NUREG/CR-3544
PNL-4886

Beta Particle Measurement and Dosimetry Requirements at NRC-Licensed Facilities

Prepared by L. A. Rathbun, G. W. R. Endres, R. A. Fox,
P. L. Roberson, R. I. Scherpelz

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

GPO Printed copy price: $4.25

NUREG/CR-3544
PNL-4886

Beta Particle Measurement and Dosimetry Requirements at NRC-Licensed Facilities

Manuscript Completed: February 1984
Date Published: August 1984

Prepared by
L. A. Rathbun, G. W. R. Endres, R. A. Fox,
P. L. Roberson, R. I. Scherpelz

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Facility Operations
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN B2448

ABSTRACT

Researchers from Pacific Northwest Laboratory (PNL) have conducted beta radiation measurements under laboratory and field conditions to assess the degree of the measurement problem and offer suggestions for possible remedies. The primary measurement systems selected for use in this study were the silicon (Si) surface barrier spectrometer system and the multielement beta dosimeter. Three boiling water reactors (BWRs), two pressurized water reactors (PWRs), and one fuel fabrication facility were visited during the course of the study. Although beta fields from cobalt-60 were the most common type found at commercial reactor facilities, higher energy beta fields were found at locations associated with spent fuel handling, liquid radioactive waste, and BWR turbine components. Commercially-available dosimeters and survey instruments were used to measure the same laboratory and licensee facility beta fields characterized with PNL's active and passive spectrometers. A prototype spectrometer was also used in the laboratory measurements. The commercial instruments and dosimeters used in this study typically responded low to the beta fields measured, especially where maximum beta energies were less than approximately 500 keV. A single calibration factor is usually not adequate for either beta dosimeters or instruments. There is a need for more refinement in beta measurement devices and training for the users of such devices.

CONTENT

ABSTRACT iii

1.0 INTRODUCTION 1

2.0 METHODOLOGY 2

2.1 MEASUREMENT SYSTEMS 2
2.1.1 Silicon Surface Barrier Spectrometer System . . 2
2.1.2 Multielement Beta Dosimeter 3

2.2 SURVEY INSTRUMENTS AND DOSIMETERS 8
2.2.1 Commercial Survey Instruments 8
2.2.2 Prototype Beta-Gamma Dose Rate Meter 11
2.2.3 Commercial Dosimeters 11

2.3 SITE SELECTION AND DESCRIPTION 12

3.0 RESULTS AND DISCUSSION 15

3.1 FIELD SPECTRA 15

3.2 MULTIELEMENT BETA DOSIMETER 36

3.3 SURVEY METER RESULTS 36

3.4 COMMERCIAL DOSIMETERS 40

4.0 CONCLUSIONS 42

4.1 BETA EXPOSURE AREAS 42

4.2 SURVEY METERS 42

4.3 DOSIMETERS 43

5.0 RECOMMENDATIONS 44

5.1 GOOD PRACTICES 44

5.2 SURVEY METER SELECTION AND USE 44

6.0 REFERENCES 46

FIGURES

1	Silicon Surface Barrier Spectrometer System Components	3
2	Silicon Detector Response to Laboratory Beta Sources	4
3	PNL Eight-Element Dosimeter	6
4	Site Q, Pump Room B, Measurement Location	15
5	Pump Room B Spectrum	16
6	Site Q, Fuel Pool Area, Measurement Location	17
7	Site Q Fuel Pool Chute Beta-Gamma Spectrum	18
8	Site Q Fuel Pool Chute Gamma Spectrum	18
9	Underwater Light Spectrum	19
10	Smear from Control Rod Drive Shaft (Beta + Gamma), Site P	20
11	Smear from Control Rode Drive Shaft (Beta Spectrum Only), Site P	20
12	Spire Bolt (Beta + Gamma) Spectrum	22
13	Spire Bolt (Beta) Spectrum	22
14	Site P, Spare Main Coolant Pump Location	23
15	Main Coolant Pump Spectrum (after subtraction)	23
16	Site R, Uranium Full Spectrum	24
17	Site R, Uranium Spectrum (with 420 mg/cm^2 shield)	25
18	Site R, Uranium Spectrum (with 1448 mg/cm^2 shield)	26
19	Site M, Turbine Room, Shutdown BWR	27
20	Site M Turbine Blade Diaphragm Spectrum	28
21	Beta-Gamma Spectrum from Cleanup Pipe Section, Site N	30
22	Beta-Only Spectrum from Cleanup Pipe Section, Site N	31
23	Contaminated Concrete Wall, Site O	32

24 Beta + Gamma Spectrum from Contaminated Concrete Wall, Site O 33

25 Liquid Holdup Tank Area, Site O 34

26 Site O, Liquid Holdup Tank Area Spectrum 35

TABLES

1 Multielement Dosimeter Filter Specifications 4

2 Responses of Dosimeters Exposed to Natural Uranium and $^{90}Sr/^{90}Y$ Sources 7

3 Typical Data for Exposed 7-Element Dosimeter 7

4 Beta Calibration Sources 8

5 Calibration Factors for Survey Instruments 10

6 Measurements of PNL K-Shell Photons with INEL Beta-Gamma Dose Rate Meter 12

7 Commercial Dosimeter Responses to Beta Calibration Sources . . 13

8 Participating Licensee Facilities 14

9 Beta and Gamma Doses Measured with Multielement Dosimeters . . 37

10 Response of Commercial Survey Instruments to Beta Fields at Licensee Facilities 38

11 Response of Licensee Survey Instruments to Beta Fields . . . 39

12 Response of Commercial Dosimeters to Beta Radiation at Licensee Facilities 41

1.0 INTRODUCTION

The considerable variations in beta energies, intensities, and source geometries found at Nuclear Regulatory Commission (NRC) licensee facilities make measurements of beta dose and dose rates difficult. The presence of gamma fields further complicates the measurement effort. In addition, most survey instruments and dosimeters used by licensees to measure beta radiation are actually gamma radiation measurement devices that have been modified for use in beta fields. This study was undertaken by the Pacific Northwest Laboratory (PNL) to determine the extent to which beta radiation measurements made at NRC-licensed facilities are accurate. The study also sought to identify the reasons for erroneous measurements and to describe appropriate corrective action.

Beta spectra and dose rates were measured at three BWRs, two PWRs, and one fuel fabrication facility. A silicon surface barrier detector with a depletion depth of 5 mm was used as an active spectrometer to characterize beta fields. A passive spectrometer consisting of thermoluminescent dosimeters (TLDs) covered by varying thicknesses of aluminum was used to complement the active spectrometer and to measure doses.

PNL used the beta fields characterized with spectrometers to study the response of commercial dosimeters and survey meters typical of those used by licensees. The commercial devices were also exposed to several calibrated beta sources in the laboratory. A prototype spectrometer was exposed to the same calibrated laboratory sources and the responses were noted.

Section 2 describes the methodology used to obtain the data. Detailed descriptions of the measurement systems and associated calibration techniques are presented. The licensee site selection process is documented, as are the methods used to collect the data in the field. The study findings are presented and discussed in Section 3. Conclusions based on the study findings are presented in Section 4. Section 5 contains specific recommendations stemming from the study results and conclusions.

2.0 METHODOLOGY

This section describes the instrumentation and dosimeters used to measure beta radiation at the selected licensee facilities. The measurement system calibration methods are documented. The selection, calibration, and use of commercial survey meters and dosimeters are discussed. Finally, steps undertaken in site selection and data collection are described.

2.1 MEASUREMENT SYSTEMS

The measurement systems selected for this study were the silicon surface barrier spectrometer system (Fox and Borkowski 1962) and the multielement beta dosimeter (Scherpelz et al. 1983).

2.1.1 Silicon Surface Barrier Spectrometer System

Deeply depleted silicon (Si) surface barrier detectors have very good detection efficiencies for beta particles up to approximately 3.5 MeV. The Si detector energy response is essentially flat. Although the energy resolution is not particularly good at room temperature, it is adequate for most field applications. The major limitation on the use of the silicon surface barrier detector is its high background due to Compton scattering and backscattering of electrons. Electronic noise and the backscattering phenomenon drive the lower limit of detection to about 70 keV for electrons. From a health physics perspective, this is not a problem because electrons below this energy will not penetrate the dead surface layer of human skin. In pure beta fields and where the photon and beta fields can be satisfactorily separated, the spectra obtained with the Si detector can be used to derive an approximate beta dose rate.

The silicon detector and associated components are shown in Figure 1. The 100-mm^2 circular silicon surface barrier detector is depleted to a depth of 5 mm. It was mounted, via a rear high voltage mount, on a low noise, charge-sensitive preamp with a high bias voltage capability. The detector was enclosed within a light-tight cardboard tube with a 0.025 mm-thick aluminized mylar window. Cables for the signal, bias voltage, and preamp power connected the preamp to a multichannel analyzer (MCA) containing an amplifier, bias voltage source, and preamp power source. The unanalyzed data were obtained directly from the MCA and stored on magnetic tape in a cassette.

Silicon detectors act as ion collectors. Their structure is basically the same as that of normal semiconductor diodes for electronic applications, with a p-type region and an n-type region separated by a junction or depletion layer where the concentration of charge carriers is practically zero. The depletion layer is the sensitive part of the detector. The detector is biased in the reverse direction. When a charged particle penetrates the depleted region, the electrons liberated there will move to the n-type zone (which has been given a positive voltage), while the positive charge carriers ("holes") move to the p-type zone. The average energy, E, necessary to create an electron-hole pair in a given semiconductor at a given temperature is independent

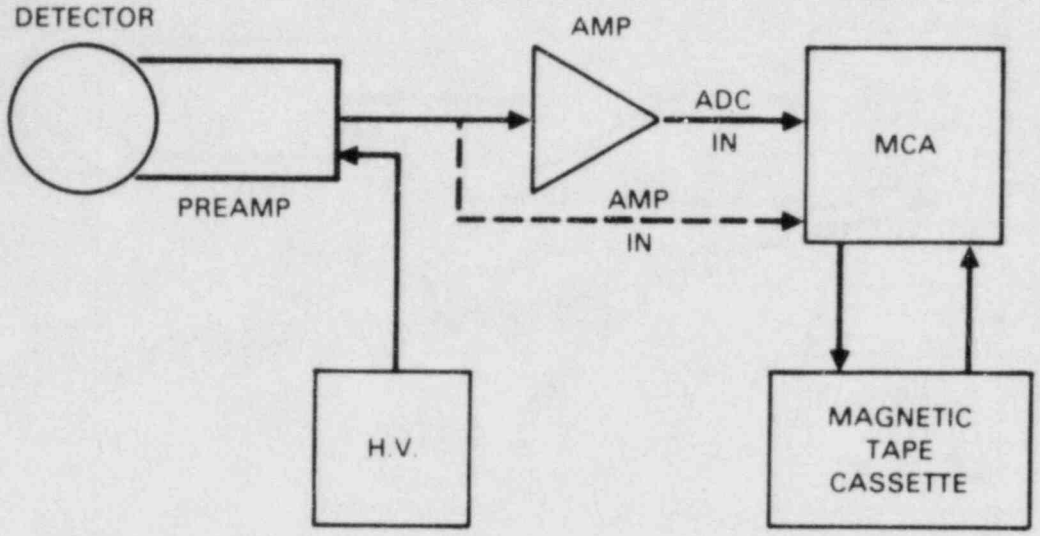

FIGURE 1. Silicon Surface Barrier Spectrometer System Components

of the type and the energy of the ionizing radiation. Therefore, the number of electron-hole pairs produced is proportional to the incident energy of the incident particle, provided the particle is fully stopped within the active volume of the detector. The value of E of silicon at room temperature is 3.62 eV, compared with about 33 eV needed to create an ion pair in a typical gas-filled detector. Although the temperatures in reactor containment areas are often 100°F to 140°F, the advantage of the silicon detector is maintained. Figure 2 illustrates the response of the silicon detector to three laboratory beta sources.

2.1.2 Multielement Beta Dosimeter

The multielement dosimeter uses thermoluminescent dosimeter (TLD) chips placed under several aluminum filters of varying thicknesses. Each filter attenuates the beta radiation to some degree, depending on the thickness of the filter. An indication of the beta energy spectrum can be obtained by examining the relative responses of the TLDs under the different filters and comparing these relative responses to those of similar dosimeters that have been exposed to calibrated sources. This spectrum indication allows the selection of a calibration factor for converting TLD response to dose. Throughout this report the exposures and exposure rates determined with the multielement dosimeters are considered to be true or reference quantities.

The multielement beta dosimeter used in this study has seven elements. Each element consists of three TLD chips covered by a filter made of either mylar or aluminum. Table 1 describes the filter materials and thicknesses used in the dosimeter.

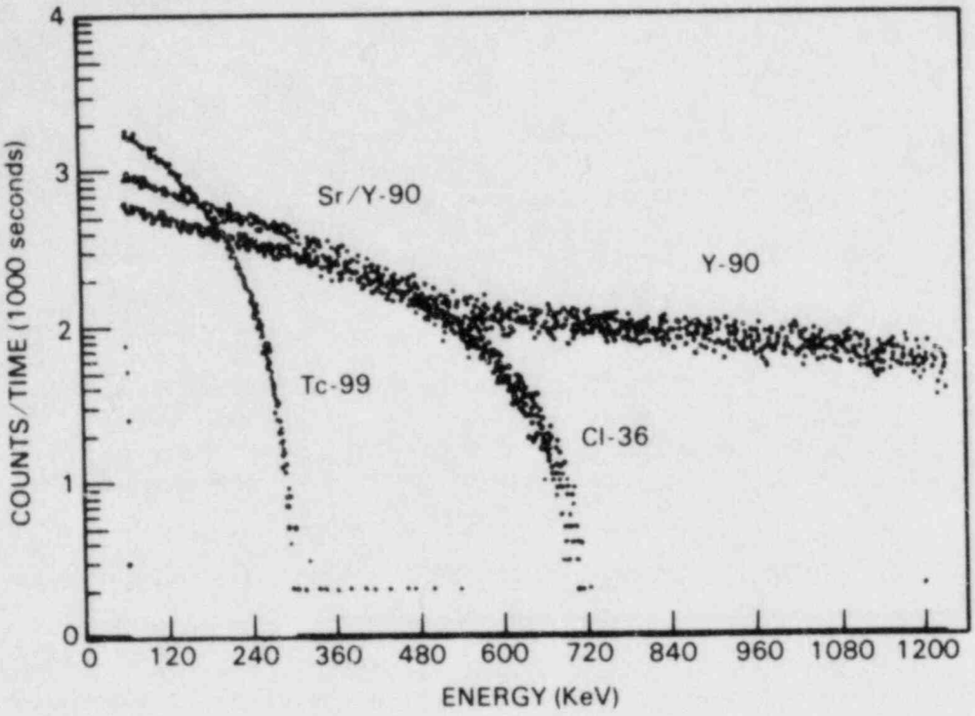

FIGURE 2. Silicon Detector Response to Laboratory Beta Sources

TABLE 1. Multielement Dosimeter Filter Specifications

Filter Material	Filter Thickness (in.)	(mm)	(mg/cm²)
Aluminized mylar	0.000002	0.00005	0.013
Al	0.005	0.127	34
Al	0.010	0.254	69
Al	0.020	0.508	137
Al	0.032	0.813	219
Al	0.064	1.626	439
Al	0.125	3.175	857

A photograph of an 8-element dosimeter is presented in Figure 3. It is identical to the 7-element dosimeter except that a 6.9 mg/cm² filter and associated element have been added.

The dosimeters used were actually double-sided: two 7-element dosimeters were placed back-to-back, separated by a 3.175-mm aluminum sheet between them. Because the aluminum separator prevents betas that strike one face of the dosimeter from striking the other face, this dosimeter package can be used to measure the beta radiation coming from two opposite directions. Thus, a dosimeter placed against a wall could determine separate doses for betas coming from wall contamination and for betas emitted by sources in the room.

The thickest dosimeter filter will stop beta particles with an incident energy of 1.9 MeV or less, and the 0.127-mm filter will stop betas with energies lower than 180 keV. The mylar filter is so thin that it will stop only betas with energies below 3 keV. Photons with energies above 40 keV are not significantly affected by any of the filters.

The response of the dosimeter described above is shown in Table 2. In our initial characterization of this device, two 7-element dosimeters were exposed to betas. One dosimeter was exposed to a $^{90}Sr/^{90}Y$ source, which has a beta spectrum with a maximum energy of 2.3 MeV. The other was exposed to a source of natural uranium, which has a beta spectrum with a virtually identical maximum energy, but with most betas having energies lower than the source (the average energy of the uranium source is about 0.8 MeV compared to 0.9 MeV for $^{90}Sr/^{90}Y$). Because of difference in intensity between the two sources, they are compared by listing the percentage of beta particles transmitted through each filter. The transmission of the aluminized mylar is assumed to be 100%.

It is evident from Table 2 that the TLD responses do decrease with increasing filter thickness, and that the rate of decrease depends on the incident beta spectrum. This feature of the multielement dosimeter is used when analyzing an exposed dosimeter to determine the incident beta energy distribution. The data in Table 2 also show that the thickest filter allows very little beta radiation to penetrate to the deepest element.

In another test, a 7-element dosimeter was exposed to radiation from a $^{137}Cs/^{137}$ Ba source that was shielded so that only gammas (no betas) were emitted. This dosimeter had TLD responses that were quite uniform--the standard deviation of the seven individual responses was only 2% of their mean response. Therefore, we may assume that all elements respond identically to moderate energy photon radiation.

The elements in the dosimeter produced TLD responses depending on the thickness of the shield covering the TLDs. For each element, the mean value of the three TLD responses (as determined by the TLD reader) was recorded. A typical set of this "raw" data, in units of nanocoulombs (nc), for a 7-element dosimeter is presented in Table 3.

