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RADIOACTIVE WASTE DISCHARGES TO THE ENVIRONMENT FROM NUCLEAR POWER FACILITIES

U.S. DEPARTMENT OF HEALTH, EDUCATION, AND WELFARE
Public Health Service
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U.S. DEPARTMENT OF HEALTH, EDUCATION, AND WELFARE
Public Health Service
Environmental Health Service
Bureau of Radiological Health
Rockville, Maryland 20852

FOREWORD

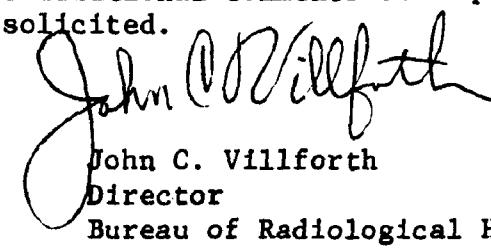
The Bureau of Radiological Health carries out a national program designed to evaluate the exposure of man to ionizing and non-ionizing radiation and to promote development of controls necessary to protect the public health and safety.

Within the Bureau, the Division of Environmental Radiation conducts programs relating to 1) public health evaluation of planned and operating nuclear facilities, 2) field studies at operating nuclear facilities to develop environmental surveillance technology, and 3) a system of national radiation surveillance projects to evaluate population exposure from all sources of environmental radioactivity.

The Bureau publishes its findings in the monthly publication Radiological Health Data and Reports, Public Health Service numbered reports, appropriate scientific journals, and Division technical reports.

The technical reports of the Division of Environmental Radiation allow comprehensive and rapid publishing of the results of intramural and contract projects. The reports are distributed to State and local radiological health program personnel, Bureau technical staff, Bureau advisory committee members, university personnel, libraries and information services, industry, hospitals, laboratories, schools, the press, and other interested groups and individuals. These reports are also included in the collections of the Library of Congress and the Clearinghouse for Federal Scientific and Technical Information.

I encourage the readers of these reports to inform the Bureau of any omissions or errors. Your additional comments or requests for further information are also solicited.



John C. Villforth
Director
Bureau of Radiological Health

PREFACE

The discharges of radioactivity to the environment from nuclear power stations contribute to the radiation dose received by the general population. The Bureau of Radiological Health provides guidance and recommendations to health agencies for the development of environmental surveillance programs related to nuclear facilities. In order to provide a better technical basis for surveillance recommendations, radiological data relative to discharges of radioactivity from nuclear power facilities have been compiled.

This report summarizes discharges of radioactive material to the environment from nine selected nuclear power facilities and relates the discharges to power produced and plant maintenance operations. The facilities included in this report represent three basic reactor types: pressurized water, boiling water, and high temperature gas. The operating facilities which are not included in this report either represent unique designs not being constructed in the present generation of power plants or they are smaller plants similar in design to those which are included.

Appreciation is expressed to the nuclear facility operators and the Atomic Energy Commission's Division of Regulation and Division of Naval Reactors who provided comments and suggestions on the draft version of this report.

Charles L. Weaver, Director
Division of Environmental Radiation

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RADIOACTIVE WASTE DISCHARGES TO THE ENVIRONMENT FROM NUCLEAR POWER FACILITIES

INTRODUCTION

Nuclear power plants produce large quantities of radioactive material as a by-product of operation of the reactor. Most of the radioactive material is contained within the fuel elements and remains there until the fuel is chemically reprocessed at a fuel reprocessing facility. The relatively small portions of radioactive material that escape from the fuel or are produced outside the fuel are contained and processed as radioactive waste at the nuclear power plant. This material is concentrated and converted to solid form and shipped offsite for burial at licensed burial sites. The very small amounts of residual material are discharged to the atmosphere or hydrosphere. The quantities and types of waste discharged vary from facility to facility depending primarily on design characteristics of the plant and on waste management practices.

Measurements of radioactivity in the environs of nuclear power plants made by State health departments, nuclear power facility operators, and the Bureau of Radiological Health, have in most cases revealed little or no increase in environmental radioactivity resulting from plant operations. In those cases where increases were measured, the levels were barely detectable. However, in consideration of the rapid expansion of the nuclear power industry, it is incumbent upon health agencies and the Bureau of Radiological Health to continually review radioactive waste disposal practices and to evaluate their potential effect on the environment.

The purpose of this report is to provide information concerning discharges of radioactive material to the environment by nuclear power facilities. Nuclear power facility operators routinely prepare operating reports which contain information concerning discharges of radioactive material to the environment. These reports are used as the principal source of data for this report.

Administrative Controls

Discharge of radioactive waste to the environment by nuclear power facilities is regulated by the Atomic Energy Commission (AEC) through an operating license issued to the nuclear facility operator. The license requires the licensee to operate the plant in accordance with written Technical Specifications that have been approved by the AEC which include, among other items, limits for radioactive liquid and gaseous discharges. Discharge limits presented in the Technical Specifications are based on

effluents. ("Exclusion area" means that area surrounding the reactor in which the licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area). Discharge limits may further be reduced by the AEC to compensate for possible re-concentration of radionuclides by environmental media. For example, the gaseous discharge limits now being applied in the licensing process to ^{131}I releases are reduced by a factor of 700 to compensate for possible reconcentration through the pasture-cow-milk exposure pathway.

SOURCES OF LIQUID AND GASEOUS WASTE

Radioactivity at nuclear power facilities is produced primarily as a by-product of the fission process or from neutron activation of structural material within the pressure vessel and impurities in the primary coolant. A combination of leakage of fission products through the fuel cladding into the primary coolant and activation of materials outside the fuel makes the primary coolant the principal source of liquid and gaseous wastes. However, leakage of primary coolant into other systems and various plant operations cause the sources to be numerous. Typical plant operations which result in liquid or gaseous radioactive waste include:

1. Refueling and maintenance
2. Control of primary coolant chemistry
3. Sampling
4. Rejection of non-condensable gases from steam condensers
5. Blowdown of steam generators
6. Expansion water when the plant goes from a cold to a hot operation
7. Decontamination of clothing, components, tools, and surfaces
8. Regeneration of demineralizer resins

The activation of impurities in systems other than the primary coolant has not been a major source of liquid radioactive wastes in light water reactors. However, at Peach Bottom Nuclear Power Station, a high temperature gas-cooled reactor, the absence of significant quantities of liquid radioactive wastes from other sources makes the primary shield cooling water the principal source of liquid wastes at this reactor.

Measurements of concentrations of specific radionuclides present in gaseous and liquid wastes are not generally available from nuclear power plants. Normally, facility operators report only gross beta-gamma activity and sometimes tritium activity in liquid wastes. Gaseous waste discharges are generally categorized and reported by facility operators as being either halogens and particulates or activation and noble gases. Some facility operating reports include results of specific radionuclide analyses of primary coolant. However, the relative abundance of radionuclides in the primary coolant may be different than the

relative abundance in liquid or gaseous waste effluents. Relative abundances of radionuclides in the primary coolant are functions of:

1. Cladding leakage
2. Temperature changes which may cause release of particles that have been attached to surface of the primary system
3. Use and effectiveness of coolant purification
4. Rate of primary system leakage
5. Chemical additives in the primary coolant
6. Type of coolant
7. Power history

Relative abundances of radionuclides in the waste effluents are primary functions of:

1. Their abundance in the primary coolant
2. Their respective half-lives
3. Design of the radwaste treatment system
4. Waste treatment practices

Waste treatment capabilities at selected operating nuclear facilities are summarized in Table 1.

Most radionuclides can be classified as either fission products or activation products. Tritium (^{3}H) however, is a special case in that it is produced both from fissioning and from neutron activation. It is also special because it is not affected by methods presently utilized in processing radioactive wastes. Therefore, the tritium released to the primary coolant or produced in the primary coolant is ultimately discharged to the environment in either liquid or gaseous form. Additional information on environmental tritium contamination from nuclear energy sources is provided in Reference 2.

OPERATING EXPERIENCE

Most experience on radioactive waste discharge to date has been with pressurized water reactors (PWR) and boiling water reactors (BWR). References 3 and 4 provide descriptions of these types of facilities. Limited operating experience has been gained from a gas-cooled reactor through the operation of Peach Bottom-1. Reference 5 describes this type of facility. Table 2 provides general information for facilities included in this report.

TABLE 1

WASTE PROCESSING CAPABILITY AT OPERATING NUCLEAR POWER FACILITIES

REACTOR	GASEOUS WASTE TREATMENT			LIQUID WASTE TREATMENT
	HOLDUP CAPACITY	PARTICULATE TREATMENT	IODINE TREATMENT	
PWRs				
Shippingport	60 days	none	none	gas scrubbing, evaporation, demineralization, filtration
Yankee	60 days	none	none	gas scrubbing, evaporation, demineralization
Indian Point-1	120 days	absolute filters	none	evaporation, demineralization, gas stripping, filtration
San Onofre	60 days	none	none	demineralization
Conn. Yankee	variable	fiberglass filter	none	evaporation, demineralization
BWRs				
Dresden-1	20 min.	absolute filters	none	filtration, evaporation, demineralization
Big Rock Point	30 min.	absolute filters	none	filtration, evaporation, demineralization
Humboldt Bay	18 min. design 40 min. actual	absolute filters	none	filtration, demineralization
HTGR				
Peach Bottom-1	variable	filtration	charcoal filters	demineralization

TABLE 2

GENERAL INFORMATION FOR FACILITIES INCLUDED IN THIS REPORT

Facility	AEC Docket No.	Power Level[6] MWt	Power Level[6] MWe Net	Location	Stack Height (Feet)	Stack Exhaust Rate (CFM)	Condenser Water for Dilution Flow Rate (GPM)	Body of Water Receiving Liquid Waste
PWRs								
Shippingport	None	505	90	Shippingport, Pa.	26[a]	9,000	114,000	Ohio River
Yankee	50-29	600	175	Rowe, Mass.	150	15,000	138,000	Deerfield River
Indian Point-1	50-3	615	265	Buchanan, N.Y.	400	280,000	300,000	Hudson River
San Onofre	50-206	1,347	430	San Clemente, Calif.	100	40,000	350,000	Pacific Ocean
Conn. Yankee	50-213	1,825	573	Haddam Neck, Conn.	175	70,000	372,000	Connecticut River
BWRs								
Dresden-1	50-10	700	200	Morris, Ill.	300	45,000	166,000	Illinois River
Big Rock Point	50-155	240	71	Charlevoix, Mich.	240	30,000	50,000	Lake Michigan
Humboldt Bay	50-133	240	68	Eureka, Calif.	250	12,000	100,000[b]	Humboldt Bay (Pacific Ocean)
HTGR								
Peach Bottom-1	50-171	115	40	Peach Bottom, Pa.	150	20,000	43,000	Susquehanna River

[a]Gas discharge stack; vapor container exhaust stack 116 feet.

[b]Flow rate for Humboldt Bay Unit 3 is 51,800 gpm. All calculations are based on a flow rate of 100,000 gpm which is the combined flow rate for Humboldt Bay Units 1, 2, and 3. Units 1 and 2 are fossil fuel plants.

Liquid Discharges

Quantities of gross beta-gamma activity (less tritium) discharged annually by each facility are shown in Table 3. Operating reports for Dresden-1 did not indicate total amount of radioactivity discharged, but gave an average contribution to the radioactivity in the condenser cooling water discharge canal. Except as noted in the table, the total annual discharge for Dresden was obtained by multiplying the facility's contribution to the concentration of radioactivity in the condenser cooling discharge canal times the annual flow rate of the canal as calculated from Table 2.

Calculations have been made based on data in Tables 2 and 3 and the liquid discharge limits for each facility to provide comparisons to discharge limits. These comparisons are provided in Table 4. These data do not include tritium which is presented later in Tables 6 and 7.

With the exception of Shippingport,* discharge limits are prescribed by Technical Specifications. All Technical Specifications limit concentrations in liquid effluents to those listed in Appendix B, Table II of 10CFR20. Without analysis for specific radionuclides, the limit is considered to be 10^{-7} μCi/ml. If liquid wastes are analyzed for specific radionuclides, discharge limits can be based on the maximum permissible concentration for the radionuclides present. These limits normally are higher than the limit for unidentified radionuclides. Most nuclear power plants discharge sufficiently small quantities of radioactivity in liquid wastes that dilution factors associated with the condenser cooling canal are sufficient to permit discharge on the basis of unidentified radionuclides. In most cases, there is no requirement for reporting radionuclide analyses of wastes in operating reports and as a result they are not normally reported. Therefore, the limit for unidentified radionuclides has been used as the basis for comparison in Table 4 except as noted.

It should be noted that the use of a limit for a mixture including unidentified nuclides which is adequate to show compliance, involves the arbitrary assignment of all the activity present to the most restrictive nuclide present. Liquid wastes necessarily involve a mixture of many fission and corrosion products. The resulting "percent of the discharge limit" therefore is artificially high. Limits based

*Shippingport has been developed and operated under AEC sponsorship. Shippingport radioactivity discharge limits are equal to or less than radiation protection standards set forth in Title 10, Code of Federal Regulations, Part 20, AEC Manual Chapter 0524 and a waste discharge permit from the Pennsylvania Sanitary Water Board.

