REPORT

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for the

ANNUAL SURVEY OF THE RETIRED
PIQUA NUCLEAR POWER FACILITY

to

THE UNITED STATES DEPARTMENT OF ENERGY

October 1988

by

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prepared by

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MASTER

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TABLE OF CONTENTS

	<u>Page</u>
INTRODUCTION	1
Surface Contamination	2
Smear Survey	2
Radiation Surveys.	2
Sump Water and Sludge Samples	10
Sump Water	10
Facility Tap Water	10
Sump Solids.	10
Visual Inspection	10
DISCUSSION AND RESULTS	12
Surface Contamination	12
Sump Water and Sludge Samples	12
Comparison of Concentrations of Activity of Water in the Sump with the Concentration Guides Listed in DOE Order 5480.1 Chapter XI	15
Visual Inspection	18
Photographic Services	18
Description of Sump Systems	18
Cathodic Protection System	18
Fire Protection	18
Other Safety-Related Systems.	23
CONCLUSIONS.	23
EXHIBIT I	
SURVEY RESULTS: PNPf ANNUAL SMEAR AND RADIATION SURVEY REPORT.	3
EXHIBIT II	
A: PIQUA NUCLEAR POWER FACILITY REACTOR BUILDING - 78'6".	6
B: PIQUA NUCLEAR POWER FACILITY REACTOR BUILDING - 56'6".	7
C: PIQUA NUCLEAR POWER FACILITY REACTOR BUILDING - 100'0".	8
EXHIBIT III	
ACCEPTABLE SURFACE CONTAMINATION LEVELS.	9
EXHIBIT IV	
FLOWPATH P-17 SUMP TO P-18 SUMP.	22

TABLE OF CONTENTS
(Continued)

LIST OF TABLES

	<u>Page</u>
Table 1. Survey Specifications	11
Table 2. Instrumentation and Calibration	13
Table 3. Water, Sludge and Soil Samples.	14
Table 4. Locations and Descriptions of Photographic Slides	19

LIST OF GRAPHS

Graph I. Total Suspended and Dissolved Activity in Sump Water by Year.	16
Graphs II. Activity of Solids in Sump Sludge by Year	17

APPENDICES

Appendix A. Cathodic Protection System.	A-1
Appendix B. Fire Protection Sprinkler System Test	B-1

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ANNUAL SURVEY OF THE RETIRED PIQUA NUCLEAR POWER FACILITY

by

G.E. Kirsch

INTRODUCTION

The annual environmental radiological survey of the Retired Piqua Nuclear Power Facility was performed on October 12, 1988, by Battelle-Columbus. The radiation survey was performed on site and samples were collected for analyses at the Battelle West Jefferson radiochemistry laboratory. Samples collected included area smears to be counted for removable surface contamination, tap water, a sump water sample, and a sump sludge sample. Analyses were performed for these media as described in the survey specifications on page 11. The analyses indicate no significant change from past year's surveys and meet all applicable radiological criteria for the Piqua facility.

The specific survey parameters are as follows:

(1) Survey Instrumentation

All survey and monitoring results were identified listing types of monitoring instruments used, and calibration certification data for each instrument, including method of establishing calibration that is traceable to the National Bureau of Standards.

(2) Survey Specifications

The following describes the annual radiological survey and analysis of the Piqua Nuclear Power Facility performed since 1970. For the 1987 survey, additional smear, air, and water samples were collected and analyzed for tritium. No tritium was observed in any of the sample media; consequently, tritium samples were not continued for 1988. Also for 1988, the soil sampling program was excluded due to the absence of radionuclides in the soil. The other specifications of the survey for 1988 are the same as surveillance activities performed in 1981, 1982, 1983, 1984, 1985, and 1986, with the exception of the optional surveys

performed in 1984 and 1987. The optional surveys performed in 1984 consisted of verifying the flow path of an effluent discharge point and analyzing samples of water and soil collected at the discharge point. Also, a sample of residual water was collected from a storage tank located in the auxiliary building and analyzed. The work performed in 1984 was required on a one-time basis only. New for 1988 is the incorporation of 35 mm slides showing interior and exterior views of the facility. Also, a discussion of the sump system is incorporated in this year's report.

SURFACE CONTAMINATION

Smear survey

Smear samples for transferrable contamination of the outer face of the reactor shielding wall at various levels below the main floor were collected. The survey consisted of sampling and monitoring points 22 through 78 as described in Exhibit I and as illustrated in Exhibits IIA and IIB. The samples were counted for gross alpha and beta/gamma activity and reported in disintegrations per minute per 100 square centimeters (dpm/100 cm²).

If the detected levels of contamination within a specific area averaged more than the levels shown in Exhibit III, an isotopic determination was to be performed, if possible, and a confirming follow-up survey of the area was to be performed as soon as possible.

Radiation Surveys

Radiation surveys were performed on the outer face of the reactor shielding wall and at various levels below the main floor, specifically the numbered locations 22-78 as described in Exhibit I and as illustrated in Exhibits IIA and IIB. All radiation readings were reported in units of millirads per hour at 1 centimeter from the monitored surfaces. The survey instrument detector has no more than 7 milligrams per square centimeter of total absorber and was calibrated to conform with the Battelle Quality Assurance Program.

