forum (AIF). 1980a. An Assessment of Engineering Techniques for Reducing Occupational Radiation Exposure at Operating Nuclear Power Plants. Washington, D.C.

Atomic Industrial Forum (AIF). 1980b. Study of Effects of Reduced Occupational Radiation Exposure Limits on the Nuclear Power Industry. AIF/NESP-017, Washington, D.C.

Atomic Industrial Forum (AIF). 1981. Compendium of Design Features to Reduce Occupational Radiation Exposures at Nuclear Power Plants. AIF/NESP-020, Washington, D.C.

Cember, H. 1969. Introduction to Health Physics. Pergamon Press, New York, New York.

Endres, G. W. R., J. M. Aldrich, L. W. Brackenbush, L. G. Faust, R. V. Griffith, and D. E. Hankins. 1983. Neutron Dosimetry at Commercial Nuclear Plants. Final Report of Subtask A: Reactor Containment Measurements. NUREG/CR-1769, PNL-3585, Pacific Northwest Laboratory, Richland, Washington.

Fenyves, E., and O. Haiman. 1969. The Physical Principles of Nuclear Radiation Measurements. Academic Press, New York.

Hajnal, F., and C. Klusek. 1974. "Semi-Empirical Efficiency Equations for Ge(Li) Detectors." NIM 122:559-565.

International Commission on Radiation Units and Measurements (ICRU). 1971. Radiation Protection Instrumentation and Its Application. ICRU Report 20, ICRU Publications, Washington, D.C.

International Commission on Radiological Protection (ICRP). 1983. Radionuclide Transformations. ICRP Publication 38, Pergamon Press, Elmsford, New York.

Kaplan, I. 1955. Nuclear Physics. Addison-Wesley, Cambridge, Massachusetts.

Moeller, M. P., G. A. Stoetzel, and L. H. Munson. 1985. Historical Summary of Occupational Radiation Exposure Experience in U.S. Commercial Nuclear Power Plants. NUREG/CR-4160, PNL-5404, Pacific Northwest Laboratory, Richland, Washington.

National Council on Radiation Protection and Measurements (NCRP). 1978. Instrumentation and Monitoring Methods for Radiation Protection. NCRP Report No. 57, Washington, D.C.

Parkhurst, M. A., L. A. Rathbun, and D. W. Murphy. 1983. Radiological Assessment of Steam Generator Repair and Replacement. NUREG/CR-3540, PNL-4877, Pacific Northwest Laboratory, Richland, Washington.

Pelletier, C. A., L. Simmons, M. Barbier, and J. H. Keller. 1974. Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants. AIF/NESP-005, SAI Services, Washington, D.C.

Rathbun, L. A., and P. L. Roberson. 1983. Beta Particle Measurement and Dosimetry Requirements at NRC-Licensed Facilities. NUREG/CR-3544, PNL-4886, Pacific Northwest Laboratory, Richland, Washington.

Roberson, P.L., G. W. R. Endres, R. A. Fox, D. L. Haggard, K. L. Holbrook, and L. A. Rathbun. 1984. Spectral and Dosimetric Measurements of Photon Fields at Commercial Nuclear Sites. NUREG/CR-3569, PNL-3585, Pacific Northwest Laboratory, Richland, Washington.

Seltzer, S. M. 1981. "Calculated Response of Intrinsic Germanium Detectors to Narrow Beams of Photons with Energies up to ∿300 keV." NIM 188:133-151.

Stephenson, R. 1954. Introduction to Nuclear Engineering. McGraw-Hill, New York, New York.

U.S. General Accounting Office (GAO). 1982. Report to Senator John Glenn, "Actions Being Taken to Help Reduce Occupational Radiation Exposures at Commercial Nuclear Power Plants. GAO/EMO-82-91, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1984. Optimization of Public and Occupational Radiation Protection at Nuclear Power Plants. NUREG/CR-3665. Office of Nuclear Regulatory Research, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1982. Safety Goals for Nuclear Power Plants: A Discussion Paper. NUREG-0880, Washington, D.C.

Warman, E. A., R. M. McMellon, and J. M. Cardito. 1981. Occupational Radiation - Exposure Reduction Technology Planning Study. EPRI NP-1862, Electric Power Research Institute, Palo Alto, California.

APPENDIX A

RADIONUCLIDES OF INTEREST

To discuss the impact of fuel cladding failures on occupational radiation exposure, it is useful to review the radionuclide composition of the liquid and gaseous source terms within a reactor system. The source terms are composed of radionuclides from two distinguishable sources, fission products and activation products. When fuel cladding failure has occurred, the concentration of fission products increases in the reactor coolant system. The concentration of activation products in the reactor coolant depends on the age of the plant, the magnitude of its feedwater pipe corrosion, and the plant's water chemistry treatment.

A fission product is a radionuclide produced either by fission or subsequent radioactive decay or by neutron activation of the nuclides formed in the fission process. The range of mass numbers is from 72, an isotope of zinc with atomic number 30, to 158, an isotope of europium with atomic number 63. The theoretical total number of possible fission fragments may be 87, which is the number of mass numbers between 72 and 158. Approximately 97 percent of uranium-235 nuclei undergoing fission yield products that fall into a light group, at a mass number from about 85 to 104, and a heavy group, at a mass number from about 130 to 149. This phenomenon is illustrated in Figure A.1. The most probable type of fission, which occurs in approximately 6 percent of the total, results in fission fragments between 95 and 139. The fission yield of a particular fission product is defined as the percentage of fissions that lead to the formation of that fission product. Generally, each fission produces two fission products resulting in a total fission yield of 200 percent.

During the typical fission reaction for ^{235}U, the most probable fission fragments are strontium (Sr) and xenon (Xe) with atomic numbers 38 and 54 and mass numbers 94 and 140, respectively. The mass numbers of these fission products correspond to the light and heavy groups. Both products of this reaction are radioactive since the heaviest stable isotope of strontium is ^{88}Sr and of xenon is ^{136}Xe. The further decay of each fission fragment is termed a fission chain and usually involves about three successive beta disintegrations. However, in addition to being beta emitters, many of the fission products release gamma photons.

Fission product decay results in a series of decay chains. A list of fission and decay products appears in Table A.1. Major radionuclides include isoptopes of strontium (Sr), yttrium (Y), zirconium (Zr), niobium (Nb), ruthenium (Ru), rhodium (Rh), iodine (I), tellurium (Te), xenon, cesium (Cs), barium (Ba), lanthanum (La), cerium (Ce), praseodymium (Pr), and neodymium (Nd). For external occupational radiation exposures, the gamma-emitting radionuclides are the most important.

Activation products are formed from neutron bombardment of the reactor's construction materials and the elements in the reactor's cooling water. The component systems of most commercial reactors are made of the stainless steel, carbon steel and inconel (nickel alloy). The activation of these metals forms

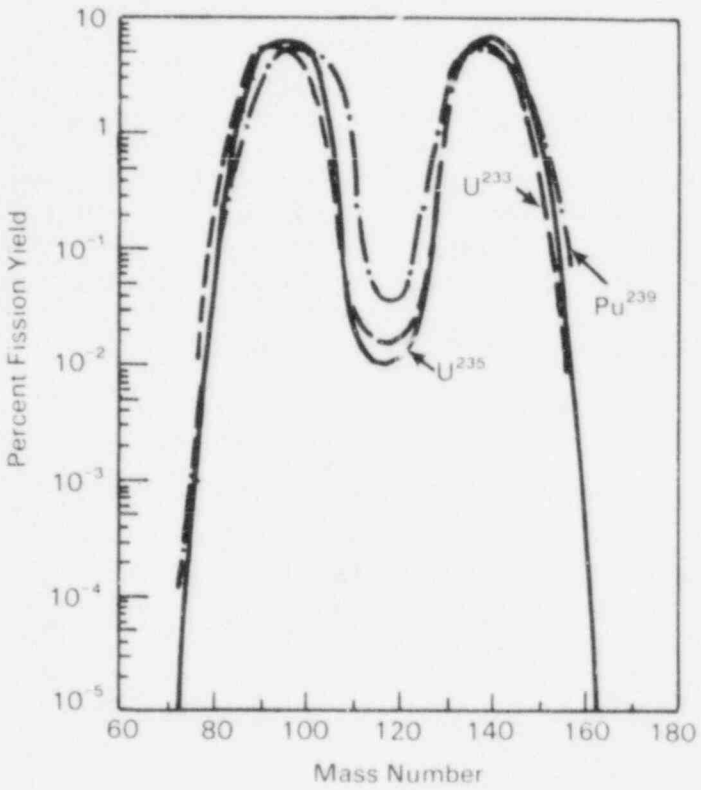

FIGURE A.1. Fission Product Yields for ^{235}U, ^{239}Pu, and ^{233}U

products that circulate in the reactor coolant water as radioactive crud. Some of these products are isotopes of sodium (Na), iron (Fe), chromium (Cr), manganese (Mn), nickel (Ni), and cobalt (Co). Activation of oxygen and hydrogen in the reactor coolant also forms activation products, including isotopes of nitrogen (N) and hydrogen (H). A list of typical activation products is presented in Table A.2.

For this study, a list of the radionuclides of interest was developed. This list, presented in Table A.3, is based on the identification of radionuclide peaks during analysis of the gamma spectroscopy data. In addition to the activation and fission products, the list also includes some naturally occuring radionuclides present in the thorium decay chain.

TABLE A.1. Production of Important Fission Products in a Reactor[1]

Fission product	Activity (curies)[a] after selected periods of continuous operation of a reactor at a power level of 1000 kW		
	100 days	1 year	5 years
Kr-85	53	191	818
Rb-86	0.25	0.26	0.26
Sr-89	28,200	38,200	38,500
Sr-90	402	1,430	6,700
Y-90[b]	402	1,430	6,700
Y-91	34,800	48,900	49,500
Zr-95	32,900	49,200	50,300
Nb-95(90 H)[b]	446	687	704
Nb-95(35 D)[b]	20,900	48,200	50,500
Ru-103	25,100	30,900	31,000
Rh-103[b]	25,100	30,900	31,000
Ru-106	753	2,180	4,220
Rh-106[b]	753	2,180	4,220
Ag-111	151	151	151
Cd-115	4.8	5.9	5.9
Sn-117	83	84	84
Sn-119	< 24	< 24	< 100
Sn-123	4	9	10
Sn-125	100	101	101
Sb-125[b]	12	43	139
Te-125[b]	5	34	136
Sb-127	787	787	787
Te-127(90 D)[b]	146	260	277
Te-127(9.3 H)[b]	808	922	939
Te-129(32 D)	1,410	1,590	1,590
Te-129(70 M)[b]	1,410	1,590	1,590
I-131	25,200	25,200	25,200
Xe-131[b]	250	252	252
Te-132	36,900	36,900	36,900
I-132[b]	36,900	36,900	36,900
Xe-133	55,300	55,300	55,300
Cs-136	52	52	52
Cs-137	300	1,080	5,170
Ba-137[b]	285	1,030	4,910
Ba-140	51,500	51,700	51,700
La-140[b]	51,300	51,700	51,700
Ce-141	43,000	47,800	47,800
Pr-143	45,000	45,300	45,300
Ce-144	9,860	26,700	44,000
Pr-144[b]	9,860	26,700	44,000
Nd-147	21,800	21,800	21,800
Pm-147[b]	1,290	4,900	16,000
Sm-151	9	37	175
Eu-155	23	74	207
Eu-156	108	109	109
Total	563,691	693,573	767,547

[a] Calculated using fission product yields. [b] Daughter product.
From *Radiological Health Handbook*, 1960.

(1) Reprinted with permission from Introduction to Health Physics, by Herman Cember, Volume 105 of the International Series of Monographs in Nuclear Energy, Copyright 1969, Pergamon Press, Ltd.

TABLE A.2. Activation Products in Reactor Coolant Water

^{3}H	Hydrogen (Tritium)	^{60}Co	Cobalt
^{7}Be	Beryllium	^{65}Ni	Nickel
^{24}Na	Sodium	^{65}Zn	Zinc
^{51}Cr	Chromium	^{69m}Zn	Zinc
^{54}Mn	Manganese	^{95}Zr	Zirconium
^{54}Mn	Manganese	^{110m}Ag	Silver
^{58}Co	Cobalt	^{113}Sn	Tin
^{59}Fe	Iron	^{189}W	Tungsten

TABLE A.3. Radionuclides of Interest

^{7}Be	Activation Product
^{24}Na	Activation Product
^{40}K	Naturally Occurring
^{51}Cr	Activation Product
^{56}Mn	Activation Product
^{58}Co	Activation Product
^{59}Fe	Activation Product
^{60}Co	Activation Product
^{65}Zn	Activation Product
^{87}Kr	Fission Product
^{88}Kr	Fission Product
^{88}Rb	Fission Product
^{89}Rb	Fission Product
^{91}Sr	Fission Product
^{95}Zr	Fission Product
^{95}Nb	Fission Product
^{98}Nb	Fission Product
^{105}Ru	Fission Product
^{113}Sn	Activation Product
^{124}Sb	Fission Product
^{125}Sb	Fission Product
^{131}I	Fission Product
^{132}I	Fission Product
^{133}I	Fission Product
^{133}Xe	Fission Product
^{134}Cs	Fission Product
^{134m}Cs	Fission Product
^{135}I	Fission Product
^{135}Xe	Fission Product
^{137}I	Fission Product
^{137}Cs	Fission Product
^{138}Xe	Fission Product
^{138}Cs	Fission Product
^{142}Xe	Fission Product
^{187}W	Activation Product
^{208}Tl	Naturally Occurring
^{212}Pb	Naturally Occurring
^{214}Pb	Naturally Occurring
^{214}Bi	Naturally Occurring

APPENDIX B

GAMMA SPECTRAL ANALYSES

The gamma spectroscopy data for the 15 measurements made in this study are presented in the graphs in Figures B-1 through B-15. The count peaks have been identified according to the associated radionuclides (ANN in the graphs stands for annihilation peaks). The number of counts under each peak corresponds to the total count information presented as part of Appendix D.