TABLE 3

TOTAL ANNUAL LIQUID WASTE DISCHARGED[a]

GROSS β-γ LESS TRITIUM
(CURIES)

	1959	1960	1961	1962	1963	1964	1965	1966	1967	1968
PWRs										
Shippingport[b]	0.083	0.21	0.129	0.09	0.19	0.53	0.14	0.06	0.07	0.08
Yankee			0.008	0.008	0.003	0.002	0.029	0.036	0.055	0.008
Indian Point-1				0.130	0.164	13.0	26.3	43.7[c]	28.0[d]	34.6[d]
San Onofre									0.32	1.6
Conn. Yankee									0.216	3.9
BWRs										
Dresden-1		0.770	2.095	2.61	2.78	3.82	8.7	11.5	4.3[d]	6.1
Big Rock Point[e]				0.2	0.63	6.22	2.80	6.12	10.1	7.9
Humboldt Bay					0.397	0.664	1.89	2.10	3.13	3.2
HTGR										
Peach Bottom-1									0.0017	0.0004

[a]Based on operators' reports except as noted.
[b]Data taken from Radiological Health Data and Reports.
[c]Data from Reference 31.
[d]Data taken from Reference 8.
[e]Data taken from Reference 32.

TABLE 4

ANNUAL AVERAGE LIQUID RADIOACTIVE WASTE DISCHARGE CONCENTRATIONS
EXPRESSED AS PERCENT OF LIMIT [a]

PERCENT OF DISCHARGE LIMIT EXCLUSIVE OF TRITIUM

	1959	1960	1961	1962	1963	1964	1965	1966	1967	1968
PWRs										
Shippingport	0.37	0.93	0.57	0.40	0.84	2.34	0.62	0.27	0.31	0.35
Yankee			0.03	0.03	0.01	0.007	0.1	0.13	0.02[b]	0.03
Indian Point-1				0.22	.26	22	43	70.1	1.55[c]	1.65[e]
San Onofre									0.46	2.35
Conn. Yankee									0.01[c]	5.35
BWRs										
Dresden-1		2.33	6.34	7.9	8.42	11.6	26.4	34.8	13.0	18.5
Big Rock Point				4.46	14.2	54.4	42.4	62.5	50.4[d]	82.3[f]
Humboldt Bay					2.77	3.37	9.52	12.15	16.89	19.7
HTGR										
Peach Bottom-1									0.02	0.005

[a]Percent of limit calculations were based on the following: (1) 10CFR20 limit for unidentified radionuclides in water of 10^{-7} μCi/ml except as noted, (2) average flow rates in the discharge canals equal to those given in Table 2.

[b]Concentration limit 10×10^{-7} μCi/ml, based on radionuclide analysis.

[c]Concentration limit 30×10^{-7} μCi/ml, based on radionuclide analysis.

[d]Concentration limit 2×10^{-7} μCi/ml, based on radionuclide analysis.

[e]Concentration limit 35×10^{-7} μCi/ml, based on radionuclide analysis.

[f]Concentration limit 1.5×10^{-7} μCi/ml, based on radionuclide analysis.

on complete analysis, if performed, would be expected to be substantially higher than those used; and the percentages in Table 4 would be substantially less.

The data in Table 3 are further expanded in Table 5 to show the ratio of gross beta-gamma radioactivity discharged to power produced for each facility. These data indicate that with the exception of Indian Point-1, BWR's have discharged more gross beta-gamma radioactivity in liquid waste per unit of power produced than other types. There is nothing inherent in a BWR which would cause it to produce more radioactivity in liquid waste than would be produced in PWR. These higher quantities from BWRs are believed to be primarily a result of fuel cladding leakage.

Figure 1 compares annual discharges of radioactivity in liquid wastes to power generation for facilities included in this report. This figure indicates that both electrical generation and liquid waste discharges are increasing with time, but that waste discharges are not increasing as rapidly as electrical generation.

Discharges of tritium in liquid wastes are generally reported separately from gross beta-gamma discharges primarily because of its high relative abundance in liquid wastes and its relatively high discharge concentration limit as compared to the concentration limit for unidentified radionuclides. Table 6 provides a summary of available data concerning tritium in liquid waste discharges. The number of curies of tritium discharged in liquid waste is high relative to the number of curies discharged of other radionuclides. However, the relative hazard per curie of tritium is low. The data in Table 6 are expanded in Table 7 to show derived average discharge concentrations and percent of the discharge limit for 1968. By comparing the data in Tables 4 and 7, it is evident that concentrations of gross beta-gamma activity (exclusive of tritium) in liquid discharges more nearly approach the limit used than do concentrations of tritium. This is significant because current methods for treatment of liquid wastes are ineffective in reducing quantities of tritium discharged.

Table 5 provides a ratio of curies of tritium discharged in liquid waste to electrical power produced by each facility. The data show that PWRs discharge much higher quantities than the other two types. Higher tritium concentrations in PWRs are due in part to neutron reactions with boron which is added in the form of boric acid to the primary coolant. Since the boron is in solution with the primary coolant, there is no cladding barrier to retain the tritium so produced. This is not the case with a BWR where the boron is used in the form of cladded plates or curtains. Other sources of tritium which are common to both PWRs and BWRs include fission product tritium and reactions with lithium, nitrogen, helium-3, poison material used in control rods or plates, and reactions with structural material.

TABLE 5

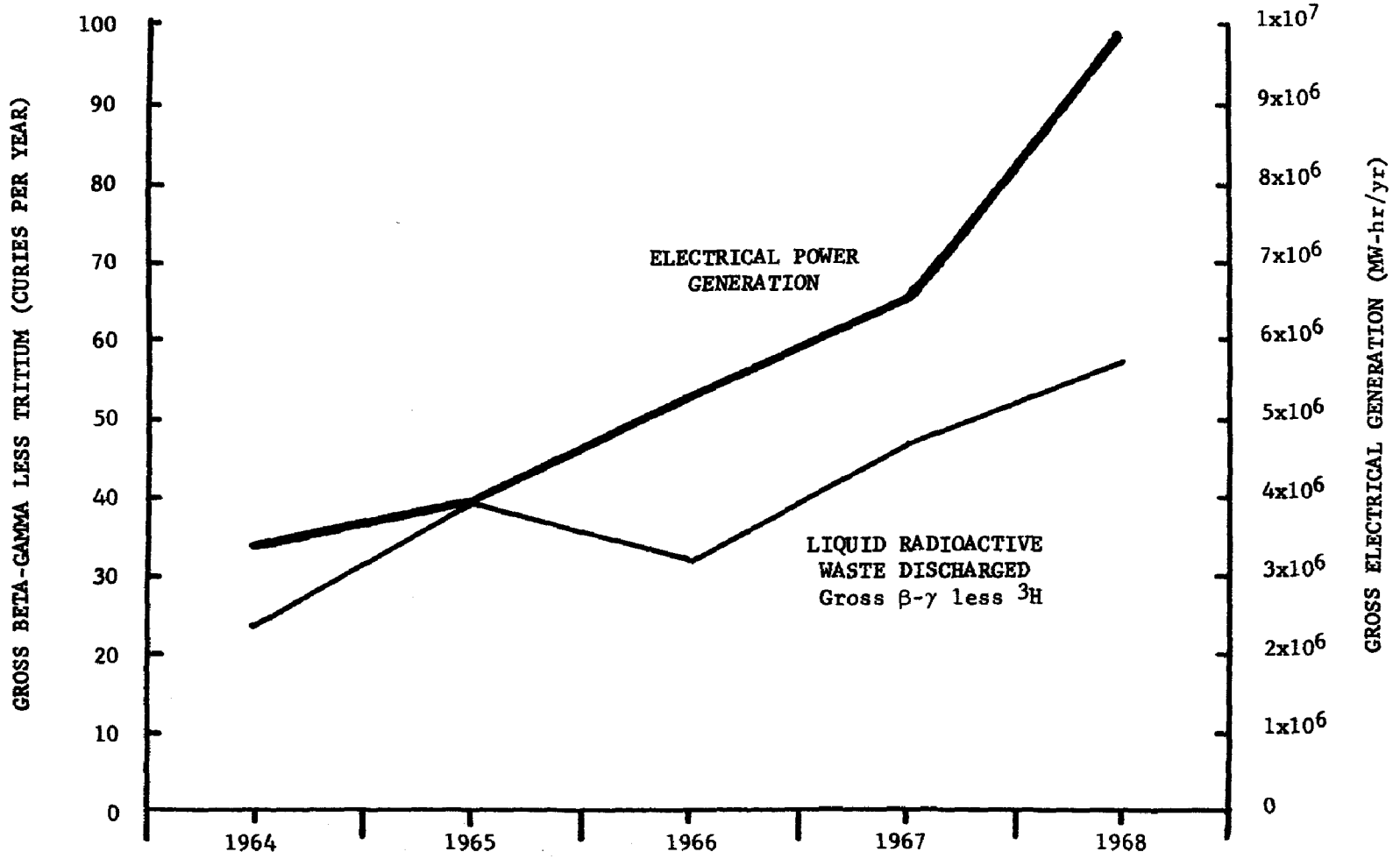
COMPARISON OF RADIOACTIVE WASTE DISCHARGES TO ELECTRICAL POWER GENERATION

Facility & Reactor Type	Period Covered[a]	Total Waste Discharges During Period			Total Gross Electrical Generation (MWe-hr)	μCi Discharges/MWe-hr Gross		
		Liquid		Gaseous		Liquid		Gaseous
		Gross β-γ[b] (Curies)	Tritium (Curies)	Gross β-γ (Curies)		Gross β-γ Less ^3H	Tritium	Gross β-γ
PWRs								
Shippingport	1959-68 (1968)	1.6 (0.08)	281 (35.2)	0.58 (0.001)	3.5×10^6 (4.1×10^5)	0.46 (0.20)	80 (86)	0.17 (0.002)
Yankee	1961-68 (1968)	0.15 (0.008)	6,080[c] (1,170)	37 (0.68)	8.9×10^6 (1.2×10^6)	0.02 (0.007)	1,220[c] (950)	4.16 (0.57)
Indian Point-1	1964-68 (1968)	112 (34.6)	1,080[d] (787)	166 (59.6)	6.2×10^6 (1.6×10^6)	17.1 (21.6)	319[d] (492)	26.77 (37.2)
San Onofre	1967-68 (1968)	1.92 (1.6)	 (2,350)	8.8 (4.83)	1.7×10^6 (1.4×10^6)	1.1 (1.1)	 (1,680)	5.18 (3.45)
Conn. Yankee	1967-68 (1968)	4.1 (3.9)	1,960 (1,740)	3.75 (3.74)	3.7×10^6 (3.2×10^6)	1.1 (1.3)	530 (544)	1.01 (1.17)
BWRs								
Dresden-1	1961-68 (1968)	41.9 (6.1)	 (2.9)	2.8×10^6 (2.4×10^5)	7.6×10^6 (9.7×10^5)	5.5 (6.3)	 (3)	3.68×10^5 (2.47×10^5)
Big Rock Point	1962-68 (1968)	33.1 (7.5)	 (34)[e]	1.33×10^6 (2.32×10^5)	1.7×10^6 (4.5×10^5)	19.5 (17.6)	 (76)	7.82×10^5 (5.16×10^5)
Humboldt Bay[f]	1963-68 (1968)	11.4 (3.20)	< 248 < (6.6)	2.23×10^6 (8.53×10^5)	1.8×10^6 (4.7×10^5)	6.3 (6.9)	< 138 (15)	1.24×10^6 (1.83×10^6)
HTGR								
Peach Bottom-1	1967-68 (1968)	0.002 (0.0004)		117 (109)	3.1×10^5 (1.5×10^5)	0.006 (0.003)		377 (727)

[a] 1968 data is in parentheses
[b] Exclusive of tritium
[c] Based on data for 1965-68 wherein electrical generation was 4.98×10^6 MWe-hr.
[d] Based on data for 1967-68 wherein electrical generation was 3.39×10^6 MWe-hr.
[e] Based on an upper limit calculation wherein all liquid waste released during 1968 was assumed to contain as much tritium as was in primary system water.
[f] Data from P.G.&E. records.

FIGURE 1

COMPARISON OF ANNUAL LIQUID RADIOACTIVE WASTE DISCHARGED TO ANNUAL ELECTRICAL POWER GENERATION[a]



[a]Totals for all facilities included in this report.

TABLE 6

TOTAL ANNUAL LIQUID TRITIUM DISCHARGES (CURIES)

	1958	1959	1960	1961	1962	1963	1964	1965	1966	1967	1968
PWRs											
Shippingport[a]	50.0	64.0	99.0	13.2	1.33	2.17	1.39	3.04	27.3	34.8	35.2
Yankee[b]				--	--	--	--	1,300	1,920	1,690	1,170
Indian Point-1[c]					--	--	--	--	125	297	787
San Onofre[c]										--	2,350
Connecticut Yankee[d]										221	1,740
BWRs											
Dresden-1[c]			--	--	--	--	--	--	--	--	2.9
Big Rock Point[c]					--	--	--	--	--	--	34[e]
Humboldt Bay[c]						< 214	< 100	< 54	< 60	< 166	< 6.6
HTGR											
Peach Bottom-1[c]										--	Neg.