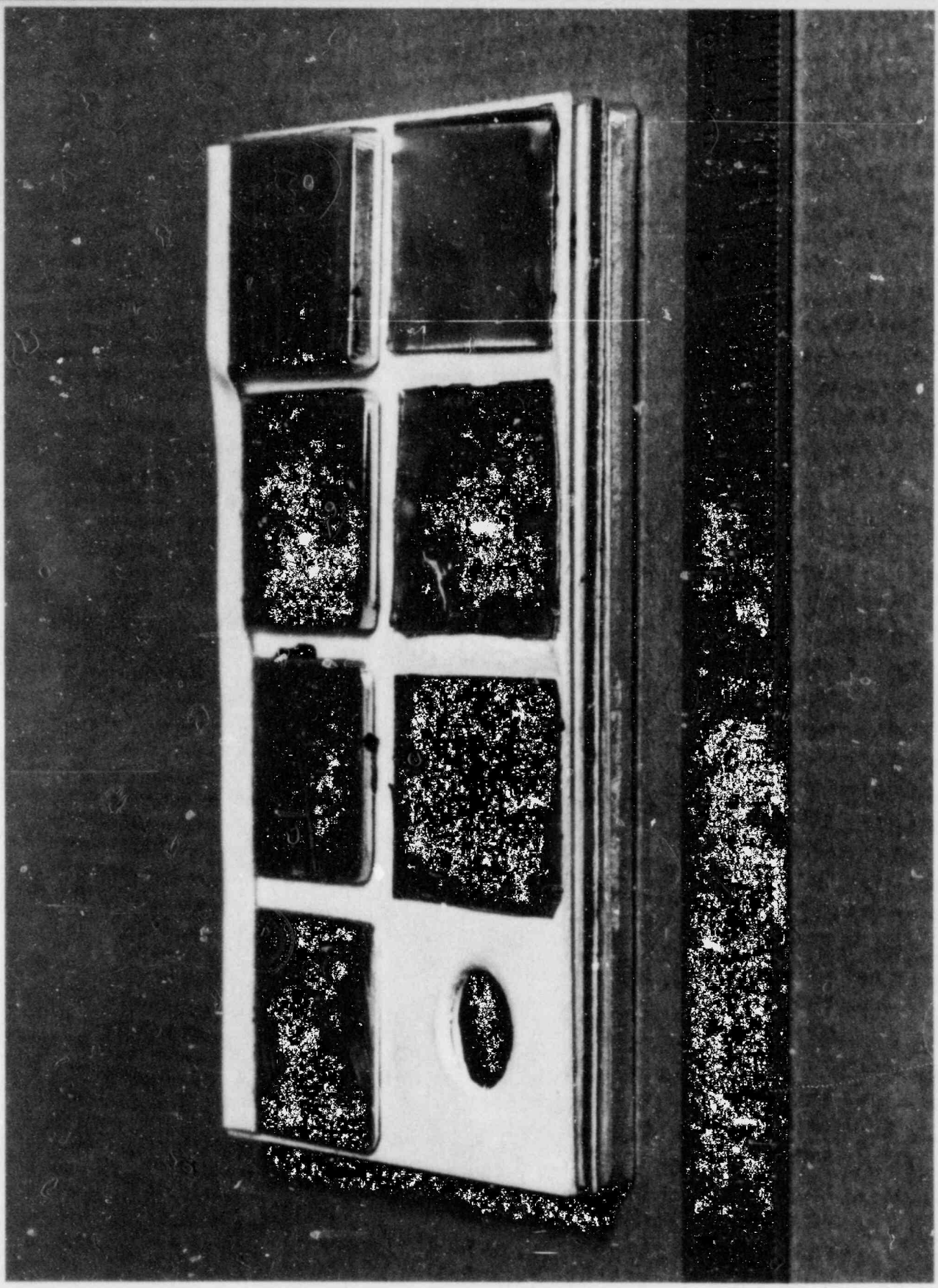

FIGURE 3. PNL Eight-Element Dosimeter

TABLE 2. Responses of Dosimeters Exposed to Natural Uranium and $^{90}Sr/^{90}Y$ Sources

Filter	U (natural)	$^{90}Sr/^{90}Y$
0.000002-in. Al (on Mylar)	2.73 (100%)	4.60 (100%)
0.005-in. Al	1.69 (62%)	3.59 (78%)
0.010-in. Al	1.46 (54%)	3.05 (66%)
0.020-in. Al	0.886 (32%)	2.38 (52%)
0.032-in. Al	0.590 (22%)	1.53 (33%)
0.064-in. Al	0.201 (7%)	0.303 (11%)
0.125-in. Al	0.098 (4%)	0.002 (0%)

TABLE 3. Typical Data for Exposed 7-Element Dosimeter

Filter	Gamma + Beta (nc)	Beta (nc)
0.000002-in. Mylar	1.99	1.83
0.005-in. Al	0.81	0.65
0.010-in. Al	0.78	0.62
0.020-in. Al	0.49	0.33
0.032-in. Al	0.30	0.14
0.064-in. Al	0.21	0.05
0.125-in. Al	0.16	0.00

The reading for the TLDs under the thickest filter was used to indicate the gamma dose. This value was subtracted from the other element readings to obtain a response for each element to betas alone.

To determine doses from the exposed dosimeters, a ratio of each element's beta response to the mylar-covered element's beta response was found. These ratios were compared to ratios for various beta spectra that were determined during calibration exposures. This comparison was used to select an energy-dependent calibration factor, which is multiplied by the mylar-covered element's beta response to determine the beta dose.

The multielement beta dosimeters were calibrated by exposing them to known radiation sources at the PNL Calibrations Laboratory. The beta sources used in these calibrations and the maximum energies of the emitted betas are presented in Table 4.

The dosimeters were also exposed to calibrated sources of x rays with effective energies ranging from 15 to 78 keV, and to a calibrated $^{137}Cs/^{137}Ba$ source emitting 662-keV gammas. These measurements characterized multielement dosimeter response to photons, and showed that photon attenuation is dependent on filter thickness only for energies less than about 40 keV. For photons with higher energies, the TLD response is fairly uniform for all elements. This dosimeter characteristic permits using the response of the element with the thickest aluminum filter as an indicator of photon dose.

The data from these beta and photon calibrations were used to derive calibration factors for determining the beta doses from exposed dosimeters. More detailed information regarding the derivation and use of multielement dosimeter calibration factors is contained in Scherpelz et al. (1983).

TABLE 4. Beta Calibration Sources

Nuclide	Maximum Beta Energy (MeV)	Dose Rate mrad/hr)	Distance (cm)
^{147}Pm	0.23	10.2	20
^{85}Kr	0.62	7750	50
^{204}Tl	0.76	64.1	30
$^{90}Sr/^{90}Y$	2.3	164	50
$^{106}Ru/^{106}Rh$	3.5	200	50

2.2 SURVEY METERS AND DOSIMETERS

Selected commercially available dosimeters and instruments were studied for performance under laboratory and field conditions. These beta measurement devices were selected to be representative of the wide variety of instruments and dosimeters available to the licensee.

2.2.1 Commercial Survey Instruments

The six survey instruments selected for study included four air ionization detectors, one plastic wall gas ionization detector, and one Geiger-Mueller tube. Five were manufactured in the United States and one was of foreign manufacture. Detector window thickness ranged from 7 mg/cm² to approximately 60 mg/cm². The detector volume varied by an order of magnitude among the six instruments. All six instruments were combined beta-gamma survey instruments and in all but one case (instrument A) the design favored

photon monitoring over beta monitoring. This is evidenced by the fact that the length of the detector on each of the five instruments exceeded the width of the detector. For instrument A, the width (diameter) was about 1.7 times the length of the detector chamber. Even though this ratio is far from optimal, it tends to favor beta monitoring more than the other designs usually found at licensee facilities. The advantages for beta monitoring of a "pancake" type of ion chamber were described by Roesch and Donaldson (1955), but no United States vendor is known to manufacture such a device.

All of the ionization chamber instruments were calibrated with ^{137}Cs photons by the manufacturers. In addition, two of the instruments were checked for response to beta sources by the manufacturers. The G-M tube manufacturer suggests that the user calibrate the detector.

Each of the field survey meters was calibrated in the PNL Calibrations Laboratory. The beta sources and distances listed in Table 4 were used for these calibrations. First the beta window of each instrument was placed at the specified distance from a given beta source and the response noted. Then the calibration was repeated with the center of the detector at the specified distance. Each instrument was also calibrated with the beta window in contact with a natural uranium slab. The responses of the instruments to the known beta dose-rates of the calibration sources were used to calculate calibration factors. This information is listed in Table 5.

The response of most of the survey meters changed rather dramatically with the energy of the beta emitter. Thus a single calibration factor is usually not adequate. The variation in calibration factors with the different beta sources ranges from less than a factor of 2, for instrument A, to a factor of 35 for instrument F. It is also interesting to note the change in calibration factors between the first and second listing in Table 5. The first listing is based on the distance between each source and the window of a survey meter, whereas for the second listing the distance is measured between the center of the detector and the source. Even for the largest detectors the change in the position of the detector is only a few centimeters, yet the change in calibration factors is quite dramatic in some cases. For ^{147}Pm, the difference in calibration factors between the first and second listing is typically about 50% of the larger value. However, instrument F (end) drops from a calibration factor of 73 to 17. The response of instrument F is also directionally dependent as shown by the two-fold change in response when changing from end-on to side-on measurements.

The variation in calibration factors described above is apparently even more dramatic at closer source-to-detector distances. A recent PNL publication (Hooker et al. 1983) deals with calibration factors for air ionization chambers. That work shows a ten-fold change in calibration factors when the survey meter's window is placed at one-half inch from rather than in contact with a small beta source.

TABLE 5. Calibration Factors for Survey Instruments

Source	^{147}Pm	^{85}Kr	^{204}Tl	$^{90}Sr/^{90}Y$	U_{nat}	$^{106}Ru/^{106}Rh$
Energy, Mev	0.23	0.62	0.76	2.3	2.3	3.5
Distance[a]	20	50	30	50	contact	50
A (air ion, 7 mg/cm²)[b]	2.8	2.6	2.5	1.2	3.2	1.8
208 cc	1.2	2.0	1.6	1.1		1.5
B (gas ion, 60 mg/cm²)	2.8	0.6	0.6	0.1	0.5	0.2
34.4 cc	2.5	0.7	0.6	0.1		0.2
C (air ion, 50 mg/cm²)	3.4	5.9	6.3	2.0	4.5	2.3
250 cc						
D (air ion, 7 mg/cm²)	0.6	1.5	1.2	0.9	2.0	1.2
300 cc	0.5	1.0	1.0	0.9		0.9
E (GM, 30 mg/cm²)	1.8	OS[c]	OS	OS	6.4	OS
17.3 cc	0.8	OS	OS	OS		OS
F (air ion, 17 mg/cm²)	73	12	6.5	2.1	7.8	2.7
end 300 cc	17	7.2	4.1	1.6		2.2
F (air ion, 17 mg/cm²)	34	3.2	3.0	1.5		1.9
side 300 cc	9.9	2.2	2.0	1.2		1.5
G INEL dose rate meter	1.0	1.6	1.0	1.0		0.8

(a) The first calibration factor listed under each source for a given instrument is for the source-to-window distance; the second calibration factor is for the source-to-instrument detector center distance.
(b) Instruments are designated by letters A-G, followed by the type of detector, window thickness, and detector volume (except INEL dose rate meter).
(c) OS = off scale.

Evidence that the source to detector distance is not so important for large area sources is also seen in Table 5. Here the window of the detector is placed virtually in contact with the uranium slab. Due to the different sizes of detectors, the source-to-detector center distance varies considerably among the different instruments. Even so, the measured values of beta radiation on the uranium slab bear about the same relationship to each other as do those for the other high energy beta emitters. The calibration factors for the uranium slab are about twice as large as those for the other high energy sources because calculating the calibration factors based on the exposure at the instrument's window rather than its center leads to much greater errors at close distances. This is offset partially, but not entirely, by the difference in area between the uranium slab and the other sources. Because

calibrating the average survey meter with a uranium slab, in the manner described above, is commonly thought to cause an overestimation of the calibration factor, licensees often use this method of calibration. However, in the case of instruments B, C, and F, the uranium slab calibration factor would not always lead to conservative measurements when compared to the other laboratory calibration sources.

2.2.2 Prototype Beta-Gamma Dose Rate Meter

The prototype beta-gamma dose rate meter was developed at Idaho National Engineering Laboratory (INEL) and described in a paper by Johnson et al. (1983). The detector system consists of a very thin plastic scintillator (NE-105) covered by a light tight aluminized mylar entrance window. The scintillator is backed by a 1-cm thick, lucite light pipe which is attached to a photomultiplier tube. Pulse shape discrimination techniques are used to minimize Cerenkov events within the lucite light pipe and photomultiplier tube glass and direct interaction within the photomultiplier tube dynode structure. The INEL beta-gamma dose rate meter is entirely portable but had not been ruggedized for routine field use at the time that it was used to measure PNL laboratory beta sources.

The INEL portable instrument was calibrated in the same manner as the commercial survey instruments described in Section 2.2.1. Because the plastic scintillator has a negligible depth compared to the commercial survey instrument detectors, only one measurement per beta source was required. The calibration factors determined for the plastic scintillator instrument are listed in Table 5. The calibration factor for the three Physikalisch-Technische Bundesanstalt (PTB) sources is unity, i.e., the instrument is direct-reading for ^{147}Pm, ^{204}Tl, and $^{90}Sr/^{90}Y$ sources. However, the indicated calibration factor for ^{85}Kr is 1.6 and for $^{106}Rh/^{106}Ru$, 0.8. Although the geometry and attenuation for the beta spectra of the latter two sources are different than that of the PTB sources, this does not adequately explain the apparent nonlinearity of the INEL dose rate meter response.

A previous test had also shown the instrument to have a low response for ^{204}Tl beta particles (Johnson et al. 1983). However, at the time of the previous test, the scintillator was an NE-102 device rather than the NE-105 scintillator now in use. The beta responses should not have been affected by the change in scintillators, but the low-energy photon response of the NE-105 device should be better. The improvement in low-energy photon response was confirmed by PNL x-ray calibrations. The results are shown in Table 6. Because the INEL instrument was not used in field environments during this study, no further comparisons were possible.

2.2.3. Commercial Dosimeters

Five commercial vendors supplied personnel dosimeters for use in this study. Three of the vendors supplied multiple types of dosimeters. The dosimeters included four types of film badge, four types of TLD badge, and one type of pocket ionization chamber. One of the TLD badges was a 4-element dosimeter. The remainder of the badges were two-element devices. The beta

TABLE 6. Measurements of PNL K-Shell Photons with INEL Beta-Gamma Dose Rate Meter

E (keV)	Calibrated Air Exposures (±5%) (mrem/hr)	Reading (mrem/hr)
100	396	440
78	420	452
58	648	700
43	1344	1530
34.3	1884	1410
23.7	4620	1210
16.1	4236	385

dosimetry problem associated with the two-element dosimeter are described in a recent report by Sherbini and Porter (1983).

The commercial dosimeters used in this study were calibrated in the same manner as were the survey instruments, except for a longer exposure period. The results are displayed in Table 7.

The processor reports for the commercial dosimeters used in this study (Table 7) indicate that all responded low to the uranium and ^{147}Pm sources. The responses of some dosimeters to $^{90}Sr/^{90}Y$ and ^{204}Tl was much closer to the doses calculated from known dose rates and exposure times.

2.3 SITE SELECTION AND DESCRIPTION

The NRC-licensed facilities chosen for beta spectra, doses and dose rates measurement were limited in number. However, the three BWRs, two PWRs, and one fuel fabrication facility were selected to be representative of typical United States commercial BWR, PWR, and fuel fabrication facilities. The criteria used to compare commercial nuclear power plants included age, location, and operating status. At commercial power reactors, the two major sources of beta radiation (^{60}Co and fission products) both typically increase with age. Therefore only middle-aged (approximately 10 years old) and older (approximately 20 years old) facilities were considered. To make certain that the ownership and management of the facilities was as varied as possible, they were selected from widely separated sections of the United States.

Because PNL researchers had previously visited most of the facilities chosen for this study, as well as numerous other similar facilities, choices could be made based on first-hand knowledge of the radiation fields present at the various facilities. Other studies which proved most useful in this regard include "Measurement of Low- and High-Energy Photon Spectra at Commercial Nuclear Sites" (Roberson et al. 1982) and "Neutron Dosimetry at Commercial Nuclear Plants" (Endres et al. 1981).

TABLE 7. Commercial Dosimeter Responses to Beta Calibration Sources

PNL Beta Source	Beta Exposure (mrad)	Dosimeter A (mrad)	B (mrad)	C (mrad)	D (mrad)	E (mrad)	F (mrad)	G (mrad)	H (mrad)	I (mrad)
U slab	100	63		31	34	5	0	0		
U slab	300	112	95	106	108	50	0	105		90
U slab	500									152
U slab	675	234	232	198	212	160	0	0		
U slab	4000	1487	1430	1295	1350	1040	60	55		
^{90}Sr	2000		1934		1057	1345	152	150	7050	0
^{90}Sr	1000	901		604						
^{90}Sr	500	605	327							
^{204}Tl	1000	1000	632	284	116	205			2400	
^{204}Tl	500	694	336							3
^{247}Pm	250	54	396	5.5	6.5	60	0	0	95	12

Although the original intent was to visit each facility during both operating and shutdown conditions, this proved to be less important than other considerations. Therefore, when it was found that Site N could not be visited during shutdown, a similar facility, Site M, was visited as a substitute shutdown plant.

Because of the similarities in product and operation, the fuel fabrication facility was chosen mainly on the basis of location. The designations and basic descriptions of the facilities chosen for this study are shown in Table 8.

2.4 DATA COLLECTION

At each licensee facility, PNL scientists reviewed site-specific routine survey data and discussed possible beta exposure sources with onsite health physics personnel. This information was used in conjunction with the known equipment capabilities and responses and potential for personnel exposure to choose specific beta measurement locations.

All devices used to measure beta radiation were wrapped in plastic prior to entry into radiation zones to prevent unnecessary contamination. This plastic acted as an additional filter for beta radiation.

In areas where the radiation field to be measured had higher dose levels than adjacent areas, the silicon detector was placed near the source to be measured and the analyzer was placed in the lower dose area, up to 30 feet

TABLE 8. Participating Licensee Facilities

Facility Code[a]	Facility Type
Q	BWR
P	PWR
R	Fuel Fabrication
M	BWR
N	BWR
O	PWR

(a) The facility code followed by the number 1 or 2 indicates first or second visit, e.g., Q-2 denotes the second visit to Site Q.

away. Because the beta radiation fields usually were accompanied by measurable gamma radiation fields, each beta spectrum was obtained by subtracting the photon portion of the beta-gamma spectrum. To obtain the gamma-only spectra, beta shields of aluminum, plastic, and lead were placed over the silicon detector.