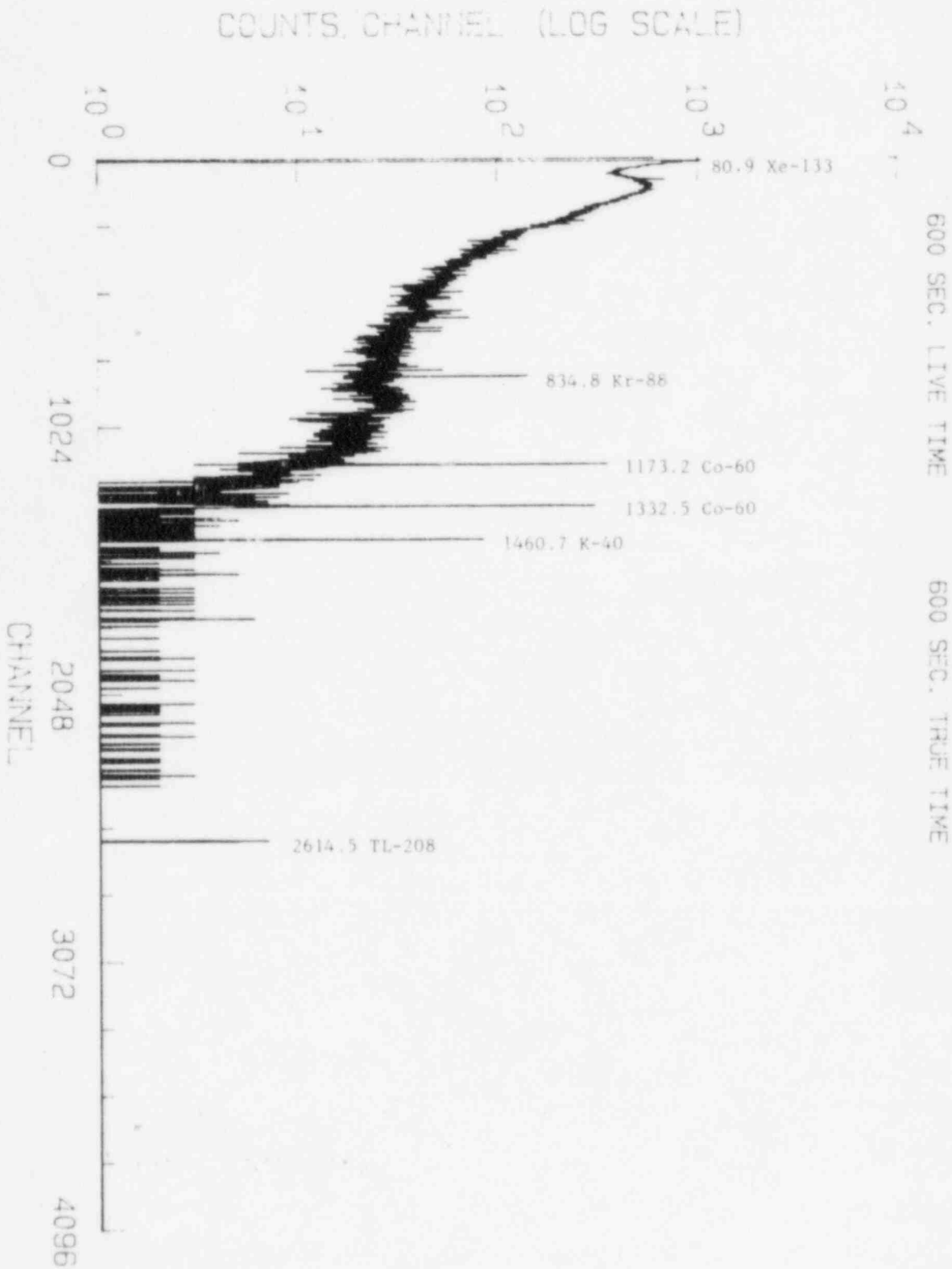

FIGURE B.1. General Area Near Letdown Heat-Exchanger Room, Unit A

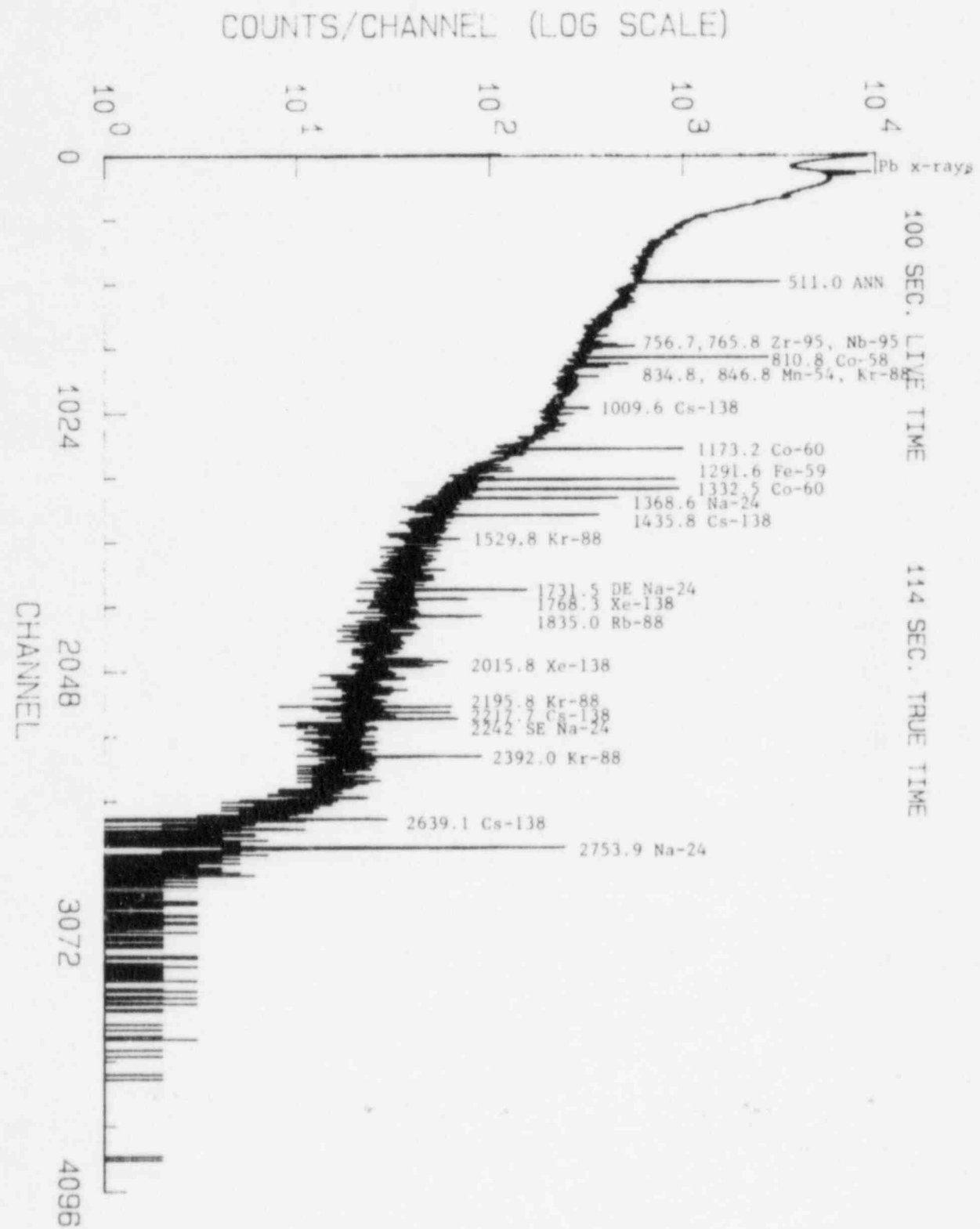

FIGURE B.2. Bypass Ion-Exchanger Valve Room/Outlet, Unit A

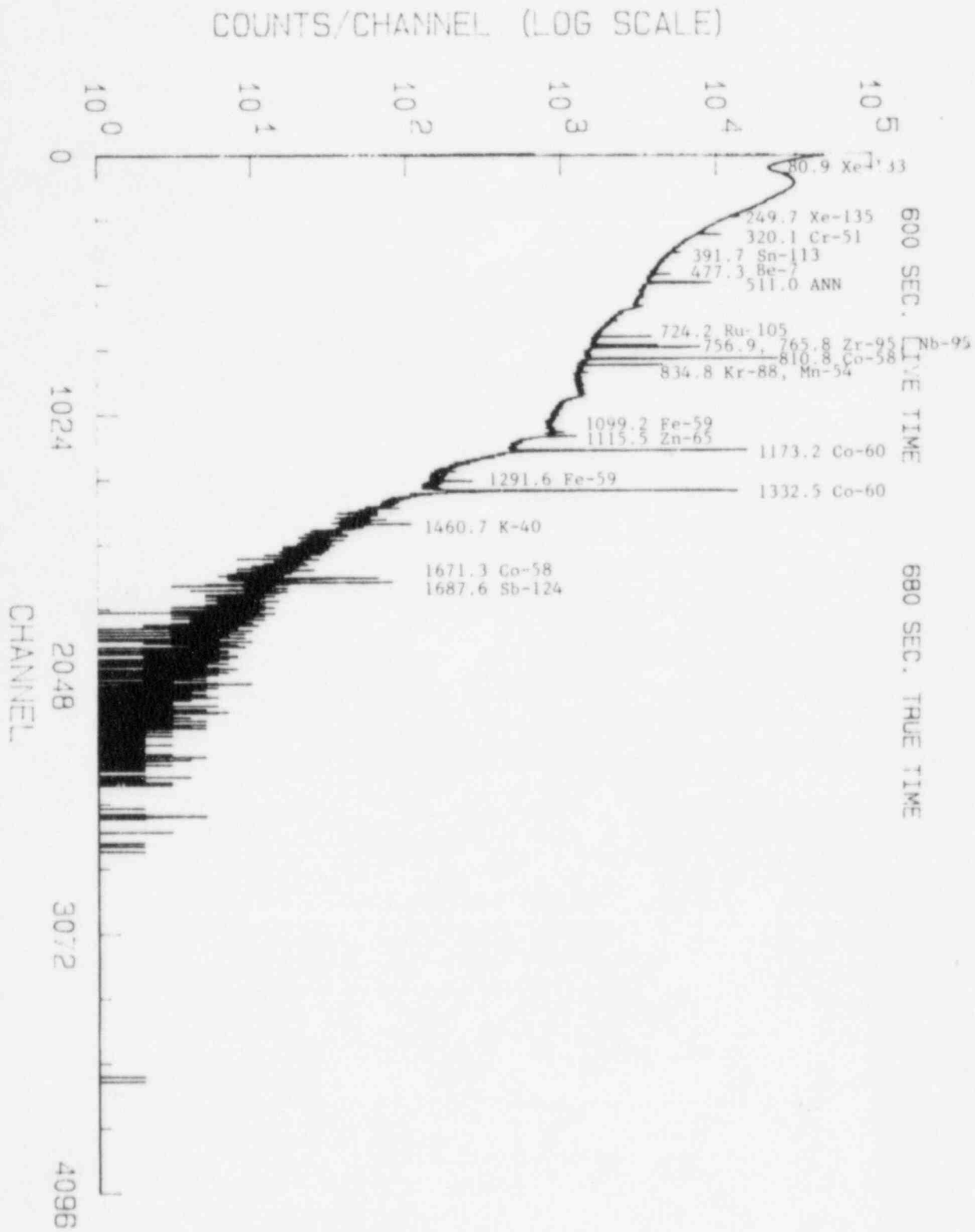

FIGURE B.3. Boric Acid Make-Up System Room, Unit A

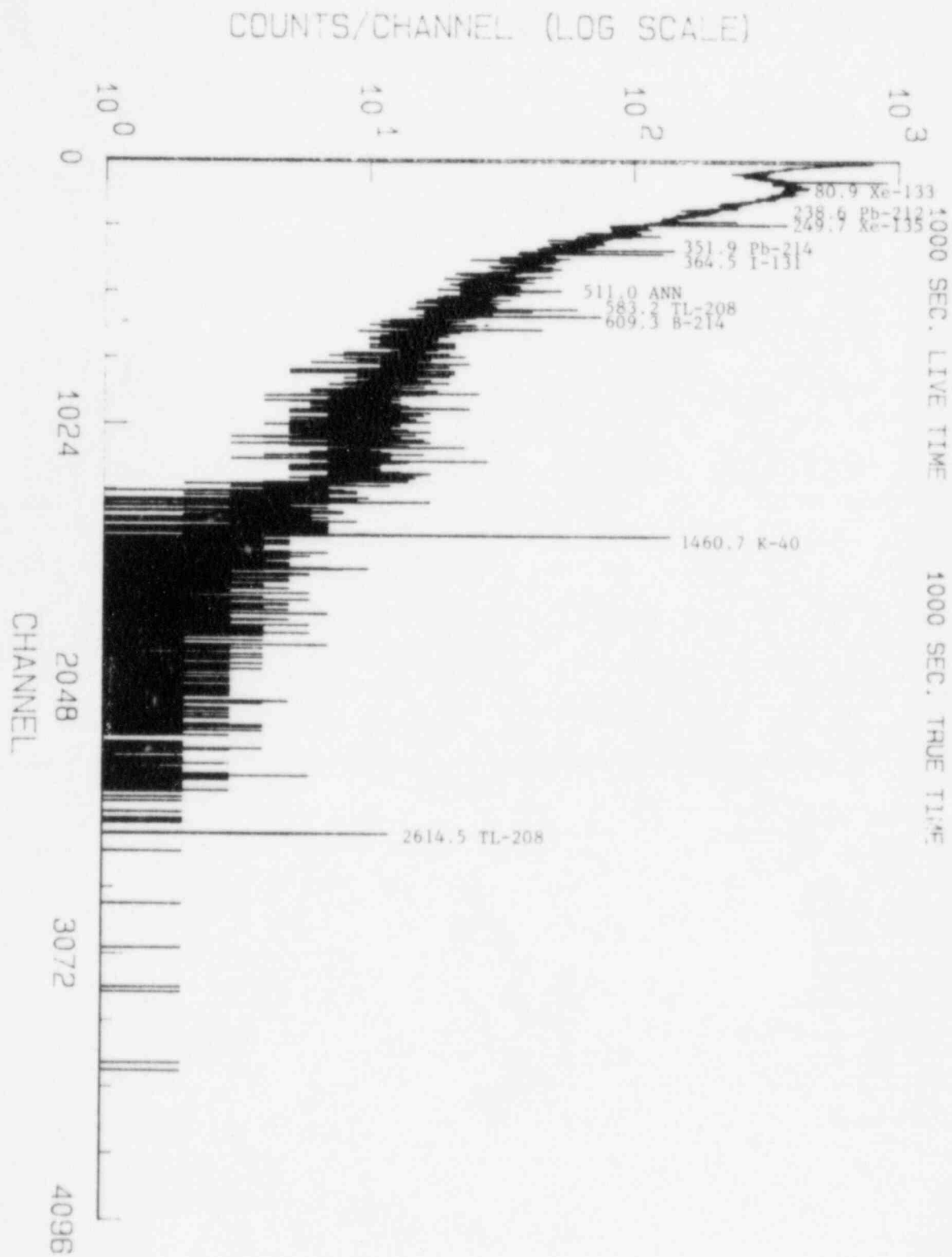

FIGURE B.4. General Area Near Letdown Heat-Exchanger Room, Unit B

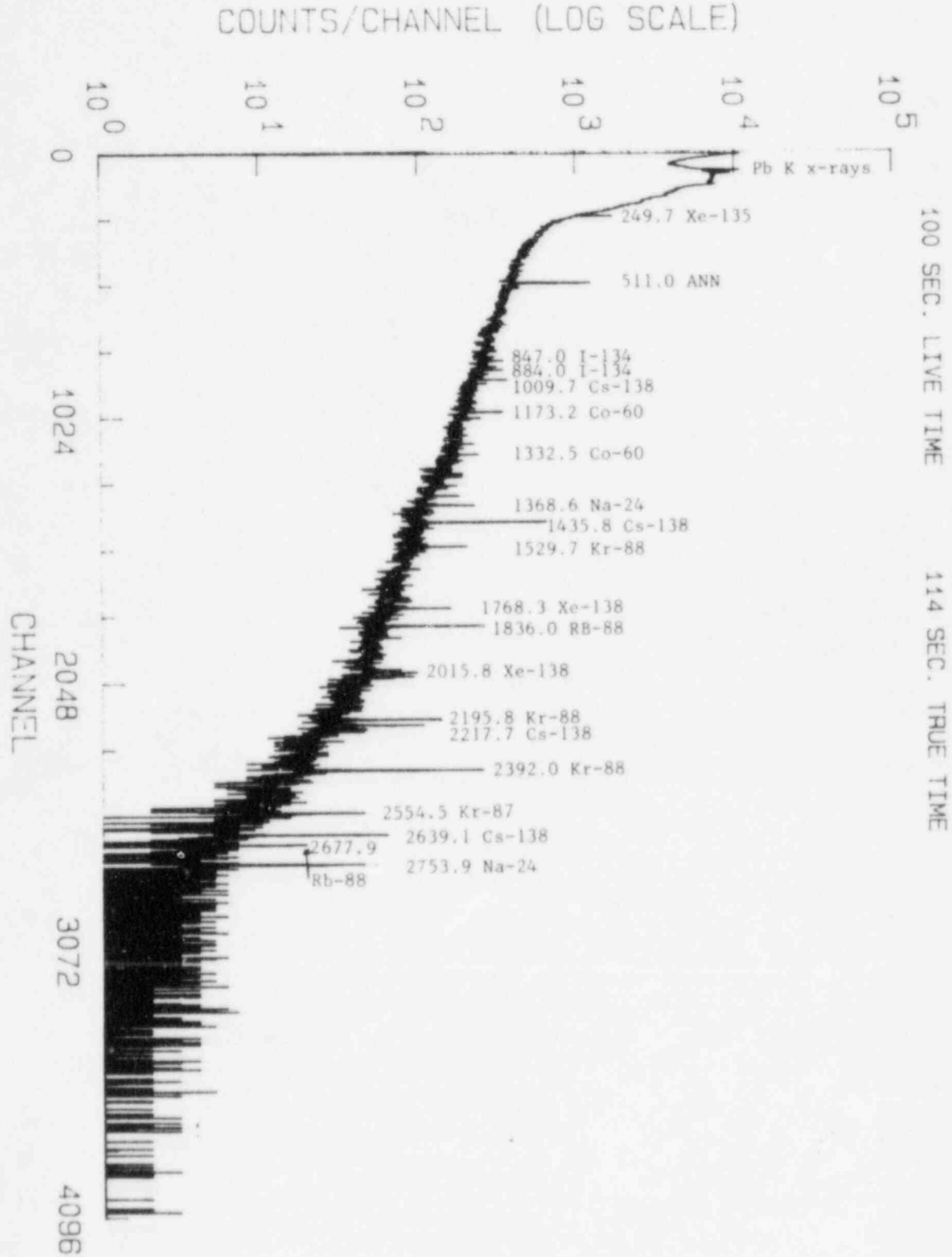

FIGURE B.5. Bypass Ion-Exchanger Valve Room/Outlet, Unit B

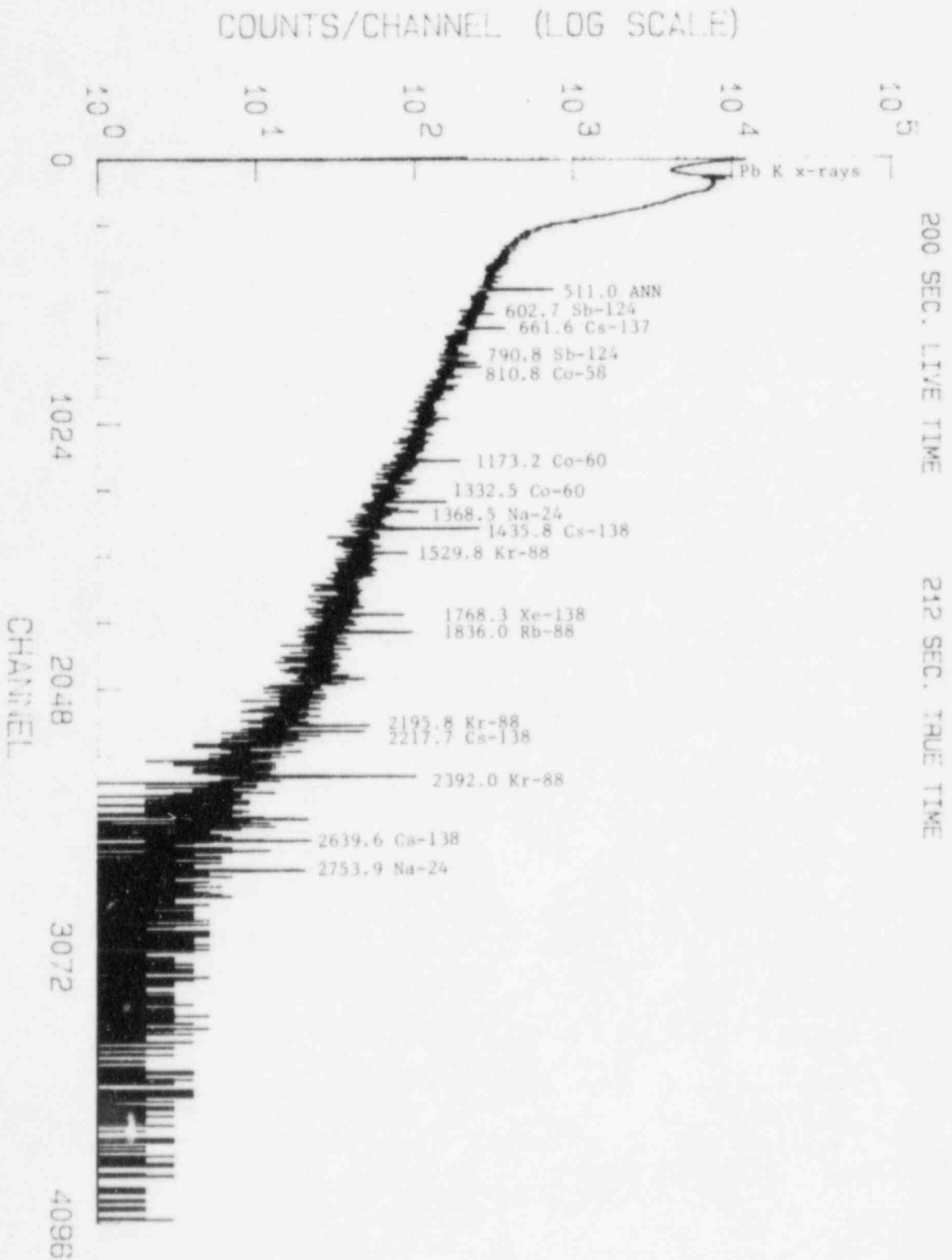

FIGURE B.6. Boric Acid Make-Up System Room, Unit B

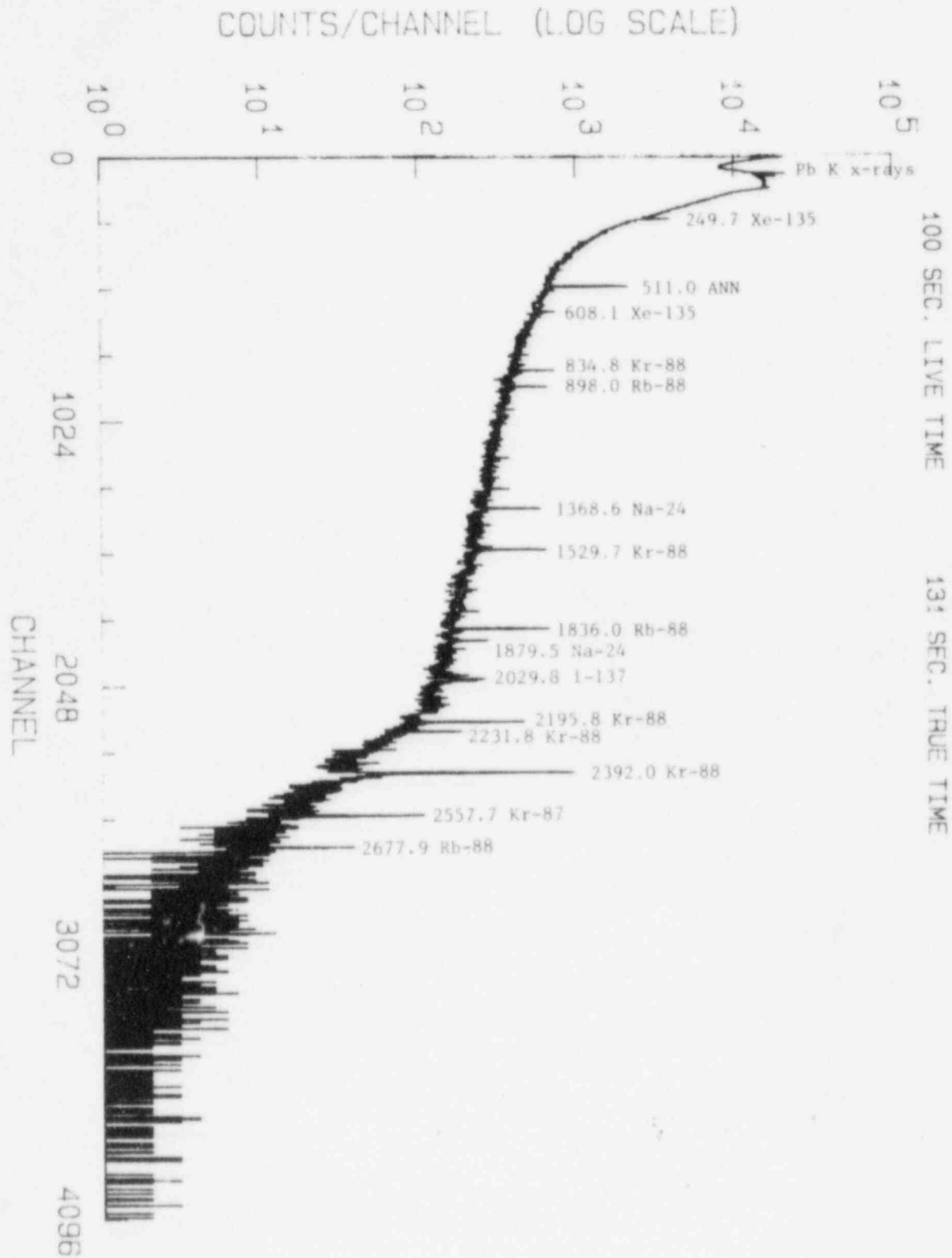

FIGURE B.7. Pressurizer Off-Gas Line, Unit B

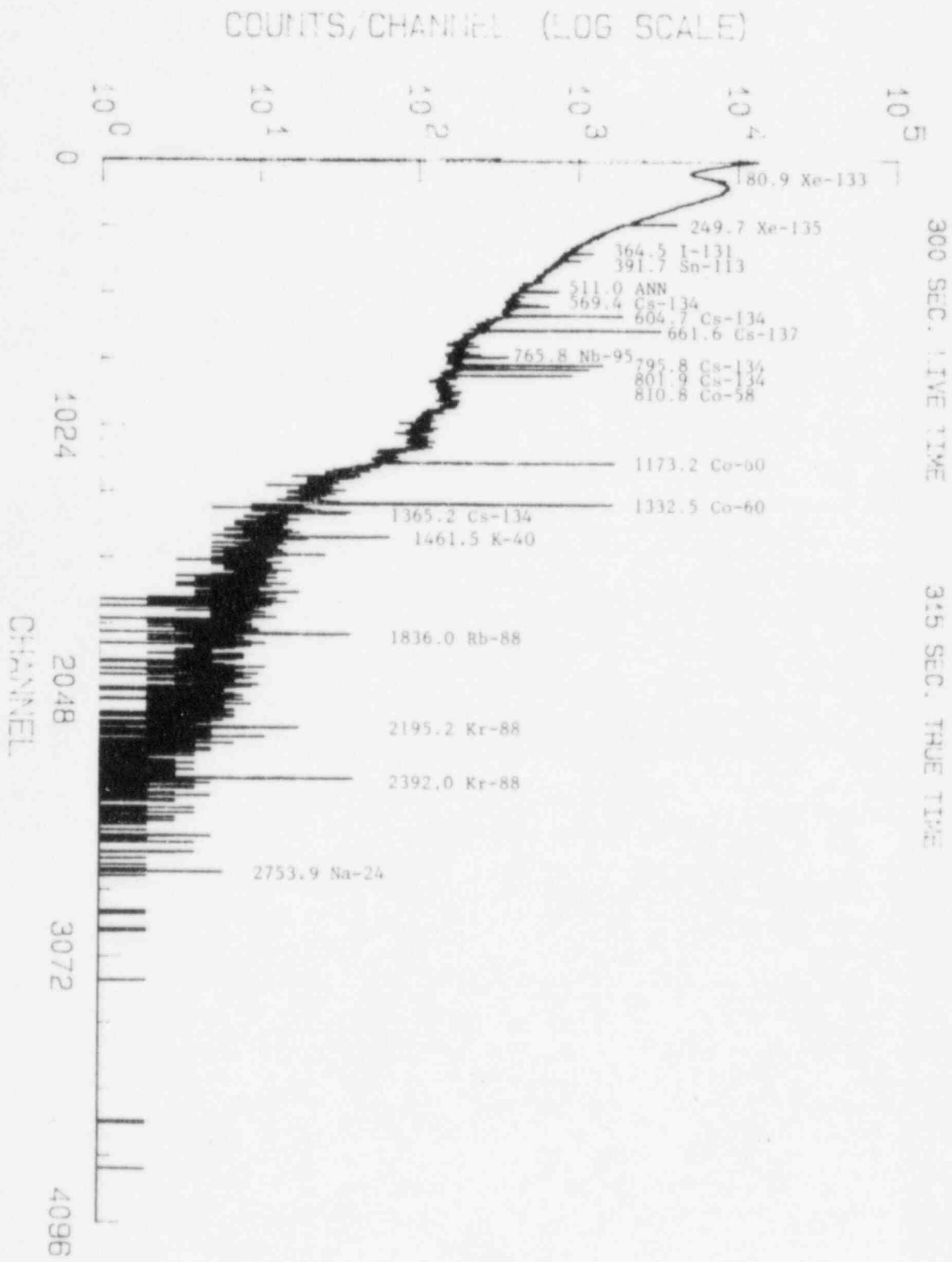

FIGURE B.8. Purifying Ion-Exchanger Valve Room/Inlet, Unit B

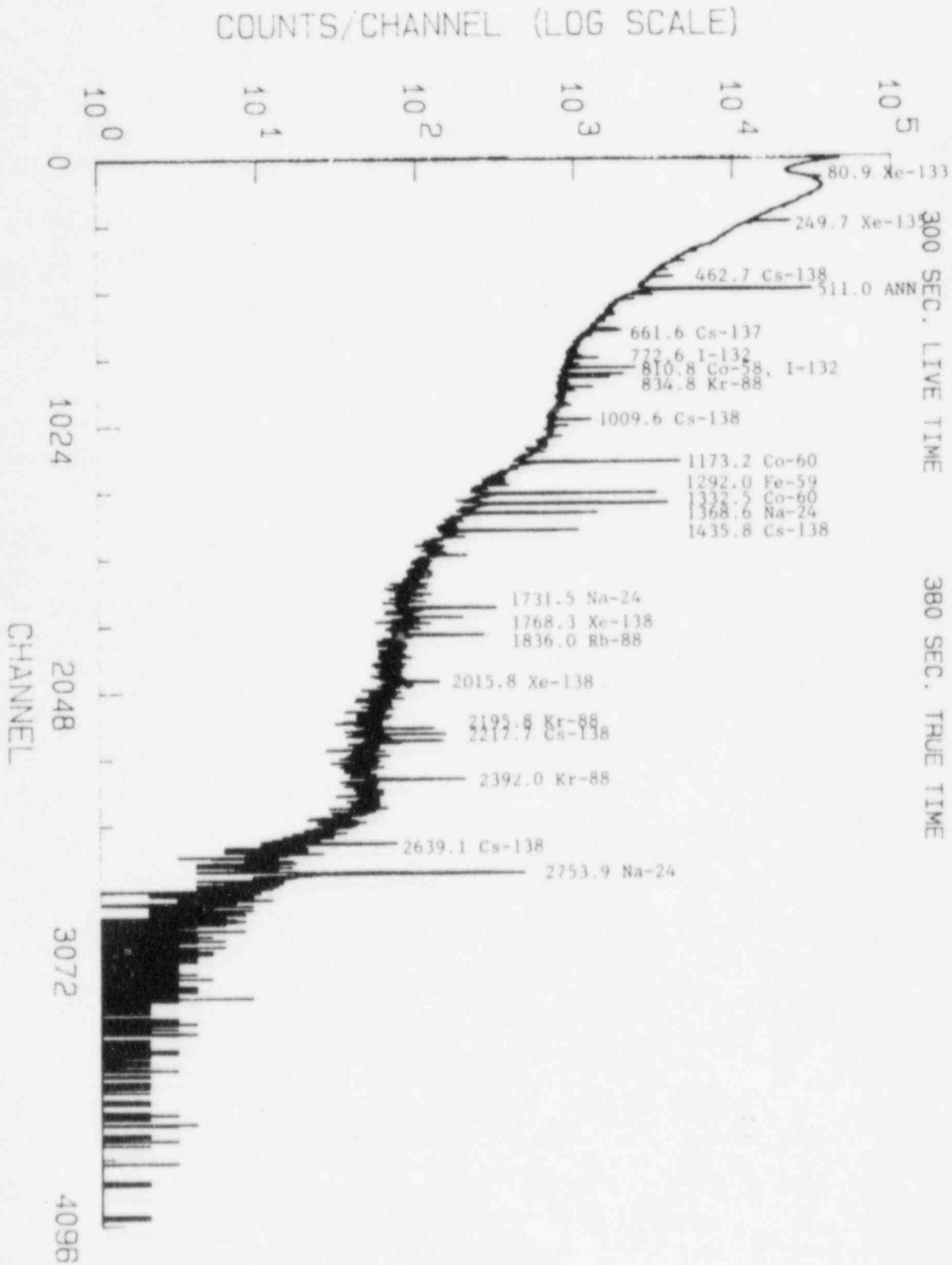

FIGURE B.9. Purifying Ion-Exchanger Valve Room/Inlet, Unit A

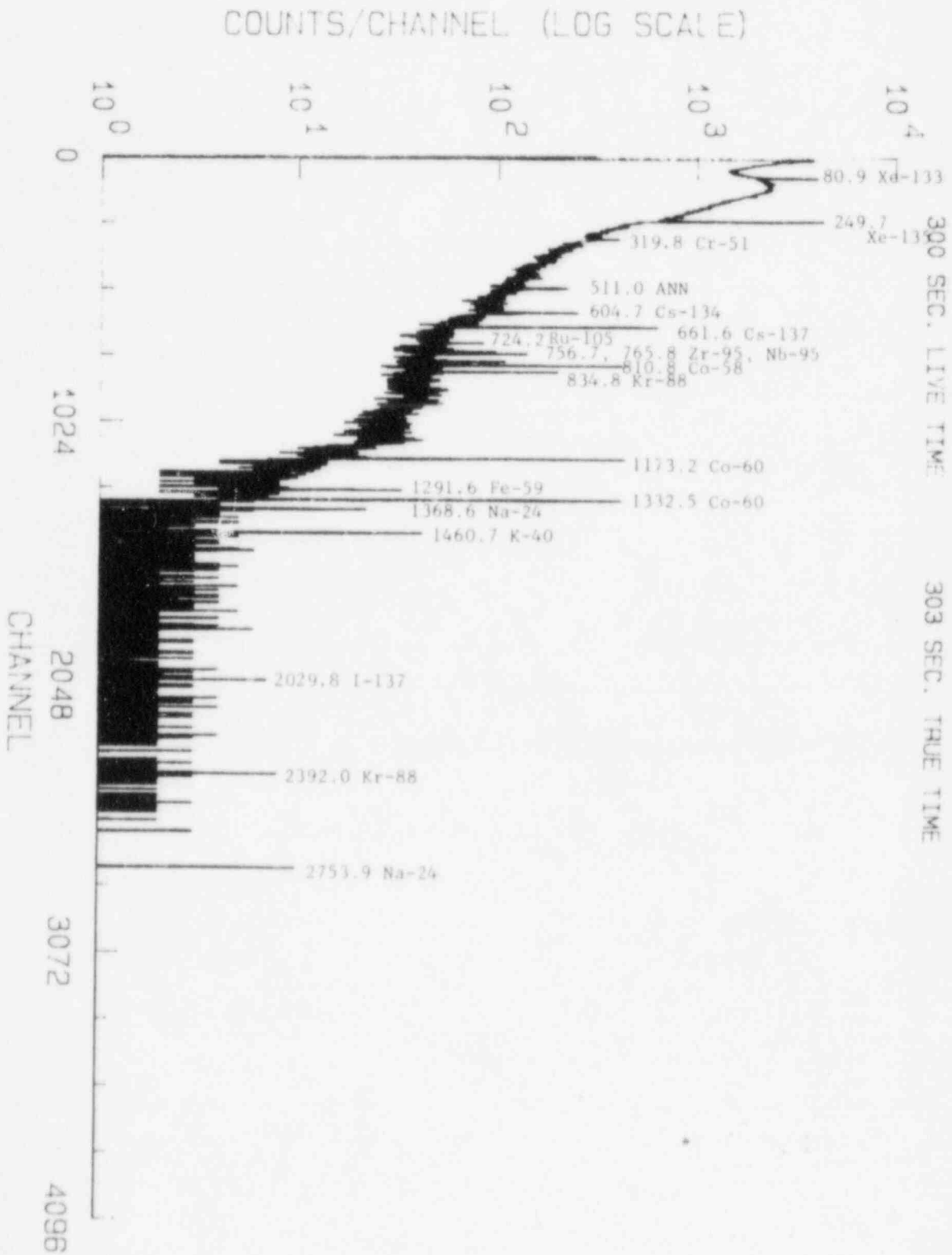

FIGURE B.10. Charging Pump Room, Unit A

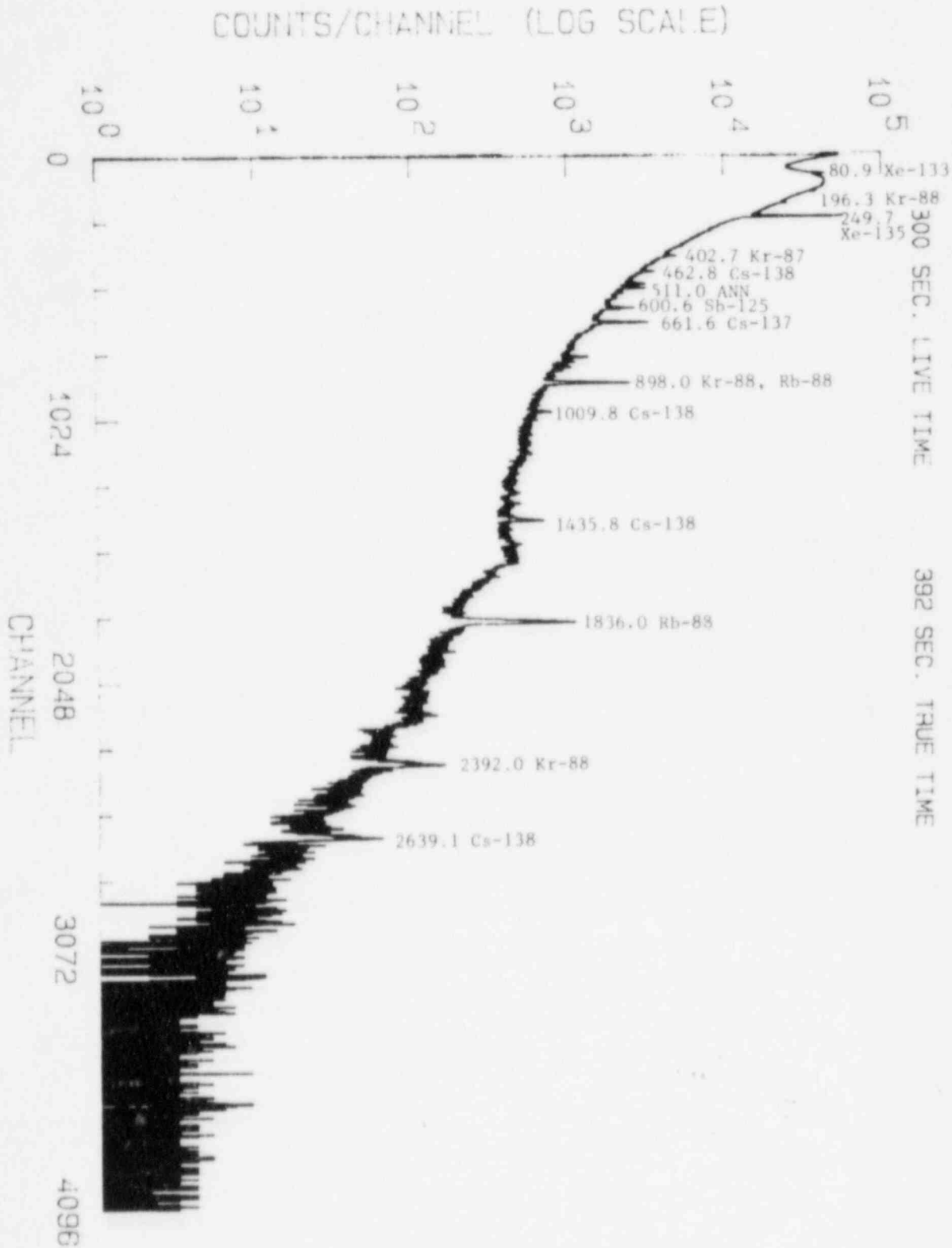

FIGURE B.11. Charging Pump Room, Unit B

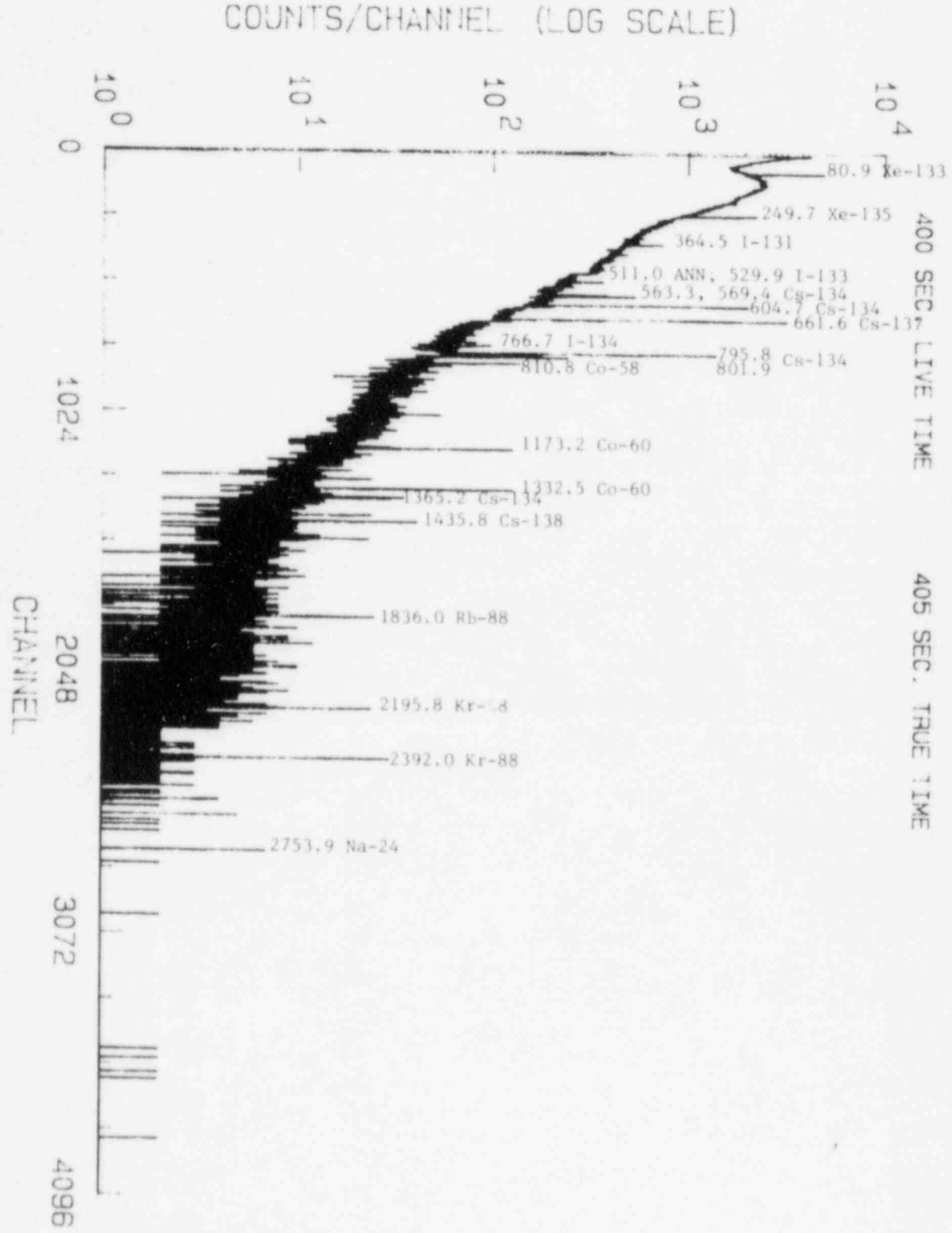

FIGURE B.12. Outside Pipe Penetration Room, Unit B

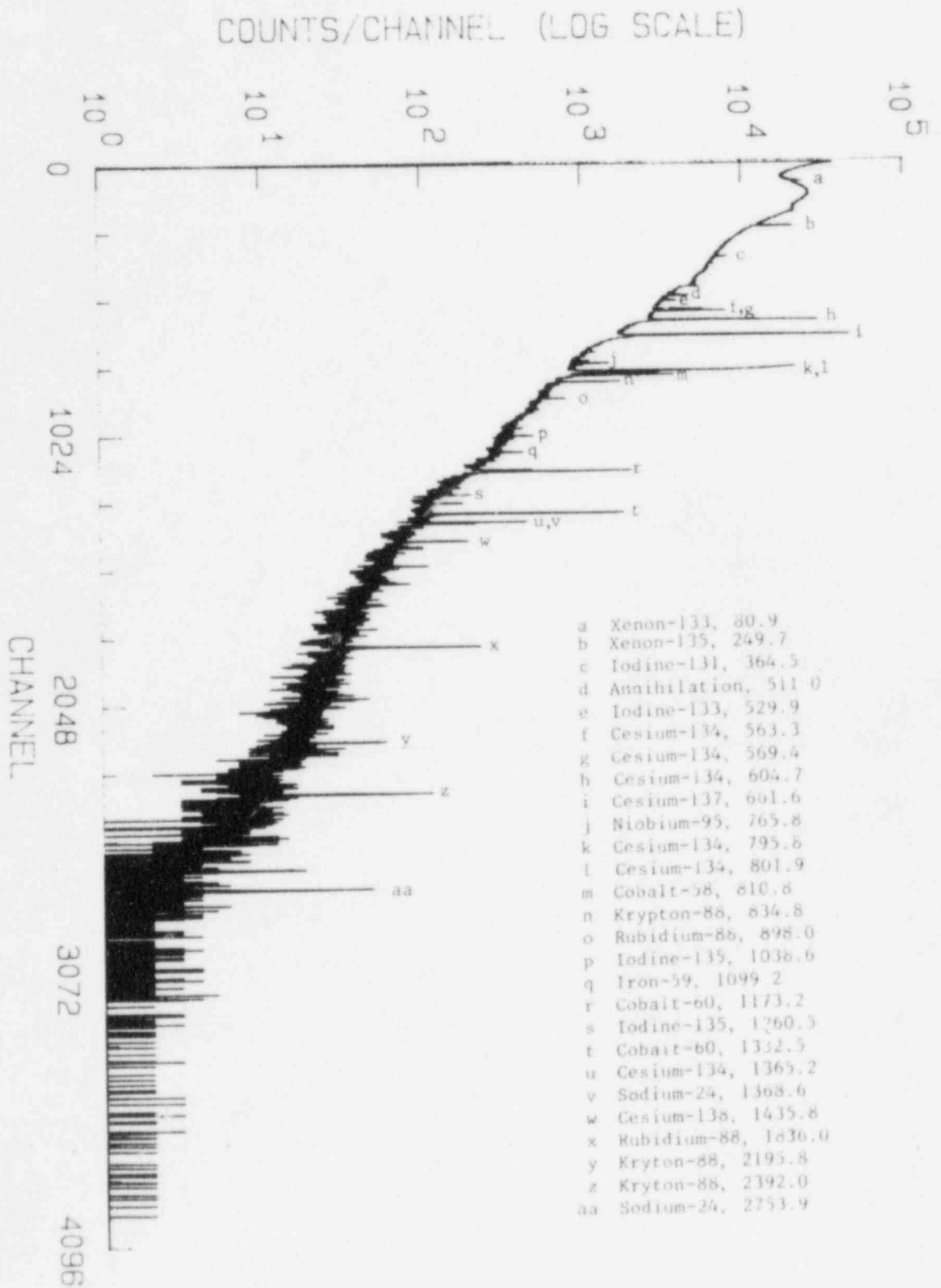

FIGURE B.13. Inside Pipe Penetration Room, Unit B

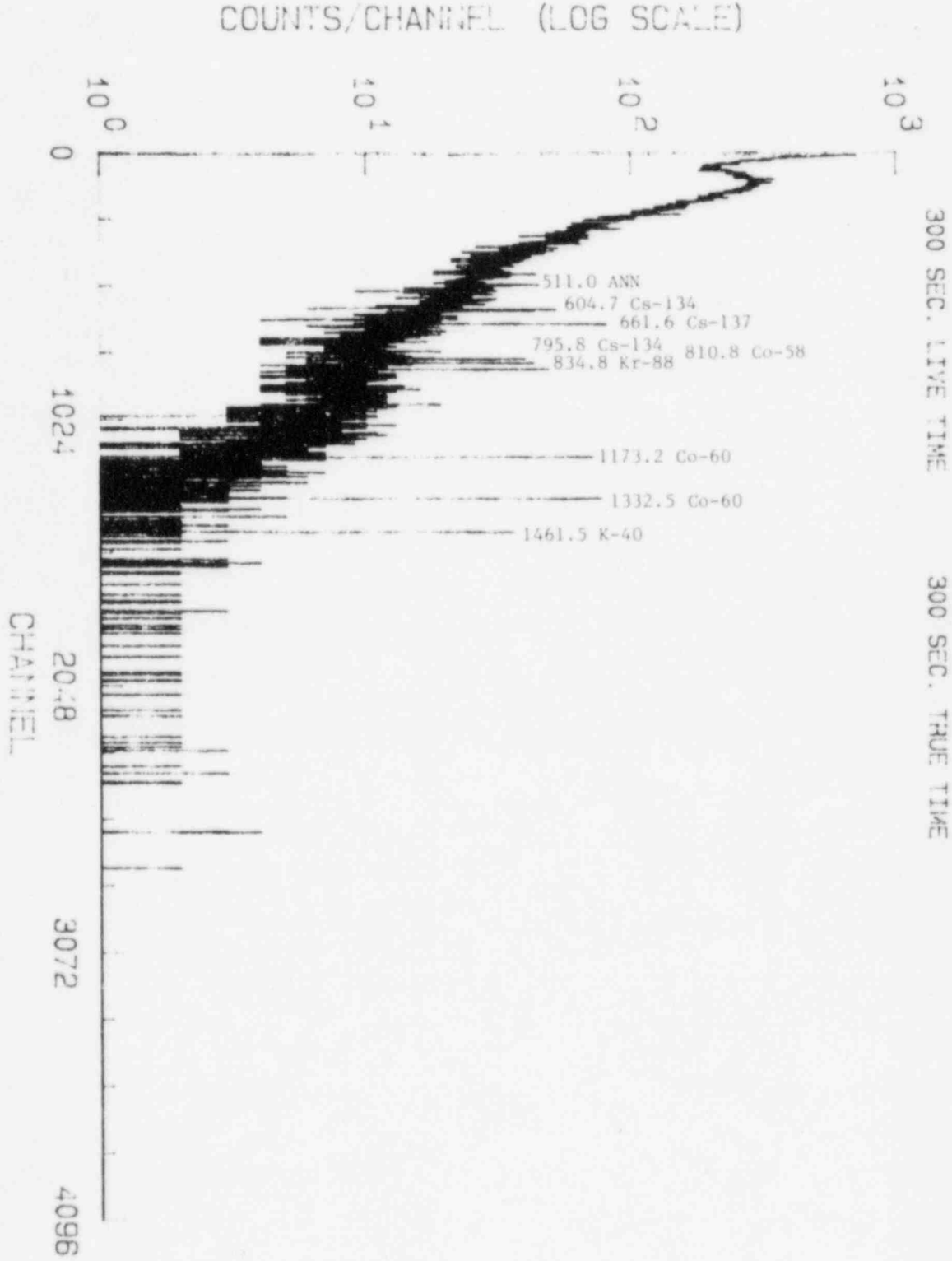

FIGURE B.14. Outside Pipe Penetration Room, Unit A

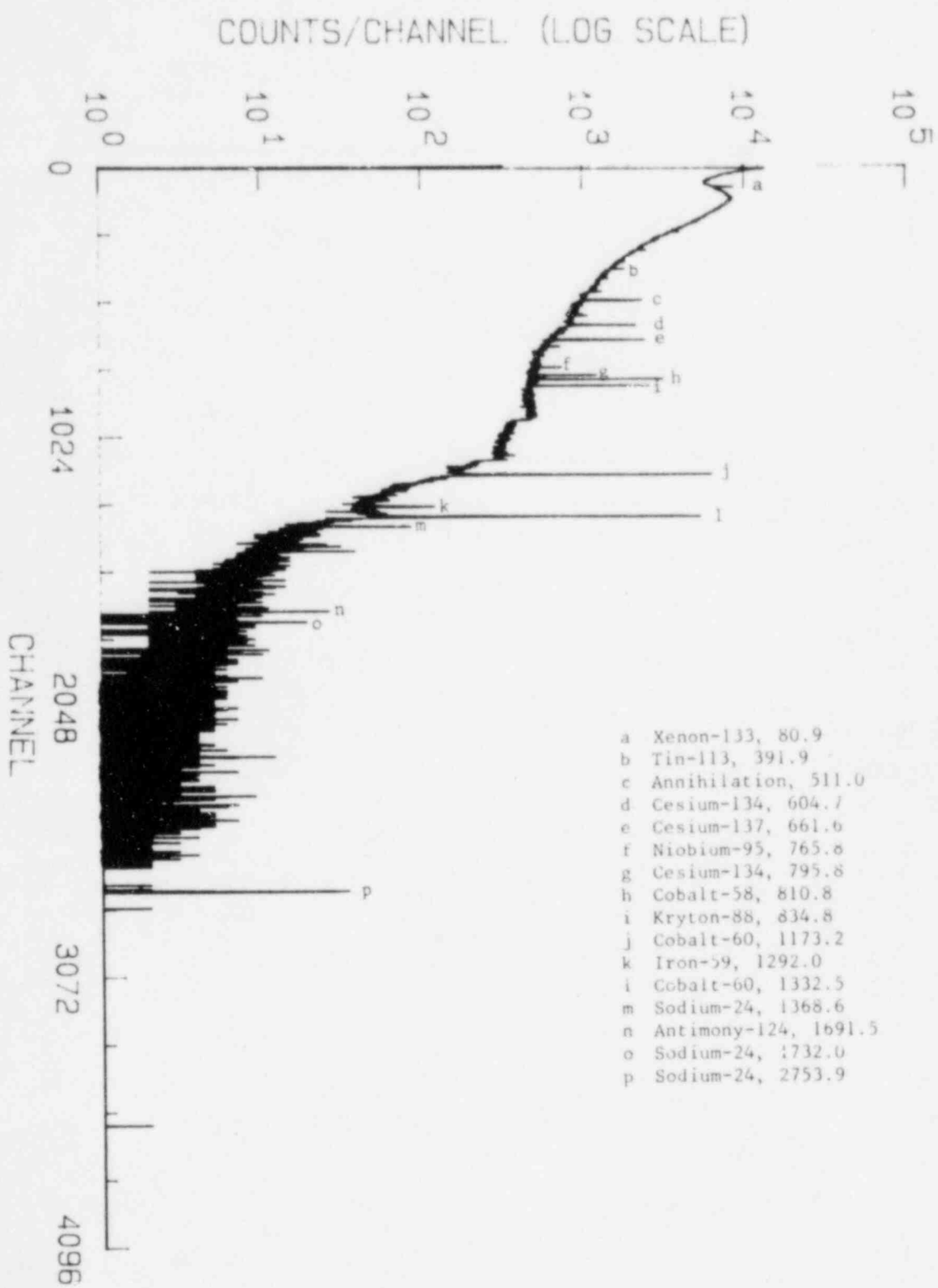

FIGURE B.15. Inside Pipe Penetration Room, Unit A

APPENDIX C

COMPUTER PROGRAM SOURCE CODE LISTING FOR THE CALCULATION OF ATTRIBUTABLE DOSE RATES

The computer program source code has two parts. First, the code KEVFIL is used to develop an input file. Then the code KEVCAL uses the input file to calculate the attributable dose rates resulting from activation products, fission products and other radionuclides.

```
100 'PROGRAM KEVFIL
110 DIM RTC(100)
120 DIM RKEV(100)
130 DEFINT I
140 DEFSTR S,Y,N
150 DEFSNG R
160 I3=92
170 DATA 79.62,80.8,102.6,128.3,172.55,184.24,238.6,249.8,319.6,351.8,364.5,391.
9,402.4,434.4,452.8,475.4,477.7,503,507,511,522.75,529.9,546.6,563.3,563.4
180 DATA 583.2,598.2,600,602.6,604.8,607.5,609.1,653.5,656.7,661.7,664.7,667.6,6
86.1,724.2,756.8,765.9,772.4,788.9,795.8,802,810.9,835,846.6,884,887.4,891.6
190 DATA 898,1002.3,1009.6,1038.6,1089.2,1115.8,1167.9,1173.4,1246.5,1260.9,1292
,1332.6,1336,1365.1,1369,1436.9,1455.9,1461.2,1523.5,1675.3
200 DATA 1691.5,1732,1767.6,1811.7,1818.1,1824.1,1836,1879.8,2015.4,2028.7,2185.
7,2217.4,2230.6,2242.7,2392.2,2554.7,2567.6,2614.7,2638.7,2678.7,2764.3
210 FOR I1=1 TO I3
220 READ RKEV(I1)
230 NEXT I1
240 PRINT:INPUT"NAME YOUR NEW INPUT FILE: ",SFILE
250 PRINT:INPUT"IDENTIFY PLANT A OR B: ",SAB
260 PRINT:INPUT"IDENTIFY MEASUREMENT LOCATION: ",SLOC
270 PRINT:PRINT"INPUT COUNTING TIME IN SECONDS."
280 INPUT"(RESPOND 1234): ",RSEC
290 OPEN "O",#1, SFILE
300 FOR I1=1 TO I3
310 PRINT:PRINT USING"FOR KEV ENERGY ####.#",RKEV(I1)
320 PRINT"INPUT TOTAL NUMBER OF COUNTS."
330 INPUT"(RESPOND 123456): ",RTC(I1)
340 NEXT I1
350 LPRINT:LPRINT USING"FILE NAME:  &";SFILE
360 LPRINT:LPRINT"  ID         KEV       TOTAL COUNTS"
370 LPRINT" ____     ________  ____________":LPRINT
380 FOR I1=1 TO I3
390 IF(RTC(I1)=0)THEN 410
400 LPRINT USING"  ##      #####.#        ######.";I1,RKEV(I1),RTC(I1)
410 NEXT I1
420 PRINT:PRINT"ARE ALL INPUT VALUES CORRECT?"
430 INPUT"(RESPOND YES OR NO): ",S