[a]Data for years 1958 through 1960 taken from Shippingport Operations from Power Operation After First Refueling to Second Refueling, (May 6, 1960 to August 16, 1961), DLCS-36402; data for years 1961 through 1968 taken from Radiological Health Data and Reports.

[b]Data taken from Yankee Nuclear Power Station Operations Reports; tritium analysis was not included in the operating reports prior to 1965.

[c]Data taken from Reference 8.

[d]Data taken from Connecticut Yankee Atomic Power Company operations reports.

[e]Based on an upper limit calculation wherein all liquid waste released during 1968 was assumed to contain as much tritium as was in primary system water.

TABLE 7

1968 TRITIUM DISCHARGES IN LIQUID

WASTES COMPARED TO AEC LIMITS[a]

Pressurized Water Reactors	Average Discharge[b] Concentration μCi/ml	Percent of Limit
Shippingport	1.6×10^{-7}	0.0053
Yankee	4.5×10^{-6}	0.15
Indian Point-1	1.56×10^{-6}	0.045
San Onofre	3.3×10^{-6}	0.11
Connecticut Yankee	2.4×10^{-6}	0.08
Boiling Water Reactors		
Dresden-1	9×10^{-9}	0.0003
Big Rock Point	3.6×10^{-7}	0.012
Humboldt Bay[c]	4.1×10^{-8}	0.0014
Gas Cooled Reactors		
Peach Bottom-1	Negligible	Negligible

[a]Based on 10CFR20 limit for unrestricted areas of 3×10^{-3} μCi/cc and dilution in the condenser cooling water discharge canal.

[b]Calculated based on annual quantity of tritium discharged and the condenser cooling water flow rate.

[c]From P.G. & E. records.

Gaseous Wastes

Gaseous wastes may be in the form of particulates, volatiles (such as iodine) or gases. The gases constitute the major portion of discharged radioactivity via the stack and are generally referred to as activation and noble gases.

Technical Specifications limit average discharge rates for radioactive gaseous wastes such that the average annual concentrations at the plants's exclusion boundary will not exceed those listed in Appendix B, Table II of 10CFR20. Additional limits (usually a factor of 10 higher than limits for average release rates) are established for maximum release rates. Limits for gaseous release rates are a function of the atmospheric dilution available between the point of release and the exclusion boundary. Since atmospheric dilution factors are affected by stack height, distance from stack to exclusion boundary, topography, and local meteorology, gaseous discharge limits vary widely from facility to facility. Limits for gaseous releases are usually expressed in μCi/sec with limits for iodines and particulates being relatively more restrictive than the limits for activation and noble gases. The reason for the more restrictive limits for iodines and particulates is their potential for reconcentration through environmental media. For example, the discharge limit for ^{131}I is generally set at a factor of 700 below the rate that would produce the 10CFR20 concentration limit of 1×10^{-10} μCi/ml at the exclusion boundary. This is to compensate for possible reconcentration through the pasture-cow-milk chain. The isotopic mixtures of noble gas discharges are such that the maximum permissible concentration in the environment as calculated from values listed in 10CFR20 range from 3×10^{-8} μCi/ml with decay of less than two hours to 3×10^{-7} μCi/ml at ages of three days and longer.[9]

Table 8 provides an annual summary of discharges of activation and noble gases. With the exception of Humboldt Bay, the facilities included in this report do not include separate discharge rates of halogens and particulates in their operating reports. However, these data have been reported by the AEC in Reference 8 for 1967 and 1968 and are reproduced in Table 9 along with the percent of discharge limit for each facility.

Table 10 provides a summary of annual average discharge rates for activation and noble gases expressed as percent of discharge limits. Unlike other facilities, the discharge limit for Shippingport does not take into consideration atmospheric dilution between the point of discharge and the exclusion boundary. The discharge limit for Shippingport is 3×10^{-7} μCi/ml in the stack. The discharge limit 1.26 μCi/sec used in Table 10 for calculating percent of limit is derived based on a stack discharge rate of 9,000 cfm.

Review of the average discharge rate for each of the facilities listed in Table 8 reveals that boiling water reactors discharge very much larger quantities of activation and noble gases than pressurized water reactors.

TABLE 8

TOTAL ANNUAL GASEOUS WASTE DISCHARGED[a]
NOBLE AND ACTIVATION GASES
(CURIES)

	1959	1960	1961	1962	1963	1964	1965	1966	1967	1968
PWRs										
Shippingport[b]	0.014	0.029	0.103	0.012	0.351	0.0024	0.032	0.030	0.002	0.001
Yankee			0.00096	21.700	7.4	0.95	1.7	2.4	2.3	0.68
Indian Point-1				--	0.0072	13.2	33.1	36.4	23.4	59.7
San Onofre									4.02	4.83
Connecticut Yankee									0.021	3.74
BWRs										
Dresden-1[c]			34,800	284,000	71,600	521,000	610,000	736,000	260,000[d]	240,000
Big Rock Point[e]				25.6	803	783	132,000	705,000	264,000	232,000
Humboldt Bay[f]					716	5,975	197,000	282,000	896,000	853,000
HTGR										
Peach Bottom-1								0.00126	7.76	109

[a]Based on operators' reports except as noted.
[b]Data from Radiological Health Data and Reports. Data corrected for 1963 and 1964 from Reference 33.
[c]1961 and 1962 based on maximum rate of noble fission gas activity discharged: 1963-1968 based on average activity discharge rate for the year while plant was operating.
[d]Data taken from Reference 8.
[e]Data taken from Ref. 32.
[f]Data from P G & E records
---indicates plant was operational but discharge data was not available.

TABLE 9

RELEASES OF HALOGENS AND PARTICULATES FROM POWER REACTORS IN GASEOUS EFFLUENTS[8]

Facility and type	1967			1968	
	Released (Curies)	Permissible[a] (Curies)	Percent of Permissible	Released (Curies)	Percent of Permissible
PRESSURIZED WATER REACTORS					
Yankee	Negligible	0.03	< 1	Negligible	< 1
Indian Point-1	Negligible	7	< 1	Negligible	< 1
San Onofre	Negligible	0.8	< 1	Negligible	< 1
Conn. Yankee	0.001	0.2	0.5	Negligible	< 1
BOILING WATER REACTORS					
Dresden-1	0.039	100	0.04	0.15	0.15
Big Rock Point	0.25	38	0.66	0.09	0.24
Humboldt Bay	0.64	5.7	11	0.45	8
HIGH TEMPERATURE GAS COOLED					
Peach Bottom-1	Negligible	0.09	< 1	Negligible	< 1

[a]Where the Technical Specifications do not state an annual limit for the iodines and particulates, an MPC value of 1×10^{-10} μCi/cc was used. This MPC is based on the most restrictive isotope normally found -- ^{131}I. The annual limit was reduced by a factor of 700 to account for reconcentration.

TABLE 10

ANNUAL GASEOUS RADIOACTIVE WASTE DISCHARGES EXPRESSED AS PERCENT OF LIMIT[a]
(NOBLE AND ACTIVATION GASES)

	1959	1960	1961	1962	1963	1964	1965	1966	1967[b]	1968[b]
PWRs										
Shippingport	0.035	0.073	0.26	0.03	0.87	0.006	0.08	0.075	0.005	0.0025
Yankee			0.000014	0.32	0.11	0.014	0.025	0.035	0.036	0.008
Indian Point-1					4.5×10^{-7}	0.00083	0.0020	0.0022	0.0015	0.0037
San Onofre									0.0024	0.00085
Conn. Yankee									0.00003	0.0039
BWRs										
Dresden-1			0.158	1.29	0.32	2.37	2.77	3.34	0.87	1.09
Big Rock Point						0.0025	0.43	2.27	0.85	0.74
Humboldt Bay					0.045	0.38	12.5	17.8	56.7	54.0
HTGR										
Peach Bottom-1								6.67×10^{-6}	0.04	0.087

[a]Percent of limit calculations were based on the following: (1) values as given in Table II for stack flow rates and Table VIII for annual quantities discharged, (2) discharge limits presented in Appendices I through IX except as noted. In cases where the discharge limit is expressed as a factor times MPC and no MPC is given, 3×10^{-8} μCi/cc is used.

[b]Limits for 1967 and 1968 from Reference 8 except for Shippingport. Shippingport limits from Reference 10 for 1959 through 1968.

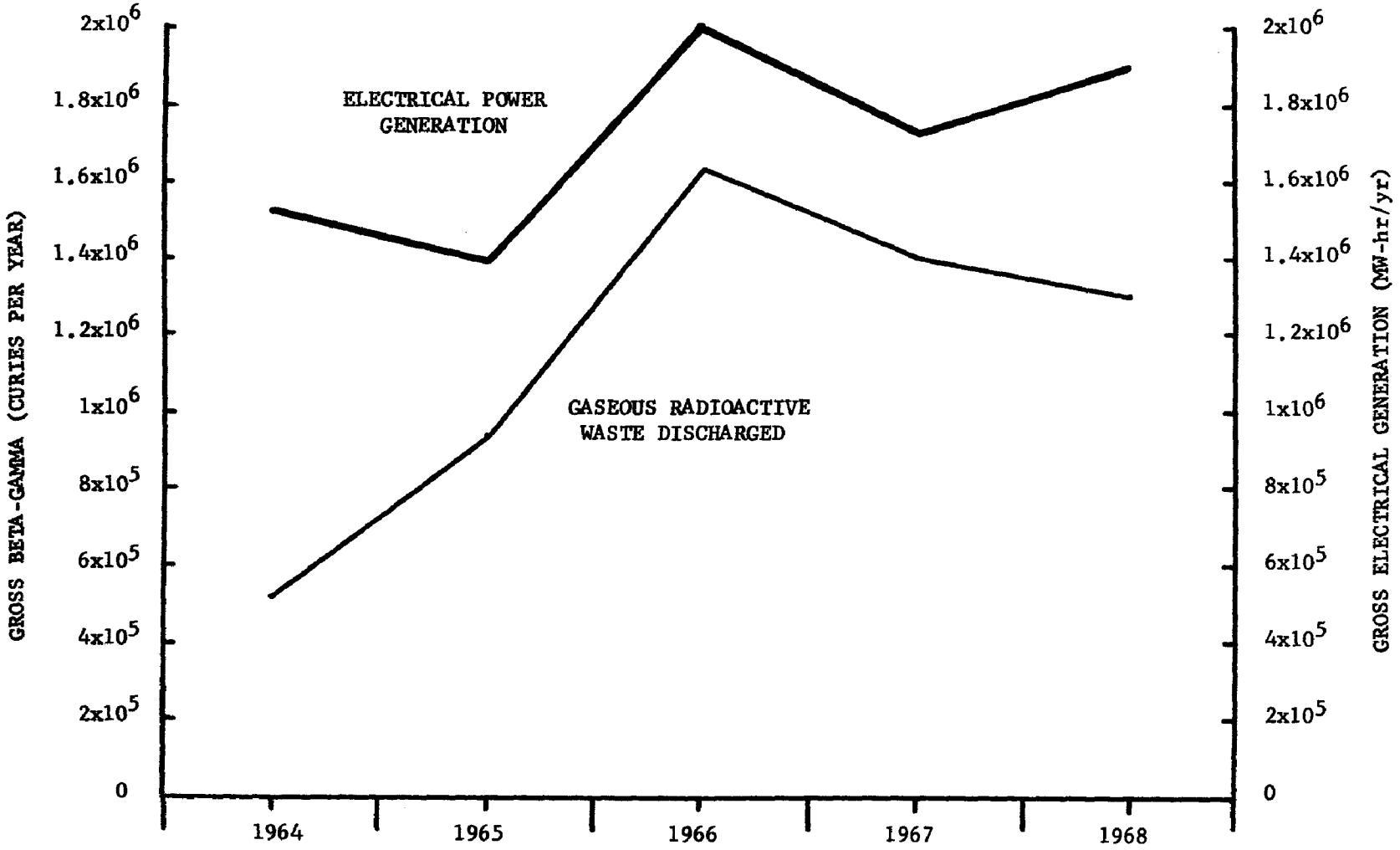
This trend is further analyzed in Table 5 where the ratio of gaseous waste discharged to electrical power produced is presented. The greater quantities of radioactive gases discharged from BWRs are a result of a shorter hold-up for decay prior to discharge to the environment. Gases in the primary coolant system of a BWR are carried over with the steam to the condenser air ejectors where they are immediately ejected as non-condensables and discharged to the environment with a hold-up time of 20-30 minutes. The radioactive gases generated in a PWR are retained for longer periods in the primary coolant system. Those that are released from the coolant system are stored in tanks for further decay prior to discharge and therefore PWRs discharge less short-lived gaseous wastes to the atmosphere. As a result, population exposure to external radiation from gaseous releases will be higher in the immediate vicinity of a BWR than in the immediate vicinity of a PWR. However, since the increased quantities discharged from a BWR are made up of short-lived radionuclides, the contribution by BWRs to general population exposures should not be greater than for other types of reactors.