3.0 RESULTS AND DISCUSSION

3.1 FIELD SPECTRA

The first NRC-licensed facility (Q) visited was an operating BWR of median age relative to other U.S. reactors. Company health physicists identified three possible areas of beta exposure. These included the turbine room, pump room B, and contaminated items near the top of the fuel pool. The first two areas were expected to contain significant beta fields only if airborne ^{16}N were present.

Beta measurements were taken in pump room B at the location shown in Figure 4. The silicon detector spectrum for pump room B is shown in Figure 5. Because both this spectrum and the nearly identical spectrum (not shown) for the turbine room exhibited a peak near the endpoint of the spectrum, it was initially suspected that high energy beta particles might be present. However, further analyses showed these peaks to be the double escape peaks for photon

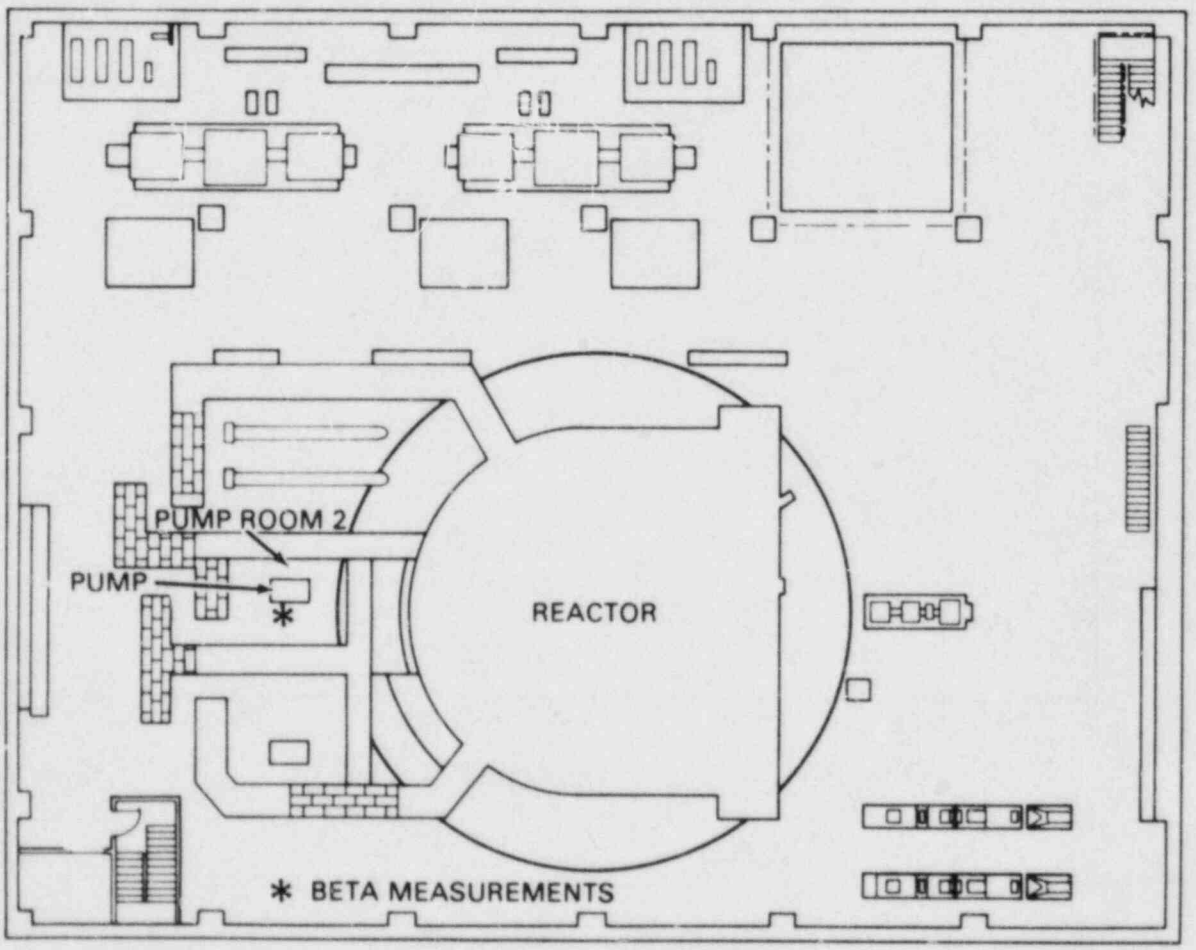

FIGURE 4. Site Q, Pump Room B, Measurement Location

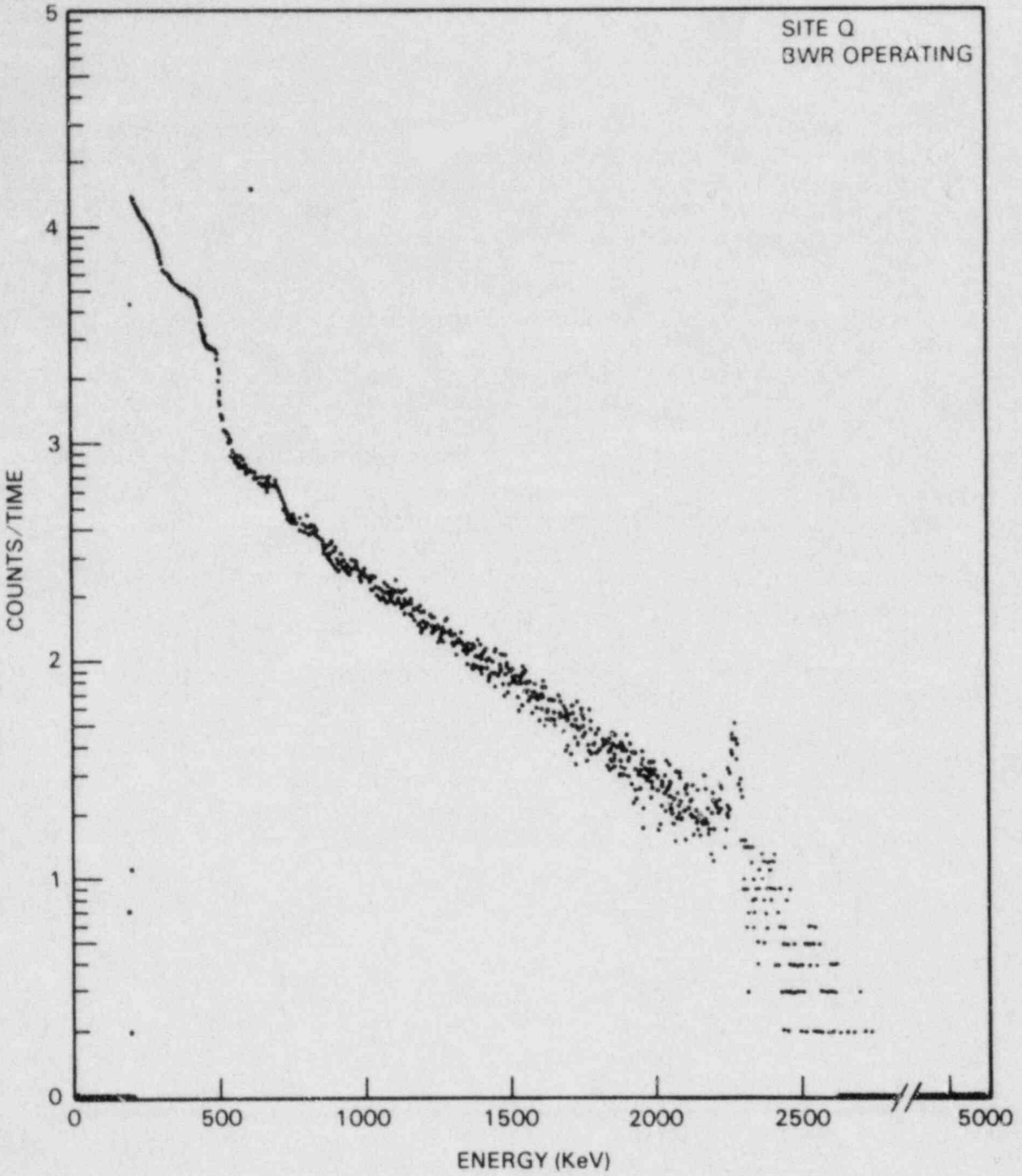

FIGURE 5. Pump Room B Spectrum

pair production in the silicon. The small electron contribution to the exposure in pump room B is thought to be due to photon interactions with air and other media.

The most significant beta radiation field found during the first visit to Site Q was associated with a metal chute placed between the fuel storage pool and the reactor pool during refueling. The fuel pool chute location is shown in Figure 6. The ^{60}Co and ^{137}Cs contamination of the metal chute is evident in Figure 7. When a 3/16-in. Nalgene® absorber was placed between the detector and the chute, the 600- to 650-keV conversion electron peak from ^{137}Cs was removed, as shown in Figure 8.

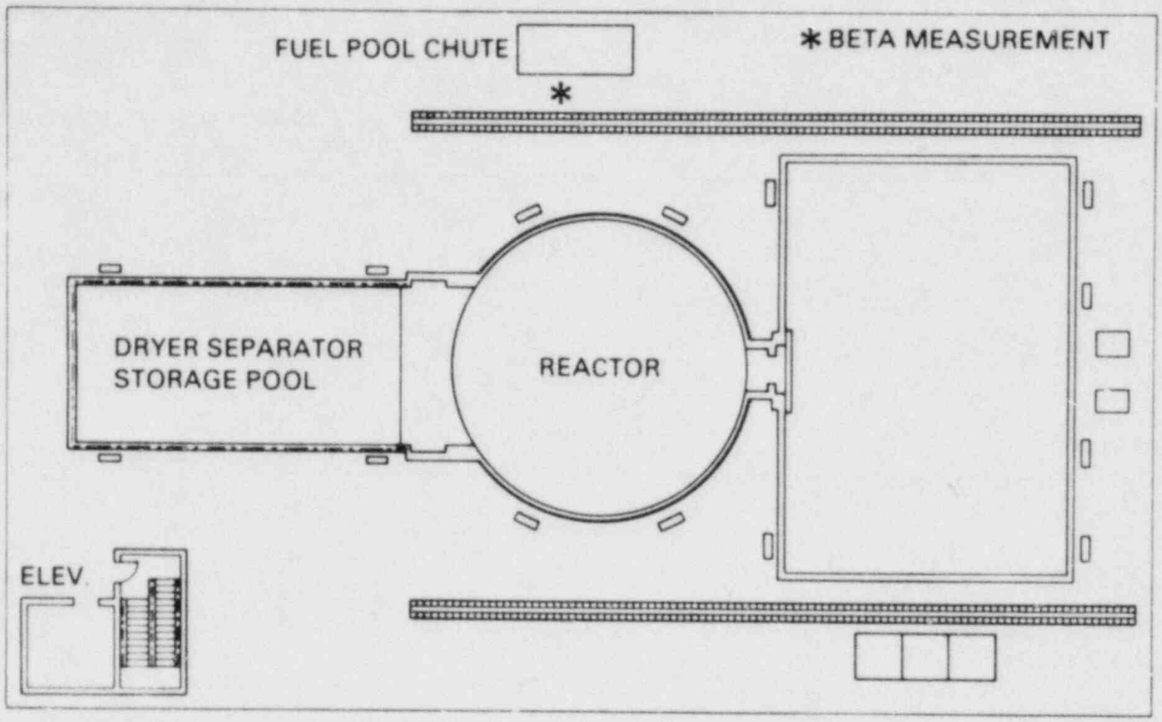

FIGURE 6. Site Q, Fuel Pool Area, Measurement Location

Because the fuel chute was in use during a second visit to Site Q, it could not be checked for beta radiation. However, with the reactor shut down for refueling, other contaminated components from the fuel pool were available. The unmoderated spectrum from an underwater light is shown in Figure 9. Once again the Compton scatter spectra for ^{60}Co photons is very evident, but no influence from ^{137}Cs can be seen. The ^{60}Co beta spectrum, with an endpoint energy at 314 keV, cannot be properly separated from backscatter and low energy photon events in this spectrum. This problem is prevalent in containment areas.