440 IF((LEFT$(S,1)="Y")OR(LEFT$(S,1)="y"))THEN 520
450 IF((LEFT$(S,1)="N")OR(LEFT$(S,1)="n"))THEN 460 ELSE 420
460 PRINT:PRINT"IDENTIFY ERROR OR ESCAPE  Y TYPING 0."
470 INPUT"(RESPOND 1, 2 ... 100, OR 0): ",I1
480 IF(I1=0)THEN 350
490 IF((I1<1)OR(I1>I3))THEN 460
500 PRINT:PRINT"INPUT CORRECT NUMBER OF TOTAL COUNTS"
510 INPUT"(RESPOND 123456): ",RTC(I1):GOTO 460
520 PRINT:PRINT USING"WRITING INPUT VALUES TO &";SFILE
530 WRITE #1, SAB
540 WRITE #1, SLOC
550 WRITE #1, RSEC
560 FOR I1=1 TO I3
570 WRITE #1, RTC(I1)
580 NEXT I1
590 CLOSE #1
600 PRINT:PRINT"PROGRAM COMPLETED - FILE CLOSED."
610 PRINT:PRINT"BYE"
620 END
```

```
100 'PROGRAM KEVCAL
110 DIM RKEV(100)
120 DIM RDOS(100)
130 DIM RTC(100)
140 DIM RTYP(100)
150 DIM RCUM(4)
160 DEFINT I
170 DEFSTR S,Y,N
180 DEFSNG R
190 I3=92
200 DATA 79.6,80.8,102.6,128.3,172.6,194.2,238.6,249.8,319.8,351.8,364.5,391.9,4
02.4,434.4,462.6,475.4,477.7,503,507,511,522.8,529.9,546.6,563.3,569.4
210 DATA 583.2,598.2,600,602.8,604.8,607.5,608.1,653.5,656.7,661.7,664.7,667.6,6
66.1,724.2,766.8,765.8,772.4,789.9,795.9,802,810.9,835,846.6,884,887.4,891.6
220 DATA 898,1002.3,1009.8,1038.6,1099.2,1115.8,1167.9,1173.4,1248.5,1260.9,1292
,1332.6,1338,1365.1,1369,1426.9,1435.9,1461.5,1529.5,1675.3
230 DATA 1691.5,1732,1767.6,1811.7,1818.1,1824.1,1836,1879.9,2015.4,2028.7,2185.
7,2217.4,2230.6,2242.7,2392.2,2554.7,2567.6,2614.7,2638.7,2678.7,2754.3
240 FOR I1=1 TO I3
250 READ RKEV(I1)
260 NEXT I1
270 'RTYP: 1=ACTIVATION  2=FISSION  3=NATURAL/ANNIHILATION/UNKNOWN
280 DATA 2,2,2,2,2,2,3,2,1,3,2,1,2,2,2,2,1,3,3,3,2,2,2,2,2,3,2,2,2,2,2,3
290 DATA 2,2,2,2,2,1,2,1,2,2,2,2,2,1,2,1,2,3,2,2,3,2,2,1,1,2,1,2,2,1,1,2
300 DATA 2,1,2,2,3,2,1,2,1,2,1,2,2,2,1,2,2,2,2,2,1,2,2,2,3,2,2,1
310 FOR I1=1 TO I3
320 READ RTYP(I1)
330 NEXT I1
340 S1="I131  I133* Xe133 Cs134MNb98  Kr88  Pb212 Xe135 Cr51  Pb214 I131  Sn113
Kr87  Xe138 Cs138 Cs134 Be7   UNKWN UNKWN ANNIH "
350 S2="I132  I133  I135* Cs134 Cs134 TL208 Xe142 Sb125 Sb124 Cs134 Xe135*Bi214
Sr91  Xe142 Cs137 Cs173 I132  W187  Ru105 Zr95  "
360 S3="Nb95  I132* Xe142 Cs134 Cs134 Co58  Kr88  Mn56  I134  UNKWN Xe142 Rb88*
UNKWN Cs138 I135  Fe59  Zn65  Cs134 Co60  Rb89  "
370 S4="I135  Fe59  Co60  Xe142 Cs134 Na24  I137  Cs138 K40   Kr88  Co58  Sb124
Na24  Xe138 Mn56  Cs134 Cs134 Rb88* Na24  Xe138 "
380 S5="I137  Kr88* Cs138 Kr88  Na24  Kr88  Kr87  Kr87  TL208 Cs138 Rb88  Na24
"
390 RCUM(1)=0:RCUM(2)=0:RCUM(3)=0:RTOT=0
400 PRINT:PRINT:INPUT"IDENTIFY DATA FILE: ",SFILE
410 OPEN "I",#1, SFILE
420 INPUT #1, SAB
430 INPUT #1, SLOC
440 INPUT #1, RSEC
450 FOR I1=1 TO I3
460 INPUT #1, RTC(I1)
470 NEXT I1
480 PRINT:PRINT"INPUT GEOMETRIC CORRECTION FACTOR."
490 INPUT"(RESPOND 12345): ",RGEO
500 'LPRINT:LPRINT"*** DIAGNOSTICS ***
510 FOR I1=1 TO I3
520 IF(RTC(I1)=0)THEN 810
530 'DETERMINE NET CPS
540 RCPS=RTC(I1)/RSEC
550 'NATURAL LOG OF ENERGY IN KeV
560 RX=LOG(RKEV(I1))
```

```
570 'DETERMINE DETECTOR EFFICIENCY BY ENERGY
580 IF(RKEV(I1)>279.2)THEN 600
590 RA=-34.4878:RB=10.1687:RC=-1.02472:RD=0:RE=0:GOTO 610
600 RA=-76.1537:RB=45.4905:RC=-11.164:RD=1.18362:RE=-.0466965
610 REFF=RA+(RB*RX)+(RC*RX^2)+(RD*RX^3)+(RE*RX^4)
620 REFF=EXP(REFF)*RGEO
630 'CORRECT NET CPS FOR EFFICIENCY
640 RCPS=RCPS/REFF
650 'DETERMINE FLUX
660 'AREA OF DETECTOR=PI*2.25^2=15.904313
670 RFLUX=RCPS/15.904313#
680 'DETERMINE DOSE FACTOR (REM/HR)/(PHOTON/CM2-SEC)