Figures 2, 3, and 4 provide plots of electrical generation and gaseous waste discharges as a function of time. The graphs have been separated into BWRs and PWRs in Figures 2 and 3. Since gaseous waste discharges at Indian Point-1 have been much higher than at other PWRs, its data were plotted separately in Figure 4. These figures show relationships for each year, but there is not enough history to establish definite trends. Several facilities have been involved in research programs utilizing the reactor to test different fuels and cladding, in some cases resulting in significant releases of fission products from the fuel elements to the primary coolant. Humboldt Bay has experienced a high percentage of leaking fuel cladding resulting in relatively large amounts of fission products being released to the primary coolant. Such releases affect the shapes of the discharge curves in Figures 2, 3, and 4, and therefore reduce their significance as far as establishing trends.

The Appendices which follow are in alphabetical order by facility title and present discussions and data pertaining to each facility. The primary sources of these data are the operating reports issued by the facility operators with supplemental data from AEC reports and from direct correspondence with facility operators. The lack of detailed data pertaining to specific radionuclides in waste discharges is apparent. The best data of this type is found in Appendix 3, which is a result of special study[11] performed around Dresden-1 by the Bureau of Radiological Health.

FIGURE 2

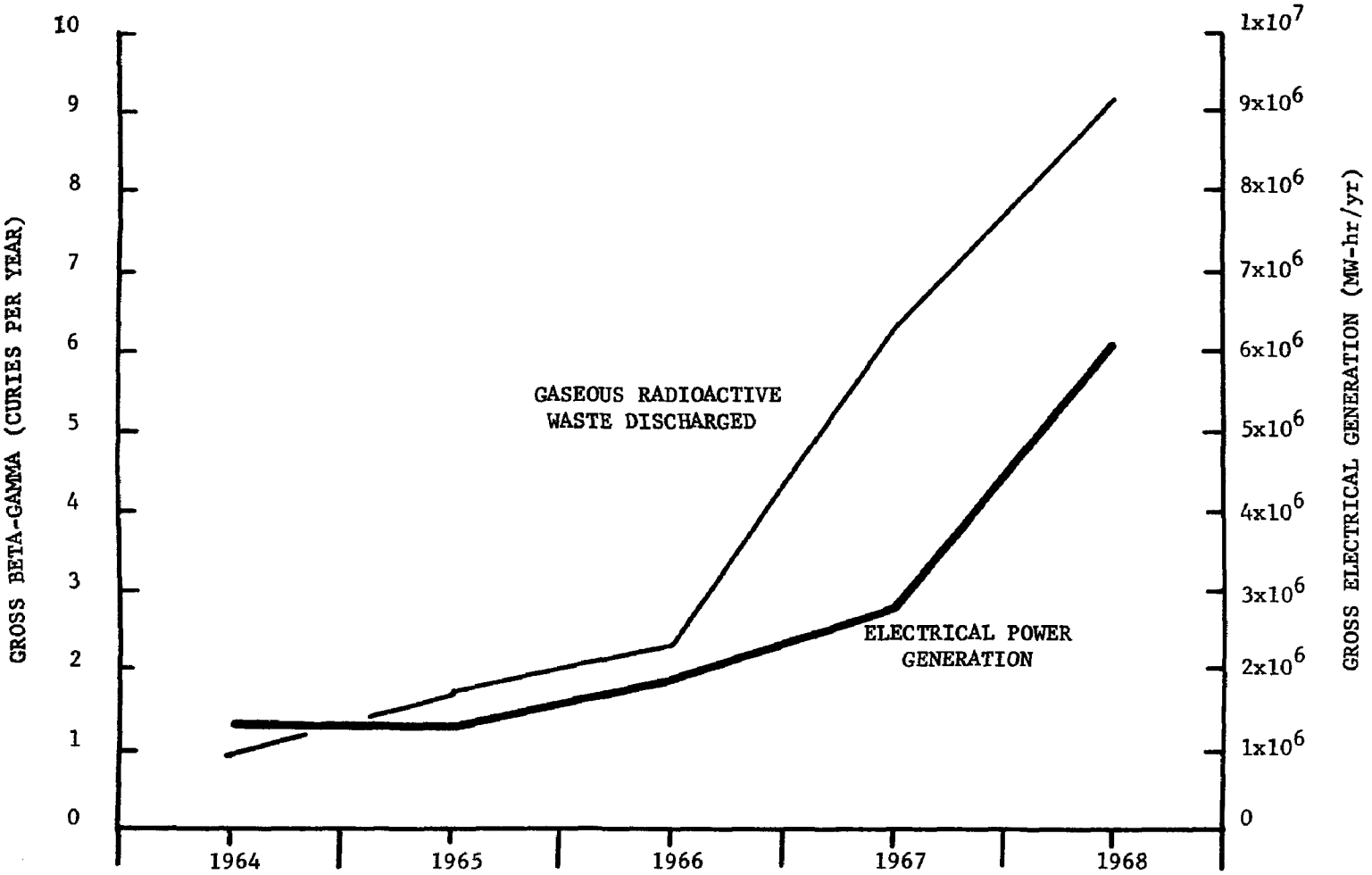
COMPARISON OF ANNUAL GASEOUS RADIOACTIVE WASTE DISCHARGED TO ANNUAL ELECTRICAL POWER GENERATION FOR BWR FACILITIES[a]



[a]Includes all BWR facilities listed in Table 5.

FIGURE 3

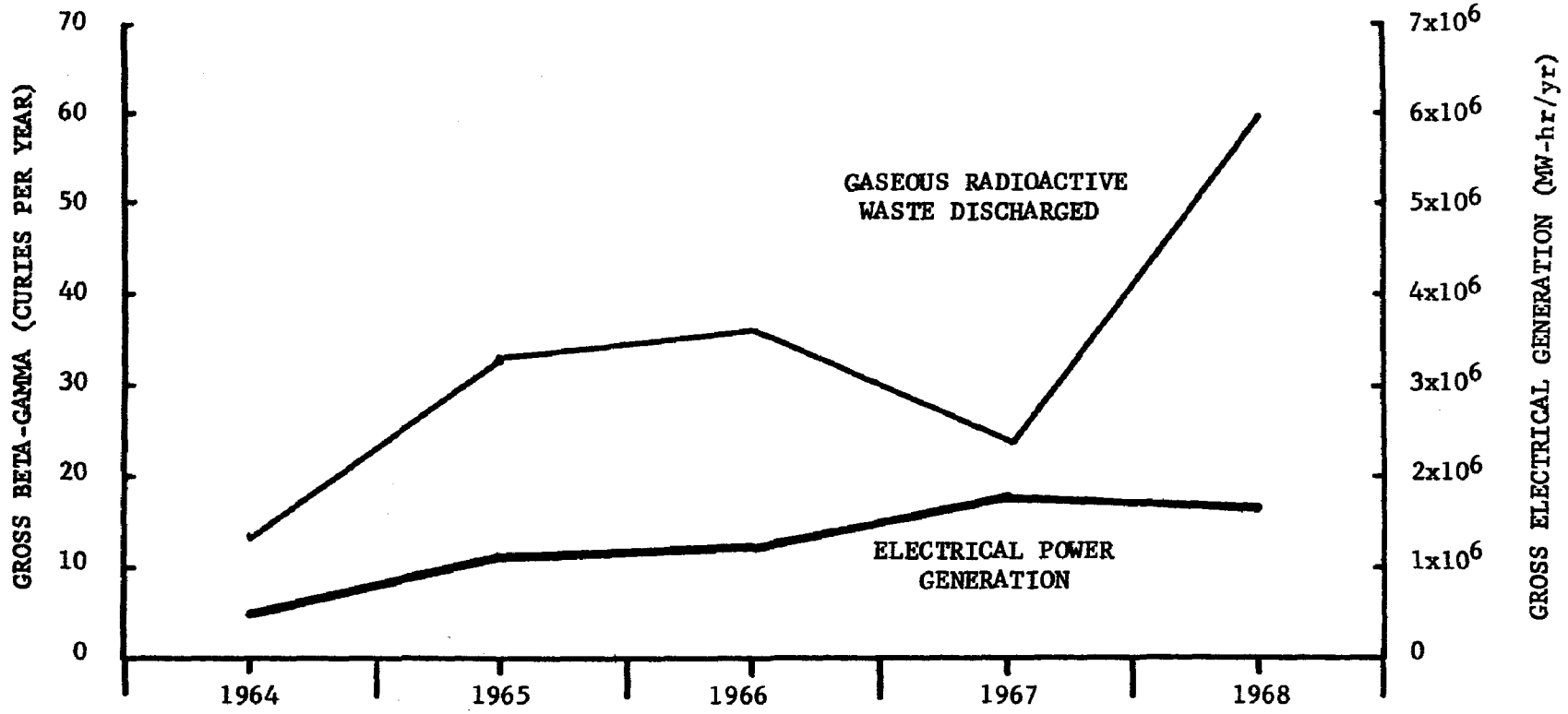
COMPARISON OF ANNUAL GASEOUS RADIOACTIVE WASTE DISCHARGED TO ANNUAL ELECTRICAL POWER GENERATION FOR PWR FACILITIES[a]



[a]Includes all PWR facilities included in this report with the exception of Indian Point Station, Unit 1

FIGURE 4

COMPARISON OF ANNUAL GASEOUS RADIOACTIVE WASTE DISCHARGED TO ANNUAL ELECTRICAL POWER GENERATION FOR INDIAN POINT STATION, UNIT 1



SUMMARY

Data pertaining to discharges of radioactive liquid and gaseous wastes from nine selected operating nuclear power facilities are presented and discussed. The following summary is based on these data:

1. Experience to date with nuclear power plants has shown that careful waste management practices, engineered safeguards, and proper operating procedures generally result in radioactivity levels in waste effluents of a few percent or less of the AEC's licensed discharge limits. Exceptions are mostly associated with either an unusually high percentage of leaky fuel elements or with liquid discharge limits which are artificially low as a result of not analyzing liquid wastes for radionuclide content.

2. Technical Specifications for all facilities limit liquid discharges such that average annual concentrations of radioactivity in the condenser cooling discharge canal will be less than values listed in Appendix B, Table II, 10CFR20. The limits for gaseous discharges vary from facility to facility, depending on available dilution factors in the atmosphere. They have varied in the manner in which they are expressed; however, the AEC is in the process of developing uniform reporting requirements. They also have varied in that the limits for halogens and particulates for some of the early reactors did not include the "700" factor to account for possible reconcentration through environmental media.

3. Facility operating reports, which are prepared to demonstrate that the facility operator is in compliance with specific requirements of the operating license, vary widely as to units used and types of information concerning discharges of radioactive wastes. In general, there is a paucity of information in these reports concerning specific radionuclides discharged. Information which is available indicates that relative concentrations of specific radionuclides in waste discharges are not constant. Therefore, assumptions which must be made (in the absence of data) in order to analyze potential exposure pathways to man are not necessarily valid.

The AEC has been developing requirements for isotopic analyses and for reporting of more detailed data on quantities and concentrations of specific radionuclides in discharges. The selection of nuclides for analysis will take into account expected release quantities and possibilities for reconcentration through environmental media.

4. A number of comparisons have been made of power produced versus liquid or gaseous waste discharges. The most predominant trends shown in these comparisons are that boiling water reactors discharge

relatively large quantities of gaseous waste and pressurized water reactors discharge relatively high quantities of tritium in liquid wastes. No obvious trend is discernible concerning quantities of waste discharged as a function of power generation. This is to be expected since fuel cladding integrity and waste treatment practices are major factors in determining the quantity of waste available for discharge.

5. Information is available concerning the type of waste treatment facilities installed at each nuclear power station. However, in operating reports, there is generally no indication as to the types of waste treatment that were used. A comparison of waste treatment practices to other parameters such as power history, primary coolant characterization, and quantities and types of waste discharged, would be useful in analyzing the possible effectiveness of proposed waste treatment facilities at nuclear power facilities.

APPENDIX I

BIG ROCK POINT NUCLEAR POWER STATION

Big Rock Point Nuclear Power Plant is located on the northeast shore of Lake Michigan near the city of Charlevoix. It uses a boiling water reactor of General Electric design with an authorized thermal power level of 240 MWt which is equivalent to about 71 net electrical megawatts. The plant attained criticality in September 1962 and first began producing significant amounts of power in January 1963. The plant is operated by Consumers Power Company.

Gaseous waste discharges are limited by Technical Specifications[12] to one curie per second for fission and activation gases. Discharges of halogens and particulates to the atmosphere are limited to 10CFR20 limits x 1.2 x 10^{10} cm^3/sec. No concentration limit was prescribed; however, assuming a concentration limit of 10^{-10} μCi/cc which is the 10CFR20 limit for ^{131}I, the limiting discharge rate for halogens and particulates is 1.2 μCi/sec. Operating reports did not include information on quantities of halogens and particulates discharged.