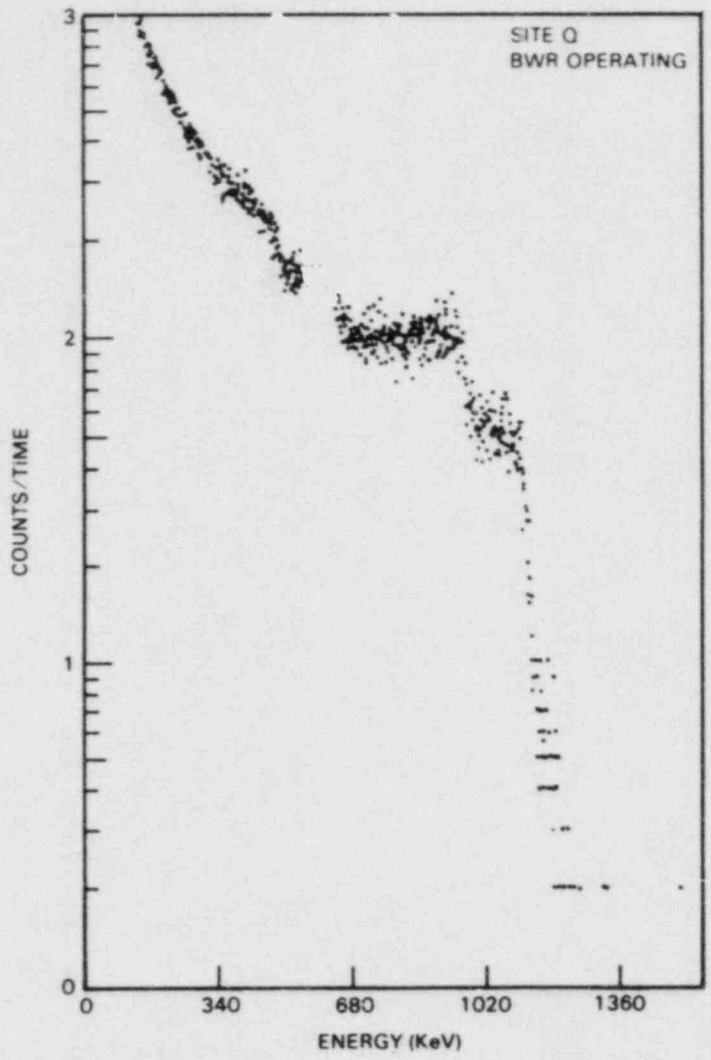

FIGURE 7. Site Q Fuel Pool Chute Beta-Gamma Spectrum

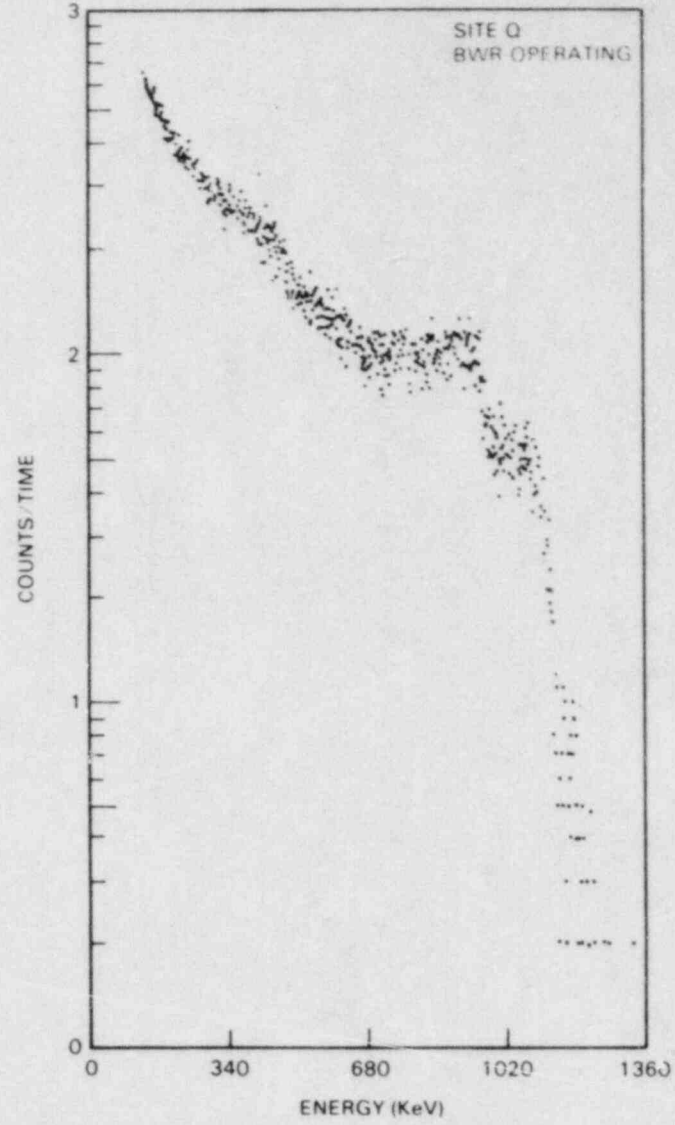

FIGURE 8. Site Q Fuel Pool Chute Gamma Spectrum

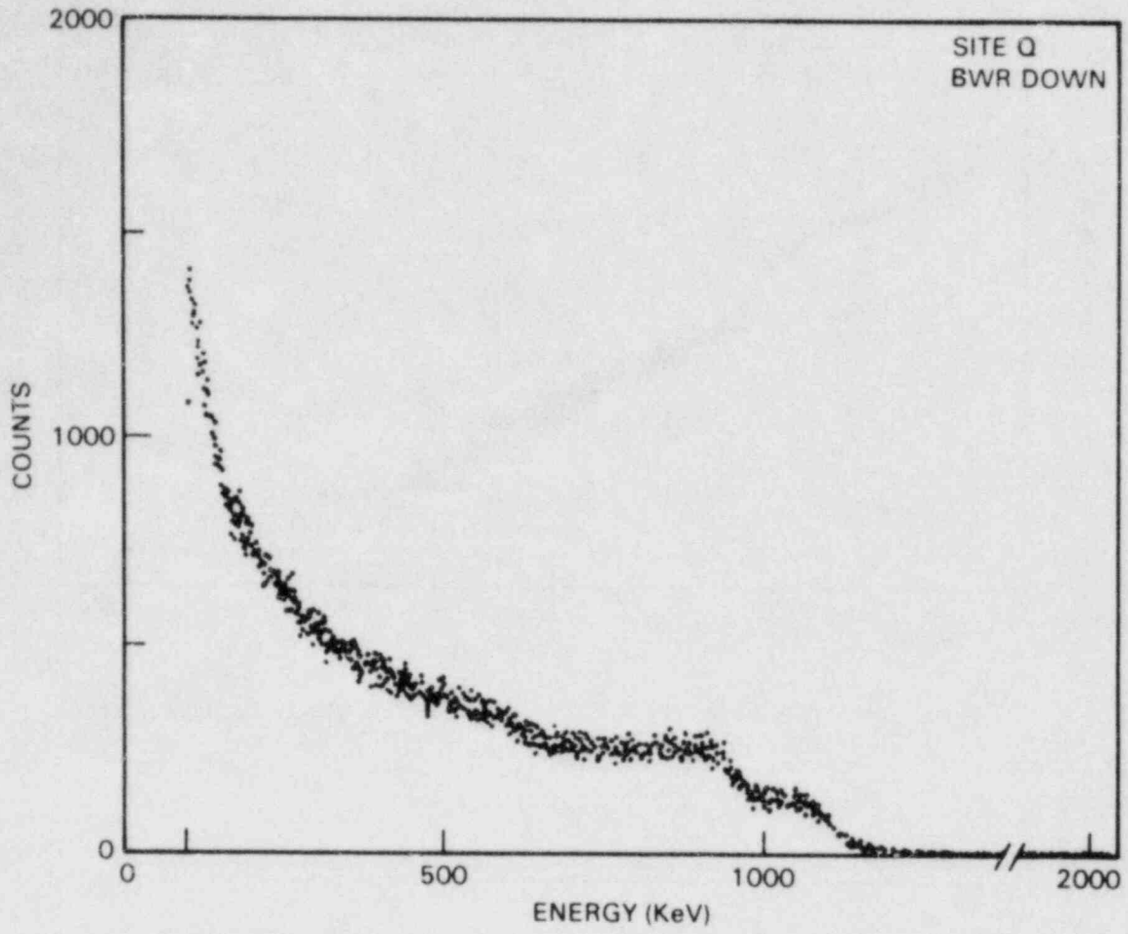

FIGURE 9. Underwater Light Spectrum

Site P, an older LWR, was the second site visited. It was shutdown at the time of the first visit and operating during the second visit. No beta field suitable for silicon detector measurements was found during the first visit to Site P. However, the silicon detector was used to obtain spectra from smear samples and air filter samples. The samples represented the environment of a steam generator, a decon tent, and an in-core instrument (ICI) package. These spectra exhibit the characteristics of both gamma and beta interactions, as seen in Figure 10, where much of the first one-third of the spectrum is due to gamma interactions and the final two-third is due entirely to beta interactions. Figure 11 shows the effect of subtracting out the photon-generated portion of the spectrum. The remaining beta spectrum has a steeply descending, low-energy component due to ^{60}Co beta particles (endpoint energy, 314 keV) and a high-energy component, probably due to $^{90}Sr/^{90}Y$ and $^{106}Rh/^{106}Ru$. Because this spectrum was obtained in a low background area, using a small, thin source 3 in. from the detector, the beta and gamma spectra could be subtracted on a one-to-one basis. Although this is typical of laboratory circumstances, it is not often found in the field.

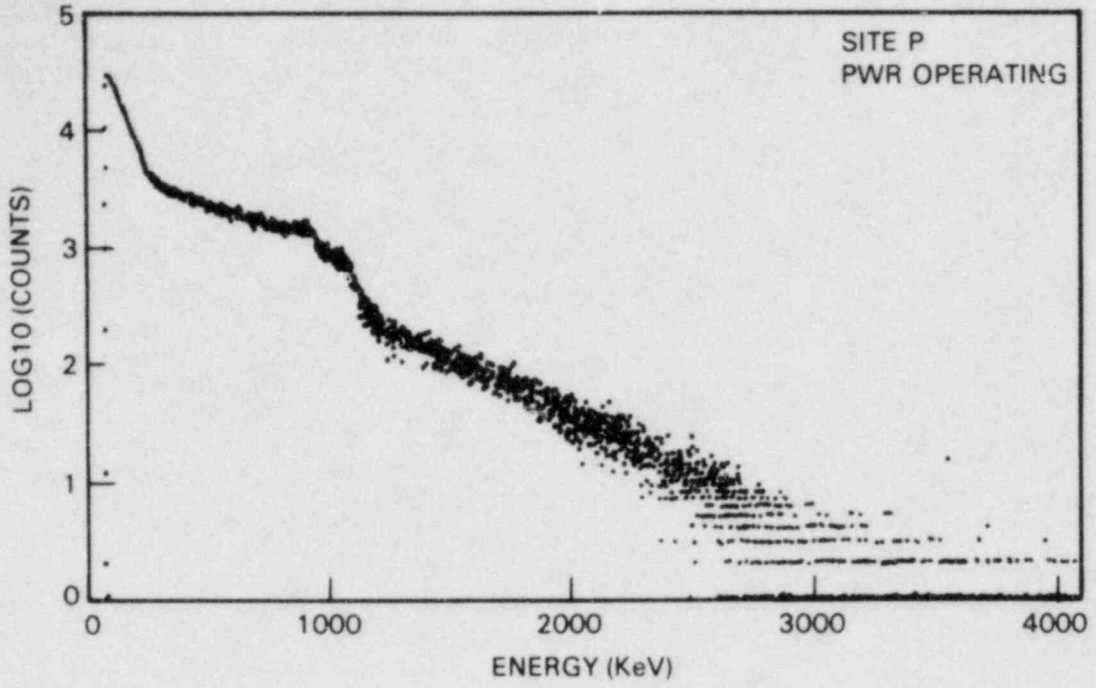

FIGURE 10. Smear from Control Rod Drive Shaft (Beta + Gamma), Site P

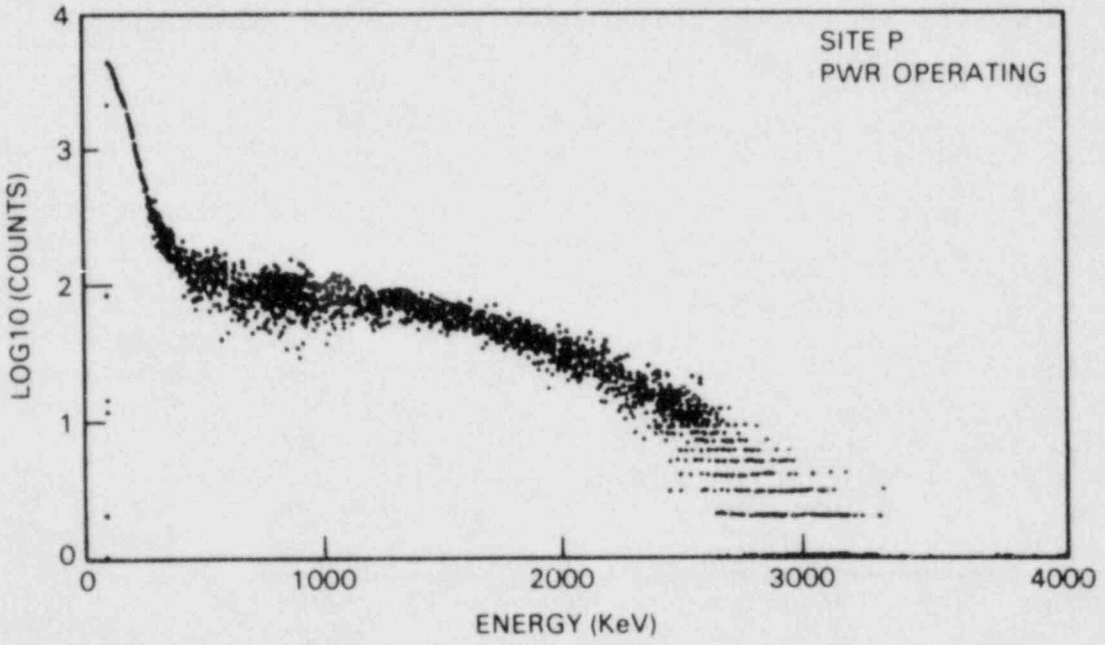

FIGURE 11. Smear from Control Rod Drive Shaft (Beta Spectrum Only), Site P

The initial spectra taken at Site P during the second visit were also obtained under laboratory conditions. Figures 12 and 13 show the beta-gamma and beta only spectra for a spire bolt that had been removed from the vicinity of the reactor core. The same results discussed for Figures 10 and 11 apply here.

Inside containment at Site P, a significant beta field was found at the bottom of the spare main coolant pump, which was sitting on the upper floor (see Figure 14). Figure 15 is a result of subtracting the main coolant pump spectrum obtained with a beta absorber (1/8-in. Pb) from the spectrum obtained without the beta absorber. Hence, if this were laboratory conditions, we would expect a nearly pure beta spectrum. However, in Figure 15, only the portion of the spectrum below about 300 keV is predominantly of beta origin (^{60}Co). The portion of the spectrum from 300 to 1,100 keV is primarily from ^{60}Co photons with the remainder due to neutron interactions in the silicon detector. Although an attempt was made to obtain beta spectra from radioactive gases that might have been present in containment, no beta emitters were identified in the Site P containment atmosphere.

One fuel fabrication facility (Site R) was visited during this study. Spectra from large buckets of powdered uranium and shallow rectangular containers (trays and boats) of pellets were obtained. All the spectra were the predicted, nearly-pure beta spectra characteristic of uranium. Figure 16 shows a virtually unattenuated beta spectrum for a boat of fuel pellets. Figure 17 shows the effect of a 1/16-in.-thick aluminum absorber. A 3/16-in. aluminum absorber plus a 3/16-in. Nalgene® absorber were placed over the silicon detector to obtain the spectrum shown in Figure 18. In Figures 16 through 18 a change in the shape of the uranium beta spectrum is noted, as the removal of the lower energy beta particles from the spectrum outweighs the attenuation of the higher energy beta particles. These results are in contrast with the experimental results (Dudley 1951) referenced in The Atomic Nucleus (Evans 1955). However, Hankins (1982a) and Simons (1982) have also shown that, in many cases, considerable spectral shift occurs during beta absorption.

Site M was the fourth facility visited by PNL during beta field characterization work. This plant, a BWR of median age, was shutdown when spectra were obtained with the silicon detector system. At this facility, the only sources found whose beta fields were strong relative to the gamma fields were the disassembled turbine components. The measurement location is shown in Figure 19. The beta spectrum for a turbine blade diaphragm is shown in Figure 20. This spectrum is one of the few examples where the gamma component was successfully subtracted from the beta-gamma spectrum in a field situation. The remaining beta spectrum is probably due to ^{131}I and ^{132}I.

Another BWR, Site N, was visited next. It was operating at the time of the visit. Spectra were taken in a heater bay and main steam isolation valve area (MSIV) in an attempt to find betas from airborne ^{16}N. However, the spectra showed only gamma-induced events. The only beta spectrum obtained at Site N was from a section of cleanup pipe, which had previously been removed and placed in a storeroom inside containment. The combined beta-gamma