690 'NATURAL LOG OF ENERGY IN MeV
700 RX=LOG(RKEV(I1)/1000)
710 IF(RKEV(I1)>500)THEN 730
720 RA=-13.626:RB=-.57117:RC=-1.0954:RD=-.24897:GOTO 740
730 RA=-13.133:RB=.72008:RC=-.033603:RD=0
740 RDF=RA+(RB*RX)+(RC*RX^2)+(RD*RX^3)
750 RDF=EXP(RDF)
760 'DETERMINE DOSE RATE (mR/hr) USING FLUX AND DOSE FACTOR
770 RDOS(I1)=1000*(RFLUX*RDF)
780 'LPRINT USING"I1= ##  RFLUX= ##.##^^^^ REFF= ##.##^^^^ RDF= ##.##^^^^ RDOS(I
1)= ##.##^^^^";I1;RFLUX;REFF;RDF;RDOS(I1)
790 RCUM(RTYP(I1))=RCUM(RTYP(I1))+RDOS(I1)
800 RTOT=RTOT+RDOS(I1)
810 NEXT I1
820 LPRINT:LPRINT"**********************************************************
************"
830 LPRINT:LPRINT USING"PLANT:  UNIT & ";SAB
840 LPRINT:LPRINT USING"LOCATION:  & ";SLOC
850 LPRINT:LPRINT USING"FILE NAME:  & ";SFILE
860 LPRINT:LPRINT USING"COUNTING TIME (SEC):  ######.";RSEC
870 LPRINT:LPRINT USING"GEOMETRIC CORRECTION FACTOR: #####.";RGEO
880 LPRINT:LPRINT"**********************************************************
************"
890 LPRINT:LPRINT"*** DATA ANALYSIS ***"
900 LPRINT:LPRINT"   ID     RADIONUCLIDE        KEV        TOTAL COUNTS      DOSE
 RATE (mR/hr)"
910 LPRINT"  ____    ___________      _______    ____________      ___________
_____":LPRINT
920 FOR I1=1 TO I3
930 IF(RTC(I1)=0)THEN 1030
940 IF(I1>80)THEN 1020
950 IF(I1>60)THEN 1010
960 IF(I1>40)THEN 1000
970 IF(I1>20)THEN 990
980 LPRINT USING"   ##        &         #####.#      #######.            ##.##^^
^^";I1;MID$(S1,(I1*6)-5,6);RKEV(I1);RTC(I1);RDOS(I1):GOTO 1030
990 LPRINT USING"   ##        &         #####.#      #######.            ##.##^^
^^";I1;MID$(S2,(I1*6)-125,6);RKEV(I1);RTC(I1);RDOS(I1):GOTO 1030
1000 LPRINT USING"   ##        &         #####.#      #######.            ##.##^
^^^";I1;MID$(S3,(I1*6)-245,6);RKEV(I1);RTC(I1);RDOS(I1):GOTO 1030
1010 LPRINT USING"   ##        &         #####.#      #######.            ##.##^
^^^^";I1;MID$(S4,(I1*6)-365,6);RKEV(I1);RTC(I1);RDOS(I1):GOTO 1030
```

```
1020 LPRINT USING"   ##        &         #####.#      #######.            ##.##
^^^^";I1;MID$(85,(I1*6)-485,6);RKEV(I1);RTC(I1);RDOS(I1)
1030 NEXT I1
1040 LPRINT:LPRINT"*************************************************************
*************"
1050 LPRINT:LPRINT"*** ATTRIBUTABLE DOSE RATE (mR/hr) ***"
1060 LPRINT:LPRINT"  ACTIVATION        FISSION        NATURAL BKGD          TOTA
L"
1070 LPRINT"          PRODUCTS        PRODUCTS        /UNKWN/ANNIH        DOSE R
ATE"
1080 LPRINT" _________   _______   ___________   _________"
1090 LPRINT:LPRINT USING"  ##.##^^^^       ##.##^^^^           ##.##^^^^         #
#.##^^^^";RCUM(1);RCUM(2);RCUM(3);RTOT
1100 LPRINT:LPRINT"*************************************************************
*************":LPRINT:LPRINT:LPRINT:LPRINT:LPRINT:LPRINT:LPRINT:LPRINT
1110 END
```

APPENDIX D

ATTRIBUTABLE DOSE RATES DERIVED FROM THE GAMMA SPECTRAL DATA

The gamma spectroscopy data have been analyzed to determine the exposure rates attributable to individual radionuclides. With this information, a summation of the exposure rates from activation or corrosion products is possible to determine their collective potential impact on occupational radiation exposures. Radionuclides that are not attributable to activation or corrosion products are listed in a group labeled natural background/annihilation/unknown. Analyses of the 15 gamma spectroscopy measurements are presented here.