EXHIBIT I

SURVEY RESULTS

PNPF ANNUAL SMEAR AND RADIATION SURVEY REPORT

Survey Date October 12, 1988 Surveyors G.E. Kirsch and P.E. Tomlin
 Survey Instrument Eberline E-120 with HP-190 Probe I.D. No. C-4171
 Smear Counting Date October 14, 1988 Counting Analyst M.J. Stenhouse
 Counting Instrument Tennelec LB-5100, I.D. No. X39265

Radiation Surveys - Surface Contamination

Location	Beta-Gamma Fixed (mrad/hr)	Removable Alpha (dpm/100 cm ²)			Removable Beta-Gamma (dpm/100 cm ²)		
		dpm	+/-2s		dpm	+/-2s	
<u>78'6" Level</u>							
22 Floor	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
23 Floor	<0.02	-0.40	4.06	<MDA (alpha)	1.28	5.23	<MDA (beta)
24 Floor	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
25 Wall	<0.02	1.62	5.74	<MDA (alpha)	-1.92	3.70	<MDA (beta)
26 Wall	<0.02	-0.40	4.06	<MDA (alpha)	0.21	4.77	<MDA (beta)
27 Wall	<0.02	-0.40	4.06	<MDA (alpha)	-1.92	3.70	<MDA (beta)
28 Floor	<0.02	-0.40	4.06	<MDA (alpha)	3.41	6.04	<MDA (beta)
29 Floor	<0.02	-0.40	4.06	<MDA (alpha)	5.55	6.75	<MDA (beta)
30 Floor	<0.02	1.62	5.74	<MDA (alpha)	0.21	4.77	<MDA (beta)
31 Wall	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
32 Metal grating catwalk	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
33 Metal grating catwalk	<0.02	-0.40	4.06	<MDA (alpha)	0.21	4.77	<MDA (beta)
34 Metal grating catwalk	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
35 Metal grating catwalk	<0.02	1.62	5.74	<MDA (alpha)	-1.92	3.70	<MDA (beta)
36 Metal grating catwalk	<0.02	-0.40	4.06	<MDA (alpha)	-1.92	3.70	<MDA (beta)

EXHIBIT 1

SURVEY RESULTS

(Page 2)

Radiation Surveys - Surface Contamination

Location	Beta-Gamma Fixed (mrad/hr)	Removable Alpha (dpm/100 cm ²)				Removable Beta-Gamma (dpm/100 cm ²)			
		dpm		+/-2s		dpm		+/-2s	
<u>78'6" Level (Continued)</u>									
37 Floor	<0.02	-0.40	4.06	<MDA	(alpha)	1.28	5.23	<MDA	(beta)
38 Stair	<0.02	-0.40	4.06	<MDA	(alpha)	4.48	6.41	<MDA	(beta)
39 Floor	<0.02	1.62	5.74	<MDA	(alpha)	-0.85	4.27	<MDA	(beta)
40 Floor drain	<0.02	-0.40	4.06	<MDA	(alpha)	1.28	5.23	<MDA	(beta)
41 Floor drain	<0.02	-0.40	4.06	<MDA	(alpha)	0.21	4.77	<MDA	(beta)
42 Floor	<0.02	-0.40	4.06	<MDA	(alpha)	-1.92	3.70	<MDA	(beta)
43 Wall	<0.02	-0.40	4.06	<MDA	(alpha)	2.35	5.65	<MDA	(beta)
44 Floor	<0.02	-0.40	4.06	<MDA	(alpha)	7.69	7.40		
45 Floor	<0.02	1.62	5.74	<MDA	(alpha)	4.48	6.41	<MDA	(beta)
46 Floor drain	<0.02	1.62	5.74	<MDA	(alpha)	1.28	5.23	<MDA	(beta)
47 Floor	<0.02	1.62	5.74	<MDA	(alpha)	3.41	6.04	<MDA	(beta)
48 Wall	<0.02	3.65	7.04	<MDA	(alpha)	2.35	5.65	<MDA	(beta)
49 Wall	<0.02	-0.40	4.06	<MDA	(alpha)	2.35	5.65	<MDA	(beta)
50 Metal grating	<0.02	-0.40	4.06	<MDA	(alpha)	-0.85	4.27	<MDA	(beta)
51 Metal grating	<0.02	1.62	5.74	<MDA	(alpha)	4.48	6.41	<MDA	(beta)
52 Wall	<0.02	-0.40	4.06	<MDA	(alpha)	3.41	6.04	<MDA	(beta)
53 Exhaust air duct	<0.02	1.62	5.74	<MDA	(alpha)	-0.85	4.27	<MDA	(beta)
54 Exhaust air duct	<0.02	-0.40	4.06	<MDA	(alpha)	3.41	6.04	<MDA	(beta)
55 Metal grating	<0.02	1.62	5.74	<MDA	(alpha)	0.21	4.77	<MDA	(beta)
56 Metal grating	<0.02	-0.40	4.06	<MDA	(alpha)	1.28	5.23	<MDA	(beta)
57 Metal grating	<0.02	1.62	5.74	<MDA	(alpha)	1.28	5.23	<MDA	(beta)
58 Metal grating	<0.02	-0.40	4.06	<MDA	(alpha)	-0.85	4.27	<MDA	(beta)
59 Metal grating	<0.02	-0.40	4.06	<MDA	(alpha)	0.21	4.77	<MDA	(beta)
60 Metal grating	<0.02	-0.40	4.06	<MDA	(alpha)	2.35	5.65	<MDA	(beta)

EXHIBIT I
SURVEY RESULTS
(Page 3)