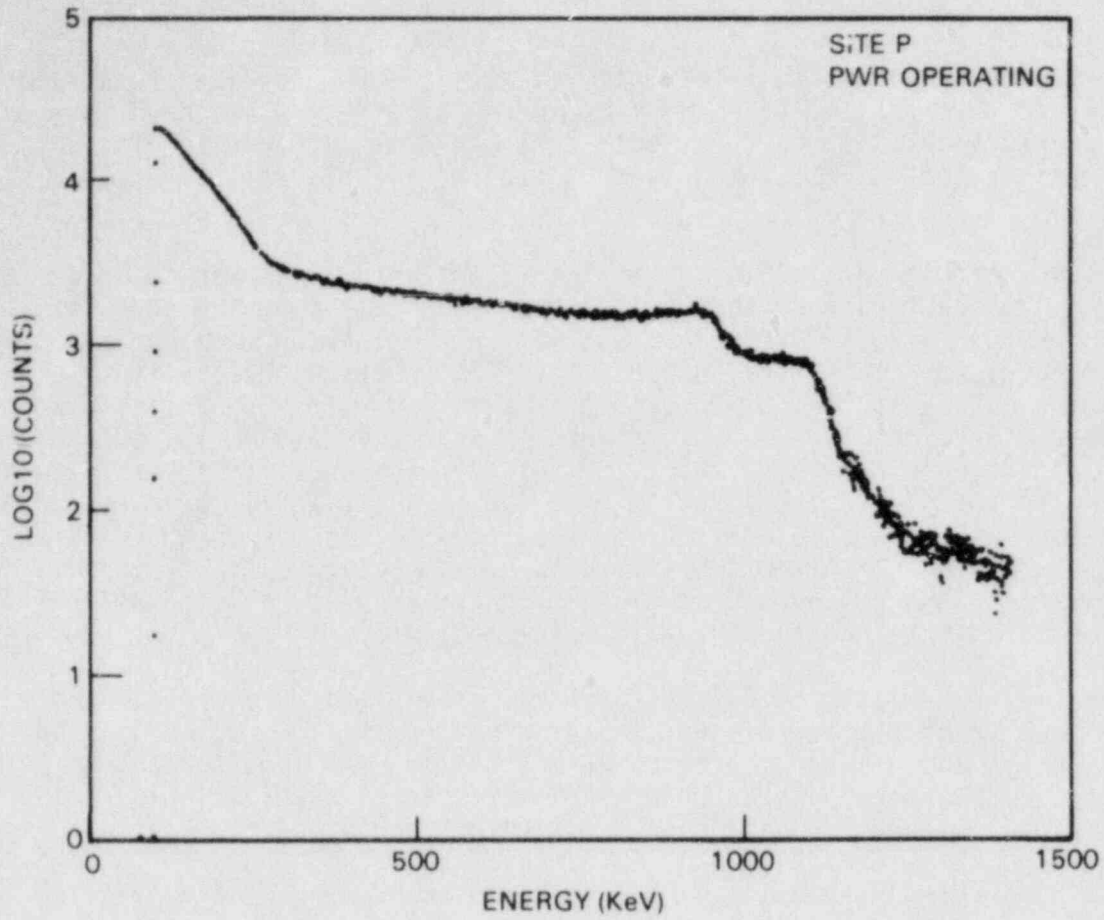

FIGURE 12. Spire Bolt (Beta + Gamma) Spectrum

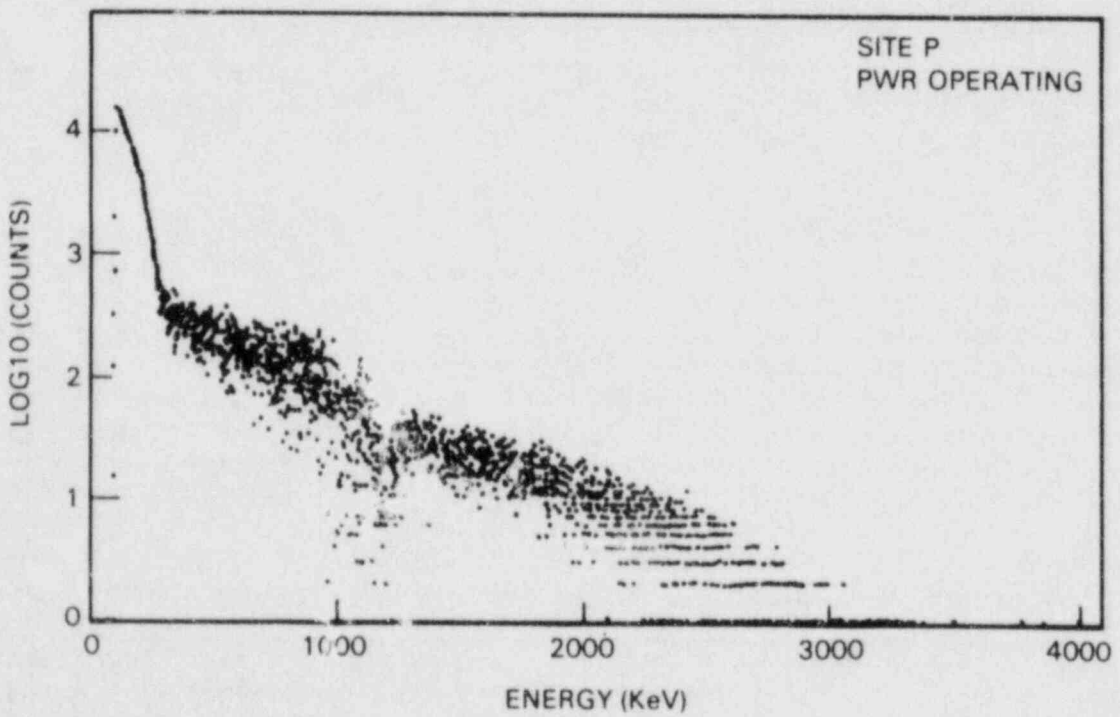

FIGURE 13. Spire Bolt (Beta) Spectrum

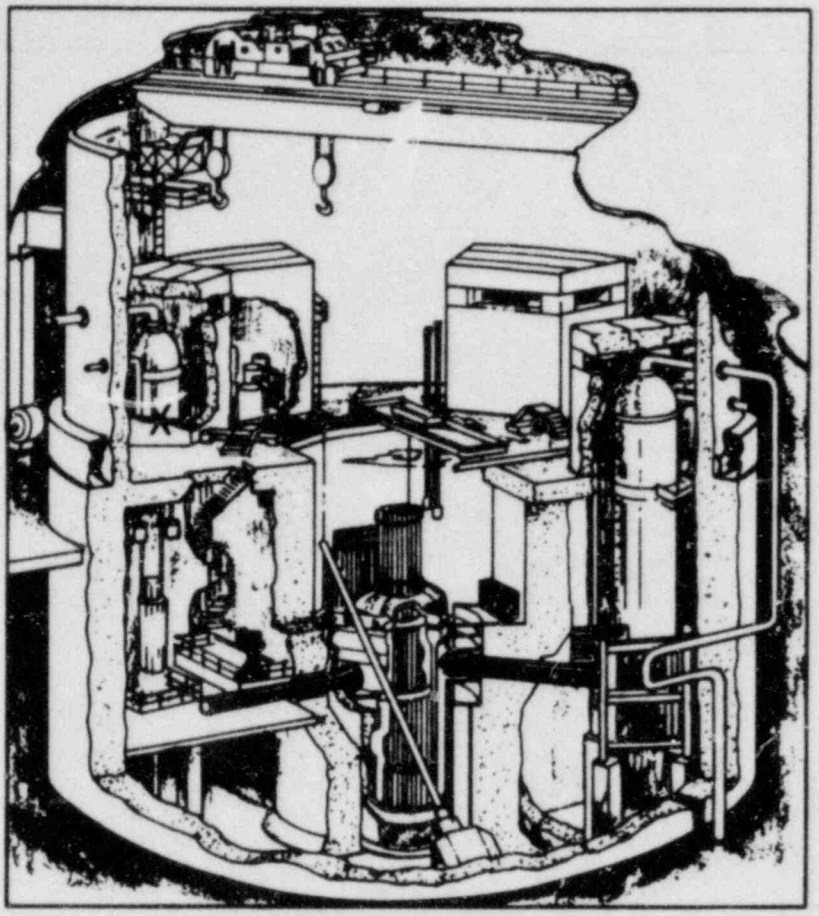

FIGURE 14. Site P, Spare Main Coolant Pump

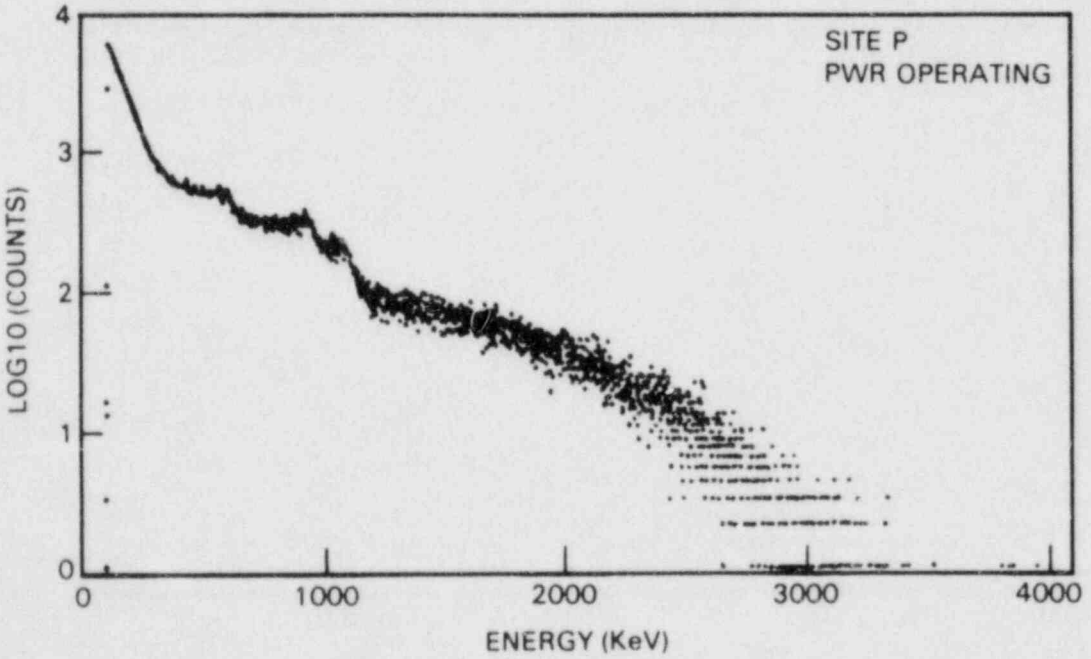

FIGURE 15. Main Coolant Pump Spectrum (after subtraction)

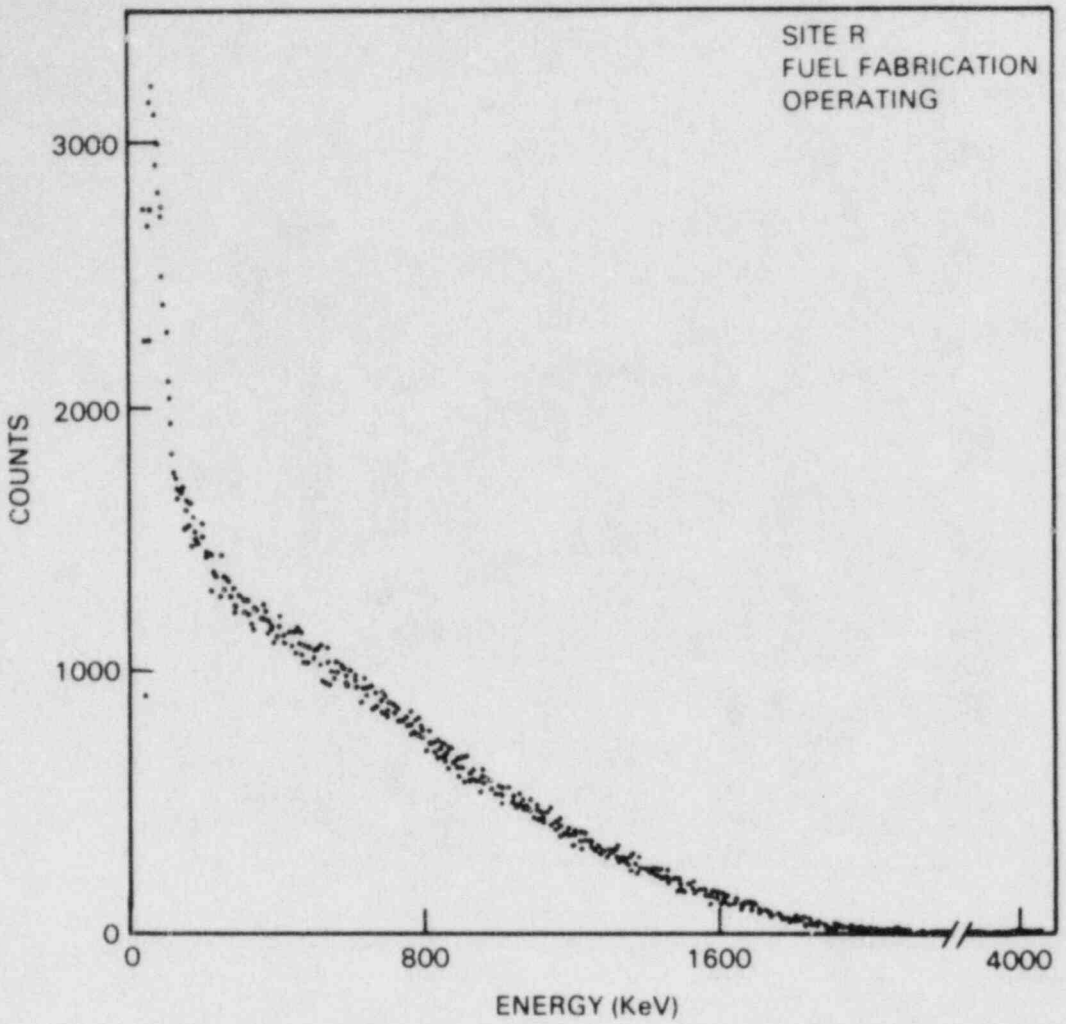

FIGURE 16. Site R, Uranium Full Spectrum

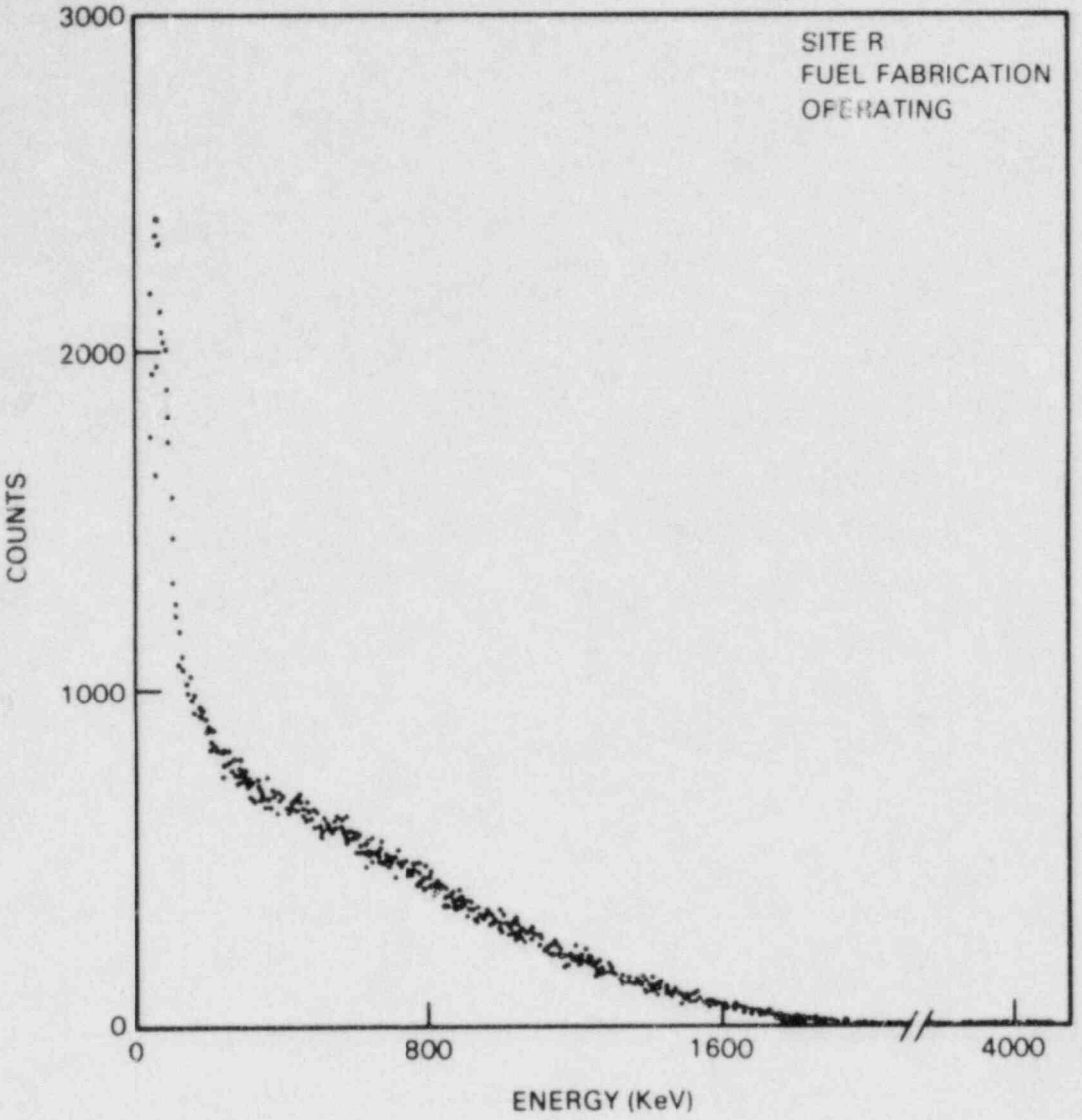

FIGURE 17. Site R, Uranium Spectrum (with 420 mg/cm^2 shield)

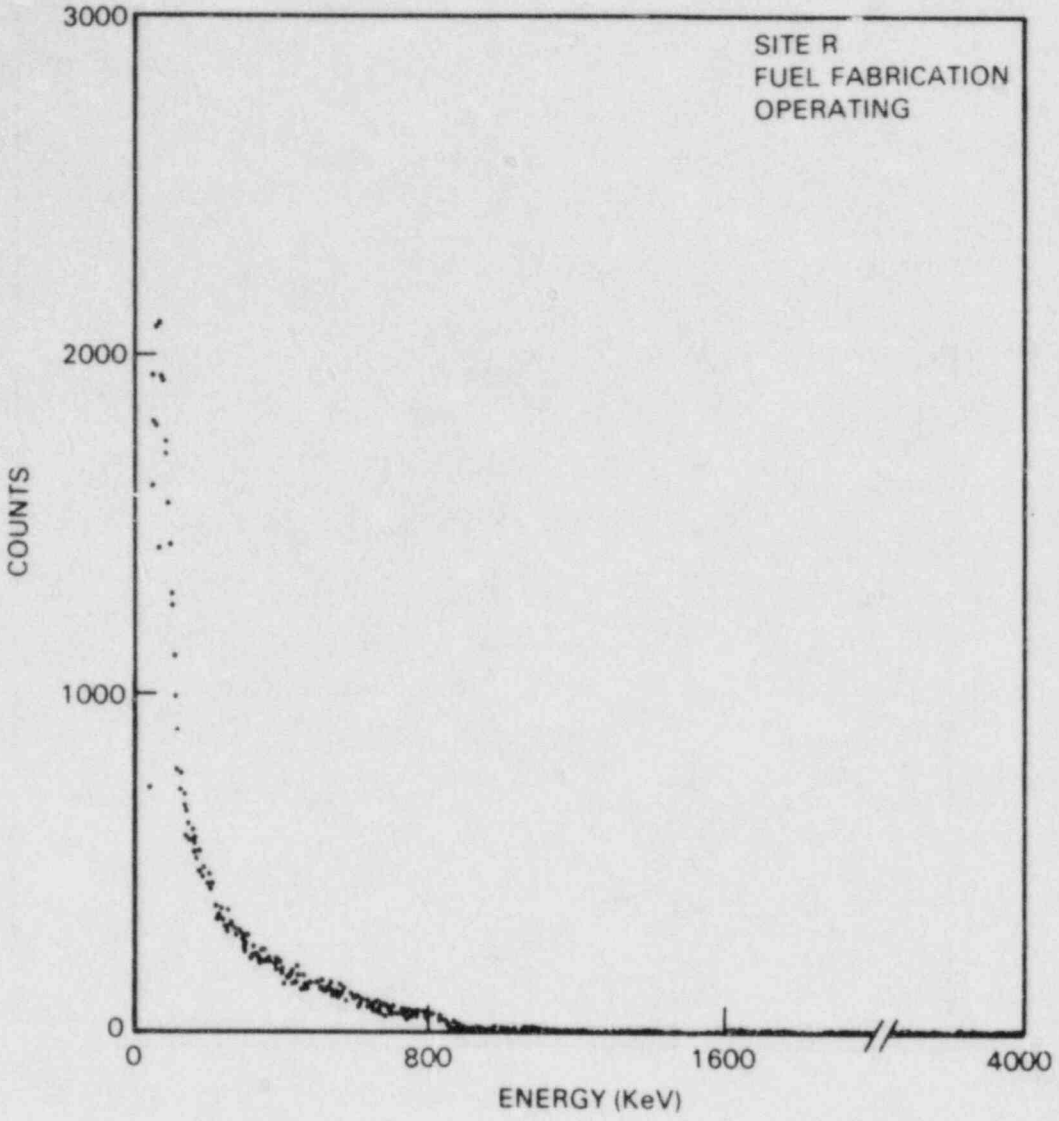

FIGURE 18. Site R, Uranium Spectrum (with 1448 mg/cm^2 shield)