PLANT: UNIT A

LOCATION: INSIDE PIPE PENETRATION ROOM

FILE NAME: GSM 15

COUNTING TIME (SEC): 300.

GEOMETRIC CORRECTION FACTOR: 900.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133+	80.8	2851.	2.33E-03
12	Sn113	391.8	763.	3.58E-03
20	ANNIH	511.0	4181.	3.16E-02
30	Cs134	604.8	2050.	2.66E-02
35	Cs137	661.7	4106.	4.80E-02
41	Nb95	765.8	498.	7.41E-03
44	Cs134	795.8	1745.	2.76E-02
45	Cs134	802.0	219.	3.51E-03
46	Co58	810.8	6585.	9.11E-02
47	Kr88	835.0	4526.	7.74E-02
59	Co60	1173.4	13258.	3.91E-01
62	Fe59	1292.0	330.	1.14E-02
63	Co60	1332.6	12634.	4.57E-01
66	Na24	1369.0	221.	8.35E-03
72	Sb124	1691.5	35.	1.87E-03
73	Na24	1732.0	51.	2.84E-03
82	Na24	2754.3	122.	1.60E-02

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
9.81E-01	1.95E-01	3.16E-02	1.21E+00

****

PLANT: UNIT B

LOCATION: INSIDE PIPE PENETRATION ROOM

FILE NAME: GSM 13

COUNTING TIME (SEC): 300.

GEOMETRIC CORRECTION FACTOR: 900.

****

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.6	10056.	8.22E-03
3	Xe133	102.6	2748.	1.97E-03
8	Xe135	248.8	17503.	3.33E-02
11	I131	364.5	2204.	9.04E-03
16	Cs134	475.4	2315.	1.51E-02
20	ANNIH	511.0	3748.	2.83E-02
22	I133	529.9	1152.	9.25E-03
24	Cs134	563.3	5570.	4.87E-02
25	Cs134	568.4	10279.	9.34E-02
30	Cs134	604.6	61688.	6.20E-01
35	Cs137	661.7	97439.	1.14E+00
36	Cs173	664.7	1111.	1.31E-02
41	Nb95	765.9	1267.	1.88E-02
44	Cs134	795.8	43320.	6.86E-01
45	Cs134	802.0	4743.	7.60E-02
46	Co58	810.8	6153.	1.00E-01
47	Kr88	835.0	2444.	4.18E-02
52	Rb88*	898.0	818.	1.19E-02
55	I135	1038.6	462.	1.12E-02
56	Fe59	1099.2	313.	8.32E-03
58	Cs134	1167.9	595.	1.74E-02
59	Co60	1173.4	4703.	1.39E-01
61	I135	1260.5	174.	5.76E-03
62	Fe59	1292.0	246.	8.47E-03
63	Co60	1332.6	4586.	1.59E-01
65	Cs134	1365.1	914.	3.44E-02
66	Na24	1369.0	360.	1.36E-02
68	Cs136	1435.9	336.	1.37E-02
70	Kr88	1529.5	127.	5.75E-03
78	Rb88*	1836.0	588.	3.68E-02
82	Kr88*	2195.7	123.	1.04E-02
86	Kr88	2392.2	358.	3.56E-02
91	Rb88	2678.7	62.	7.70E-03
92	Na24	2754.3	152.	2.00E-02

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
4.49E-01	3.01E+00	2.83E-02	3.48E+00

****

PLANT: UNIT A

LOCATION: OUTSIDE FIFE PENETRATION ROOM

FILE NAME: GSN 14

COUNTING TIME (SEC): 300.

GEOMETRIC CORRECTION FACTOR: 900.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
30	Cs134	604.6	25.	2.51E-04
35	Cs137	661.7	112.	1.31E-03
44	Cs134	795.9	76.	1.20E-03
46	Co58	810.8	62.	1.01E-03
47	Kr88	835.0	90.	1.54E-03
58	Co60	1173.4	172.	5.07E-03
63	Co60	1332.6	164.	5.83E-03
69	K40	1461.5	89.	3.74E-03

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
1.20E-02	4.30E-03	3.74E-03	2.01E-02

PLANT: UNIT B

LOCATION: OUTSIDE PIPE PENETRATION ROOM

FILE NAME: GSM 12

COUNTING TIME (SEC): 400.

GEOMETRIC CORRECTION FACTOR: 900.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.6	4196.	2.57E-03
8	Xe135	249.8	2253.	3.22E-03
11	I131	364.5	427.	1.31E-03
20	ANNIH	511.0	316.	1.79E-03
24	Cs134	563.3	342.	2.29E-03
25	Cs134	569.4	637.	4.34E-03
30	Cs134	604.8	3820.	2.68E-02
35	Cs137	661.7	5846.	5.12E-02
44	Cs134	795.9	2775.	3.29E-02
45	Cs134	802.0	238.	2.86E-03
46	Co58	810.8	415.	5.08E-03
47	Kr88	835.0	237.	3.04E-03
59	Co60	1173.4	273.	6.04E-03
63	Co60	1332.6	285.	7.73E-03
65	Cs134	1365.1	28.	7.90E-04
69	K40	1461.5	98.	3.09E-03
78	Rb88*	1836.0	46.	2.12E-03
82	Kr88*	2195.7	71.	4.50E-03
86	Kr88	2392.2	89.	6.63E-03
92	Na24	2754.3	15.	1.48E-03

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
2.03E-02	1.47E-01	4.68E-03	1.72E-01

****

PLANT: UNIT A

LOCATION: GENERAL AREA NEAR LETDOWN HEAT EXCHANGER ROOM

FILE NAME: GSM 1

COUNTING TIME (SEC): 600.

GEOMETRIC CORRECTION FACTOR: 1000.

****

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.6	292.	1.07E-04
47	Kr88	835.0	275.	2.12E-03
59	Co60	1173.4	781.	1.04E-02
63	Co60	1332.6	718.	1.17E-02
69	K40	1461.5	206.	3.89E-03

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
2.21E-02	2.22E-03	3.89E-03	2.82E-02

****

****

PLANT: UNIT 8

LOCATION: GENERAL AREA NEAR LETDOWN HEAT EXCHANGER ROOM

FILE NAME: GSM 4

COUNTING TIME (SEC): 1000.

GEOMETRIC CORRECTION FACTOR: 1000.

****

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.8	764.	1.69E-04
7	Pb212	238.6	181.	8.45E-05
8	Xe135	249.8	566.	2.91E-04
10	Pb214	251.8	119.	1.23E-04
11	I131	364.5	146.	1.62E-04
20	ANNIH	511.0	96.	1.96E-04
26	TL208	583.2	44.	1.12E-04
32	Bi214	609.1	82.	2.25E-04
69	K40	1461.5	336.	3.81E-03
89	TL208	2614.7	37.	1.18E-03

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
0.00E+00	6.21E-04	5.73E-03	6.35E-03

****

PLANT: UNIT A

LOCATION: PURIFYING ION EXCHANGER VALVE ROOM

FILE NAME: GSN 9

COUNTING TIME (SEC): 300.

GEOMETRIC CORRECTION FACTOR: 600.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.8	18525.	2.27E-02
8	Xe135	249.8	22120.	6.32E-02
11	I131	364.5	1962.	1.21E-02
13	Kr87	402.4	1099.	8.11E-03
14	Xe138	434.4	860.	7.26E-03
15	Cs138	462.6	2201.	2.07E-02
20	ANNIH	511.0	88476.	1.00E+00
22	I133	529.9	2850.	3.19E-02
23	I135*	546.6	748.	9.51E-03
35	Cs137	661.7	1337.	2.34E-02
37	I132	667.6	1604.	2.85E-02
41	Nb95	766.8	457.	1.02E-02
42	I132*	772.4	1109.	2.51E-02
46	Co58	810.9	3577.	8.76E-02
47	Kr88	835.0	3079.	7.90E-02
48	Mn56	846.6	1887.	4.98E-02
49	I134	884.0	1183.	3.33E-02
52	Rb88*	898.0	350.	1.13E-02
54	Cs136	1009.8	1364.	4.75E-02
59	Co60	1173.4	10114.	4.48E-01
61	I135	1260.5	406.	2.01E-02
62	Fe59	1292.0	7596.	3.92E-01
63	Co60	1332.6	9073.	4.92E-01
66	Na24	1368.0	3141.	1.78E-01
68	Cs138	1435.9	2407.	1.47E-01
70	Kr88	1529.5	252.	1.71E-02
73	Na24	1732.0	698.	5.83E-02
74	Xe138	1767.6	334.	2.89E-02
78	Rb88*	1836.0	497.	4.59E-02
80	Xe138	2015.4	276.	3.00E-02
82	Kr88*	2195.7	264.	3.35E-02
83	Cs138	2217.4	355.	4.59E-02
85	Na24	2242.7	363.	4.78E-02
86	Kr88	2392.2	569.	8.48E-02
90	Cs138	2638.7	167.	3.02E-02
92	Na24	2754.3	1789.	3.53E-01

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
2.11E+00	5.17E-01	1.00E+00	4.03E+00

****

PLANT: UNIT 8

LOCATION: PURIFYING ION EXCHANGER VALVE ROOM

FILE NAME: GSM 8

COUNTING TIME (SEC): 300.

GEOMETRIC CORRECTION FACTOR: 850.

****

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.8	4320.	3.35E-03
8	Xe135	248.8	4500.	8.12E-03
11	I131	364.5	680.	2.64E-03
12	Sn113	391.9	489.	2.17E-03
20	ANNIH	511.0	1146.	8.19E-03
25	Cs134	569.4	210.	1.81E-03
30	Cs134	604.8	3544.	3.38E-02
35	Cs137	661.7	6202.	6.87E-02
41	Nb95	765.9	417.	5.87E-03
44	Cs134	795.9	2625.	3.94E-02
45	Cs134	802.0	255.	3.87E-03
46	Co58	810.9	1995.	3.08E-02
47	Kr88	835.0	1515.	2.46E-02
59	Co60	1173.4	3921.	1.10E-01
63	Co60	1332.6	3847.	1.32E-01
65	Cs134	1365.1	24.	8.55E-04
69	K40	1461.5	119.	4.74E-03
78	Rb88*	1836.0	105.	6.12E-03
82	Kr88*	2195.7	67.	5.37E-03
86	Kr88	2392.2	110.	1.04E-02
92	Na24	2754.3	13.	1.62E-03

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
2.76E-01	2.15E-01	1.29E-02	5.04E-01

****

****

PLANT: UNIT A

LOCATION: BYPASS ION EXCHANGER VALVE ROOM

FILE NAME: GSM Z

COUNTING TIME (SEC): 100.

GEOMETRIC CORRECTION FACTOR: 63.

****

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
5	Nb98	172.6	917.	3.90E-02
20	ANNIH	511.0	7768.	2.51E+00
40	Zr95	756.8	256.	1.60E-01
41	Nb95	765.9	509.	3.24E-01
46	Co58	810.9	5254.	3.67E+00
47	Kr88	835.0	578.	4.24E-01
48	Mn56	846.6	374.	2.80E-01
54	Cs138	1009.6	296.	2.94E-01
59	Co60	1173.4	2159.	2.73E+00
62	Fe59	1292.0	2324.	3.43E+00
63	Co60	1332.6	2282.	3.54E+00
66	Na24	1369.0	1110.	1.80E+00
68	Cs138	1435.8	775.	1.36E+00
73	Na24	1732.0	401.	9.57E-01
74	Xe138	1767.6	193.	4.77E-01
78	Rb88*	1836.0	122.	3.22E-01
80	Xe138	2015.4	68.	2.11E-01
82	Kr88*	2195.7	114.	4.13E-01
83	Cs138	2217.4	153.	5.65E-01
85	Na24	2242.7	177.	6.67E-01
86	Kr88	2392.2	187.	7.96E-01
90	Cs138	2638.7	60.	3.10E-01
92	Na24	2754.3	823.	4.63E+00

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
2.19E+01	5.53E+00	2.51E+00	2.99E+01

****

****

PLANT: UNIT 8

LOCATION: BYPASS ION EXCHANGER VALVE ROOM

FILE NAME: GSM 5

COUNTING TIME (SEC): 100

GEOMETRIC CORRECTION FACTOR: 42.

****

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
4	Cs134M	128.3	3409.	1.64E-01
5	Xe135	249.8	1034.	1.27E-01
20	ANNIH	511.0	2607.	1.26E+00
48	Mn56	846.8	291.	3.27E-01
48	I134	884.0	228.	2.75E-01
54	Cs138	1009.8	336.	5.01E-01
59	Co60	1173.4	251.	4.76E-01
60	Rb89	1248.5	202.	4.23E-01
63	Co60	1332.8	143.	3.33E-01
66	Na24	1369.0	369.	8.96E-01
68	Cs138	1435.9	1372.	3.60E+00
70	Kr88	1529.5	382.	1.11E+00
74	Xe138	1767.8	259.	9.60E-01
78	Rb88*	1836.0	578.	2.29E+00
80	Xe138	2015.4	130.	6.05E-01
81	I137	2028.7	126.	5.94E-01
82	Kr88*	2195.7	370.	2.01E+00
83	Cs138	2217.4	289.	1.60E+00
86	Kr88	2392.2	735.	4.72E+00
87	Kr87	2554.7	156.	1.15E+00
90	Cs138	2638.7	148.	1.15E+00
91	Rb88	2678.7	29.	2.31E-01
92	Na24	2754.3	133.	1.12E+00

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
3.16E+00	2.15E+01	1.26E+00	2.59E+01

****

PLANT: UNIT 8

LOCATION: OFF-GAS LINE FROM PRESSURIZER IN VOLUME CONTROL TANK VALVE ROOM

FILE NAME: GSM 7

COUNTING TIME (SEC): 100.

GEOMETRIC CORRECTION FACTOR: 80.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.8	10403.	2.87E-01
4	Cs134M	128.3	25867.	6.53E-01
8	Xe135	249.8	2817.	1.81E-01
20	ANNIH	511.0	4598.	1.17E+00
31	Xe135*	607.5	513.	1.75E-01
47	Kr88	835.0	877.	5.06E-01
52	Rb88*	898.0	783.	5.12E-01
58	Co60	1173.4	272.	2.71E-01
62	Fe59	1292.0	328.	3.81E-01
66	Na24	1369.0	1060.	1.35E+00
70	Kr88	1529.5	1103.	1.81E+00
78	Rb88*	1836.0	1535.	3.19E+00
79	Na24	1879.9	430.	9.30E-01
81	I137	2028.7	70.	1.73E-01
82	Kr88*	2195.7	1261.	3.60E+00
84	Kr88	2230.6	366.	1.08E+00
86	Kr88	2392.2	3263.	1.09E+01
87	Kr87	2554.7	287.	1.03E+00
88	Kr87	2557.6	106.	4.05E-01
91	Rb88	2678.7	79.	3.31E-01

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
2.93E+00	2.49E+01	1.17E+00	2.90E+01

PLANT: UNIT A

LOCATION: BORIC ACID TANK PRESSURE LEVEL CONTROL ROOM

FILE NAME: GSM 3

COUNTING TIME (SEC): 600.

GEOMETRIC CORRECTION FACTOR: 980.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.8	14906.	5.59E-03
8	Xe135	249.8	2716.	2.38E-03
9	Cr51	318.8	6473.	9.48E-03
12	Sn113	391.9	1585.	3.41E-03
16	Cs134	475.4	2565.	7.71E-03
20	ANNIH	511.0	17503.	6.07E-02
29	Sb124	602.8	1141.	5.24E-03
39	Ru105	724.2	4471.	2.78E-02
40	Zr95	756.8	5260.	3.52E-02
41	Nb95	765.9	12861.	8.76E-02
46	Co58	810.9	47667.	3.57E-01
47	Kr88	835.0	6562.	5.16E-02
56	Fe59	1099.2	533.	6.51E-03
57	Zn65	1115.8	1262.	1.58E-02
59	Co60	1173.4	34698.	4.70E-01
62	Fe59	1292.0	336.	5.31E-03
63	Co60	1332.6	32359.	5.37E-01
64	Xe142	1338.0	152.	2.54E-03
69	K40	1461.5	141.	2.72E-03
71	Co58	1675.3	163.	3.94E-03
72	Sb124	1691.5	225.	5.53E-03

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
1.44E+00	1.96E-01	6.34E-02	1.70E+00

PLANT: UNIT 6

LOCATION: BORIC ACID TANK PRESSURE LEVEL CONTROL ROOM

FILE NAME: GSM 6

COUNTING TIME (SEC): 200.

GEOMETRIC CORRECTION FACTOR: 24.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
20	ANNIH	511.0	1701.	7.22E-01
38	Cs137	661.7	305.	2.01E-01
44	Cs134	795.9	267.	2.38E-01
46	Co56	810.8	366.	3.36E-01
58	Co60	1173.4	352.	5.84E-01
63	Co60	1332.6	317.	6.45E-01
66	Na24	1369.0	186.	3.85E-01
68	Cs138	1435.8	673.	1.55E+00
74	Xe138	1767.5	218.	7.07E-01
78	Rb88*	1836.0	273.	9.45E-01
82	Kr88*	2195.7	148.	7.04E-01
83	Cs138	2217.4	128.	6.20E-01
86	Kr88	2392.2	370.	2.07E+00
90	Cs138	2638.7	71.	4.81E-01
92	Na24	2754.3	72.	5.32E-01

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
2.49E+00	7.51E+00	7.22E-01	1.07E+01

PLANT: UNIT A

LOCATION: CHARGING PUMP ROOM

FILE NAME: GSM 10

COUNTING TIME (SEC): 300.

GEOMETRIC CORRECTION FACTOR: 1000.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
2	I133*	80.8	3654.	2.69E-03
8	Xe135	249.8	6828.	1.17E-02
20	ANNIH	511.0	392.	2.86E-03
30	Cs134	604.8	289.	2.62E-03
35	Cs137	661.7	1241.	1.31E-02
40	Zr95	756.8	120.	1.58E-03
41	Nb95	765.8	217.	2.90E-03
44	Cs134	795.9	110.	1.57E-03
46	Co58	810.9	920.	1.35E-02
47	Kr88	835.0	385.	5.93E-03
59	Co60	1173.4	1073.	2.35E-02
62	Fe59	1292.0	62.	1.92E-03
63	Co60	1332.6	1077.	3.51E-02
66	Na24	1369.0	52.	1.77E-03
69	K40	1461.5	113.	4.27E-03
76	Rb88*	1836.0	11.	6.09E-04
81	I137	2028.7	10.	6.59E-04
82	Na24	2754.3	29.	3.43E-03

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
8.58E-02	4.17E-02	6.94E-03	1.34E-01

PLANT: UNIT B

LOCATION: CHARGING PUMP ROOM

FILE NAME: GSM 11

COUNTING TIME (SEC): 300.

GEOMETRIC CORRECTION FACTOR: 400.

*** DATA ANALYSIS ***

ID	RADIONUCLIDE	KEV	TOTAL COUNTS	DOSE RATE (mR/hr)
1	I131	79.6	35394.	6.61E-02
6	Kr88	194.2	6529.	1.74E-02
8	Xe135	249.8	123182.	5.26E-01
13	Kr87	402.4	2277.	2.52E-02
15	Cs138	462.6	2463.	3.47E-02
18	UNKWN	503.0	459.	7.58E-03
20	ANNIH	511.0	2314.	3.83E-02
21	I132	522.6	3161.	5.58E-02
27	Xe142	598.2	758.	1.66E-02
28	Sb125	600.0	2920.	6.52E-02
30	Cs134	604.8	1536.	3.48E-02
33	Sr91	653.5	1477.	3.80E-02
34	Xe142	656.7	8101.	2.10E-01
43	Xe142	789.9	1512.	5.32E-02
47	Kr88	835.0	923.	3.57E-02
50	UNKWN	887.4	1914.	8.13E-02
51	Xe142	891.6	9579	4.10E-01
53	UNKWN	1002.3	903.	4.66E-02
67	I137	1426.9	1193.	1.08E-01
75	Mn56	1811.7	817.	1.11E-01
76	Cs134	1818.1	2083.	2.84E-01
77	Cs134	1824.1	6242.	8.54E-01

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	NATURAL BKGD /UNKWN/ANNIH	TOTAL DOSE RATE
1.11E-01	2.84E+00	1.75E-01	3.12E+00

APPENDIX E

PLANT PARAMETERS AND REACTOR COOLANT DATA

Presented in this appendix are plant parameter and reactor coolant data. These data have been used to quantify the impact of plant age and reactor power level on coolant activity so that the measurement data between the two units could be normalized. This has facilitated better comparison of the baseline and elevated failed fuel levels. The remainder of this appendix presents the plant parameter and reactor coolant data graphed as a function of time. Rather than present the data by month, day and year, each calendar day has been assigned a number. Measurements in support of this project were taken on day 775.