Radiation Surveys - Surface Contamination

Location	Beta-Gamma Fixed (mrad/hr)	Removable Alpha (dpm/100 cm ²)			Removable Beta-Gamma (dpm/100 cm ²)		
		dpm	+/-2s		dpm	+/-2s	
78'6" Level (Continued)							
61 Supply air duct	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
62 Emergency personnel air lock	<0.02	3.65	7.04	<MDA (alpha)	-0.85	4.27	<MDA (beta)
63 Emergency personnel	<0.02	-0.40	4.06	<MDA (alpha)	-2.99	4.06	<MDA (beta)
56'6" Level							
64 Stairwell	<0.02	1.62	5.74	<MDA (alpha)	0.21	4.77	<MDA (beta)
65 Stairwell	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
66 Stairwell	<0.02	-0.40	4.06	<MDA (alpha)	0.21	4.77	<MDA (beta)
67 Floor	<0.02	-0.40	4.06	<MDA (alpha)	1.28	5.23	<MDA (beta)
68 Floor	<0.02	3.65	7.04	<MDA (alpha)	-0.85	4.27	<MDA (beta)
69 Floor	<0.02	-0.40	4.06	<MDA (alpha)	0.21	4.77	<MDA (beta)
70 Wall	<0.02	-0.40	4.06	<MDA (alpha)	-0.85	4.27	<MDA (beta)
71 Wall	<0.02	-0.40	4.06	<MDA (alpha)	4.48	6.41	<MDA (beta)
72 Floor	<0.02	1.62	5.74	<MDA (alpha)	-1.92	3.70	<MDA (beta)
73 Wall	<0.02	1.62	5.74	<MDA (alpha)	0.21	4.77	<MDA (beta)
74 Drain F4	<0.02	1.62	5.74	<MDA (alpha)	-0.85	4.27	<MDA (beta)
75 Floor	<0.02	1.62	5.74	<MDA (alpha)	0.21	4.77	<MDA (beta)
76 Floor	<0.02	-0.40	4.06	<MDA (alpha)	2.35	5.65	<MDA (beta)
77 Floor	<0.02	-0.40	4.06	<MDA (alpha)	-1.92	3.70	<MDA (beta)
78 Floor	<0.02	-0.40	4.06	<MDA (alpha)	0.21	4.77	<MDA (beta)

*MDA = Less than minimum detectable activity.

EXHIBIT IIA

PIQUA NUCLEAR POWER FACILITY REACTOR BUILDING

EXHIBIT IIB

PIQUA NUCLEAR POWER FACILITY REACTOR BUILDING

EXHIBIT IIC

**PIQUA NUCLEAR POWER FACILITY
REACTOR BUILDING**

EXHIBIT III

ACCEPTABLE SURFACE CONTAMINATION LEVELS

(Reference U.S. NRC Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors")

Nuclide(a)	Average(b,c)	Maximum(b,d)	Removable(b,e)
U-nat, U-235, U-238 and associated decay products	5000 dpm α /100 cm ²	15000 dpm α /100 cm ²	1000 dpm α /100 cm ²
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm ²	300 dpm/100 cm ²	20 dpm/100 cm ²
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm ²	3000 dpm/100 cm ²	200 dpm/100 cm ²
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	5000 dpm $\beta\gamma$ /100 cm ²	15000 dpm $\beta\gamma$ /100 cm ²	1000 dpm $\beta\gamma$ /100 cm ²

- Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.
- As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.
- Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.
- The maximum contamination level applies to an area of not more than 100 cm².
- The amount of removable radioactive material per 100 cm² of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent level should be reduced proportionally and the entire surface should be wiped.

SUMP WATER AND SLUDGE SAMPLES

Sump Water

One water sample was collected from the lower level sump and analyzed for gross alpha and gross beta/gamma activity for both suspended and dissolved components. In addition, isotopic identification was performed for cesium-137 and cobalt-60. A gamma spectral analysis for energies between 65 KeV and 2.0 MeV was also performed.

Facility Tap Water

One facility tap water sample was collected and analyzed for suspended and dissolved gross alpha and gross beta/gamma activity in order to provide background comparison data with the sump water sample.

Sump Solids

One sump solids sample was collected from the lower level sump and analyzed for gross alpha and gross beta/gamma activity. In addition, isotopic identification was performed for cesium-137 and cobalt-60. A gamma spectral analysis was also performed for gamma energies between 65 KeV and 2.0 MeV.

VISUAL INSPECTION

- o The entombed structure was examined to determine the occurrence of changes which might affect the integrity of structures and components. Photographs of suspect areas were taken.
- o Photographic 35 mm slides were taken and incorporated as part of this year's report. (Refer to Table 4 regarding descriptions and locations of the slides.)
- o The flowpath between P-17 and P-18 is discussed in the report. (Refer to Exhibit IV.)

A summary of the survey specifications is provided in Table 1.

TABLE 1. SURVEY SPECIFICATIONS

Type	Location	Analyses Performed
I. Smear Survey	Locations 22-78	- Gross alpha and beta/gamma
Radiation Survey	Locations 22-78	- Beta/gamma radiation survey
II. Water Sample "A"	Building Tap H ₂ O	- Gross alpha and beta (dissolved and suspended)
Water Sample "B"	Lower Level Sump	- Gross alpha and beta (dissolved and suspended)
		- Gamma spectral including ¹³⁷ Cs and ⁶⁰ Co
Sludge Sample "B"	Lower Level Sump	- Gamma spectral including ¹³⁷ Cs and ⁶⁰ Co
		- Gross alpha and gross beta/gamma
III. Visual Inspection	Outer Containment Vessel	- Visual examination to detect changes to outer containment
Photographic Services	Building Interior and Exterior	- 35 mm slides
Characterization of Sump Plumbing	P-17 Reactor Building Sump Flow Path To P-18 Auxiliary Building	

DISCUSSION AND RESULTS

Table 2 provides a listing of instruments used for survey and analysis, along with the date of calibration, the number of the Battelle Quality Assurance (QA) procedure used to calibrate the instrument, the source identification and certification number for the National Bureau of Standards traceability.

SURFACE CONTAMINATION

Removable surface contamination levels, as recorded in Exhibit I, were appreciably less than the criteria described in Exhibit III. The highest observed transferrable activity at the 2 sigma confidence level was less than 10 alpha and 16 beta disintegrations per minute per 100 square centimeters (dpm/100 cm²). Previous annual surveys indicated approximately the same count levels.

Fixed surface contamination levels, as recorded in Exhibit I, were <0.02 mrad/hr at 1 cm for beta/gamma emitters. Previous annual surveys indicated essentially the same radiation levels.