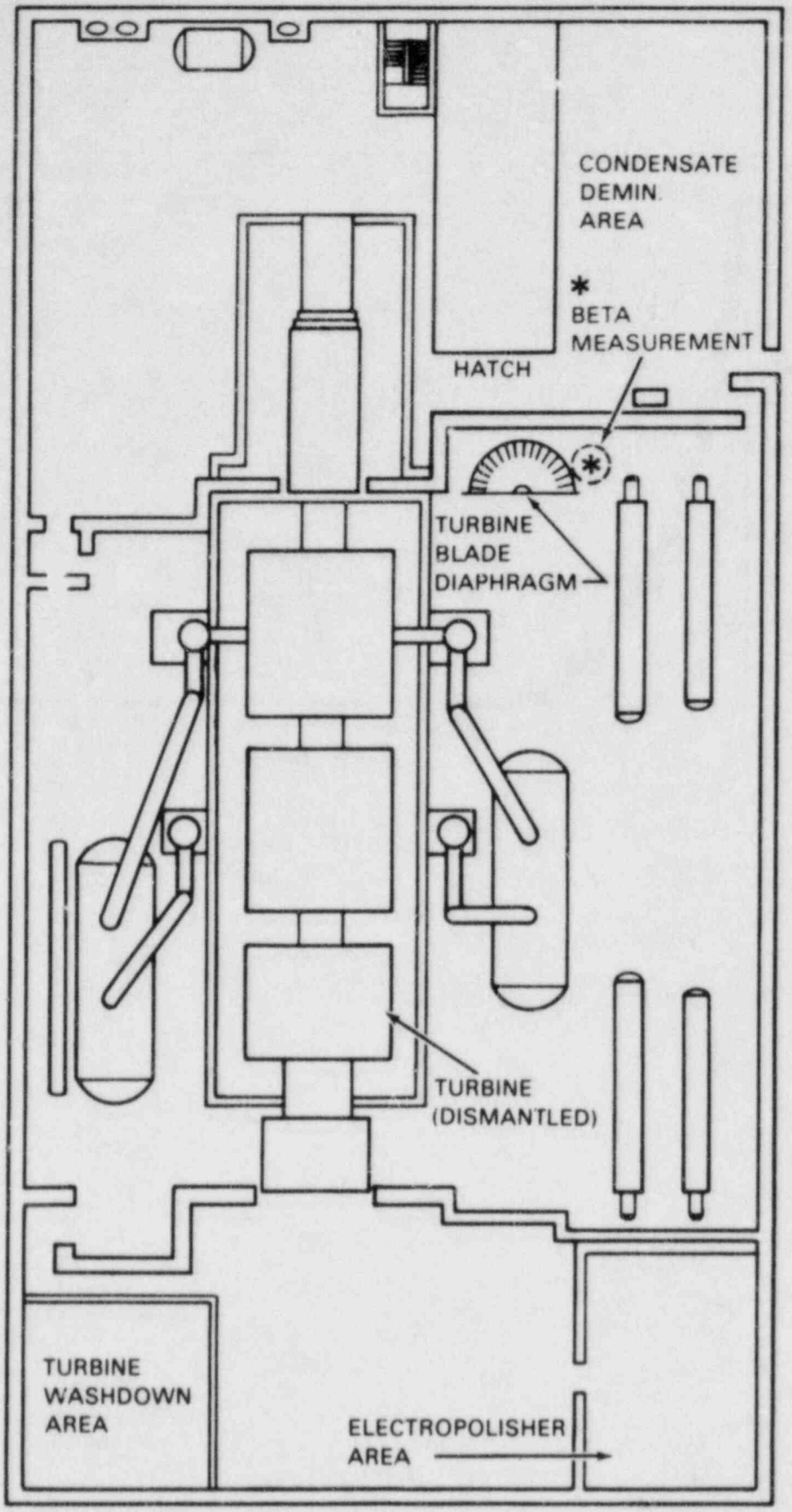

FIGURE 19. Site M, Turbine Room, Shutdown BWR

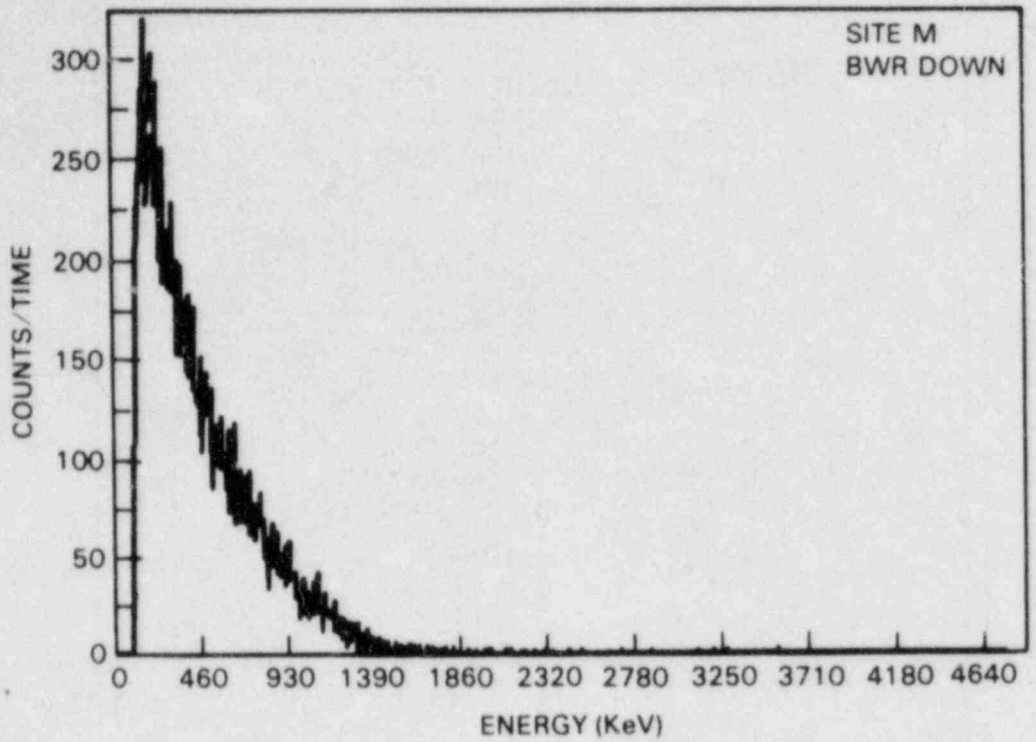

FIGURE 20. Site M Turbine Blade Diaphragm Spectrum

spectrum and beta only spectrum are shown in Figures 21 and 22. Although the subtraction process caused considerable degradation of the low-energy portion of the spectrum, the high-energy portion is useful. The presence of $^{106}Rh/^{106}Ru$ is indicated.

Finally, another older PWR (Site O) was visited. Two areas of possible beta exposure were identified. The first area shown in Figure 23 was a large concrete wall that had become contaminated with ^{137}Cs and ^{60}Co. After soaking into the concrete the ^{137}Cs had leached out again on the surface. The wall had been subsequently covered with paint and other sealants to shield against beta exposures. This was partially successful, as shown by Figure 24. Most of the spectrum is from the Compton scattering of ^{60}Co and ^{137}Cs photons. Only the portion of the spectrum from 450 to 650 keV shows evidence of the ^{137}Cs beta spectrum. The conditions present on the contaminated concrete wall illustrate the potential difficulties of using wipe samples and isotropic analyses to determine beta/gamma ratios.

Beta exposures were also measured at the Site O radioactive waste liquid holdup tank area. The holdup tank location is shown in Figure 25. The concrete floor and a portion of the wall had been contaminated by tank overflows. The spectrum shown in Figure 26 indicates that the contamination is predominately ^{60}Co.

In summary, it should be noted that the primary radionuclide source of beta fields at nuclear power plants is ^{60}Co. Because of the low energy (314 keV maximum) of ^{60}Co beta particles, they are easily shielded. Although the silicon detector is more sensitive to beta particles than to incident photons, the ^{60}Co photon-induced events dominated the spectra in all except four cases for in-containment locations. The spectra that did not show ^{60}Co domination were obtained from areas associated with fuel handling, liquid radioactive waste, and a turbine blade diaphragm. It is understandable that the ^{137}Cs contamination often found with fuel handling operations and the high energy beta emitters in liquid radioactive waste could cause the beta exposure to be higher than the photon exposure, in some cases. Although it is not entirely clear why the turbine blade diaphragm at Site M had a high beta to gamma ratio, it is suspected that a very thin film containing radioactive iodine had plated out on the surfaces of the diaphragm.

The beta fields at fuel fabrication facilities are in sharp contrast to the situation just described for nuclear power plants. At fuel fabrication facilities, each spectrum is clearly due to uranium beta emissions. The bremsstrahlung and other photons are of such low energy and intensity that they are virtually inseparable from the backscatter in the silicon detector.