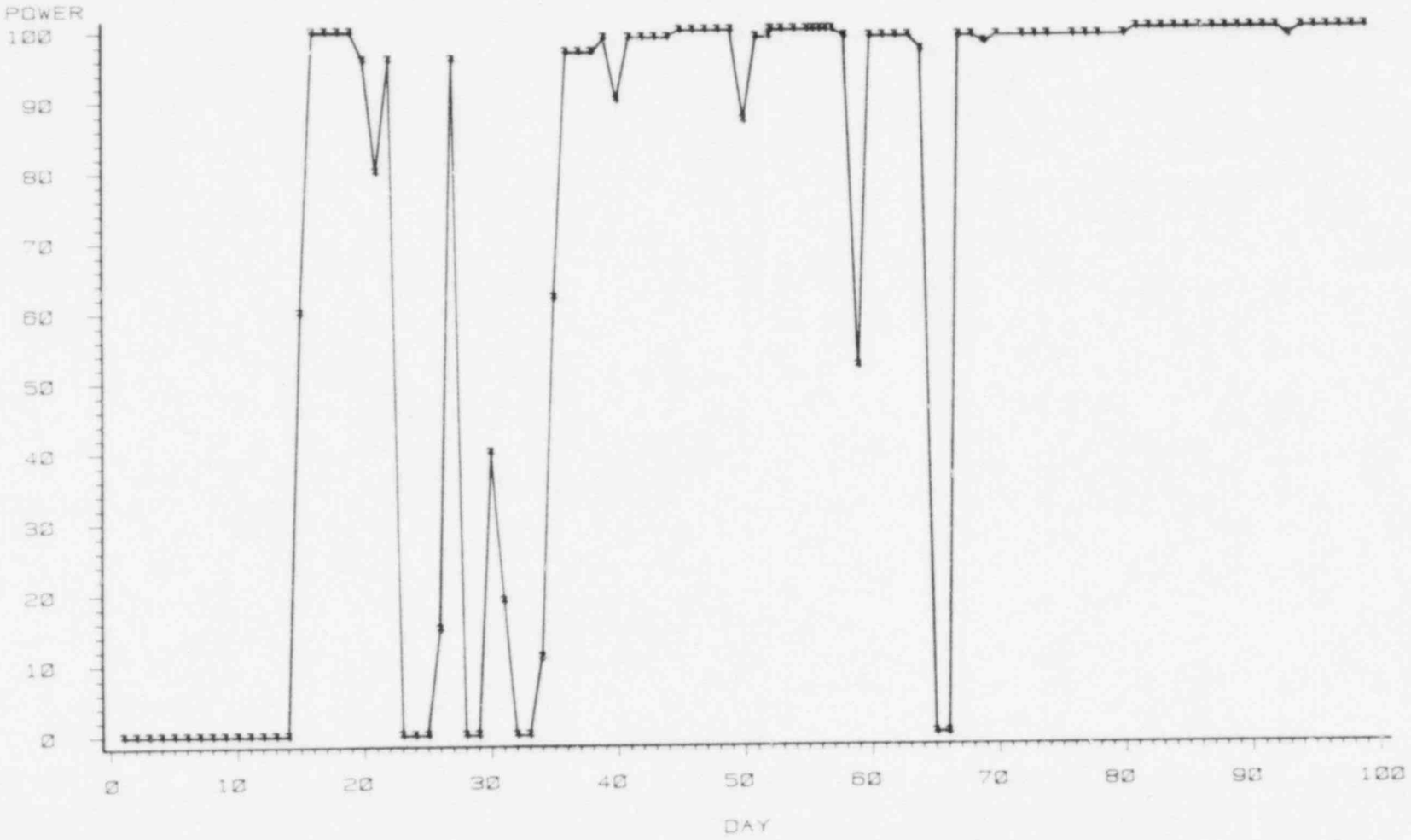

FIGURE E.1. Reactor Power Level (Day 0-100), Unit A

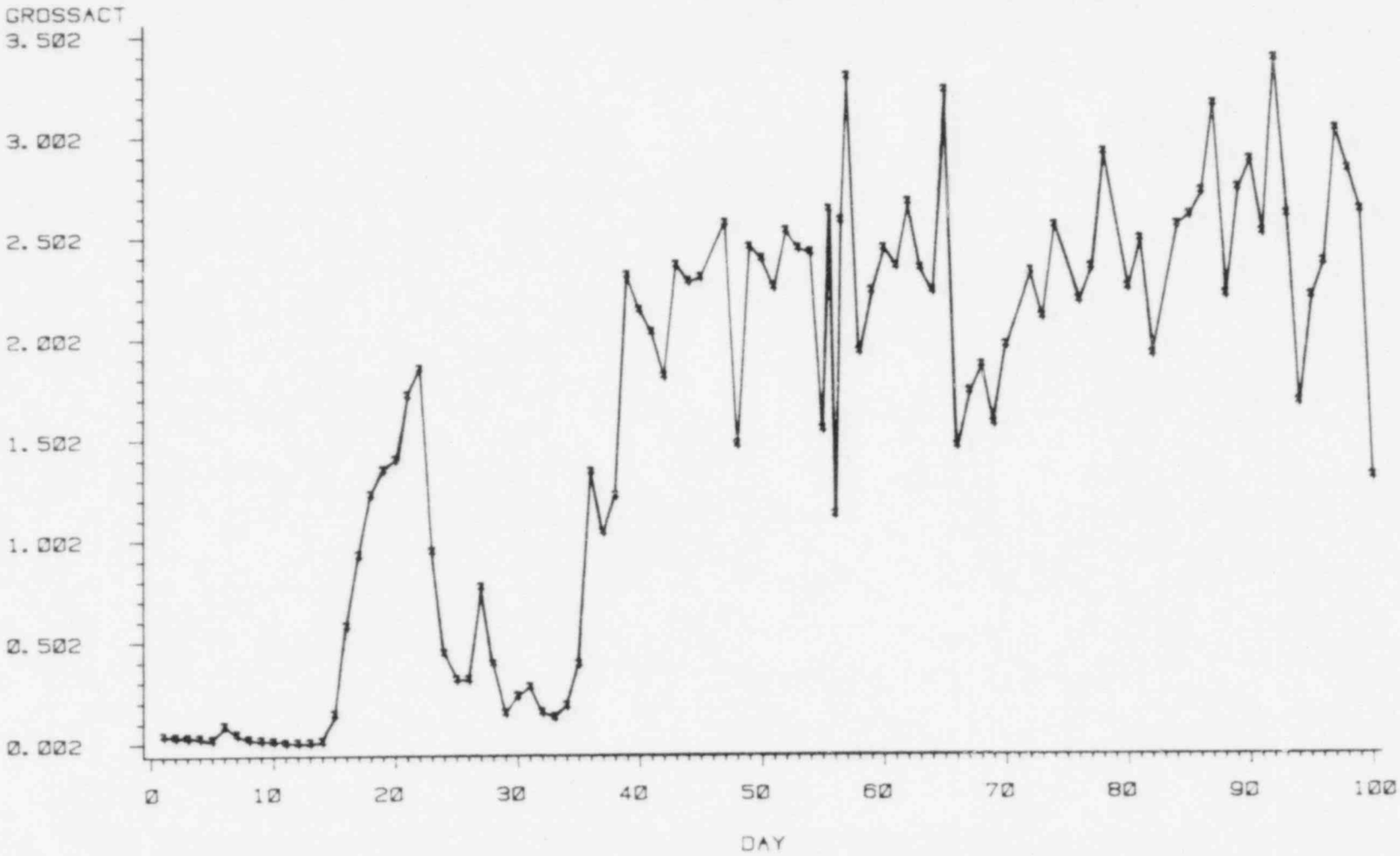

FIGURE E.2. Gross Activity (Day 0-100), Unit A

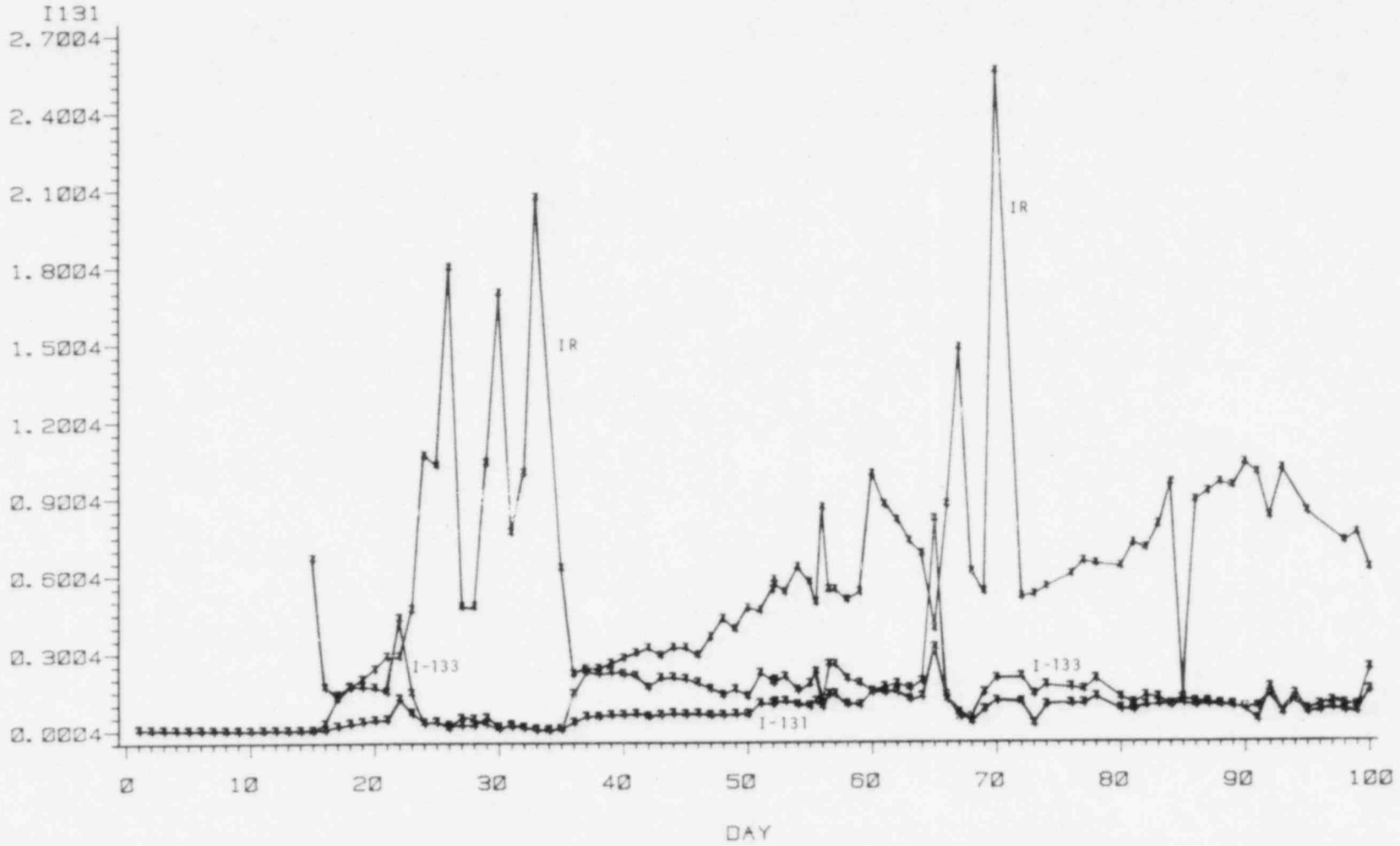

FIGURE E.3. ^{131}I, ^{133}I and Iodine Ratio (IR) (Day 0-100), Unit A

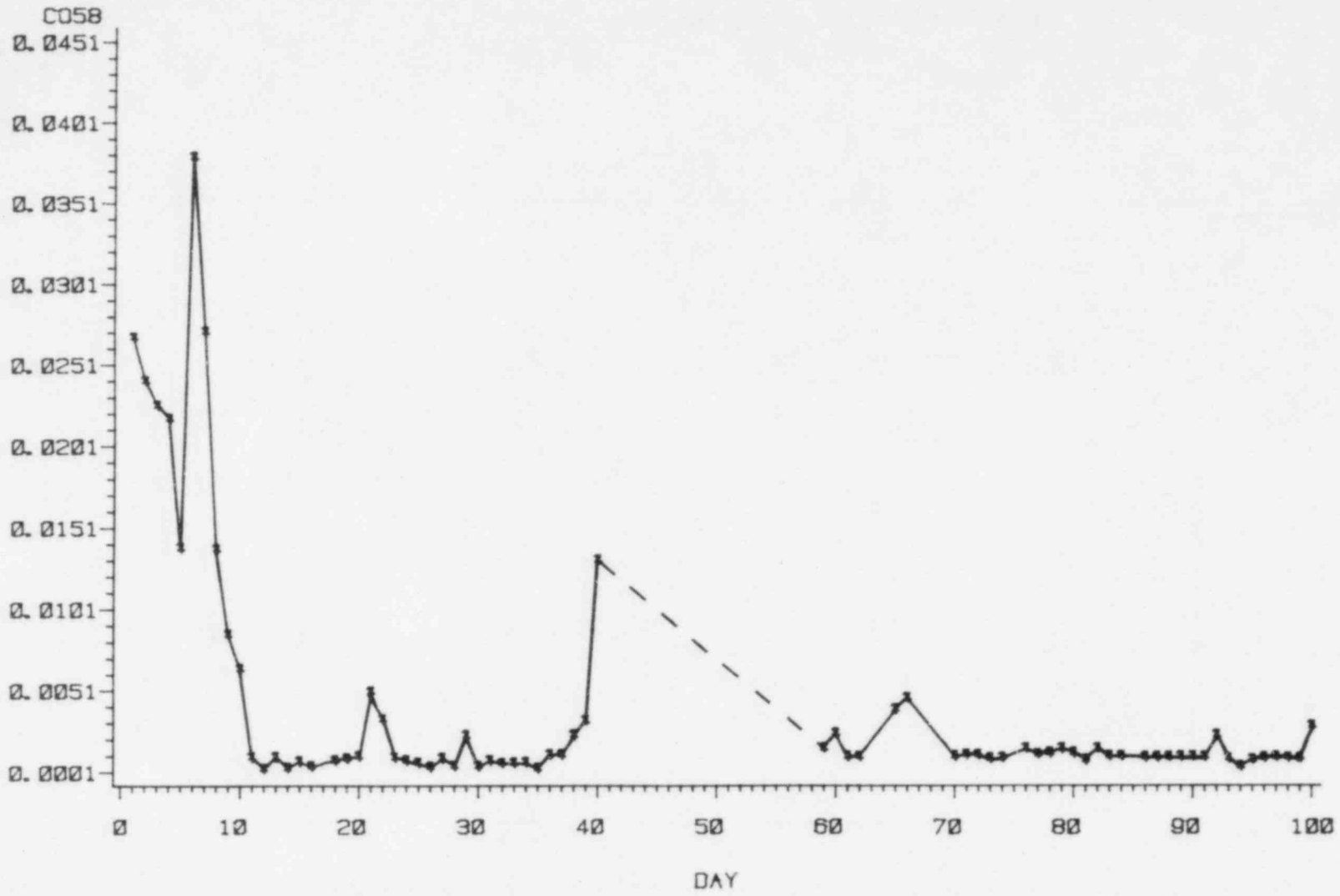

FIGURE E.4. ^{58}Co (Day 0-100), Unit A

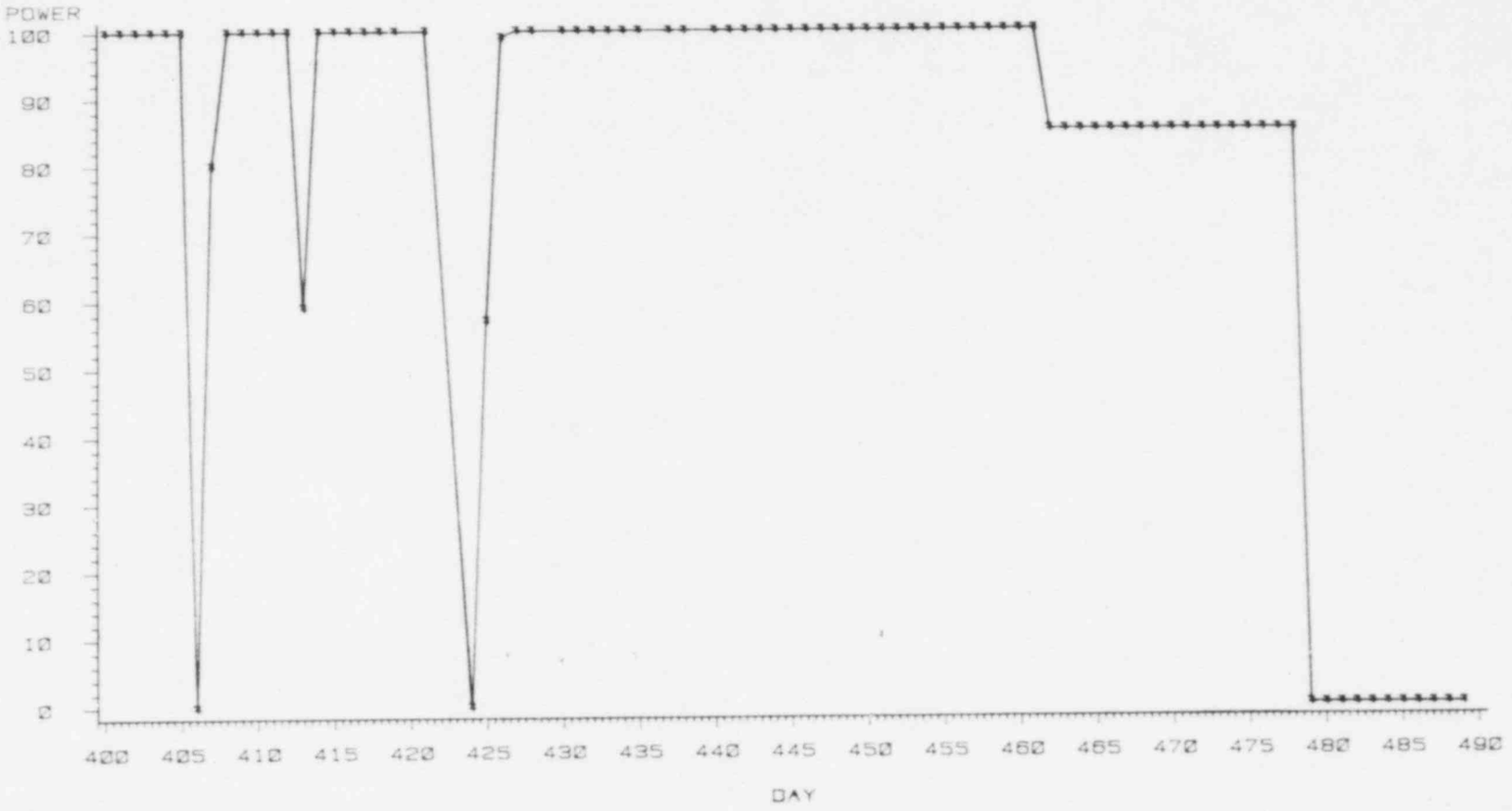

FIGURE E.5. Reactor Power Level (Day 400-490), Unit A

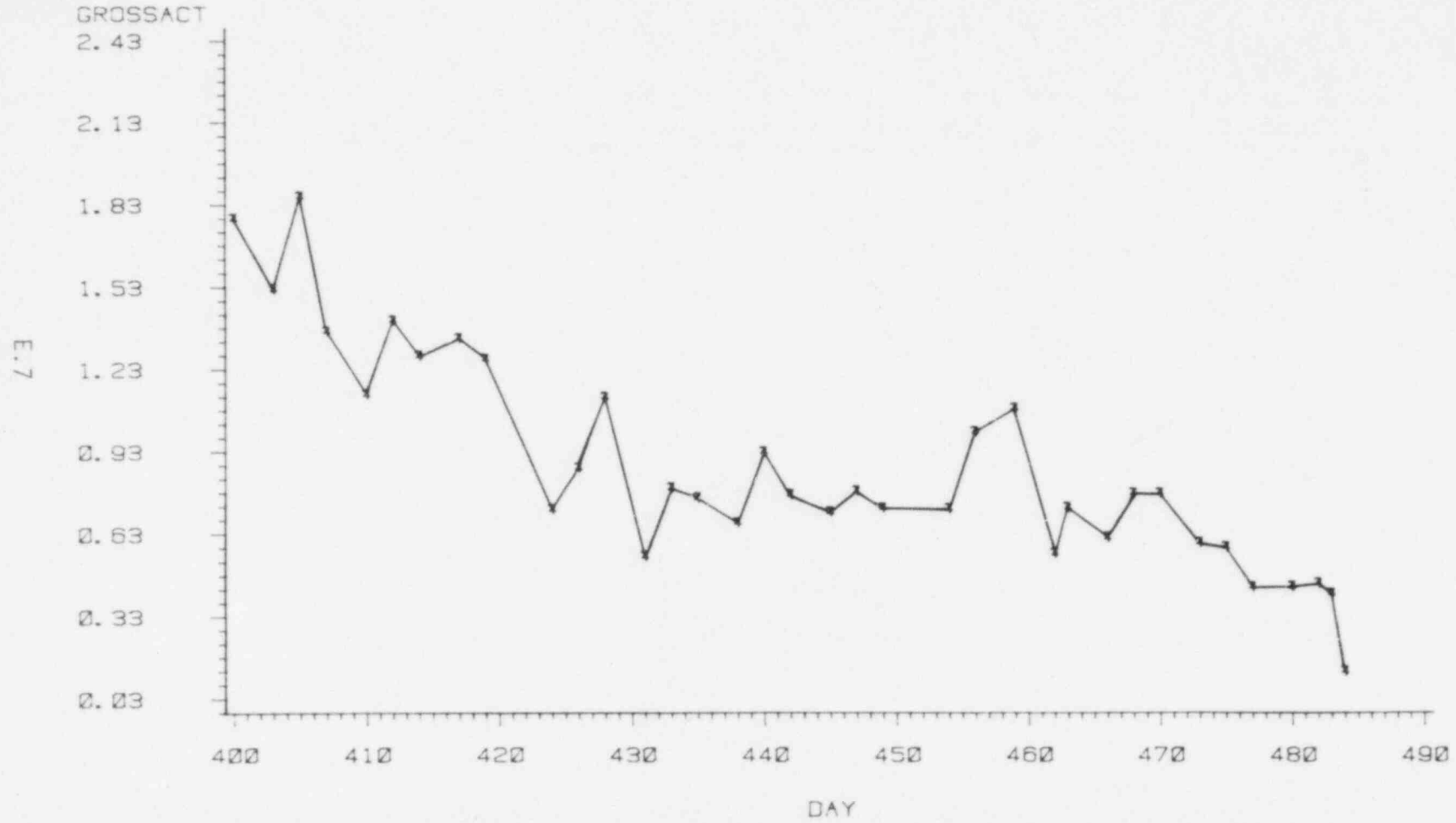

FIGURE E.6. Gross Activity (Day 400-490), Unit A

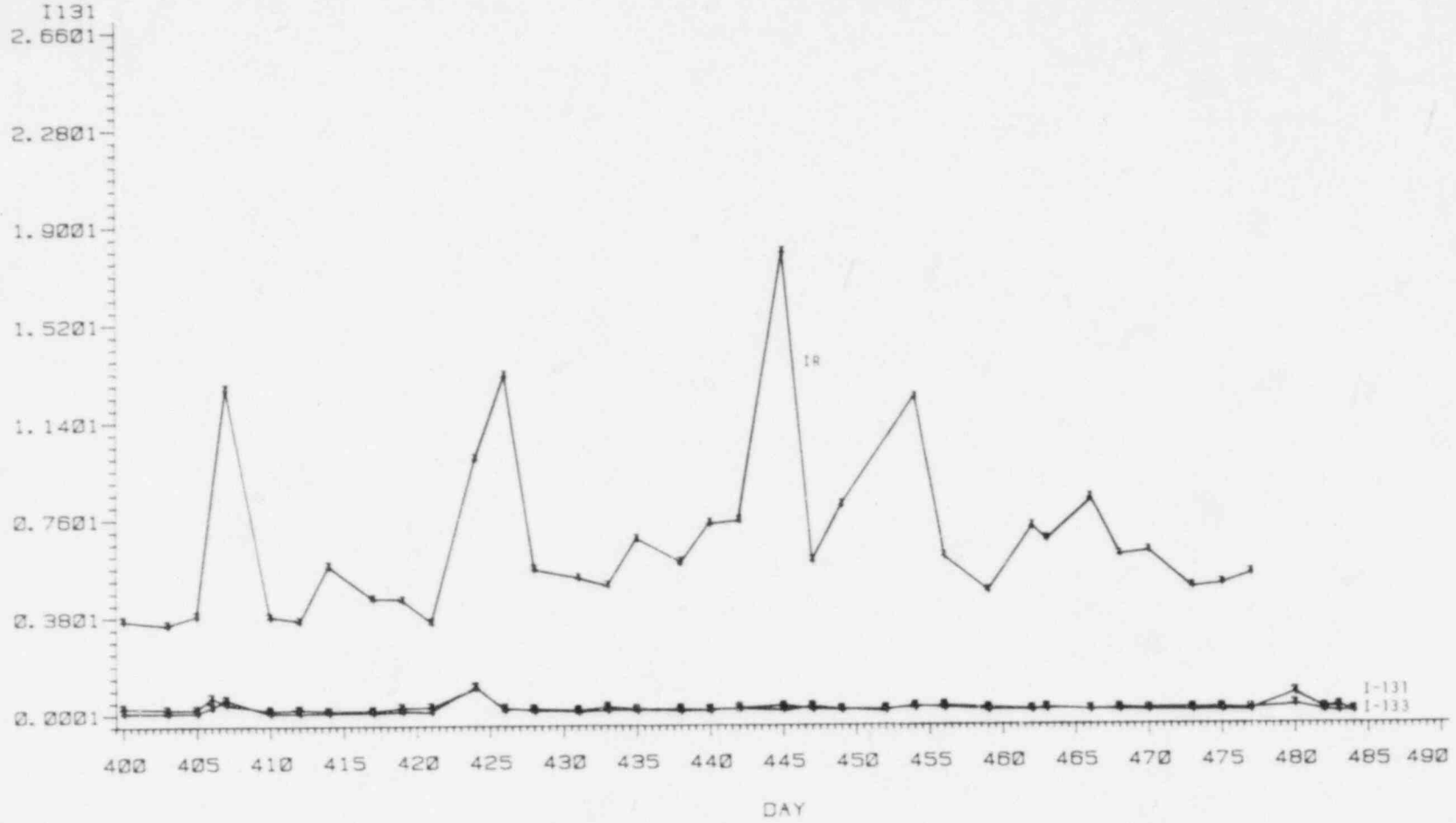

FIGURE E.7. ^{131}I, ^{133}I and Iodine Ratio (IR) (Day 400-490), Unit A

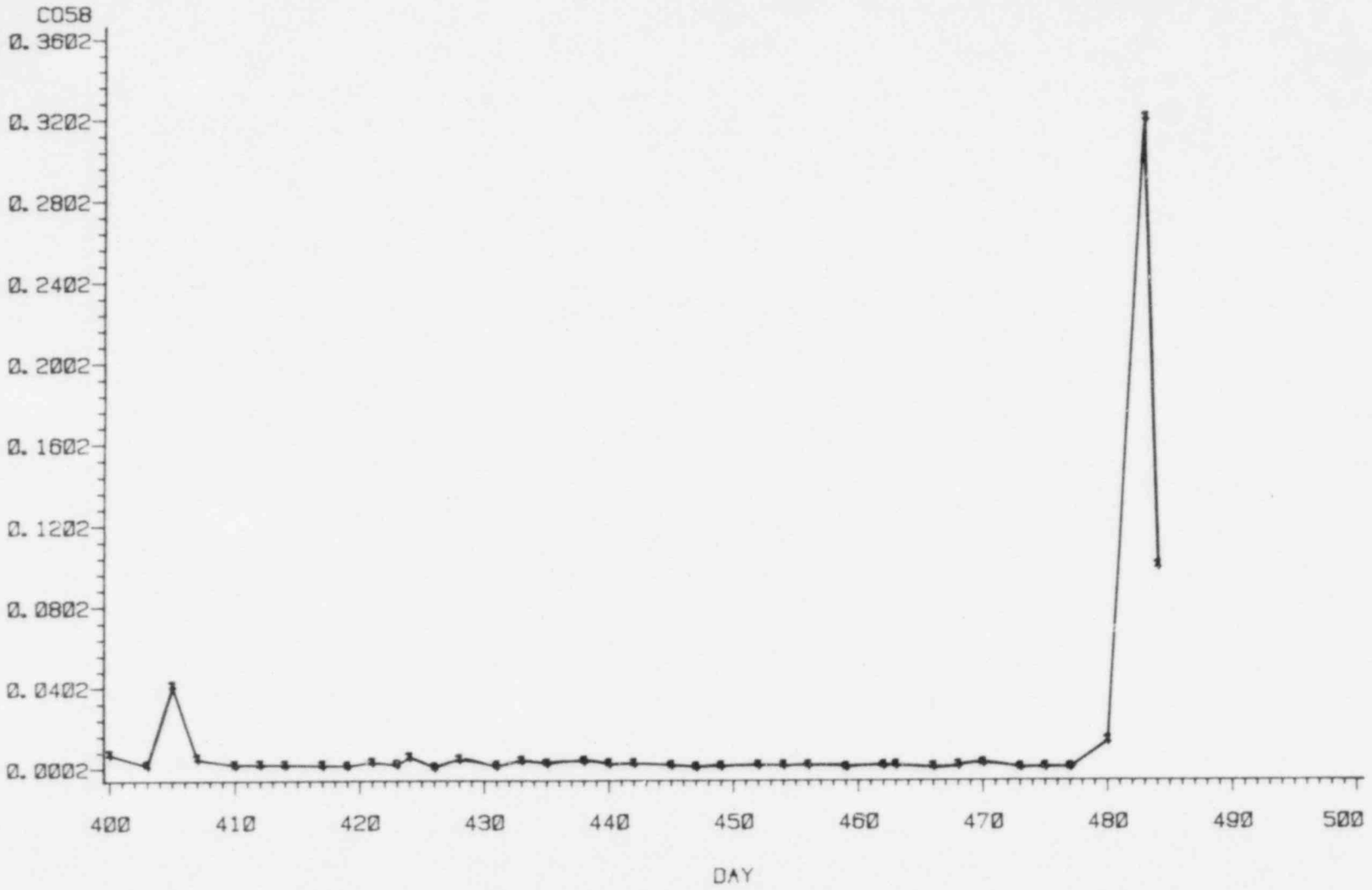

FIGURE E.8. ^{58}Co (Day 400-490), Unit A

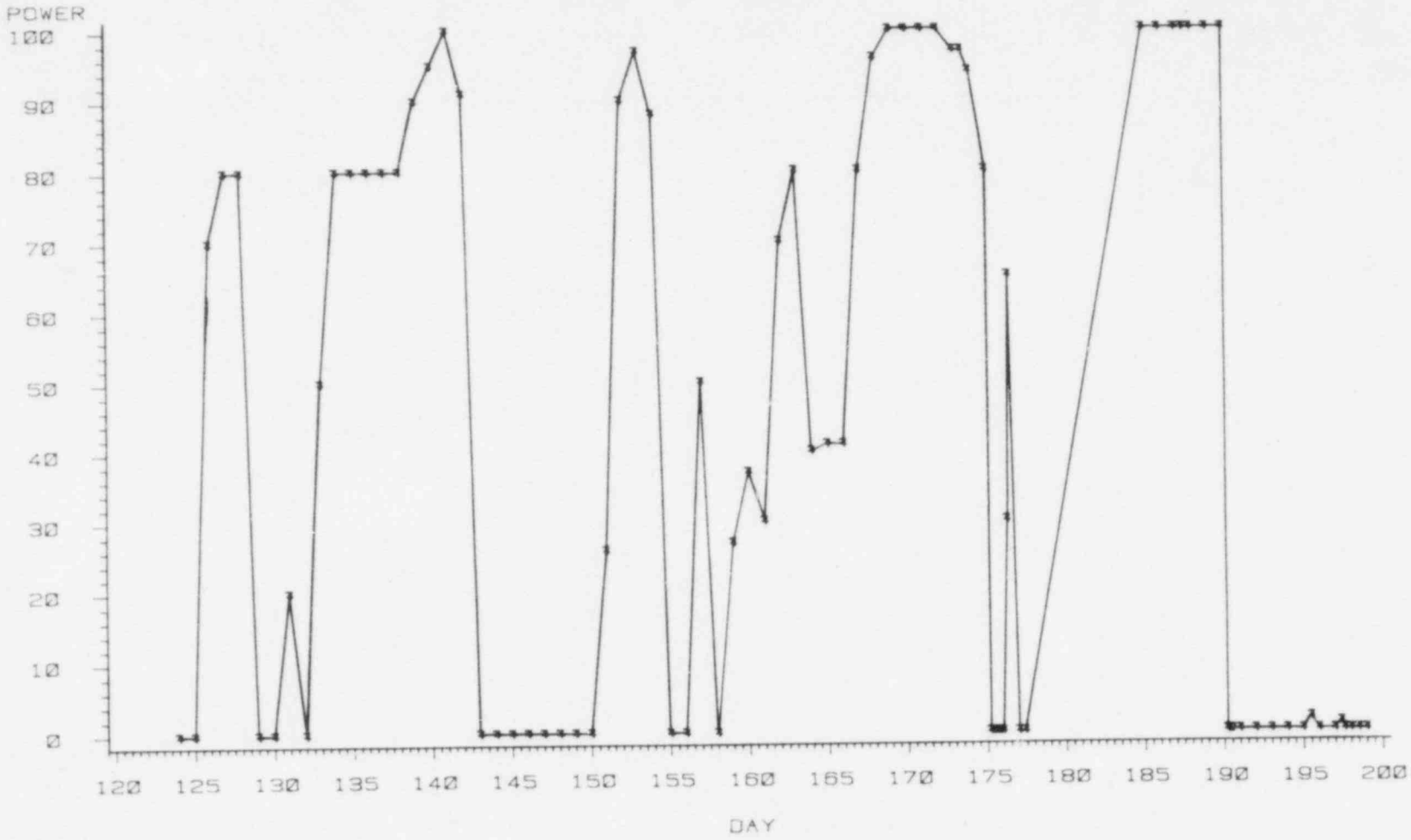

FIGURE E.9. Reactor Power Level (Day 120-200), Unit B

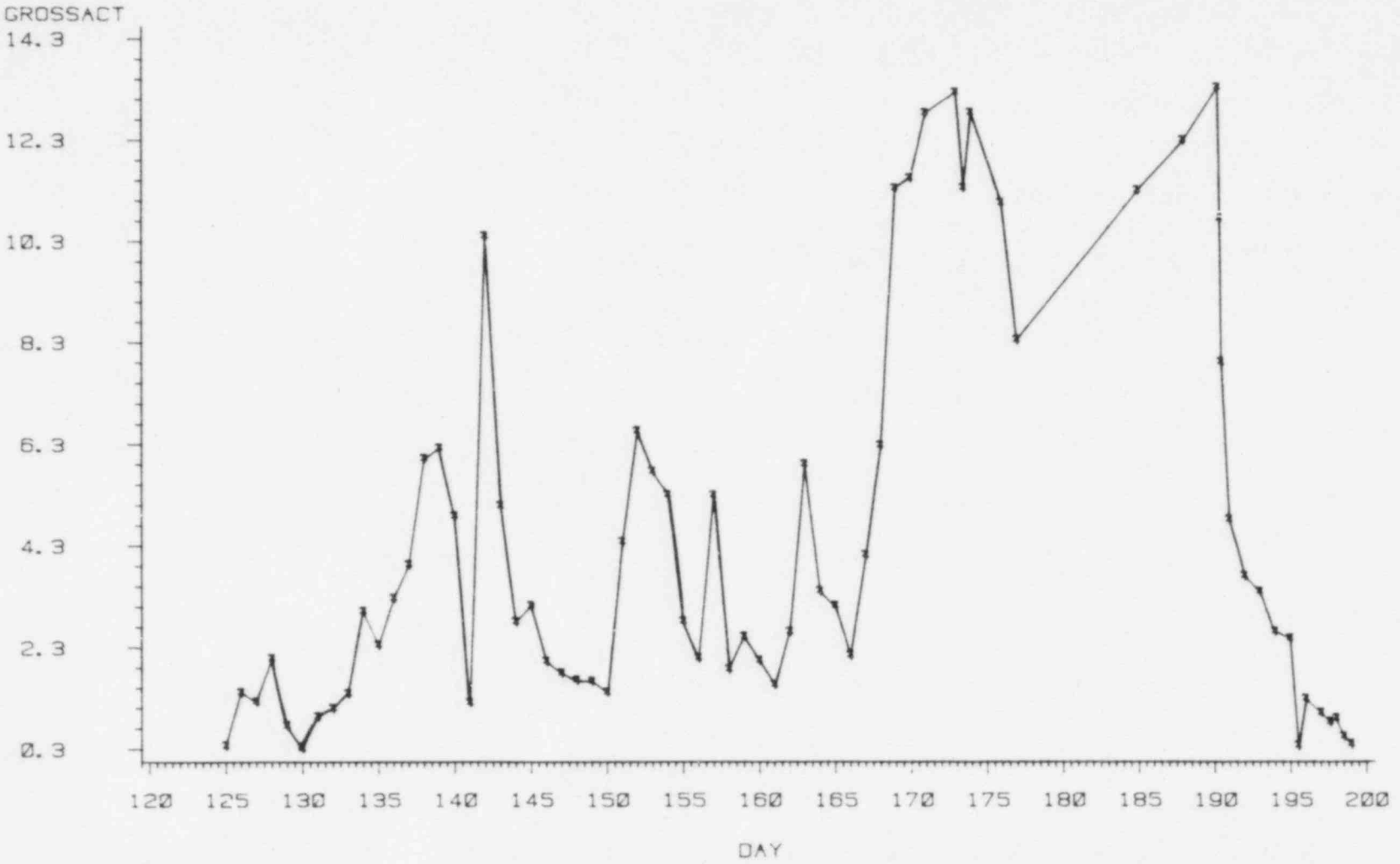

FIGURE E.10. Gross Activity (Day 120-200), Unit B

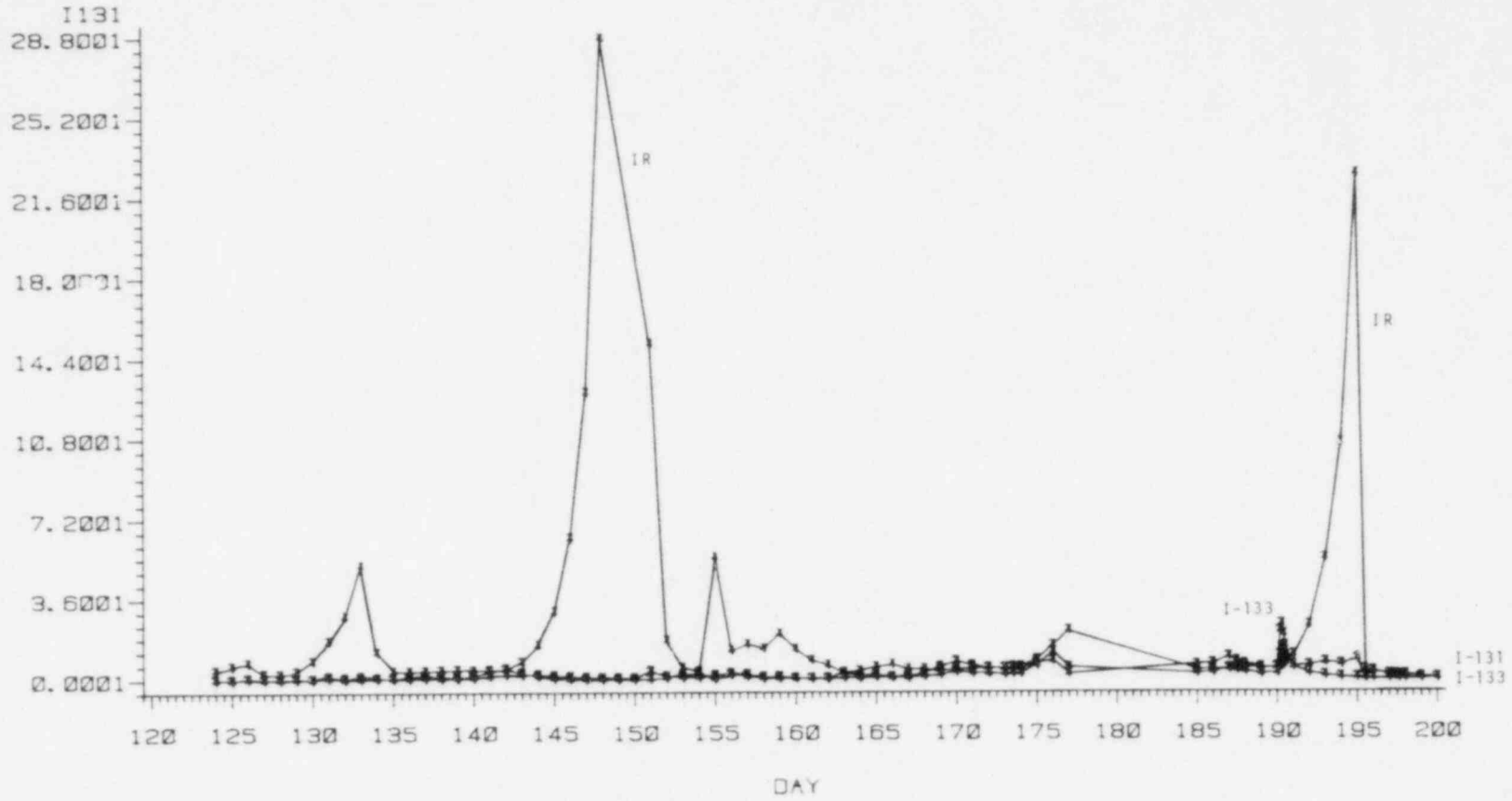

FIGURE E.11. ^{131}I, ^{133}I and Iodine Ratio (IR) (Day 120-200), Unit B

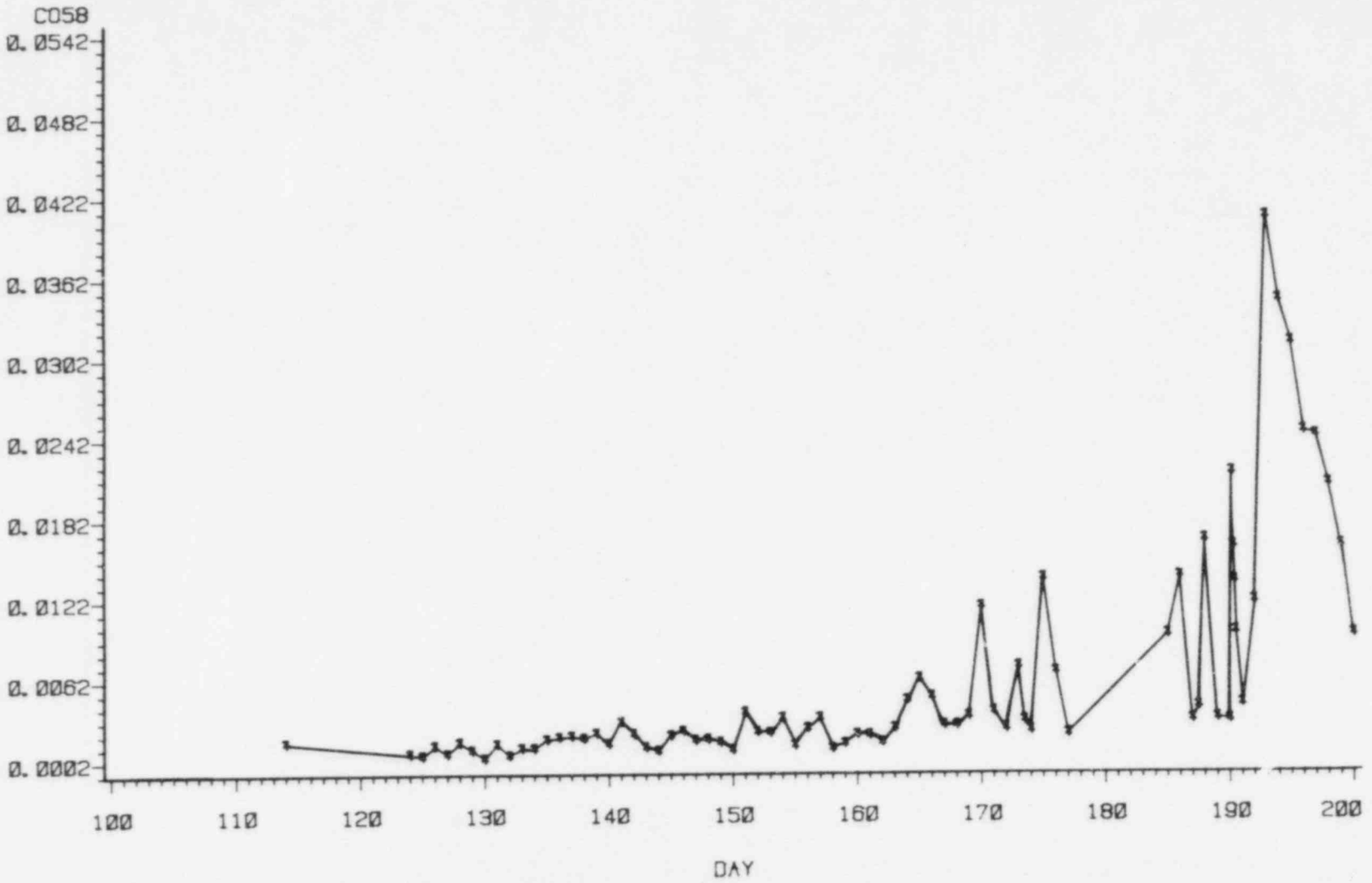

FIGURE E.12. ^{58}Co (Day 120-200), Unit B

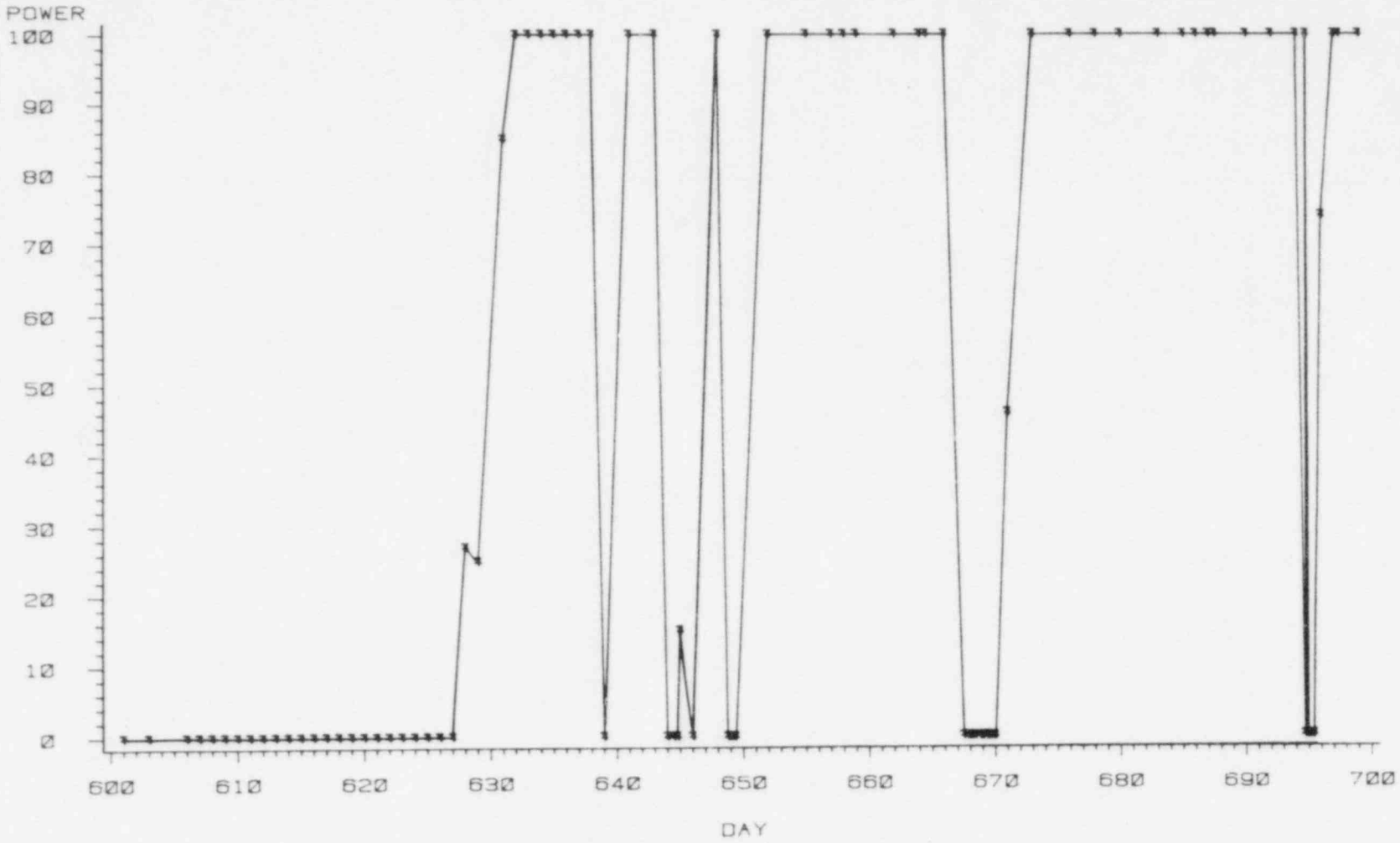

FIGURE E.13. Reactor Power Level (Day 600-700), Unit B

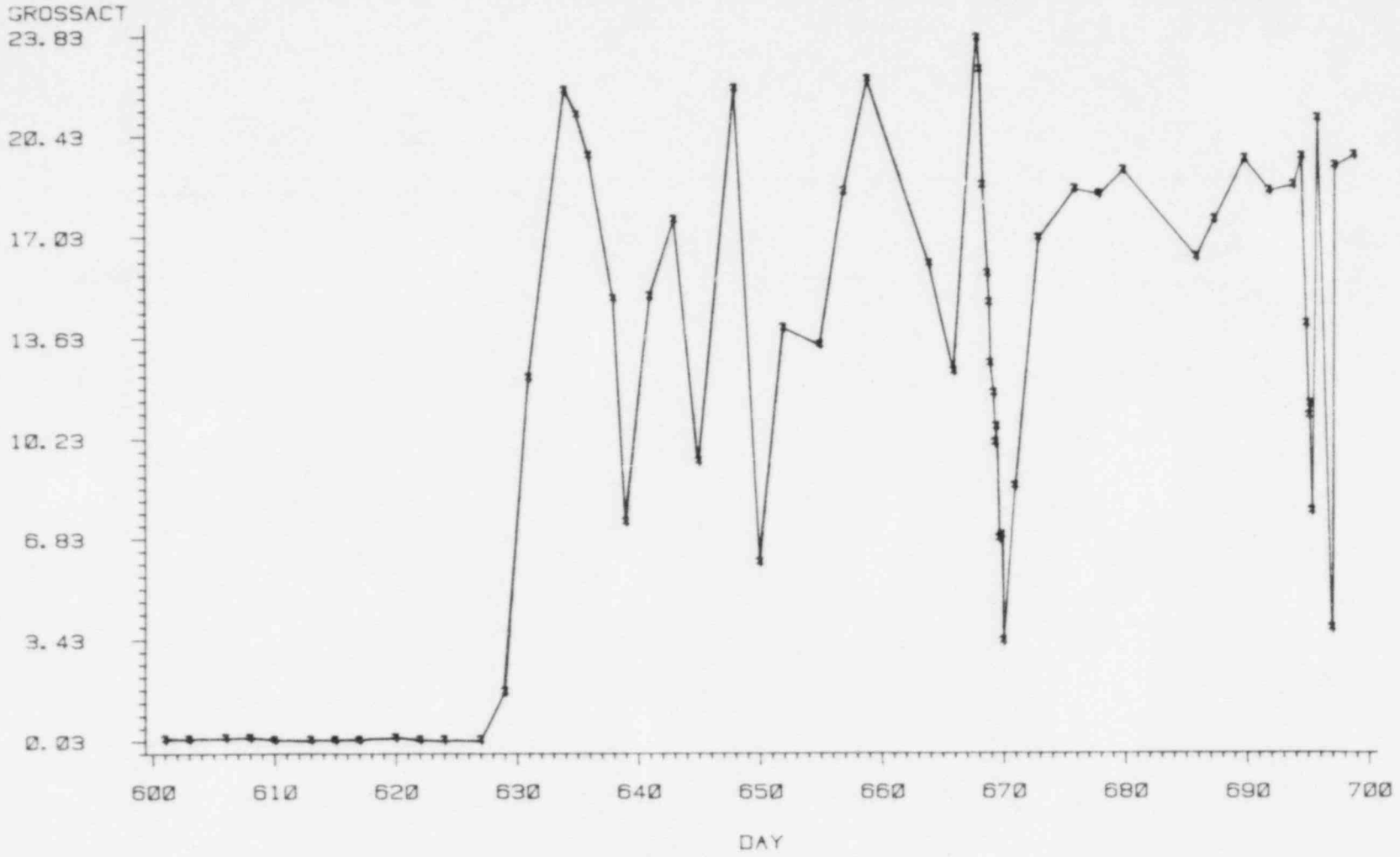

FIGURE E.14. Gross Activity (Day 600-700), Unit 5

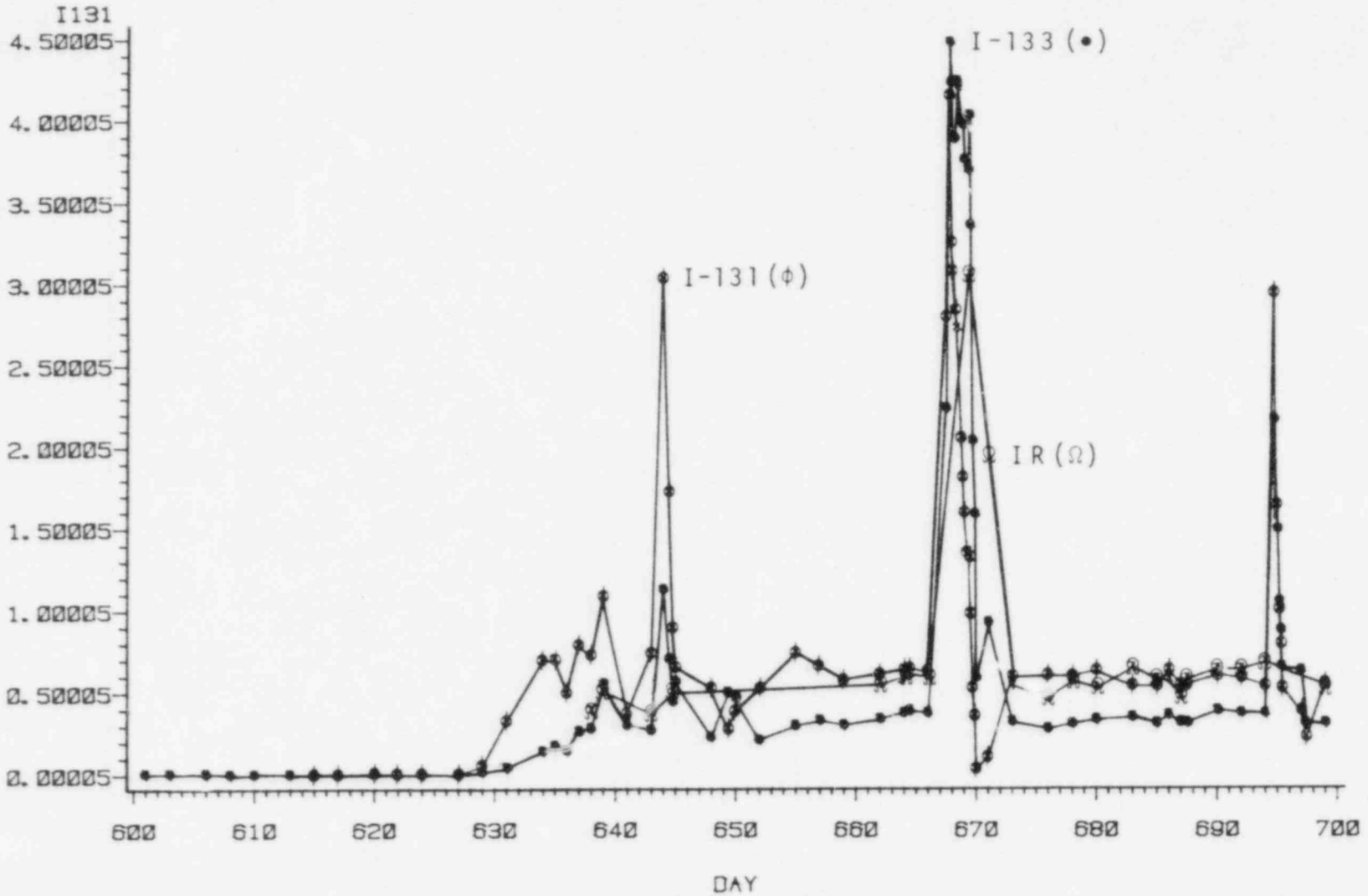