SUMP WATER AND SLUDGE SAMPLES

Refer to Table 3 (page 14) regarding water and sludge samples. Gross alpha and beta counting was performed for the water and sludge in the lower level sump (location B). For water, the alpha concentrations were $3.6 \pm 1.9 \times 10^{-9}$ and $1.3 \pm 2.8 \times 10^{-10}$ $\mu\text{Ci/ml}$, respectively, for dissolved and solid components. The combined dissolved and solids alpha activity was $3.7 \pm 1.9 \times 10^{-9}$ $\mu\text{Ci/ml}$ and the combined alpha activity for the year 1987 was $<8 \times 10^{-9}$ $\mu\text{Ci/ml}$. Combined alpha activity for samples collected prior to 1988 ranged from $<8 \times 10^{-9}$ to 1×10^{-7} $\mu\text{Ci/ml}$.

The beta concentrations in sump water for this survey were $3.4 \pm 0.2 \times 10^{-8}$ and $<1.6 \times 10^{-9}$ $\mu\text{Ci/ml}$, respectively, for dissolved and solid components. The combined dissolved and solids beta activity was $<4.0 \times 10^{-8}$ $\mu\text{Ci/ml}$. The combined beta activity for the survey performed in 1987 was $<5 \times 10^{-8}$ $\mu\text{Ci/ml}$. Combined beta activity for samples collected prior to 1987 ranged from 1×10^{-8} to 1×10^{-7} $\mu\text{Ci/ml}$.

The total dissolved and solid alpha and beta gross activity in the sump water for the 1988 survey was approximately 3.9×10^{-8} $\mu\text{Ci/ml}$. The total for the previous survey performed in 1987 was approximately 1.5×10^{-8} $\mu\text{Ci/ml}$.

The sump water was also analyzed for cesium-137 and cobalt-60. The total soluble and insoluble concentration of cesium-137 was $<3.1 \times 10^{-8}$ $\mu\text{Ci/ml}$. The soluble concentration was $<2.9 \times 10^{-8}$ $\mu\text{Ci/ml}$ and the insoluble was $<1.6 \times 10^{-9}$ $\mu\text{Ci/ml}$. The total concentration for the survey conducted in 1987 was $1.9 \pm 1.4 \times 10^{-8}$ $\mu\text{Ci/ml}$ with the respective soluble and insoluble concentrations of $1.8 \pm 1.4 \times 10^{-8}$ $\mu\text{Ci/ml}$ and $<1 \times 10^{-9}$ $\mu\text{Ci/ml}$.

TABLE 2. INSTRUMENTATION AND CALIBRATION

Instrument	Calibration Date	Sources/ID Standards	QA Procedure
Tennelec LB5100 Series 2 Automatic Counting System	10-14-88	Sr-90/Y-90 Analytics, Inc. SRS 15886-84	HL-PP-248 Revision 1 HL-PP-98 Revision 0
		Sr-90/Y-90 IPL #8521-20	
		Pu-239 Eberline S-1868	
PGT Intrinsic Germanium Detector Nuclear Data 680 Gamma Spectrometer	9-21-88	Mixed Gamma Std. NES-602, New England Nuclear	HL-PP-248.1 Revision 1 HL-PP-98 Revision 0
		Mixed Gamma Std. NBS/SRM4276B	
Eberline E-120 with HP-190 Probe ID No. C-4171	10-11-88	Ra-226 HP14 NBS No. 26812	IL-CP-129 Revision 1

TABLE 3. WATER, SLUDGE AND SOIL SAMPLES*

($\pm 1 \sigma$ FOR WATER; $\pm 2 \sigma$ FOR SOLIDS)

Identification and Location	Dissolved Alpha	Solids Alpha	Dissolved Beta	Solids Beta	Dissolved Cs-137	Solids Cs-137	Dissolved Co-60	Solids Co-60	Gamma Spectral
Sample A - Bldg. Tap Water	$6.5 \pm 1.5 \times 10^{-9}$	$7.5 \pm 3.9 \times 10^{-9}$	$7.6 \pm 0.9 \times 10^{-9}$	$1.1 \pm 0.4 \times 10^{-9}$	N/A	N/A	N/A	N/A	N/A
Sample B - Lower Level Sump Water	$3.6 \pm 1.9 \times 10^{-9}$	$1.3 \pm 2.8 \times 10^{-10}$	$3.4 \pm 0.2 \times 10^{-8}$	$<1.6 \times 10^{-9}$	$<2.9 \times 10^{-8}$	$<1.6^{**} \times 10^{-9}$	$<2.8 \times 10^{-8}$	$<1.6 \times 10^{-9}$	All <MDA
Sample B - Lower Level Sump Sludge	N/A	11.3 ± 4.3	N/A	104 ± 8	N/A	83 ± 8	N/A	43 ± 4	All <MDA Except Cs-137 and Co-60

* - Concentrations are reported as $\mu\text{Ci/ml}$ for water samples and pCi/gram for sludge samples.

< - Less than minimum detectable activity (MDA)

N/A - Not Applicable or Required

** - Based on the Tennelec LB5100 Counter MDA for beta solids, which provides a more sensitive value than gamma spectrometry.

The cobalt-60 total concentration for 1988 was $<3.0 \times 10^{-8}$ with the respective soluble and insoluble concentrations of $<2.8 \times 10^{-8}$ $\mu\text{Ci/ml}$ and $<1.5 \times 10^{-9}$ $\mu\text{Ci/ml}$. For the 1987 survey, the total concentration was $<3.3 \times 10^{-8}$ $\mu\text{Ci/ml}$ and the respective soluble and insoluble concentrations were $<3.2 \times 10^{-8}$ $\mu\text{Ci/ml}$ and $<1.0 \times 10^{-8}$ $\mu\text{Ci/ml}$.

Solids in the sludge contained in the sump were analyzed for gross alpha, gross beta, cesium-137, and cobalt-60.

The respective gross alpha and beta concentrations were 11.4 ± 4.3 and 104 ± 8 pCi/gram. The respective concentrations reported for 1987 were 16 ± 5 and 93 ± 4 pCi/gram. Cesium-137 was 83 ± 8 pCi/gram and cobalt-60 was 43 ± 4 pCi/gram. Respective results for 1987 for both radionuclides were 94 ± 5 pCi/gram for cesium-137 and 29 ± 2 pCi/gram for cobalt-60. Other gamma emitting radionuclides were not detected for the 1988 survey.