Liquid waste discharge concentrations are limited to concentrations specified in Appendix B, Table II of 10CFR20. For unidentified gross beta-gamma activity this would be 10^{-7} μCi/cc. Occasional partial analyses have indicated that 90% of the activity has consisted of a combination of ^{65}Zn, ^{58}Co, ^{137}Cs, ^{140}Ba, and ^{140}La.

Discharges of radioactivity to the environment are shown on Figures I-1 through I-3. These plots compare primary coolant concentrations and waste discharges to electrical power generation.

Since discharge data are not available on a monthly or shorter basis, it is not practical to compare discharges to plant maintenance and operations. It can be seen from Figure I-1 that the primary coolant activity increased significantly during the first few years of operation and then began to decrease with further operation. A similar trend is noted in the gaseous waste discharge as shown in Figure I-3. The decrease occurred even though there was no reduction in power generation. Review of Figure I-2 indicates no particular trend of this type in liquid waste discharge.

The Big Rock Point reactor has been used in an extensive R & D program involving the testing of fuel and fuel cladding.[13] Tests have included different metals for fuel cladding, thick and thin claddings with various heat treatments, powder and pellet forms, plutonium fuel, and UO_2 molten fuel.[14] As a result of the research, the reactor has

experienced more fuel failures than would otherwise be expected. Table I-1 summarizes fuel failure experience at the Big Rock Point plant, and relates it to gaseous discharge rates.

Several of the shutdowns were associated with insertion and removal of fuel rods for research programs. A long shutdown from 11-64 to 4-65 was for testing and repair work on the thermal shield hold-down assemblies.

FIGURE I - 1

BIG ROCK POINT NUCLEAR PLANT

PRIMARY COOLANT ACTIVITY
(GROSS BETA-GAMMA, LESS TRITIUM)

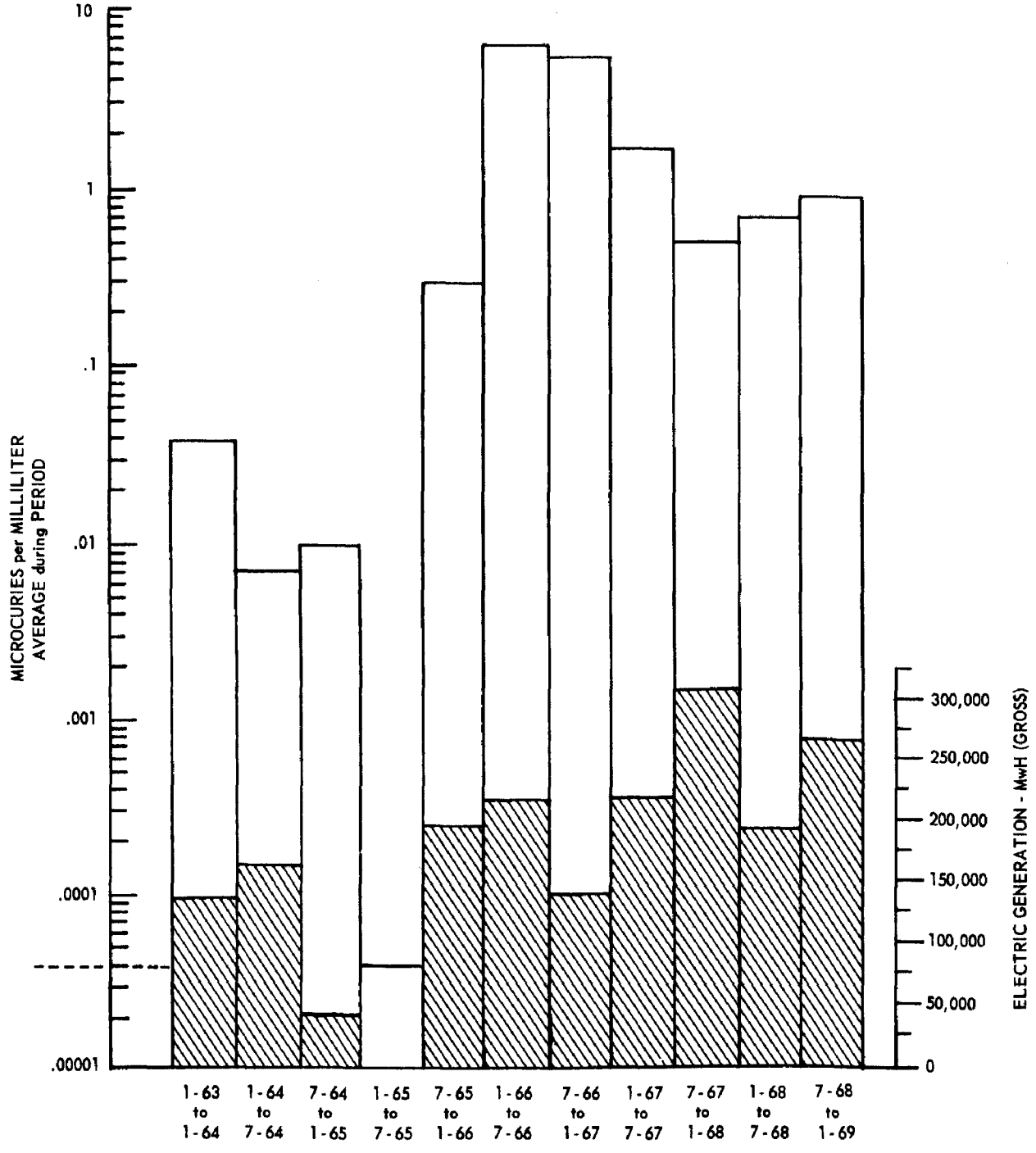


FIGURE I - 2

BIG ROCK POINT NUCLEAR PLANT

LIQUID WASTE DISCHARGED
(GROSS BETA-GAMMA, LESS TRITIUM)

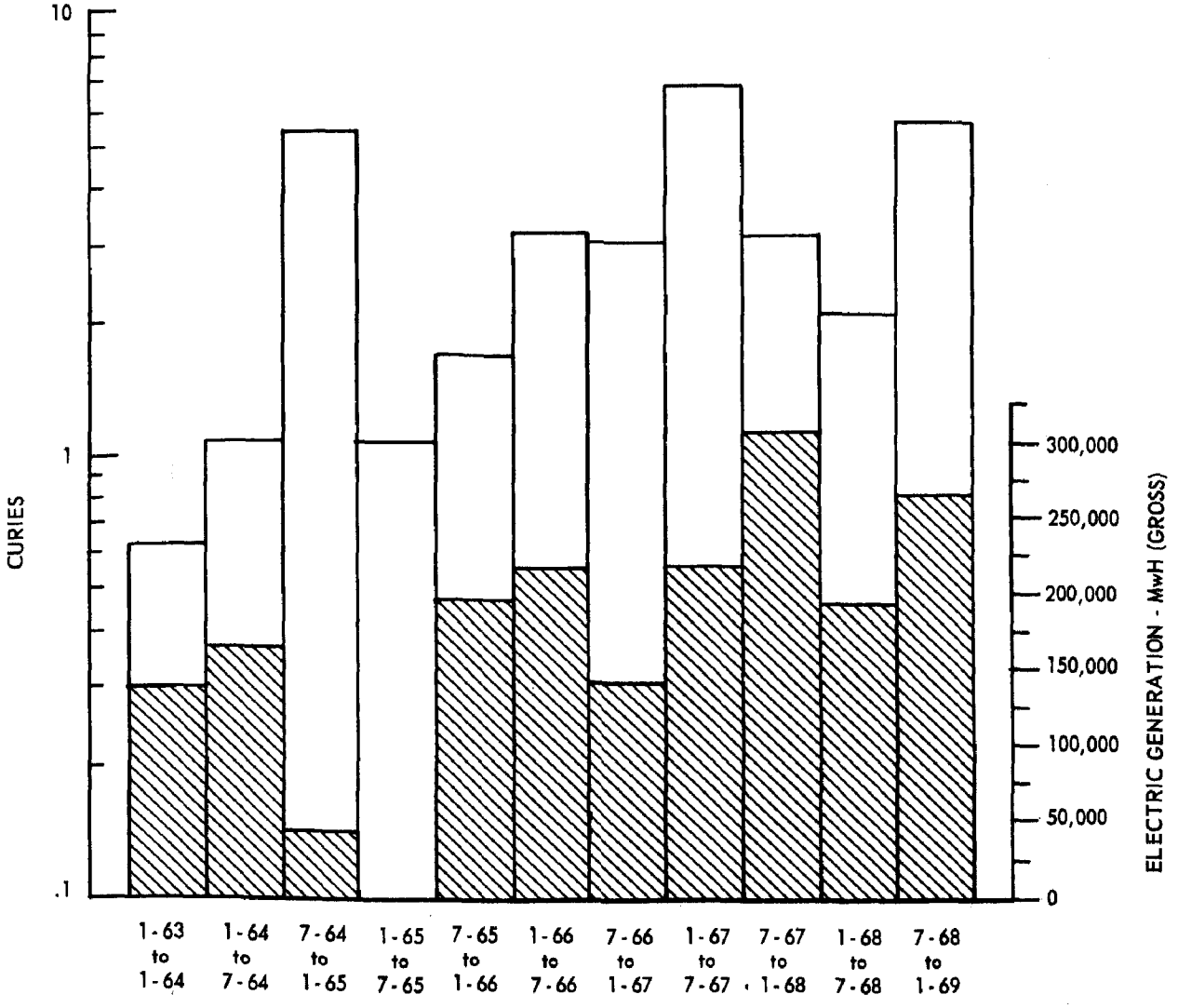


FIGURE I - 3

BIG ROCK POINT NUCLEAR PLANT

GASEOUS WASTE DISCHARGED AND ELECTRICAL GENERATION
(ACTIVATION AND FISSION GASES)

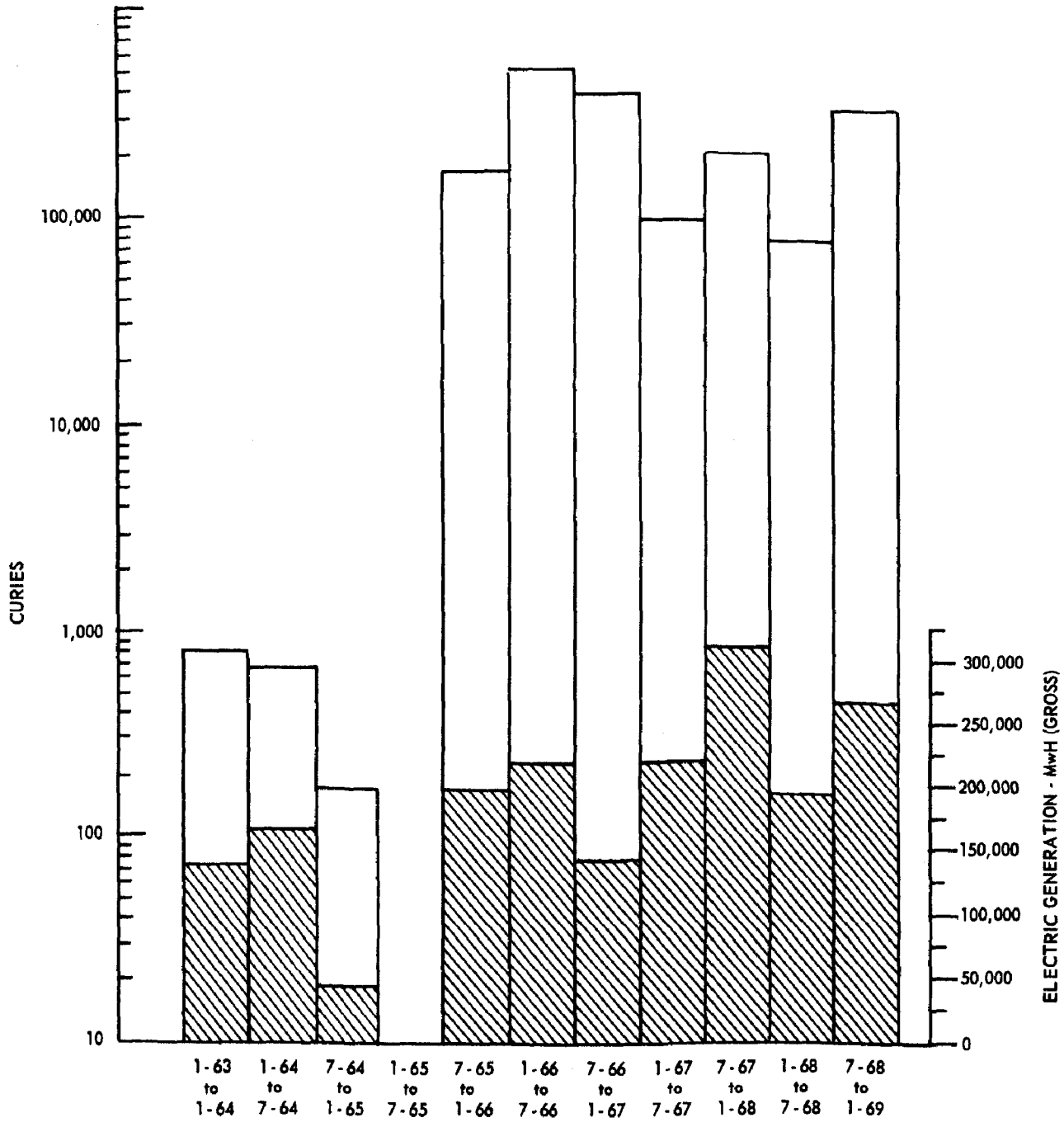


TABLE I-1[14]

FUEL FAILURE EXPERIENCE AT THE BIG ROCK POINT NUCLEAR PLANT

Run Number	Time Interval	Power Produced MWh(e) gross	Off-Gas Activity (μCi/sec)		Fuel Failures at End of Run		
			Average	Peak	Bundles	Rods-Known	Rods-Est
I	12/62-4/66	662,287	6,000	50,000	4	9	23
II	5/66-9/66	161,844	40,000	75,000	11	34	112
III	10/66-5-67	260,906	12,000	20,000	1	1	4
IV	6/67-2/68	390,851	10,000	15,000	22	43	110
V	3/68-6-68	124,696	4,000	6,000	9	16	37
VI	7/68-5/69	404,427	25,000	40,000	9	42	82
Total	-	2,005,011	-	-	56	145	368

Note: A normal core loading consists of 84 fuel bundles, containing ~ 11,000 fuel rods. Through 5/69, a total of 258 fuel bundles, containing 29,066 rods, have been irradiated in the Big Rock Point Reactor.