FIGURE E.15. ^{131}I, ^{133}I and Iodine Ratio (IR) (Day 600-700), Unit B

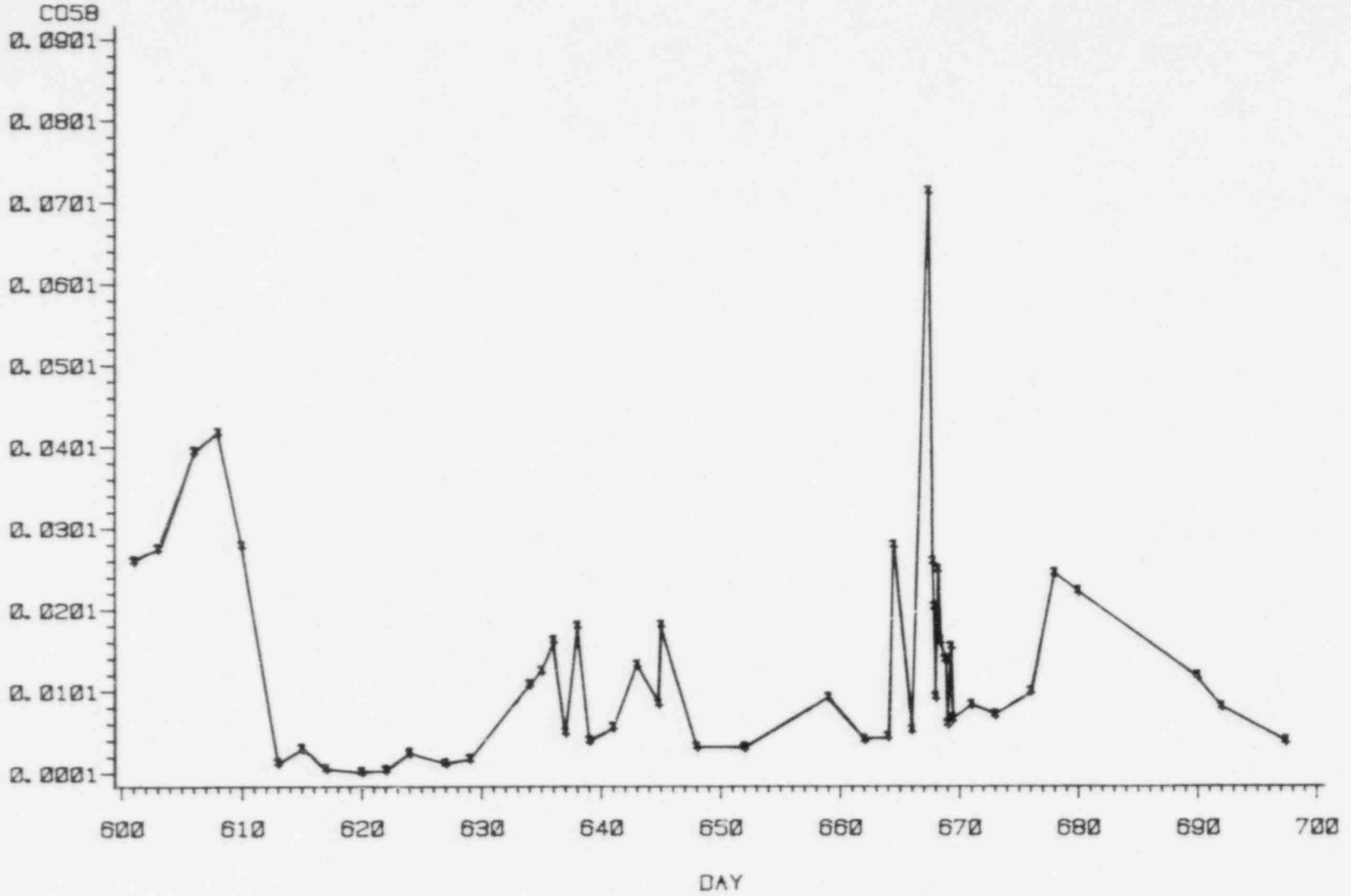

FIGURE E.16. ^{58}Co (Day 600-700), Unit B

NRC FORM 335 (2 84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC add Vol. No., if any)
NUREG/CR-4485
PNL-5606

2. TITLE AND SUBTITLE
The Impact of Fuel Cladding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants
Case Study: PWR During Routine Operations

3. LEAVE BLANK

4. DATE REPORT COMPLETED
MONTH: November
YEAR: 1985

5. AUTHOR(S)
M. P. Moeller, G. F. Martin and D. L. Haggard

6. DATE REPORT ISSUED
MONTH: January
YEAR: 1986

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Pacific Northwest Laboratory
P. O. Box 999
Richland, WA 99352

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
B2987

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Safety Review and Oversight
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

11a. TYPE OF REPORT
Technical

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This report presents data in support of evaluating the impact of fuel cladding failure events on occupational radiation exposure. To determine quantitatively whether fuel cladding failure contributes significantly to occupational radiation exposure, radiation exposure measurements were taken at comparable locations in two mirror-image pressurized-water reactors (PWRs) and their common auxiliary