The two graphs on pages 16 and 17 illustrate the cesium-137 and cobalt-60 activity in sump liquids and solids for the 12 year period 1977 through 1988.

**Comparison of Concentrations of Activity in Water in
the Sump with the Concentration Guides Listed in
DOE Order 5480.1 Chapter XI**

Where either the identity or the concentration of any radionuclide in the mixture is not known, the concentration guide is 4×10^{-7} $\mu\text{Ci/ml}$ for a controlled area. The total soluble and insoluble alpha and beta concentration of 3.9×10^{-8} $\mu\text{Ci/ml}$ for the sump water for 1988 is approximately 10 percent of the controlled area concentration guide.

From January 1, 1988 through October 12, 1988, no water was pumped from the reactor building (pump P-17) to the auxiliary building (pump P-18). Consequently for calendar year 1988, there were no releases recorded for reactor building sump water to the Great Miami River.

The respective concentration guides for soluble and insoluble cesium-137 for controlled areas are 4.0×10^{-4} $\mu\text{Ci/ml}$ and 1.0×10^{-3} $\mu\text{Ci/ml}$. The concentrations of cesium in the sump water for 1988 were $<2.9 \times 10^{-8}$ $\mu\text{Ci/ml}$ and $<1.6 \times 10^{-9}$, respectively, for soluble and insoluble cesium-137.

Cobalt-60 was not detected above background levels, $<2.8 \times 10^{-8}$ and $<1.6 \times 10^{-9}$ $\mu\text{Ci/ml}$, respectively, for the dissolved and solid components. The soluble and insoluble concentration guide for releases to controlled areas is 1×10^{-3} $\mu\text{Ci/ml}$.

GRAPH I

TOTAL SUSPENDED AND DISSOLVED ACTIVITY IN SUMP WATER BY YEAR

REPRODUCED FROM
BEST AVAILABLE COPY

GRAPH II
ACTIVITY OF SOLIDS IN SUMP SLUDGE BY YEAR

Visual Inspection

The outer containment vessel was visually examined for indications of changes to the integrity of the structure and components. No evidence of structural changes were observed.

Photographic Services

Photographic 35 mm slides were taken of the interior and exterior of the facility/site. Four sets of slides are provided with this report, and the locations where the photographs were taken are described in Table 4, pages 19, 20, and 21.

Description of Sump System

The Reactor Building at the 56'6" Level contains a sump pit that is 6 feet long, 6 feet wide and 6 feet deep, that is designed to collect water from floor drains and water that could come in contact with the containment wall as a result of a water line break or other flooding situation. This sump, known as P-17, automatically pumps water over to another sump (P-18) located in the auxiliary building in the event that an increase in the water level in the P-17 sump occurs. The P-18 sump located in the auxiliary building is 4 feet long, 4 feet wide and 6 feet deep. In addition to being the receptacle for any effluents from the P-17 sump, the P-18 sump also serves as a collection device for water from floor drains located in the auxiliary building. Effluents from the P-18 sump are discharged to the river.

In summary, the two sumps are interconnected, and any discharge from the P-17 sump to the P-18 sump activates an alarm to alert facility personnel.

A diagram of the system is illustrated in Exhibit IV, page 22.

Cathodic Protection System

Records of the cathodic protection system inspection were reviewed. Applicable portions of the survey performed by the Harco Corporation in September, 1988 are incorporated in Appendix A.

Fire Protection

The fire protection system has been upgraded with the installation of a digital communicator that is activated by any flow in the sprinkler system.

Copies of the forms of the fire protection inspection performed by the Grinnel Fire Protection Systems Company are provided in Appendix B.

TABLE 4. LOCATIONS AND DESCRIPTIONS OF PHOTOGRAPHIC SLIDES

<u>Slide No.</u>	<u>Location</u>	<u>Description</u>
1	Outside	South side of facility.
2	Outside	West side of facility.
3	Outside	Northwest side of facility.
4	Outside	East side of facility.
5	Outside	Southwest side of facility showing personnel airlock.
6	Outside	South side storage area with sewage plant in background.
7	Outside	North storage area.
8	Outside	North side of facility.
9	Outside	Armco plant northeast of facility.
10	Outside	East overhead door entrance to auxiliary building.
11	Inside Reactor Bldg. Upper 10'0" Level	Overhead crane.
12	Inside Reactor Bldg. Upper 100'0" Level	Access port to lower levels.
13	Inside Reactor Bldg. Upper 100'0" Level	Plaque describing information regarding the Reactor.
14	Inside Reactor Bldg. Upper 100'0" Level	East storage room.
15	Inside Reactor Bldg. Upper 100'0" Level	View through access port to two lower levels.
16	Inside Reactor Bldg. Upper 100'0" Level	Office.
17	Inside Reactor Bldg. Middle 78'6" Level	West view.
18	Inside Reactor Bldg. Middle 78'6" Level	North side upper deck.

<u>Slide No.</u>	<u>Location</u>	<u>Description</u>
19	Inside Reactor Bldg. Middle 78'6" Level	South side upper deck.
20	Inside Reactor Bldg. Middle 78'6" Level	Motor/pump pad.
21	Inside Reactor Bldg. Middle 78'6" Level	Motor/pump pad.
22	Inside Reactor Bldg. Middle 78'6" Level	View of 56'6" level below.
23	Inside Reactor Bldg. Middle 78'6" Level	Personnel air lock.
24	Inside Reactor Bldg. Lower 56'6" Level	Sump access area P-17.
25	Inside Reactor Bldg. Lower 56'6" Level	North view from west.
26	Inside Reactor Bldg. Lower 56'6" Level	West view along south wall.
27	Inside Reactor Bldg. Lower 56'6" Level	Information plaque on north wall of containment regarding the reactor.
28	Inside Reactor Bldg. Lower 56'6" Level	North side of building, wall by door to circular stairway.
29	Inside Reactor Bldg. Lower 56'6" Level	Wall surface inside stairwell.
30	Inside Reactor Bldg. Lower 56'6" Level	View southeast from northwest corner.
31	Inside Reactor Bldg. Lower 56'6" Level	Overhead pipe system including sump pipe from P-17 sump.
32	Inside Reactor Bldg. Upper 100'0" Level	Stairway to lower levels.
33	Auxiliary Building	Equipment room adjacent to stairway.
34	Auxiliary Building	Room behind P-18 sump room.
35	Auxiliary Building	P-18 sump area.