APPENDIX II

CONNECTICUT YANKEE ATOMIC POWER PLANT

Connecticut Yankee Atomic Power Plant is located in Haddam, Connecticut on the Connecticut River. It has a pressurized water reactor with an authorized power level of 1,825 megawatts thermal, which corresponds to 573 net megawatts electrical. The plant is operated by Connecticut Yankee Atomic Power Corporation.

Primary coolant activity and monthly liquid and gaseous waste discharges reported in monthly operating reports[15] have been plotted versus time in months (see Figures II-1 through II-5) such that these data may be compared to electrical generation and to other plant operations. Both liquid and gaseous discharges have been broken down into gross beta-gamma and tritium.

Technical Specifications[16] limit average annual concentrations in liquid waste discharges to those published in Appendix B, Table II of 10CFR20. Without isotopic analyses, this is 10^{-7} μCi/ml. Gaseous waste discharges are limited as follows:

> "When averaged over any calendar year, the release rate of radioactivity consisting of noble gases and other isotopes with half-lives less than eight days discharged at the plant stack shall not exceed $3 \times 10^4 \times$ (MPC) curies per second, where MPC is the value in microcuries per cubic centimeter given in Appendix B, Table II, Column 1 of 10 CFR Part 20. The maximum release rate when averaged over any one hour shall not exceed 10 times the yearly averaged limit.
>
> At any time when the average release rate for a week exceeds 30% of the annual average limit given above, the licensee shall make provisions for sampling iodine-131 to assure that its release rate averaged over any calendar year does not exceed 66 (MPC) curies per second."

Assuming an MPC of 3×10^{-8} μCi/ml for activation and noble gases and 1×10^{-10} μCi/ml for ^{131}I, the discharge limits would be 900 μCi/sec for activation and noble gases and 6.6×10^{-3} μCi/sec for ^{131}I. Figures II-2 through II-5 provide plots of discharge data versus time as reported in operating reports.

Insufficient information has yet been accumulated to establish trends for waste discharges. However, there appears to be a trend towards a release of more liquid tritium with increased power history.

The discontinuous plots of gaseous waste in Figure II-4 represent periodic discharges from decay tanks. Gaseous tritium releases plotted in Figure II-5 are results of purges of the containment volume. The following notes refer to Figures II-1 through II-5:

A Plant shutdown 621 hrs. for repairs

B Plant shutdown 463 hrs. for repairs

C Plant shutdown 80 hrs. for turbine control valve inspection

D Plant shutdown 208 hrs. for repairs, inspection, and modification

E Plant shutdown 456 hrs. for plant maintenance and turbine valve modification

F Plant shutdown 258 hrs. for turbine valve modification

G Core loading

H Initial criticality

I First power generation

J Lithium 7 hydroxide added

K Lithium 7 concentration reduced

L Boron recovery system not operated

M Purification system shutdown

N Waste liquid processed through evaporator

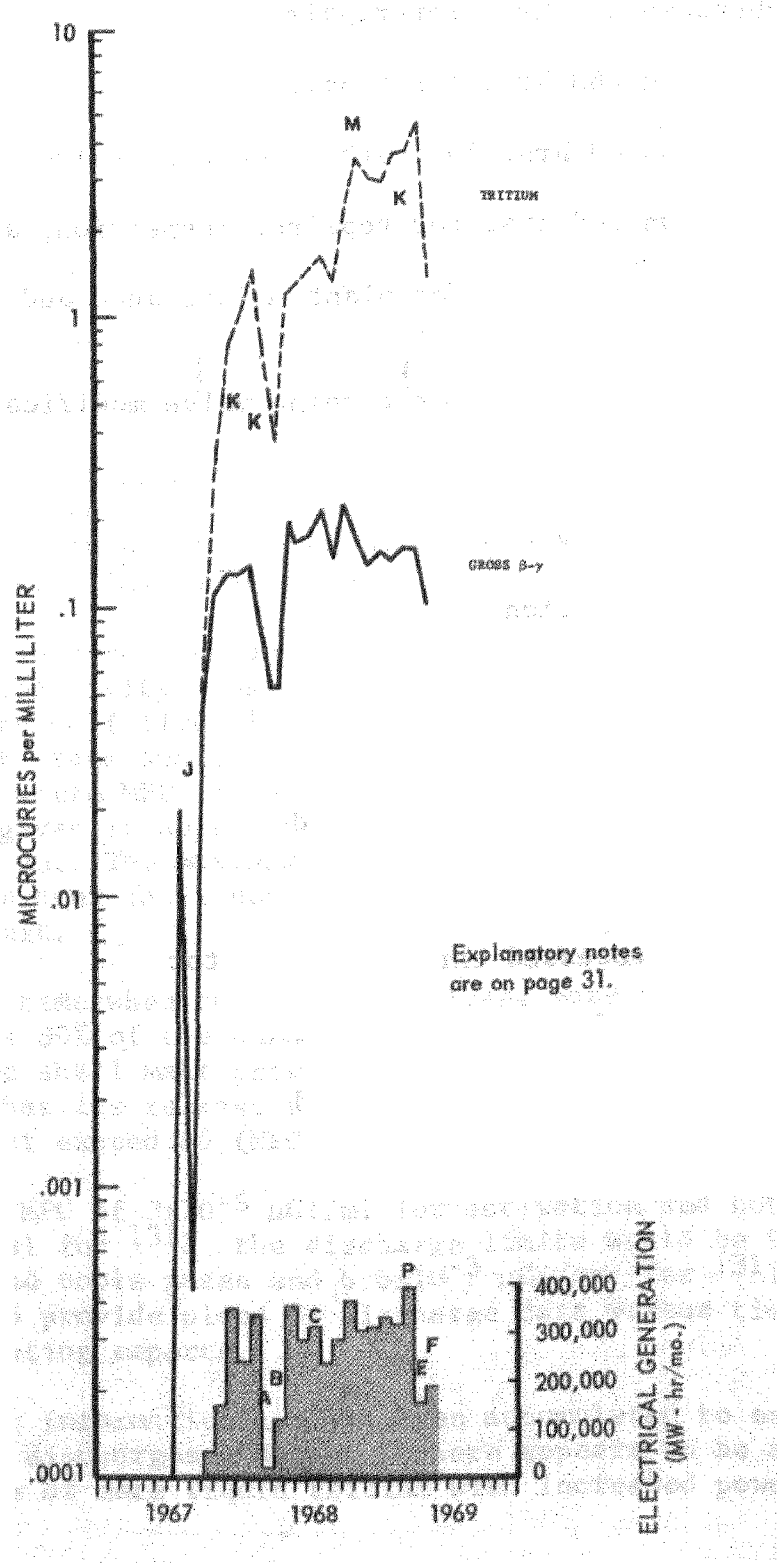
O Primary to secondary leakage noted

P Power increase to 600 MWe approved

FIGURE II - 1

CONNECTICUT YANKEE ATOMIC POWER PLANT

PRIMARY COOLANT ACTIVITY

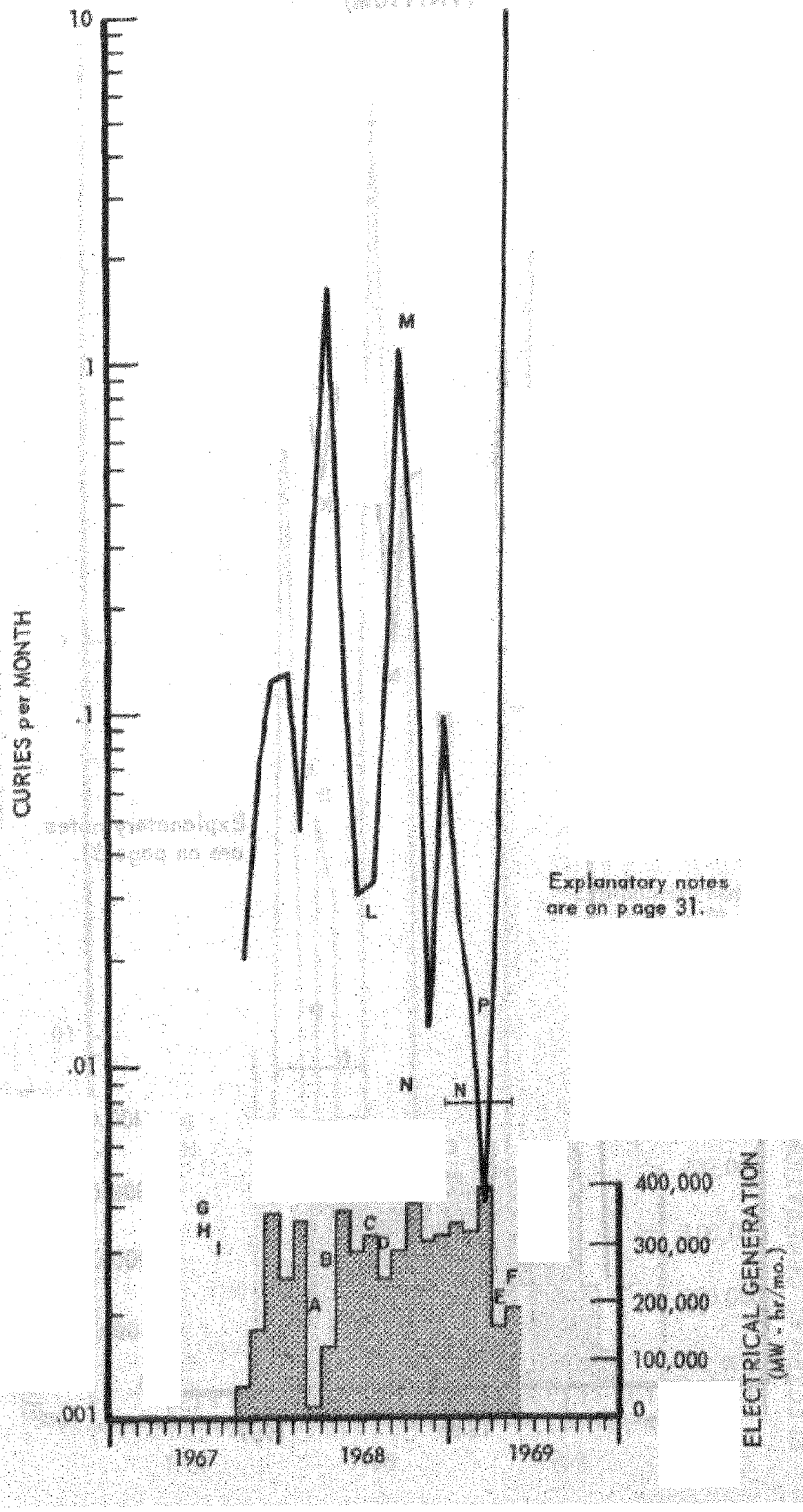


Explanatory notes are on page 31.