building. One reactor, Unit B, was experiencing degraded fuel characterized as 0.125 percent fuel pin-hole leakers and was operating at approximately 55 percent of the reactor's licensed maximum core power, while the other reactor, Unit A, was operating under normal conditions with less than 0.01 percent fuel pin-hole leakers at 100 percent of the reactor's licensed maximum core power. Measurements consisted of gamma spectral analyses, radiation exposure rates and airborne radionuclide concentrations. In addition, data from primary coolant sample results for the previous 20 months on both reactor coolant systems were analyzed. The results of the measurements and coolant sample analyses suggest that a 3560-megawatt-thermal (1100 MWe) PWR operating at full power with 0.125 percent failed fuel can experience an increase of 540 percent in radiation exposure rates as compared to a PWR operating with normal fuel. In specific plant areas, the degraded fuel may elevate radiation levels even more.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
Fuel element failure - Accidents
Radiation Hazards
Reactor Safety

b. IDENTIFIERS/OPEN ENDED TERMS
Barrier Degradation Events
Occupational Radiation Exposure

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

NUREG/CR-4405

THE IMPACT OF FOSSIL-FUEL BURNING PLANT EVENTS ON OCCUPATIONAL RADIATION EXPOSURES AT NUCLEAR POWER PLANTS

JANUARY 1986