<u>Slide No.</u>	<u>Location</u>	<u>Description</u>
36	Auxiliary Building	P-18 sump area storage tank (not in use).
37	Auxiliary Building	South side - west room.
38	Auxiliary Building	Corridor/truck bay between above rooms.
39	Auxiliary Building	South side, middle room.
40	Auxiliary Building	South side, east room.
41	Auxiliary Building	North side, east room.
42	Auxiliary Building	Electrical switch room west of main office area.
43	Auxiliary Building	Office area corridor.

FLOWPATH P-17 SUMP TO P-18 SUMP

Other Safety-Related Systems

The facility safety systems have been expanded by the City of Piqua since the last survey. New high-pressure sodium lights have been installed in strategic locations, and new emergency lights have been installed in the stairwells. Also, fire extinguisher locations have been labelled with new signs.

CONCLUSIONS

The survey did not reveal the existence of any significant changes within the facility. Smears collected for removable alpha and beta-gamma surface contamination were all minimal in activity.

All beta-gamma radiation readings were essentially at background levels. Liquid and solid components collected from the sump were analyzed for gross alpha, gross beta, beta and gamma emitters, including Cs-137 and Co-60. No significant changes were observed or releases to the environment were recorded, and the results were comparable to the last survey performed in November, 1987.

APPENDIX A

CATHODIC PROTECTION SYSTEM

RESURVEY
OF THE
CATHODIC PROTECTION SYSTEM
ON THE
NUCLEAR CONTAINMENT SHELL
PIQUA, OHIO
FOR THE
CITY OF PIQUA
PIQUA, OHIO

VI. DISCUSSION:

On September 22, 1988 Harco Technologies Corporation conducted a resurvey of the cathodic protection system provided the inactive nuclear containment shell located in The City of Piqua, Ohio. The work was conducted in accordance with The City of Piqua Purchase Order No. H-7088 dated September 7, 1988.

The nuclear containment shell is described as a steel vessel, 72' in diameter. The cathodic protection system, installed by Harco in 1964, is an impressed current system basically consisting of a Harco air-cooled rectifier unit rated at 40 volts and 24 amps DC and ten (10) 3" diameter by 60" long graphite anodes distributed around the containment shell. Anode Nos. 9 and 10 were replaced in 1984 by Harco as per the recommendations in Harco's 1984 annual resurvey report. During anode replacement four (4) test holes were installed at various locations through the asphalt and concrete. The test holes were installed so that more accurate and reproducible potential measurements could be obtained. The last annual resurvey was conducted in September 1987. The system was operating within design considerations at that time.

The cathodic protection system and its relationship to the nuclear containment shell is shown on Harco Corporation Drawing No. L-23441 which is included with this report.

To determine the effectiveness of the cathodic protection system, structure-to-soil potential measurements were taken around the perimeter of the containment shell at approximately the same locations as the previous survey. The structure-to-soil potentials were obtained with a Miller Model LC-1 variable input impedance digital voltmeter in conjunction with a stable copper-copper sulfate reference electrode. The connection necessary to complete the measuring circuit was made by direct contact to the containment shell. The test holes were utilized for half cell placement where available.

The data obtained during this survey is tabulated in Appendix "A". The specific tests and measurements used during this survey are described in greater detail in Appendix "B" of this report entitled "Test Procedures". The results of the testing are described in the following section of this report.

VII. RESULTS:

Upon arrival at the site the rectifier unit was visually inspected and the DC voltage and current outputs were read and recorded on the enclosed Cathodic Protection Rectifier Maintenance Sheet. The rectifier was found to be operating properly with the DC outputs essentially the same as those found in previous years. The rectifier output has remained within design considerations since the 1984 anode replacement.

One (1) criterion for cathodic protection of steel structures as established by the National Association of Corrosion Engineers (NACE) Standard RP-01-69, 1983 Revision is:

6.3 Criteria

6.3.1 Steel and cast iron structures.

6.3.1.1 A negative (cathodic) voltage of at least 0.85 volt as measured between the structure surface and a saturated copper-copper sulfate reference electrode contacting the electrolyte. Determination of this voltage is to be made with the protective current applied.

Table I of Appendix "A" lists the structure-to-soil potential measurements obtained around the containment shell. As can be seen from the data all structure-to-soil potential measurements meet the NACE -850 millivolt criterion for cathodic protection.

The test holes repairs, as recommended in previous resurvey reports, were performed by the City of Piqua prior to the 1988 survey. The test holes were found in good condition during this survey. The only maintenance not performed was the replacement of the test hole No. 9 enclosure. Therefore, it is recommended that the location of test hole No. 9 be identified and the enclosure reinstalled. No further repairs or maintenance are needed at this time.

To insure the continued operation of the cathodic protection system the rectifier DC voltage and current output should be read and recorded monthly and Harco Corporation should continue to conduct an annual resurvey.