FIGURE II - 2

CONNECTICUT YANKEE ATOMIC POWER PLANT

LIQUID WASTE DISCHARGED
(GROSS BETA GAMMA LESS TRITIUM)

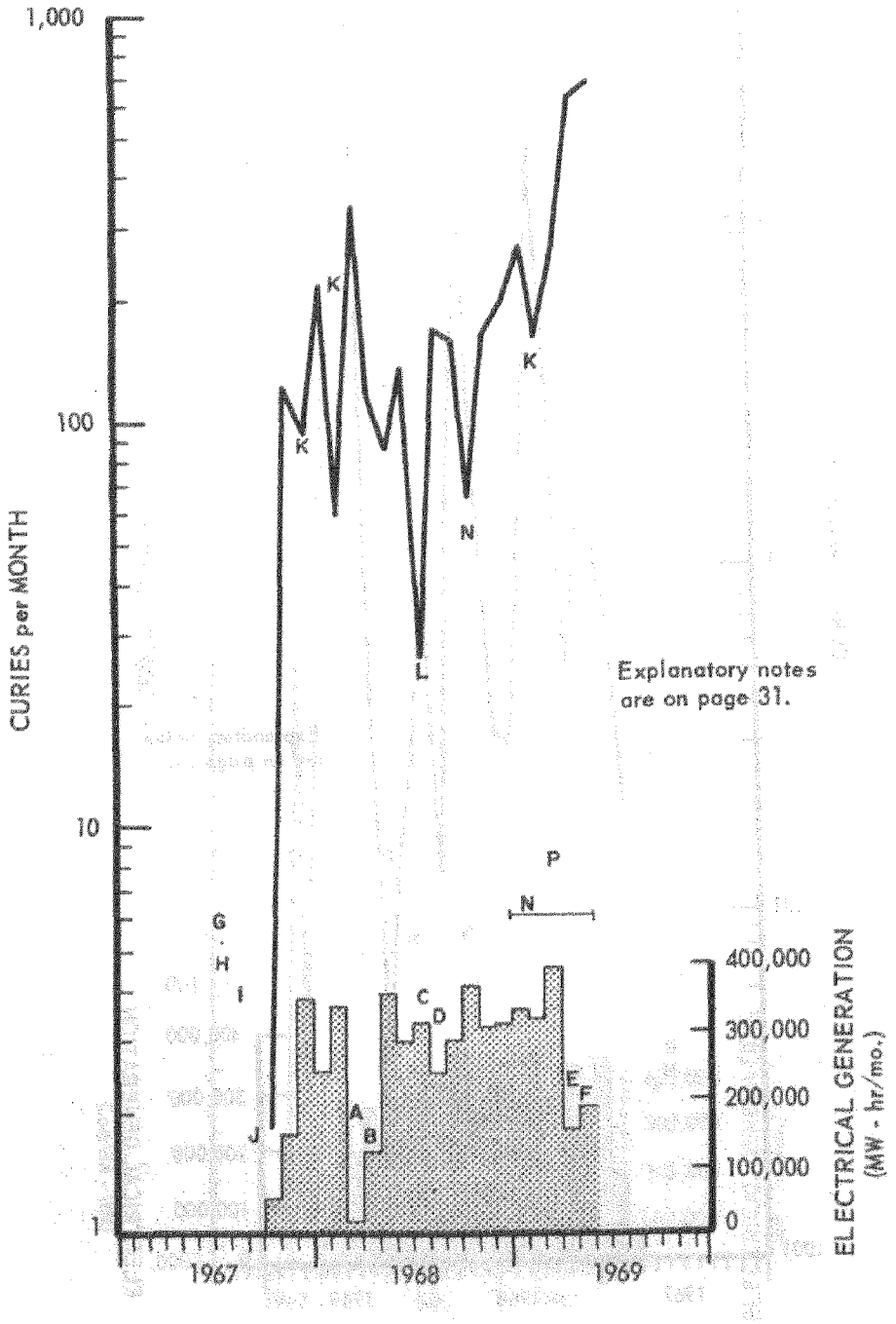


Explanatory notes are on page 31.

FIGURE II - 3

CONNECTICUT YANKEE ATOMIC POWER PLANT

LIQUID WASTE DISCHARGED
(TRITIUM)

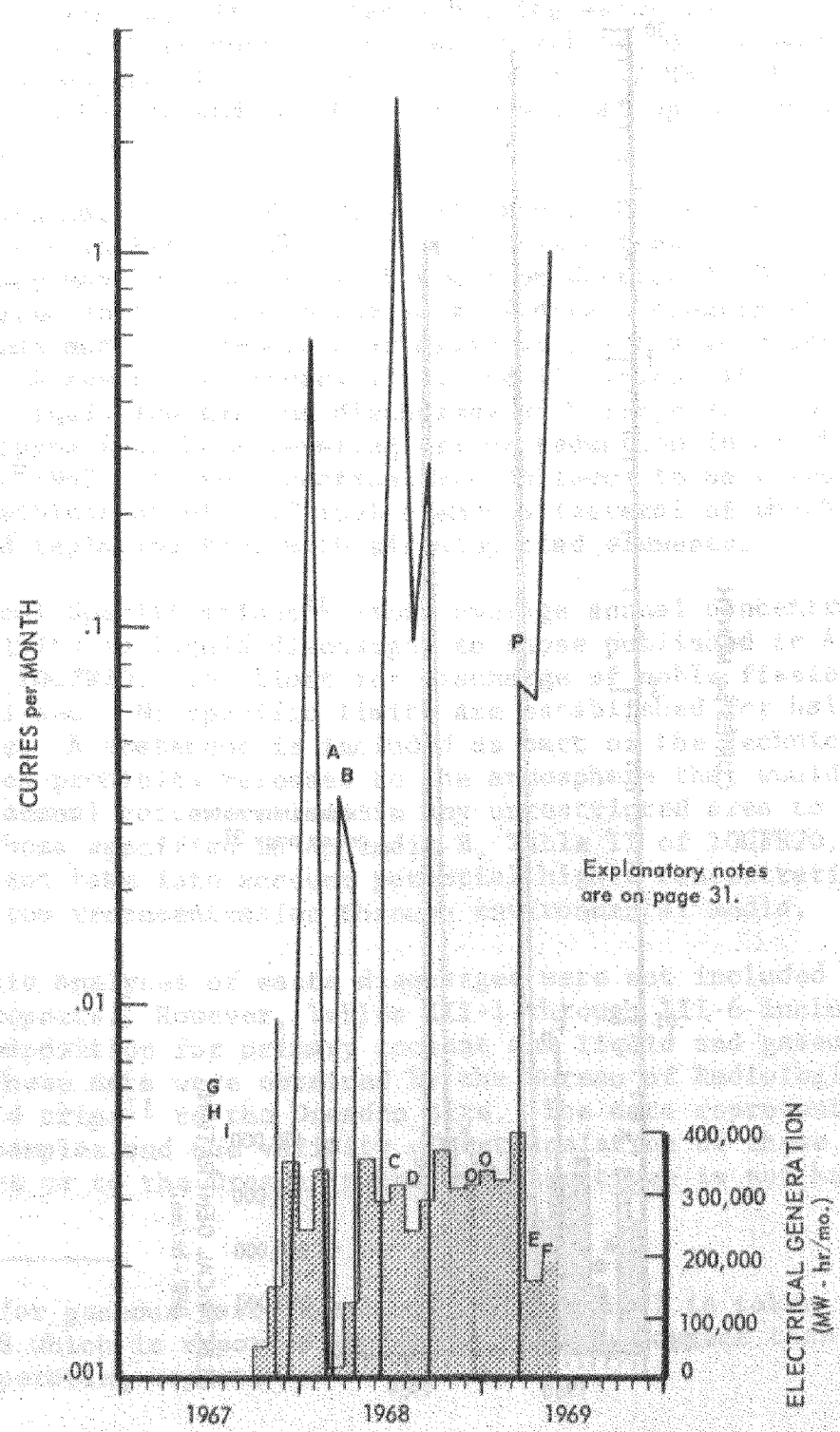


Explanatory notes
are on page 31.

FIGURE II - 4

CONNECTICUT YANKEE ATOMIC POWER PLANT

GASEOUS WASTE DISCHARGED
(GROSS BETA GAMMA LESS TRITIUM)

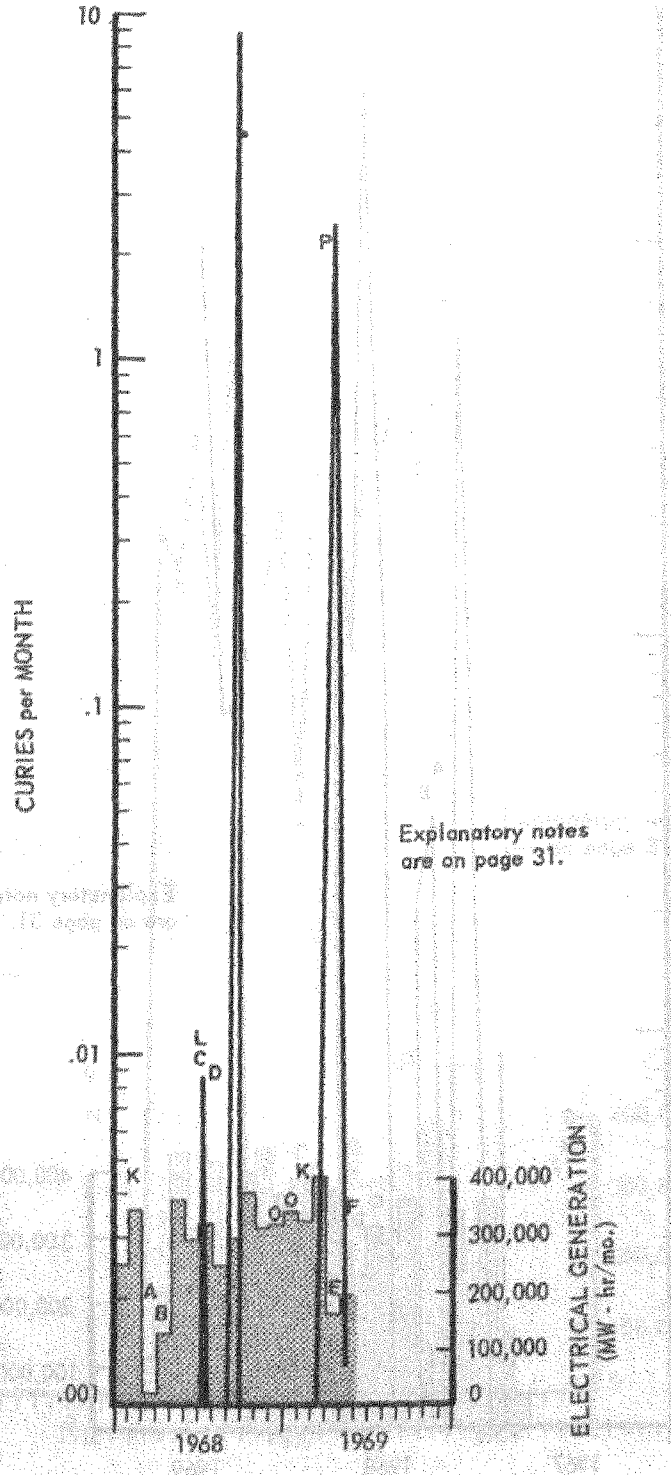


Explanatory notes
are on page 31.

FIGURE II - 5

CONNECTICUT YANKEE ATOMIC POWER PLANT

GASEOUS WASTE DISCHARGED
(TRITIUM)



Explanatory notes
are on page 31.

APPENDIX III

DRESDEN NUCLEAR POWER STATION UNIT 1

Dresden Nuclear Power Station is located on the Illinois River near Morris, Illinois. It utilizes a boiling water reactor of General Electric design with an authorized power level of 700 megawatts thermal, which corresponds to a net power level of 200 MWe. It is operated by Commonwealth Edison and has been in commercial operation since August 1960.

Waste discharges from Dresden-1 are reported[17] on an annual basis and are plotted in Figures III-1 and III-2 as a function of time and such that they may be compared with power production.* Due to the long periods covered in the reports and lack of detail concerning discharges, no attempt was made to compare discharges to plant maintenance and operations. A review of Figures III-1 and III-2 indicates a general increase in liquid and gaseous discharges with increased power history. There also appears to be a leveling off or reduction in discharges beginning in 1967. These reductions are believed to be a result of removing stainless steel clad fuel elements (several of which were leaking) and replacing them with zircaloy clad elements.

Technical Specifications[18] limit average annual concentrations of radioactivity in liquid discharges to those published in Appendix B, Table II of 10CFR20. The limit for discharge of noble fission gases is 7×10^{5} μCi/sec. No specific limits are established for halogens and particulates. A statement is included as part of the Technical Specifications which prohibits releases to the atmosphere that would result in average annual concentrations in any unrestricted area to levels in excess of those specified in Appendix B, Table II of 10CFR20. This limit does not take into account potential higher concentrations resulting from reconcentration through environmental media.

Isotopic analyses of waste discharges were not included in the operating reports. However, Tables III-1 through III-6 include detail isotopic composition for primary coolant and liquid and gaseous discharges. These data were obtained by the Bureau of Radiological Health during field trips[11] to the Dresden site. The data represent a limited number of samples and the validity of extrapolation of these data to other plants or to the Dresden plant at other times is not known.

---

*The data for gaseous waste discharged during 1967 is taken from Reference 8 which is reported by AEC to be more correct than the data from the operating report.