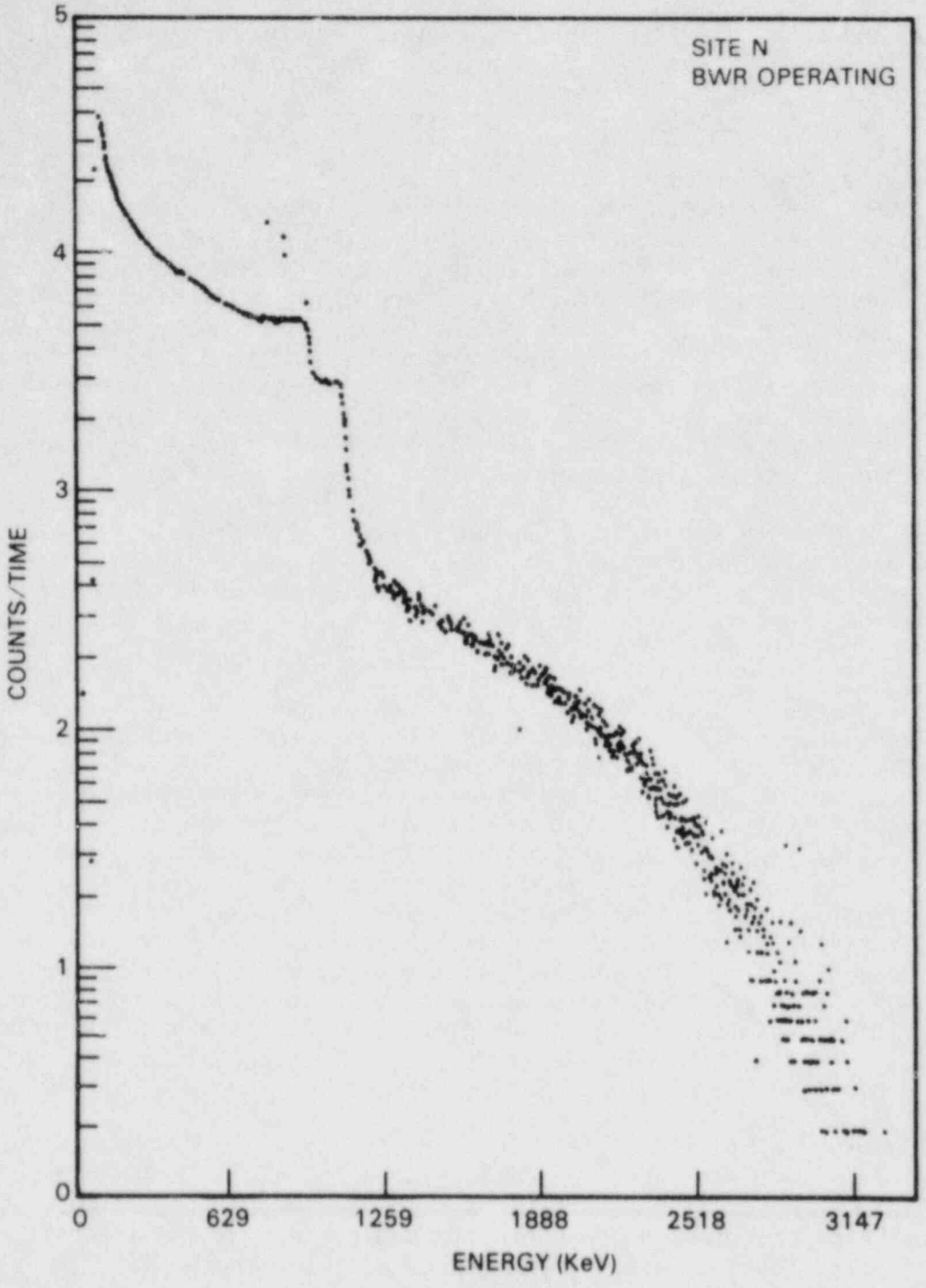

FIGURE 21. Beta-Gamma Spectrum from Cleanup Pipe Section, Site N

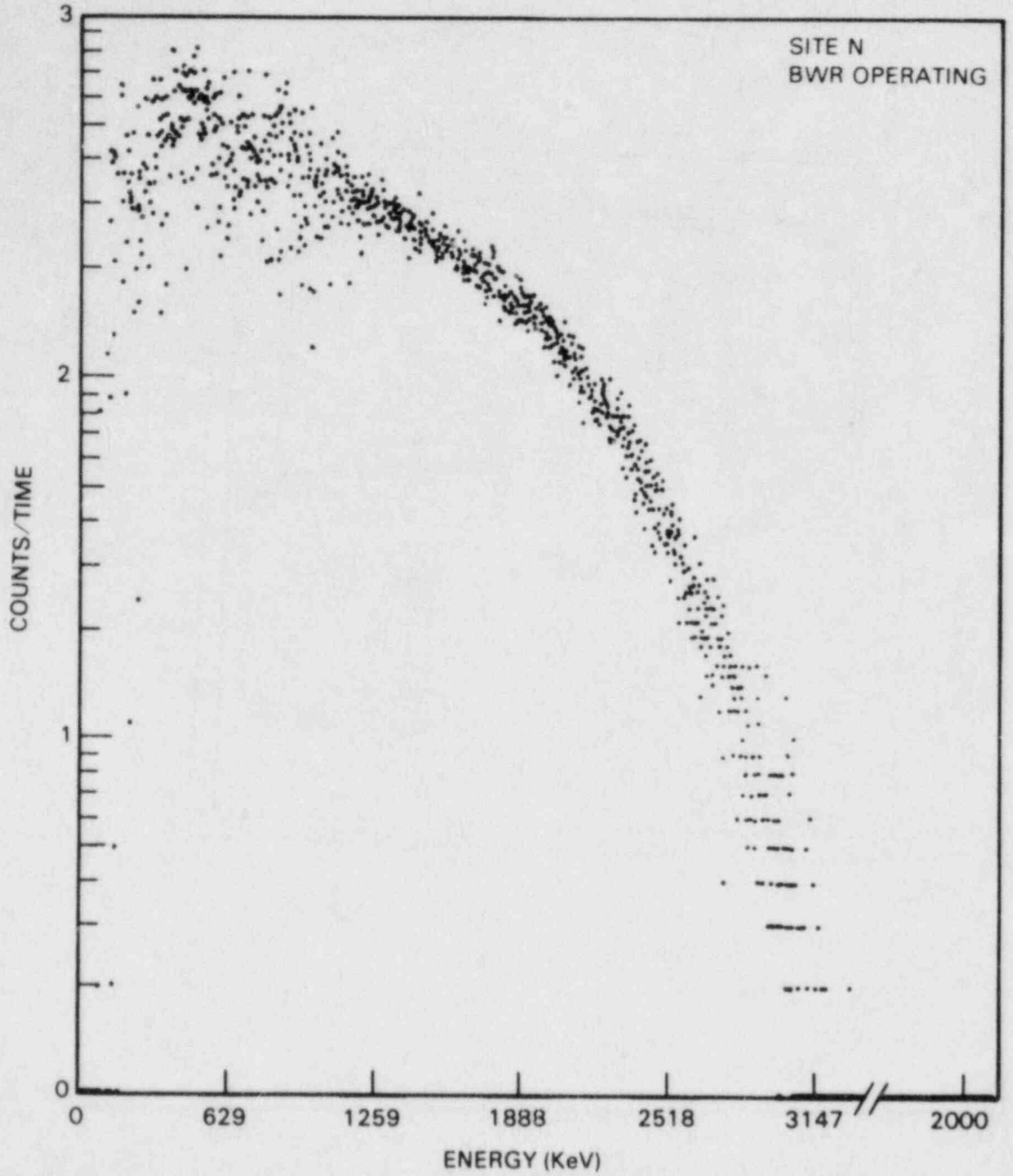

FIGURE 22. Beta-Only Spectrum from Cleanup Pipe Section, Site N

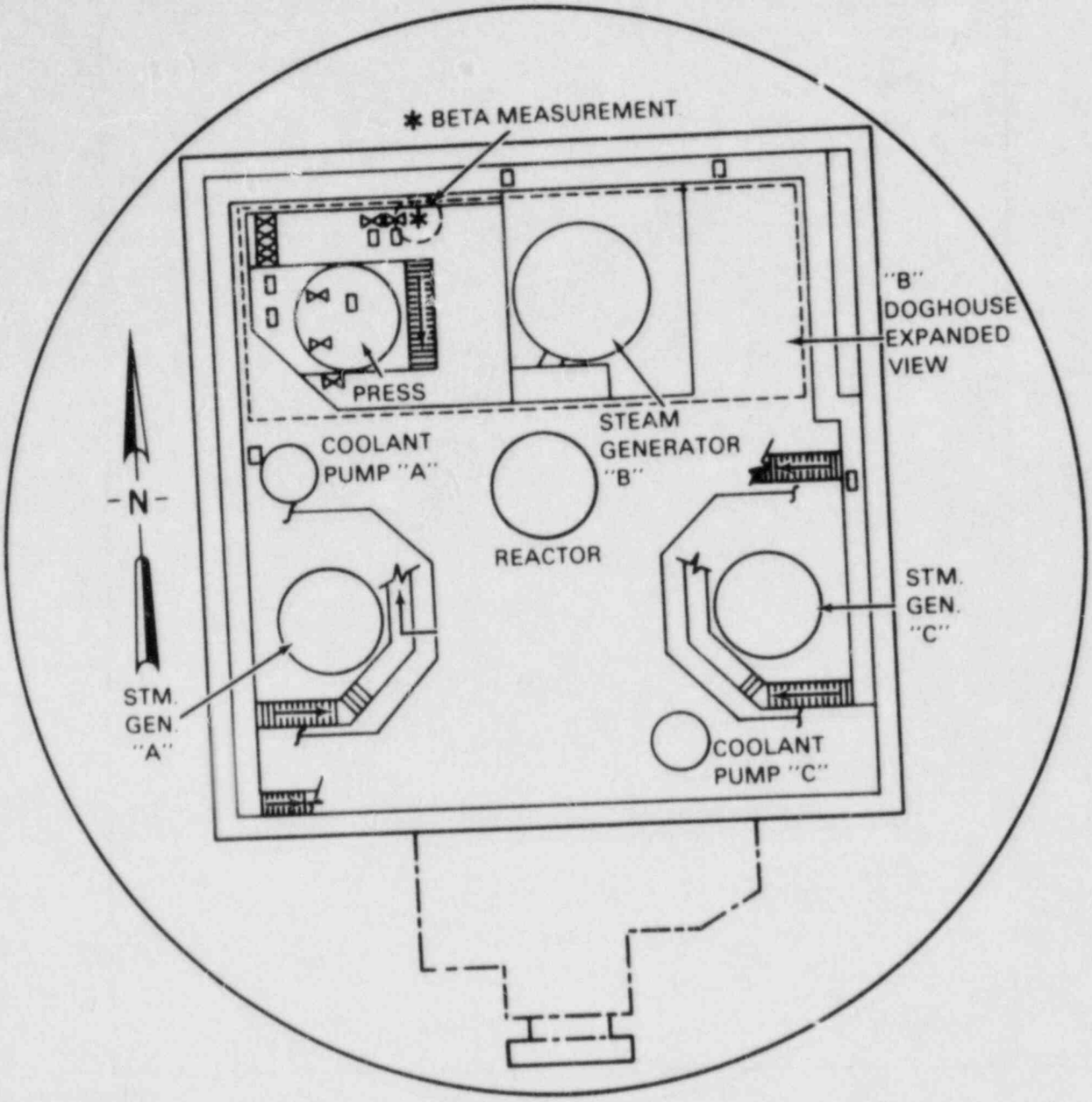

FIGURE 23. Contaminated Concrete Wall, Site 0

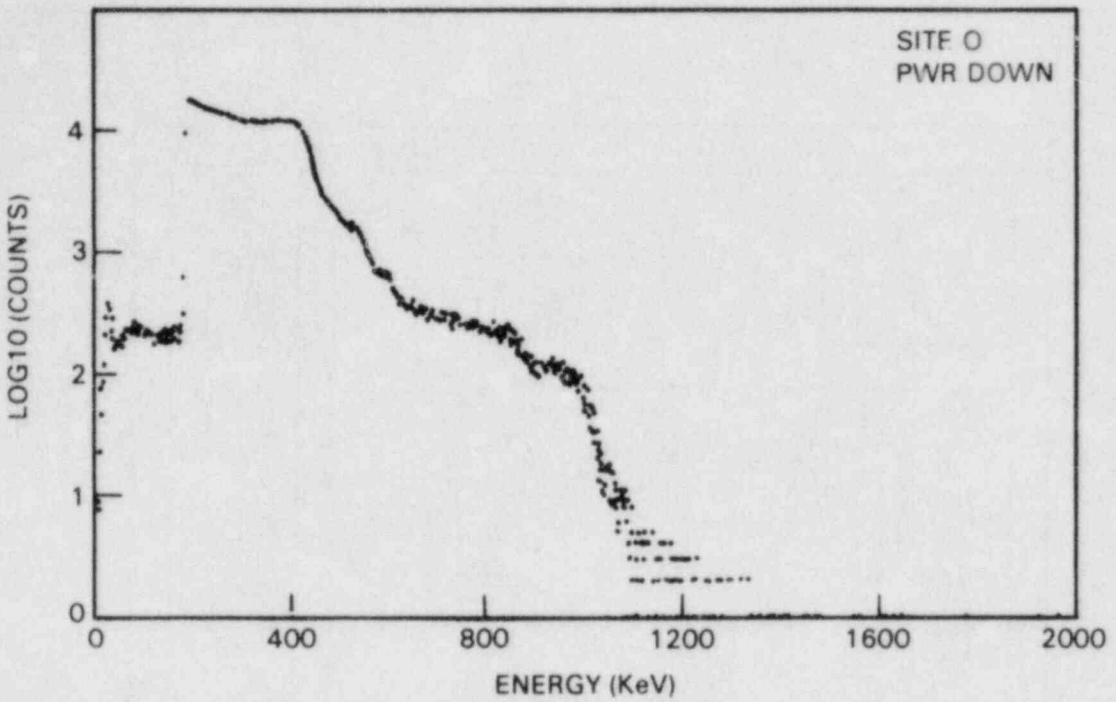

FIGURE 24. Beta + Gamma Spectrum from Contaminated Concrete Wall, Site O

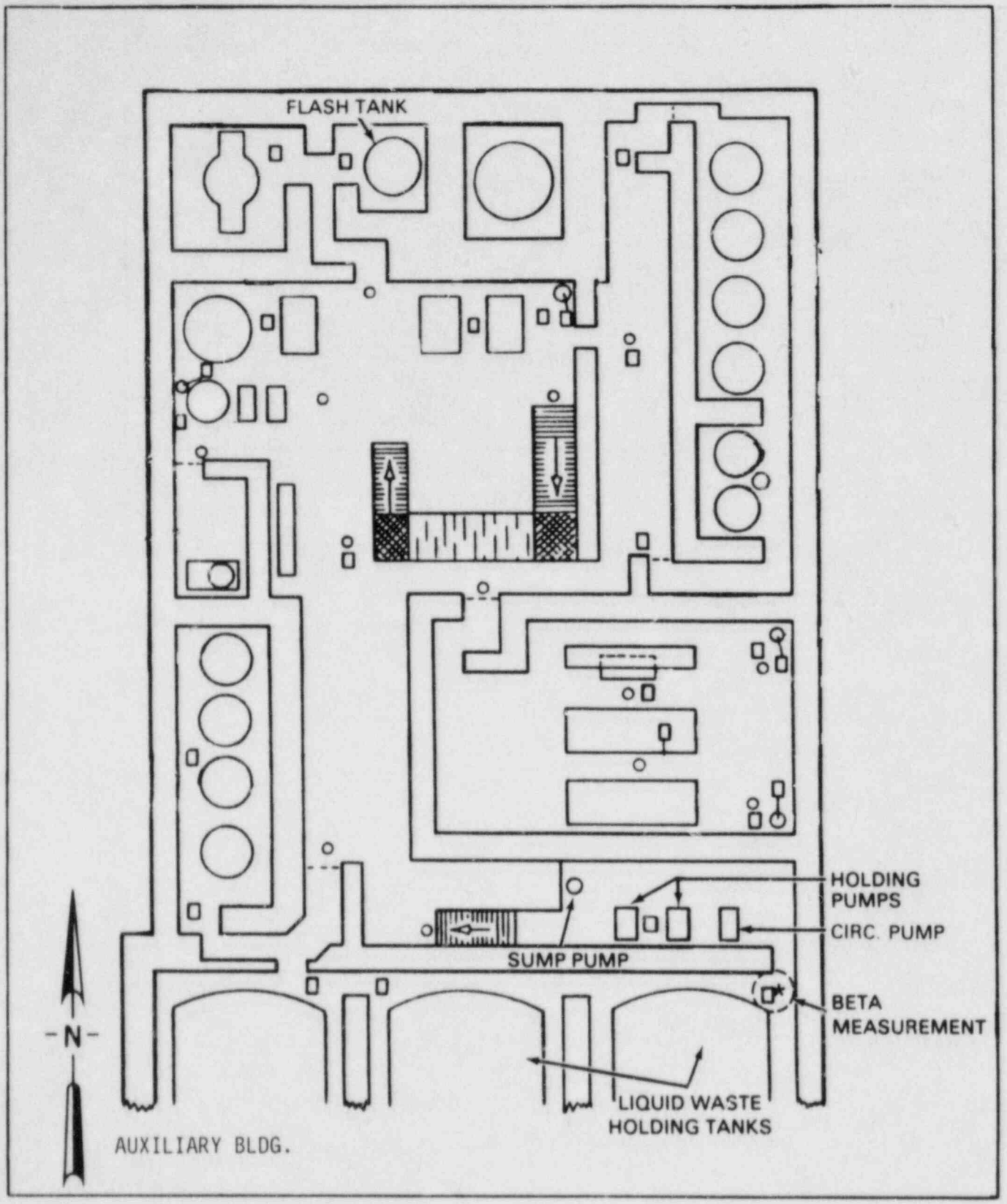

FIGURE 25. Liquid Holdup Tank Area, Site 0

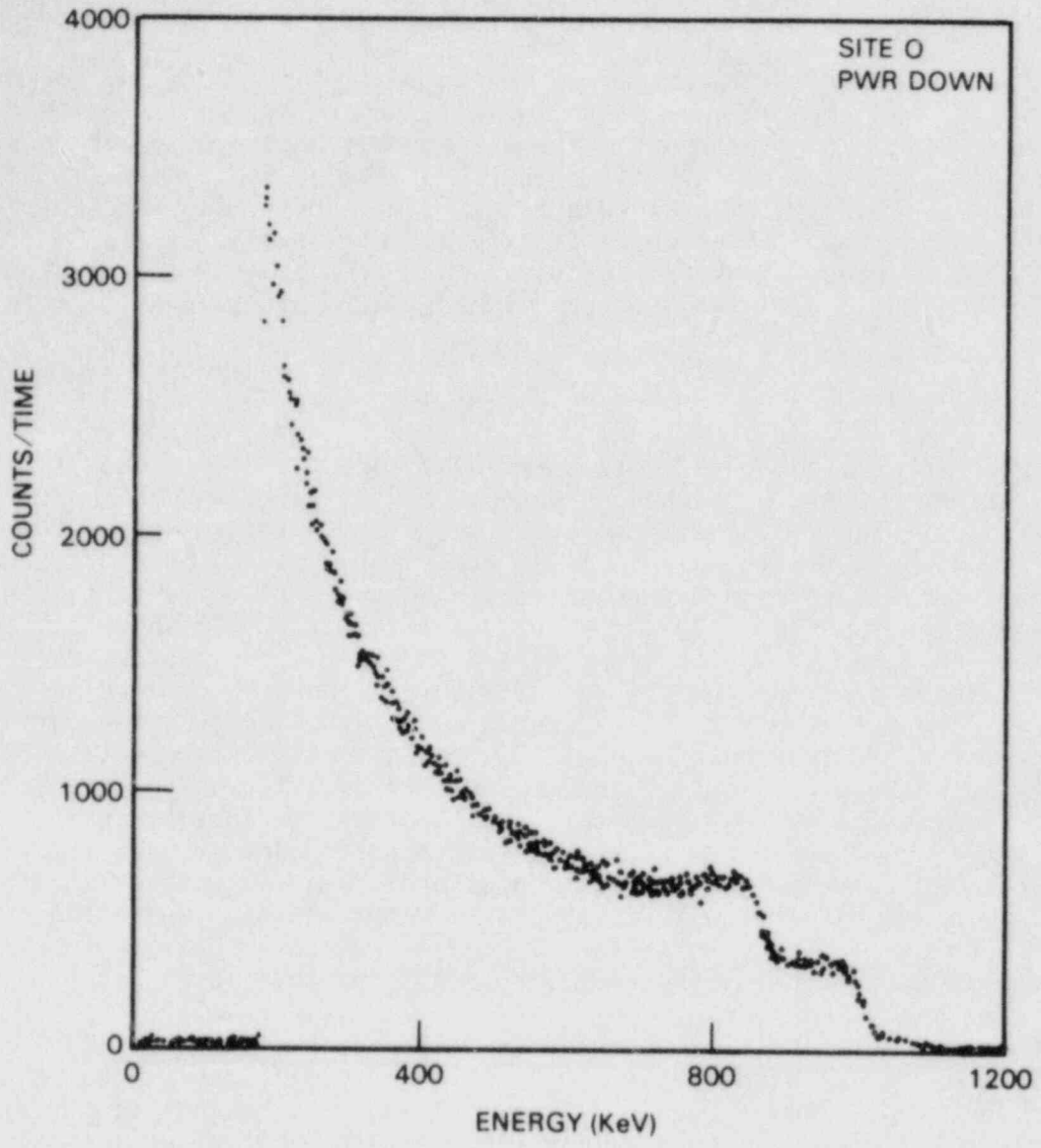

FIGURE 26. Site O, Liquid Holdup Tank Area Spectrum

3.2 MULTIELEMENT DOSIMETER

PNL's multielement dosimeters were used to measure the beta and gamma radiation doses at most of the in-plant locations where silicon detector measurements were made. Several additional multielement dosimeter measurements were made at locations where the silicon detector system could not be used. Table 9 contains a listing of beta dose rates, gamma dose rates, and beta/gamma ratios for most locations where significant beta radiation fields existed. The wide range of beta/gamma ratios found in Table 9 illustrates the fallacy of using beta/gamma ratios with gamma measurements to predict beta dose rates.

3.3 SURVEY METER RESULTS

Following the collection of beta spectra at each field location, the commercial survey meters, described in section 2.2.1, were exposed to the beta and gamma fields. Table 10 lists the results for these instruments. The values listed as beta readings in Table 10 are simply "window open minus window closed" beta dose rate determinations. No calibration factors have been applied. Instruments A, B, and D responded fairly predictably to the uranium slab and other uranium sources listed in Table 10. However, the responses of these same instruments to beta sources found at commercial reactors was much more erratic. Instruments C, E, and F responded erratically in all types of field spectra. Some of the problems with instrument response can be attributed to poor geometry and low dose rates. Thus, the survey instrument results for the Q-2 pump part, the spire bolt, looking-up and over-cavity measurements for P-2, the M-1 fuel handling device, and the turbine floor and pump impeller measurements for N-2 may be suspect. With these latter locations omitted there are still many instances where the survey instrument results are not consistent. On the average, the best calibration factors to correct the survey instrument readings in Table 10 are probably those provided by the contact natural uranium calibrations in Table 5. However, there are a few higher exposure rates cases, such as the beta fields from the Site Q contaminated wheel, the Site P main coolant pump, and the Site M turbine blade diaphragm, where a uranium slab calibration of survey meters would not yield applicable calibration factors. For each of these locations, the beta-gamma ratio was significantly greater than unity and the survey meter measurements were taken close to the source.

When possible, the licensee's own equipment was also used. The response of the licensee survey instruments to beta fields (Table 11) followed the same trend as that of the instruments PNL used. Because survey meter models varied among the different licensees, the responses are likely to be more varied than the responses for a single meter in Table 10. However, it is interesting to note that the survey meters overresponded for the Site M turbine blade diaphgram and Site P main coolant pump beta measurements.

TABLE 9. Beta and Gamma Doses Measured with Multielement Dosimeters

Site	Location	Beta Dose Rate (mrad/hr)	Gamma Dose Rate (mrad/hr)	Beta/Gamma Ratio	Comment
Q-1	Turbine room without wall	5.6	32	0.2	BWR operating
Q-1	Pump room B	1.1	23	0.1	BWR operating
Q-1	Fuel pool chute	18	3	6.2	BWR operating
Q-2	Contaminated steel wheel	28	18	1.5	BWR down
Q-2	Contaminated light	198	153	1.3	BWR down
P-1	Fuel Transfer crane hook	2.2	6.5	0.33	PWR down
P-2	Spare main coolant pump	212	153	1.4	PWR operating
M-1	Turbine blade diaphragm	15	0.06	>200	BWR down
N-1	Fuel handling device	1.7	18	0.1	BWR down
N-1	Clean up pipe	834	313	2.7	BWR operating
N-2	Low pressure heater bay	7,880	40,100	0.2	BWR operating
O-1	Contaminated concrete wall	116	65	1.8	PWR down
O-1	Liquid waste holdup area	63	99	0.6	PWR down
O-2	Contaminated concrete wall	71	105	0.7	PWR down
R-1	Bulletin board on pellet rack	2.9	0.63	4.7	Fuel Fab. operating
R-1	Under #3 furnace boat rack	23	0.63	37	Fuel Fab. operating
R-2	Pellet boat storage FE-1	44	1.1	42	Fuel Fab. operating
R-2	Pellet boat storage RP-3	20	1.0	24	Fuel Fab. operating

TABLE 10. Response of Commercial Survey Instruments to Beta Fields at Licensee Facilities

Site	Location	Beta Dose (mrad/hr)	Survey Instruments Beta Readings (mrad/hr) A	B	C	D	E	F	Window to Source Distance	Comment
Q-2	Uranium slab in laboratory	152	55	560	68		37	17	3 in.	BWR operating
Q-2	Refuel floor contaminated wheel	28	100	20	8		5	80	3 in.	BWR down
Q-2	Refuel floor contaminated light	197	50	-	0		10	0	3 in.	BWR down
Q-2	Pump part in machine shop	0.1	1	1.5	0.2	0	0	0.2	1 in.	BWR down, outside containment
P-2	Top floor looking up	4.6	4	10	0.6		3	1	Unknown	PWR operating
P-2	Main coolant pump	212	1400	>1000	160	820		245	6 in.	PWR operating
P-2	Top floor over cavity	2.3	10	5	3.6	13	2	11	20 ft	PWR operating
R-1	Uranium pellet boat	200	69	420	38	59			3 in.	Fuel Fab. Facility operating
R-1	Uranium pellet tray	210		400	465	76		70	3 in.	Fuel Fab. Facility operating
R-2	Uranium powder bucket 3	67	20	155	13	24		9	9 in.	Fuel Fab. Facility operating
R-2	Uranium powder bucket 4	97	32	184	23	34		15	7 in.	Fuel Fab. Facility operating
R-2	Uranium pellet boat 3	230	74	530	45	45		29	3 in.	Fuel Fab. Facility operating
R-2	Uranium pellet boat 4	155	46	400	28	68		17	3 in.	Fuel Fab. Facility operating
M-1	Turbine blade diaphragm	15		95					3 in.	BWR down
M-1	Fuel handling device	6.7		20					3 in.	BWR down
N-2	Turbine floor	4.0	0	0	0	0	0		10 ft	BWR operating
N-2	Low pressure heat bay	59	0	0	0	30	0	0	20 ft	BWR operating
N-2	MSIV	108	50		20	20		50	4 ft	BWR operating
N-2	Pump impeller	120	25	5	3.5	13	7	2	16 in.	BWR Operating
O-2	Contaminated concrete wall	116[a]	480		60			205	3 in.	PWR down
O-2	Liquid waste area	63[a]	20		0			2	6 in.	PWR down

(a) The reference beta dose rates for Site O are taken from the first trip rather than the second, because problems with the multielement dosimeters made the measured doses suspect on the second trip.

TABLE 11. Response of Licensee Survey Instruments to Beta Fields

Site	Location	Beta Dose Rate (mrad/hr)	Survey Meter (mrad/hr)	Window to Source Distance	Comment
O-1	Turbine floor with concrete shield	0	0	10 ft	BWR operating
O-1	Turbine floor without concrete shield	5.6	2	15 ft	BWR operating
O-1	Pump room B	11	10	5 in.	BWR operating
O-1	Shroud near fuel pool 4th floor	18	12	4 in.	BWR operating
O-2	Refuel floor contaminated wheel	28	2	3 in.	BWR down
O-2	Refuel floor contaminated light	198	50	3 in.	BWR down
O-2	Pump part in machine shop	0.1	0.5	1 in.	BWR down
P-2	Top floor looking up	4.6	2	Unknown	PWR operating
P-2	Main coolant pump	212	1060	6 in.	PWR operating
P-2	Top floor over cavity	2.3	6	20 ft	PWR operating
R-1	Uranium powder bucket 1	28	6.7	7 in.	Fuel Fab. Facility operating
R-1	Uranium pellet boat	200	80	3 in.	Fuel Fab. Facility operating
R-2	Uranium powder bucket 3	67	14	9 in.	Fuel Fab. Facility operating
R-2	Uranium powder bucket 4	98	26	7 in.	Fuel Fab. Facility operating
R-2	Uranium pellet boat 3	230	63	3 in.	Fuel Fab. Facility operating
R-2	Uranium pellet boat 4	155	31	3 in.	Fuel Fab. Facility operating
M-1	Turbine blade diaphragm	15	38	3 in.	BWR down
N-2	MSIV	108	50	4 ft	BWR operating

3.4 COMMERCIAL DOSIMETERS

For beta exposures at NRC-licensed facilities, all except one of the commercial dosimeters used in this study typically responded low, as seen in Table 12. However, at Site P the in-containment dosimeter exposures caused several dosimeters to respond high. This apparent overresponse is very likely due to the presence of a significant neutron flux noted in containment with the reactor at power. Many of the dosimeters also responded high to beta fields at Site O. Because of the considerable spatial variation in beta doses at both Site O measurement locations, it is probable that the commercial dosimeters were not exposed to a beta field comparable to that experienced by the PNL multielement dosimeter, even though they were separated by only a few inches. An example of this spatial variation is seen in the comparison between the contaminated concrete exposures A and B, for O-2. Here the two multielement dosimeters were separated by only a few inches, but the beta exposures differ by over 1000 mrad. Dosimeter H typically responded higher than the other dosimeters at those locations where the dosimeter H response was non-zero. However, there were numerous cases where each of the dosimeters had a zero response. The zero responses were especially prevalent at doses below 600 mrad.