CATHODIC PROTECTION RECTIFIER MAINTENANCE SHEET

Sheet ____ of ____

Rectifier Unit No.: Serial No. 6551

Location of Rectifier Unit: North Wall of Recreation Room

Type of Rectifier Unit: Air-Cooled

Type of Anodes: Graphite Type of Ground Bed: Conventional Distributed

No. of Anodes: 10 Size: 3" ϕ X 60" Long

Ground Bed Location: Around Parameter of Containment Shell

Rectifier Mfg. By: Harco Corporation

Rectifier AC Input: 110 Volts 1 Phase 60 Cycles

Rectifier DC Output: 40 Volts 24 Amperes

Installed: 9/19/64

Rectifier Setting	DC Output		Date	By	Remarks
	Volts	Amps			
C-1	19	20	10/13/64		
B-4	14.5	10.5	12/08/64		
B-4	15	9	1/13/71	JCM	
B-4	15	7.75	12/20/76	JAK	Found at: Left at
B-5	15	10	9/18/79	WTL	Found at: Left at
B-4	0	0	5/25/83	PVR	Found at: AC Disconnect Switch Off
B-4	15	6	5/25/83	PVR	Left at
C-1	21	4.5	5/03/84	PVR	Left at
B-4	14.65	7.7	6/07/84	RJM	New Tap Setting
B-4	14.5	7.7A	5/22/85	EWB	As Found, No adjust necessary
B-4	15	7	9/30/86	PVR	Left as found
B-4	15.6	6.5	9/24/87	JLF	Left as found
B-4	15.0	6.0	9/22/88	JLF	Panel Meter
B-4	15.5	6.2	9/22/88	JLF	H Meter

TABLE I
STRUCTURE-TO-SOIL POTENTIAL MEASUREMENTS
ON THE
NUCLEAR CONTAINMENT SHELL
CITY OF PIQUA POWER PLANT
PIQUA, OHIO

Rdg. No.	Location	Structure-to-Soil Potentials (mvs)	
		9/24/87	9/22/88
1.	Test Point 1A (Test Hole - Could Not Open)*	-1190	-1310
2.	Test Point 1 (Test Hole)	-1370	-1900
3.	Test Point 2 (Test Hole)	-960	-1290
4.	Test Point 3	-1040	-910
5.	Test Point 4	-960	-970
6.	Test Point 5	-1440	-1370
7.	Test Point 6	-2870	-2470
8.	Test Point 7	-1890	-3150
9.	Test Point 8 (Thru Asphalt)	-1240	-960
10.	Test Point 9 (Test Hole)	REMOVED	-----
11.	Fire Hydrant North of Reactor	-860	-875

*Reading obtained approximately 5' from structure in 1987.

APPENDIX B

FIRE PROTECTION SPRINKLER SYSTEM TEST

GRINNELL FIRE PROTECTION SYSTEMS COMPANY

B-1

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INSPECTION REPORT

No. 1-88/85

CONFERRED WITH

PAULA STEINKE

REPORT OF INSPECTION

INSPECTION CONTRACT

NO. 10590

BUREAU FILE

NO.

SET 1 OF 2

REPORT TO CITY OF PLOUA

STREET 123 BRIDGE ST

CITY & STATE PLOUA, OH

ZIP 45352

ATT.

BUILDING OR LOCATION INSPECTED OLD ATOMIC PLANT

INSPECTOR FISHER

GRINNELL OFFICE CINC PHONE NO. 771-2010

DATE 10-13-88

1. GENERAL

A. (To be answered by the Owner or Owner's representative)

- Have there been any changes in the occupancy classification, machinery or operations since the last inspection?
- Have there been any changes or repairs to the fire protection systems since the last inspection?
- If a fire has occurred since the last inspection, have all damaged sprinkler system components been replaced?
- Has the piping in all dry systems been checked for proper pitch within the past five years?
Date last checked _____ (checking is recommended at least every 5 years)
- Has the piping in all systems been checked for obstructive materials? OWNERS RESPONSIBILITY
Date last checked _____ (checking is recommended at least every 5 years)
- Have all fire pumps been tested to their full capacity through the use of hose streams or flow meters within the past 12 months?
- Are gravity, surface or pressure tanks protected from freezing?
- Are any of the sprinklers 50 years old or older? _____ (testing and/or replacement is recommended for such sprinklers)
- Are any extra high temperature solder sprinklers regularly exposed to temperatures near 300°F?

B. (To be answered by the inspector)

- Have the sprinkler systems been extended to all visible areas of the building?
- Does there appear to be proper clearance between the top of all storage and the sprinkler deflector?
- Are the building areas protected by a wet system, heated, including its blind attics and perimeter areas, where accessible?
- Are all visible exterior openings protected against the entrance of cold air?

2. CONTROL VALVES

- Are all sprinkler system main control valves and all other valves in the appropriate open or closed position?
- Are all control valves sealed or supervised in the open position?

Control Valves	No. of Valves	Type	Easily Accessible		Signs		Valve Open		Secured? (Sealed?) (Locked?) (Supvd.?)		Supervision Operational	
			Yes	No	Yes	No	Yes	No	Yes	No	Yes	No
CITY CONNECTION	<u>2</u>	<u>OSY/PIV</u>	<u>-</u>						<u>-</u>		<u>SEAL/SUPV</u>	
TANK												
PUMP												
SECTIONAL												
SYSTEM	<u>2</u>	<u>OSY</u>	<u>-</u>						<u>-</u>		<u>SEAL/SUPV</u>	
ALARM LINE												

3. WATER SUPPLIES

- Water supply source? City 90 Gravity Tank _____

Pressure Fire Pump & Tank _____

Pressure Fire Pump & City _____

Waterflow Test Results Made During This Inspection

Pressure Fire Pump & Pond _____

Test Pipe Located	Size Test Pipe	Static Pressure Before	Flow Pressure	Static Pressure After	Test Pipe Location	Size Test Pipe	Static Pressure Before	Flow Pressure	Static Pressure After
<u>MD#1</u>	<u>2"</u>	<u>90</u>	<u>65</u>	<u>8</u>					
<u>MD#2</u>	<u>2"</u>	<u>90</u>	<u>50</u>	<u>90</u>					

4. TANKS, PUMPS, FIRE DEPT. CONNECTIONS

- Do fire pumps, gravity, surface or pressure tanks appear to be in good external condition?
- Are gravity, surface and pressure tanks at the proper pressure and/or water levels?
- Are fire dept. connections in satisfactory condition, couplings free, caps or plugs in place and check valves tight?
- Are fire dept. connections visible and accessible?