TABLE 12. Response of Commercial Dosimeters to Beta Radiation at Licensee Facilities

Site	Location	Beta Exposure (mrad)	A (mrad)	B (mrad)	C (mrad)	D (mrad)	E (mrad)	F (mrad)	G (mrad)	H (mrad)
Q-1	Turbine floor, near turbine	1200	0	(a)						
Q-2	Uranium slab	100	0	636	76		30	0	0	0
Q-2	Contaminated wheel	500	270	300	65	0	250	0	0	1650
Q-2	Contaminated light	9,910	0	8,772	0		0	1,350	120	
P-1	Top floor, fuel trough	0	535	0						
P-2	Top floor, looking up	560	1,298	0	0		90	0	630	0
P-2	Top floor, edge of cavity	280	627							0
P-2	Main coolant pump	25,000	4,000	0	9,600	0	80	1,750	5,600	15,900
M	Turbine blade diaphragm	370	0	0						
N-1	Cleanup pipe	20,000	208	495						
N-1	MSIV	270	0	0						
N-2	Turbine floor, near turbine	20	0	0	0	0	0	0	120	
N-2	MSIV	220	0	0	0		0	220	240	0
N-2	Low pressure heater bay	130	0	0	0		0	210	240	0
O-1	Contaminated concrete wall	2,600	3,568	8,772	635		1,530	2,600	630	
O-1	Liquid waste hold up tank area (A)	40		732	151		810	1,320	1.050	
O-1	Liquid waste hold up tank area (B)	170	500	0	0		1,055	1,760	865	
O-1	Liquid waste hold up tank area (C)	450	92		0		1,280	1,850	1,200	
O-2	Liquid waste hold up tank area, average	230	0	0	110		0	81	74	38
O-2	Contaminated concrete (A)	1,800	3,013	0	804	608	540	0	1,300	4,900
O-2	Contaminated concrete (B)	750	1,500	3,195	0	0	905	1,500	1,500	2,000
R-2	Pellet boat storage 1-FE	2,100	708	440	592		35	30	1,300	
R-2	Pellet boat storage 3-FE	690	226	222	262		20	20	250	
R-2	Pellet boat storage RP-3	940	376	365	271		20	30	275	
R-2	Pellet boat storage RP-1	1,460	94	48	30		20	30	40	

(a) No data indicates that the dosimeter was not exposed.

4.0 CONCLUSIONS

4.1 BETA EXPOSURE AREAS

Both the spectra obtained with a silicon detector and the doses measured with multielement dosimeters indicate that significant beta radiation fields exist in NRC-licensed facilities. Although many of the beta fields at the PWRs and BWRs visited were due to the presence of ^{60}Co, important exceptions were found.

The potential exposures of personnel to ^{60}Co beta fields are greatly reduced by the health physics practices commonly employed at licensee facilities. These practices include wearing protective clothing, gloves, and face gear, as well as skin decontamination. However, certain activities may expose personnel to higher energy beta fields at power reactors. These include activities associated with removing spent fuel and other core components, and maintenance activities involving the opening of cleanup system piping, valves, pumps, and, in BWRs, steam turbine components. There are, no doubt, other sources of significant beta exposures not found in this study. These other beta exposure sources may include steam generators and exposure to radioactive gases. In contrast, the source of exposure at fuel fabrication facilities is known to be a high energy beta emitter. Thus, individuals in the vicinity of the fuel material are subject to beta exposure.

4.2 SURVEY METERS

Two methods are commonly employed to calibrate beta survey meters. In the first, the survey meter is placed in contact with a uranium slab (covered by a 7 mg/cm^2 film) and the survey meter measurement is compared with the known surface exposure rate (approximately 225 mrad/hr) to derive a calibration factor. Typical calibration factors range from 2 to 4. The second calibration method involves using small calibrated beta sources at a specified distance (20 to 50 cm) from the survey meter. The known beta exposure at the specified distance is used to derive a calibration factor.

In a common check procedure, the survey meter is calibrated as just described. A small check source is then placed in contact with the instrument's window and the reading is recorded for future reference. This check procedure is not a calibration, but is useful to check general instrument performance.

It appears that many licensees use the first method when they receive an instrument and later rely on the check source reading procedure. However, one licensee determined the calibration factor by calculating the exposure at the center of the survey meter for a survey meter sitting on a uranium slab. This would, of course, result in a smaller calibration factor. Hooker et al. (1983) have previously noted that some survey meter users apparently assume that the calibration factor is unity. This study has shown that it is important to use the type of survey meter and calibration geometry that will best approximate the conditions found in the expected field applications. Thus, if the source of exposure is a single high-energy beta emitter with

relatively large areal extent, such as a tray of uranium pellets, an accurate calibration factor can be determined using a uranium slab. It does not matter if the window on the survey meter's detection chamber is relatively thick (i.e., 70 mg/cm^2) if it is known that all doses will be from uranium.

The uranium slab can also be used to calibrate instruments that are to be used to survey ^{60}Co contamination. However, this can lead to conservative results for thin-window instruments. A more accurate calibration would be obtained with a calibrated ^{147}Pm source at a distance of 20 cm. Survey meters with windows thicker than approximately 20 mg/cm^2 are difficult to calibrate for ^{60}Co beta fields.

The problem with beta survey meters seems to be not so much in their calibration as in their use. For example, some survey meters have beta windows on both the end and sides. If such an instrument were calibrated for use with the end window open, and the user measured the beta field with the side window open, the measured beta exposure rate could differ from the true exposure rate by a factor of 2 or more. More importantly, the beta exposure rate changes very rapidly in the vicinity of small sources. Hence, moving the survey meter 1 to 2 in. farther than intended from the source can result in an erroneously low reading. Such a change in the position of the survey meter can happen easily in field measurements. In other instances, such as when maintenance personnel are working quickly in cramped quarters, it is not feasible to make frequent survey meter measurements. Finally, there is no way that a survey meter can measure the dose to a person who picks up a small object contaminated with a beta emitter, or places his hand inside a contaminated valve.

4.3 DOSIMETERS

Short-term beta exposures that cannot be adequately measured with a survey meter and all long-term beta exposures are commonly measured with either film badge or thermoluminescent dosimeters. Many of these have only two elements, one of which is under a thin window intended to admit both beta and gamma radiation. Of most dosimeters, it is assumed that the window for the second element stops all beta radiation. In practice, the two-element dosimeters yield good results only for the type of beta field in which they have been calibrated. This calibration is usually performed with a relatively high-energy beta source, such as ^{90}Sr/^{90}Y. Therefore, the dosimeters respond low to ^{60}Co beta fields. In this study, the response to uranium beta fields was also low for the dosimeters tested. If the so-called "thin" dosimeter window is too thick, the dosimeter will not respond to low energy beta particles at all. The one commercial four-element dosimeter used in this study typically responded high by a factor of approximately three, except where its response was zero.

5.0 RECOMMENDATIONS

5.1 GOOD PRACTICES

This study has helped to identify the kinds of locations in nuclear power plants where beta radiation fields may be found. Because of differences in ages, designs, and secondary water chemistry, different plants will likely have considerable variations in beta dose rates. Therefore, the licensee should identify the areas of possible beta exposures at his plant. It is not necessary for the licensee to conduct a study separate from other radiation monitoring efforts. It should be adequate to simply ensure that the worker performing radiation surveys has the proper type of survey meter and that he takes appropriate measurements. Special care should be taken when cleanup pipes and valves are opened so that potential areas of high beta exposure are surveyed immediately after the opening is made. Once a significant beta radiation field is identified, it should be characterized through the use of multielement dosimeters or other spectrometers. This will be helpful in predicting doses to individuals. All personnel who work in significant beta fields should wear dosimeters, including extremity dosimeters, that can accurately measure the beta dose received. This may require very thin dosimeters or a multiple filter arrangement for conventional dosimeters.

The multielement dosimeter (Scherpelz et al. 1983) used as a passive spectrometer in this study seems to be sufficiently accurate for personnel dosimetry. However, the size of this dosimeter may be less convenient for the wearer than those now in use. Until a more suitable dosimeter can be perfected, the dosimeter user should characterize his dosimeters in laboratory beta fields representative of the beta fields he normally encounters. A correction factor determined in this manner should be applied to personnel dosimeter readings. If the dosimeter does not respond to low-energy beta fields, a thinner dosimeter window or different TLD material may be required.

5.2 SURVEY METER SELECTION AND USE

The beta survey meters used should be reliable, durable, and easy to read. They should be nearly energy-independent for beta energies between 70 keV and 4 MeV. Ion chamber instruments should be calibrated in contact with a uranium slab and 20 cm from a calibrated ^{147}Pm source. Geiger-Mueller instruments should be calibrated only with the isotope that they will monitor and only in the configuration where they will be used. Almost any ion chamber instrument with a beta window of 70 mg/cm^2 or less can be used to measure large sources consisting of a single high-energy beta emitter. However, for small area or low-energy (<500 keV endpoint) beta emitters, the detection chamber should be as shallow as possible (consistent with effective signal production) and covered by a window of 7 mg/cm^2 density thickness. Among commercially available survey meters, instrument A most nearly met these specifications. A modified CP survey meter (Roesch and Donaldson 1955) should provide even better results. Although not commercially available, this instrument would not be difficult to fabricate. Another investigator (Hankins 1982b) has studied the response of a beta survey meter which may be useful at

high dose rates. The study reported here did not include the latter type of instrument. All the commercial instruments except D required both an "open window" and "closed window" measurement to determine beta dose rates in mixed fields. In some instances, an accurate beta dose rate could be read directly from instrument D, but the open window vs. closed window procedure often improved results. Maintenance problems with instrument D limited the number of measurements made with this instrument. The protoype INEL dose rate meter or other direct reading survey meters based on the thin scintillator concept may be the beta survey meters of choice in the near future.

The survey meter user should be properly trained in making beta radiation surveys. The training should include supervised field and laboratory measurements of single-point and spattered points of beta contamination, uniform and nonuniform but continuous areal sources, flat surface sources, small contaminated objects, the inside of a contaminated pipe and, most importantly, mixed beta-gamma sources. It is particularly important that the instrument user understand that beta measurements made very close to small sources require very large correction factors and may be highly inaccurate.

Beta-gamma ratios are highly variable in commercial reactor environments and should not be used to determine beta exposures. Because of differential leaching and adhesion, surface wipes and isotopic analyses are of limited value in predicting beta dose rates.

6.0 REFERENCES

Dudley, R. A. 1951. "Measurement of Beta Radiation Dosage with Photographic Emulsions." Ph.D. Thesis, Massachusetts Institute of Technology.

Endres, G. W. R., J. M. Aldrich, L. W. Brackenbush, L. G. Faust, and R. V. Griffith and D. E. Hankins. 1981. Neutron Dosimetry at Commercial Nuclear Plants. Final Report of Subtask A: Reactor Containment Measurements. NUREG/CR-1769, PNL-3585, Pacific Northwest Laboratory, Richland, Washington.

Evans, R. O. 1955. The Atomic Nucleus. McGraw-Hill.

Fox, R. J., and C. J. Borkowski. 1962. "Silicon Surface Barrier Detectors with High Reverse Breakdown Voltages." IRE Trans. Nucl. Sci., NS-9, No. 3, 213.

Hankins, D. E. 1982a. Evaluation of Beta Energy (E_{max}) and Spectral Type Using Survey Instruments. UCRL-88275, Lawrence Livermore National Laboratory, Livermore, California.

Hankins, D. E. 1982b. Beta-Energy Response of the Eberline RO-7 Survey Instrument. UCID-19484, Lawrence Livermore National Laboratory, Livermore, California.

Hooker, C. D., D. E. Hadlock, W. N. Herrington and R. L. Gilchrist. 1983. Interim Calibration and Field Use Guidance for Beta Portable Survey Instruments. PNL-4378, Pacific Northwest Laboratory, Richland, Washington.

Johnson, L. C., J. L. Alvarez and J. M. Hoggan. 1982. "Pulse Shape Discrimination in a Portable Beta-Gamma Dose Rate Meter." IRE Trans. Nucl. Sci., NS-29, No. 5.

Roberson, P. L., G. W. R. Endres, R. A. Fox, D. L. Haggard, K. L. Holbrook and L. A. Rathbun. 1982. "Spectra Dosimetry for High and Low Energy Photons." PNL-SA-10743. Presented at the 10th Water Reactor Safety Research Information Meeting.

Roesch, W. C., and E. E. Donaldson. 1955. "Portable Instruments for Beta-Ray Dosimetry." In: Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Volume 14, General Aspects of the Use of Radioactive Isotopes: Dosimetry. Geneva, August 8-20, 1955.

Scherpelz, R. I., G. W. R. Endres and L. A. Rathbun. 1983. The Use of Multi-Element Beta Dosimeters for Measuring Dose Rates in the TMI-2 Containment Building. PNL-4714, Pacific Northwest Laboratory, Richland, Washington.

Sherbini, S., and S. W. Porter. 1983. A Review of the Current Deficiencies in Personnel Beta Dosmetry, With Recommendations. NUREG/CR-3296, PCI-TR-224, U.S. Nuclear Regulatory Commission, Washington, D.C.

Simons, G. G., T. M. DeBey, R. B. Stuewe and K. D. Stansbury. 1982. "Beta Dosimetry and Spectroscopy, Annual Report." Kansas State University, Department of Nuclear Engineering, Manhattan, Kansas.

DISTRIBUTION

No. of Copies

OFFSITE

U.S. Nuclear Regulatory Commission
Division of Technical Information and Document Control
7920 Norfold Avenue
Bethesda, MD 20014

R. E. Alexander
U.S. Nuclear Regulatory Commission
Office of Radiation Protection
NL-5650
Rockville, MD 20852

2 F. M. Costello
U.S. Nuclear Regulatory Commission
Region 1
631 Park Avenue
King of Prussia, PA 19406

Margaret Federline
U.S. Nuclear Regulatory Commission
Office of Radiation Protection
NL-5650
Rockville, MD 20852

20 R. B. Neel
U.S. Nuclear Regulatory Commission
Office of Radiation Protection
NL-5650
Rockville, MD 20852

M. Erlich
U.S. National Bureau of Standards
Quince Orchard Road
Gaithersburg, MD 20760

No. of Copies

F. Hajnal
U.S. National Bureau of Standards
Quince Orchard Road
Gaithersburg, MD 20760

E. J. Vallario
DOES
Department of Energy
Washington, DC 20545

T. F. Gesell
Radiological and Environmental Laboratory
U.S. Department of Energy
550 Second Street
Idaho Falls, ID 83401

V. P. Gupta
Idaho National Engineering Laboratory
EG&G Idaho Inc.
P.O. Box 1625
Idaho Falls, ID 83415

D. E. Hankins
Hazards Control Department
Lawrence Livermore National Laboratory
P.O. Box 5505
Livermore, CA 94550

B. L. Rich
EG&G Idaho Inc.
P.O. Box 1625
Idaho Falls, ID 83415

No. of
Copies

ONSITE

44 Pacific Northwest Laboratory

L. W. Brackenbush
F. M. Cummings
G. W. R. Endres
L. G. Faust
J. J. Fix
R. A. Fox
R. T. Hadley
D. L. Haggard
G. R. Hoenes
C. D. Hooker
P. O. Jackson
J. J. Jech
J. L. Kenoyer
J. C. McDonald
L. F. Munson
L. A. Rathbun (15)
W. D. Reece
P. L. Roberson
W. C. Roesch
R. I. Scherpelz
J. M. Selby
K. L. Soldat
K. L. Swinth
Technical Information (5)
Publishing Coordination (2)

NRC FORM 335 (11 81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG/CR-3844
PNL-4886

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Beta Particle Measurement and Dosimetry Requirements at NRC-Licensed Facilities

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S) L.A. Rathbun, G.W.R Endres, R.A. Fox, P.L. Roberson, R.I. Scherpelz

5. DATE REPORT COMPLETED
MONTH February YEAR 1984

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

DATE REPORT ISSUED
MONTH August YEAR 1984

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Facility Operations
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. PROJECT TASK/WORK UNIT NO.

11. FIN NO.
B2448

13. TYPE OF REPORT
Research Report

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

Researchers from Pacific Northwest Laboratory (PNL) have conducted beta radiation measurements under laboratory and field conditions to assess the degree of the measurement problem and offer suggestions for possible remedies. The primary measurement systems selected for use in this study were the silicon (Si) surface barrier spectrometer system and the multielement beta dosimeter. Three boiling water reactors (BWRs), two pressurized water reactors (PWRs), and one fuel fabrication facility were visited during the course of the study. Although beta fields from cobalt-60 were the most common type found at commercial reactor facilities, higher energy beta fields were found at locations associated with spent fuel handling, liquid radioactive waste, and BWR turbine components. Commercially-available dosimeters and survey instruments were used to measure the same laboratory and licensee facility beta fields characterized with PNL's active and passive spectrometers. A prototype survey meter was also used in the laboratory measurements. The commercial instruments and dosimeters used in this study typically responded low to the beta fields measured, especially where maximum beta energies were less than approximately 500 keV. A single calibration factor is usually not adequate for either beta dosimeters or instruments. There is a need for more refinement in beta measurement devices and training for the users of such devices.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a DESCRIPTORS

beta particles, dosimetry
radiation measurement, beta spectrometers

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
unclassified

21. NO. OF PAGES

20. SECURITY CLASS (This page)
unclassified

22. PRICE
$

NRC FORM 335 (11 81)

120555078877 1 1AN
US NRC
ADM-DIV OF TIDC
POLICY & PUB MGT BR-PDR NUREG
W-501
WASHINGTON DC 20555

NUREG/CR-3544

BETA PARTICLE MEASUREMENT AND DOSIMETRY REQUIREMENTS
AT NRC-LICENSED FACILITIES

AUGUST 1984