5. WET SYSTEMS

- No. of systems 2 Make & Model 6" AUTOMATIC 153
- Are cold weather valves in the appropriate open or closed position?
If closed, has piping been drained?
- Has the owner or owner's representative been advised that cold weather valves are not recommended by NFPA?
- Have all the antifreeze systems been tested?
- Date antifreeze systems were tested _____
- The antifreeze tests indicate protection to:
system 1 _____ 2 _____ 3 _____ 4 _____ 5 _____ temperature
- Did alarm valves, waterflow alarm indicators and retards test satisfactorily?

†Not Applicable

*Explain (No) Answers on Back of Sheet 2

GRINNELL FIRE PROTECTION SYSTEMS COMPANY

B-2

INSPECTION CONTRACT

NO. 10594

BUREAU FILE

NO.

SET 2 OF 2

INSPECTION REPORT

No. 1-88/89

REPORT OF INSPECTION

6. DRY SYSTEMS

- a. No. of systems NONE Make & Model _____
Date last trip tested _____
- b. Is the air pressure and priming water levels normal? _____
- c. Did the air compressor operate satisfactorily? _____
- d. Were all low points drained during this inspection? _____
- e. Did all quick opening devices operate satisfactorily? _____
- f. Did all the dry valves operate satisfactorily during this inspection? _____
- g. Do dry valves appear to be protected from freezing? _____
- h. Is the dry valve house heated? _____

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7. SPECIAL SYSTEMS

- a. No. of systems NONE Make & Model _____
Type _____
- b. Were valves tested as required? _____
- c. Did all heat responsive systems operate satisfactorily? _____
- d. Did the supervisory features operate during testing? _____

Heat Responsive Devices: Type _____ Type of test _____

Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....	Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....
Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....	Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....
Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....	Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....
Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....	Valve No. _____	1.....	2.....	3.....	4.....	5.....	6.....

Auxiliary equipment: No. _____ Type _____
Location _____
Test results _____

8. ALARMS

- a. Did the water motors and gong operate during testing? _____
- b. Did the electric alarms operate during testing? _____
- c. Did the supervisory alarms operate during testing? _____

9. SPRINKLERS — PIPING

- a. Do sprinklers generally appear to be in good external condition? _____
- b. Do sprinklers generally appear to be free of corrosion, paint, or loading and visible obstructions? _____
- c. Are extra sprinklers available on the premises? _____
- d. Does the exterior condition of piping, drain valves, check valves, hangers, pressure gauges, open sprinklers and strainers appear to be satisfactory? _____
- e. Does the hand hose on the sprinkler system appear to be in satisfactory condition? _____

10. EXPLANATION OF "NO" ANSWERS (For Sections 1B thru 9):

1B-2 SOME AREAS NOT SPRINKLERED, OLD MULTITROL SYSTEM DISCONNECTED

* SEC 22 LIFESIDE HEAD IN PENDANT POSITION, SECRETARY'S OFFICE
SHOULD HAVE SIDEWALL HEAD

11. THE INSPECTOR SUGGESTS THE FOLLOWING NECESSARY IMPROVEMENTS. HOWEVER, THESE SUGGESTIONS ARE NOT THE RESULT OF AN ENGINEERING SURVEY:

12. ADJUSTMENTS OR CORRECTIONS MADE:

FLOW TEST M.D. & I.T. OPERATED & SEALED CONTROL VALVES
1 USED 054 STEEL

13. LIST CHANGES IN THE OCCUPANCY HAZARD OR FIRE PROTECTION EQUIPMENT, AS ADVISED BY THE OWNER IN SECTION 1A:

14. INSPECTION AND SUGGESTED IMPROVEMENTS WERE DISCUSSED WITH THE UNDERSIGNED OWNER OR OWNER'S REPRESENTATIVE

Signature of owner or owner's representative [Signature] Date 10/14/88

DUPLICATE TO:

STREET _____
CITY & STATE _____ ZIP _____
ATT. _____

G4550-2

†Not Applicable
*Explain (No) Answers on Back of Sheet

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Includes (43) scanned images from 35 mm colored slides.

ADJUSTABLE GANTHER

GT
H-ADJUTANT LE SHANTIE

2

FIQUA NUCLEAR POWER FACILITY

- NOTICE -

BEHIND THIS PLAQUE IS A METAL BOX CONTAINING
DETAILED INFORMATION CONCERNING THE REACTOR
COMPLEX STRUCTURE AND CONTENTS. THIS BOX

BEHIND THIS PLAQUE IS A METAL BOX CONTAINING
DETAILED INFORMATION CONCERNING THE REACTOR
COMPLEX STRUCTURE AND CONTENTS. THIS BOX
IS NOT TO BE OPENED WITHOUT THE PERMISSION
OF THE UNITED STATES GOVERNMENT.

UNITED STATES
ATOMIC ENERGY COMMISSION
- 1969 -

14A

VINCE STALLS

27

3

2

01 INS-045
:V3

FRAGILE 01 INS-045
LOW VOLTAGE
PIN TYPE
PORCELAIN INSULATORS

FRAGILE 01 INS-045
LOW VOLTAGE
PIN TYPE
PORCELAIN INSULATORS

TRADE OF INSULATORS
LOW VOLTAGE
PIN TYPE
PORCELAIN INSULATORS

44

1

44

6
MHDWP

GENERAL ELECTRIC
MOTOR
MOTOR
MOTOR

11
6
EL

ORTH
DE
MENT

T 16

#1

7/

39A

40A

INTEGRATIONS AND
WHICH NOT IN US
SOME AND (X) TAY

REGULATIONS MUST BE RECEIVED
WHEN PUT IN USE & MUST
BE RETURNED TO TANGS.

42B