FIGURE III - 1

DRESDEN NUCLEAR POWER STATION, UNIT 1

LIQUID WASTE DISCHARGED
(GROSS BETA GAMMA LESS TRITIUM)

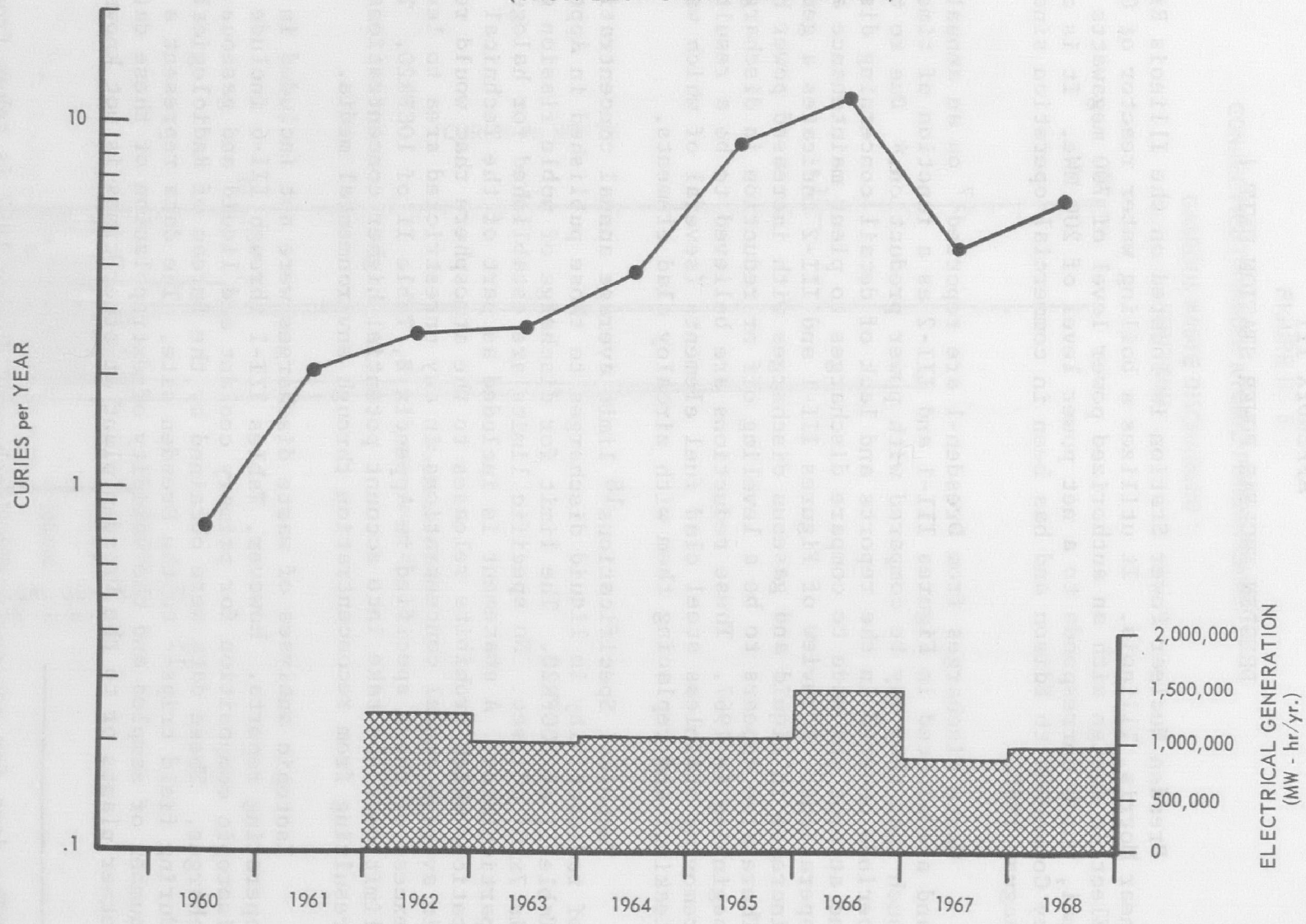


FIGURE III - 2

DRESDEN NUCLEAR POWER STATION, UNIT 1

GASEOUS WASTE DISCHARGED
(NOBLE FISSION GASES)

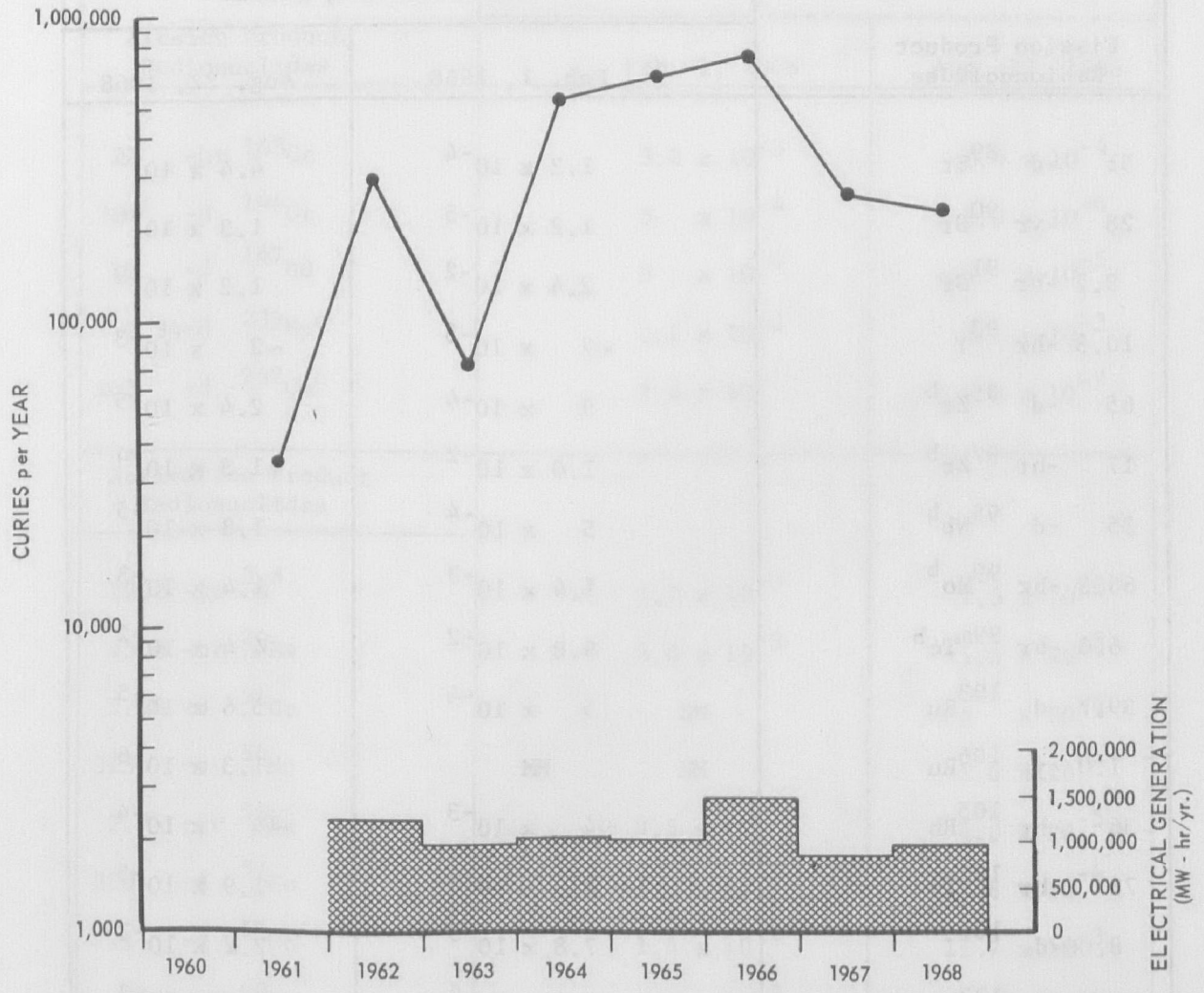


TABLE III-1

DRESDEN NUCLEAR POWER STATION UNIT 1

RADIONUCLIDE CONCENTRATION IN PRIMARY COOLANT[a]

Fission Product Radionuclides	Concentration, μCi/ml	
	Feb. 1, 1968	Aug. 22, 1968
51 -d ^{89}Sr	1.2×10^{-4}	4.4×10^{-5}
28 -yr ^{90}Sr	1.2×10^{-5}	1.3×10^{-6}
9.7 -hr ^{91}Sr	2.4×10^{-2}	1.2×10^{-2}
10.3 -hr ^{93}Y	$\sim 7 \times 10^{-3}$	$\sim 2 \times 10^{-3}$
65 -d ^{95}Zr[b]	9×10^{-4}	2.4×10^{-5}
17 -hr ^{97}Zr[b]	1.0×10^{-2}	1.3×10^{-4}
35 -d ^{95}Nb[b]	5×10^{-4}	1.8×10^{-5}
66.3 -hr ^{99}Mo[b]	1.4×10^{-3}	1.4×10^{-3}
6.0 -hr ^{99m}Tc[b]	9.0×10^{-2}	4.4×10^{-2}
39.7 -d ^{103}Ru	5×10^{-4}	5.6×10^{-5}
1.0 -yr ^{106}Ru	NM	1.3×10^{-6}
36 -hr ^{105}Rh	$\sim 4 \times 10^{-3}$	$\sim 4 \times 10^{-4}$
78 -hr ^{132}Te	8×10^{-4}	1.9×10^{-5}
8.06-d ^{131}I	7.8×10^{-3}	2.2×10^{-3}
20.9 -hr ^{133}I	6.6×10^{-2}	2.3×10^{-2}
6.7 -hr ^{135}I	1.3×10^{-1}	3.6×10^{-2}
2.07-yr ^{134}Cs[c]	1.3×10^{-5}	2.3×10^{-5}
13 -d ^{136}Cs[c]	3 x 10-5	2.4×10^{-5}
30 -yr ^{137}Cs	3×10^{-5}	4.4×10^{-5}
12.8 -d ^{140}Ba	1.4×10^{-3}	1.0×10^{-3}
32.5 -d ^{141}Ce	9×10^{-4}	4.4×10^{-5}

TABLE III-1 (Cont'd)

DRESDEN NUCLEAR POWER STATION UNIT 1

RADIONUCLIDE CONCENTRATION IN PRIMARY COOLANT[a]

Fission Product Radionuclides	Concentration, μCi/ml	
	Feb. 1, 1968	Aug. 22, 1968
33 -hr ^{143}Ce	3.2×10^{-3}	1.0×10^{-4}
284 -d ^{144}Ce	3×10^{-4}	8.0×10^{-6}
11 -d ^{147}Nd	3×10^{-4}	$\sim 3 \times 10^{-5}$
2.34-d ^{239}Np[c]	2.1×10^{-2}	4.1×10^{-3}
163 -d ^{242}Cm[c]	2.2×10^{-7}	1.2×10^{-7}
Activation Product Radionuclides		
12.3 -yr ^{3}H[d]	1.7×10^{-3}	1.3×10^{-3}
15.0 -hr ^{24}Na	3.0×10^{-3}	1.6×10^{-3}
27.8 -d ^{51}Cr	NM	$\sim 5 \times 10^{-4}$
313 -d ^{54}Mn	NM	1.6×10^{-6}
2.7 -yr ^{55}Fe	9.5×10^{-5}	4.0×10^{-5}
270 -d ^{57}Co	$\sim 1 \times 10^{-5}$	1.6×10^{-6}
71 -d ^{58}Co	1.4×10^{-2}	1.7×10^{-3}
5.26-yr ^{60}Co	2.2×10^{-3}	2.6×10^{-4}
12.7 -hr ^{64}Cu	$\sim 1 \times 10^{-2}$	2.2×10^{-3}
244 -d ^{65}Zn	NM	4.0×10^{-6}
253 -d ^{110m}Ag	NM	1.9×10^{-6}
115 -d ^{182}Ta	2×10^{-5}	$\sim 7 \times 10^{-7}$

[a]Concentrations at time of sampling. [b]Also activation products.
[c]Formed by (n,γ) reactions with uranium or fission products.
[d]Also from ternary fission. NM: Not measured.

TABLE III-2

DRESDEN NUCLEAR POWER STATION UNIT 1

RADIONUCLIDE CONCENTRATIONS IN HIGH CONDUCTIVITY LIQUID WASTE, pCi/ml[a]

Radio-Nuclide	Nov. 12, 1967	Jan. 13, 1968	March 16, 1968	June 25, 1968	Aug. 20, 1968	Average
^{3}H	900[b] (NM)	950 (NM)	520 (NM)	770 (NM)	1,100 (NM)	850
^{54}Mn	NM (< 1)[b]	NM (< 1)	NM (< 1)	NM (< 1)	< 1 (22)	(4)
^{55}Fe	NM (NM)	NM (NM)	NM (NM)	NM (45)	NM (50)	(48)
^{58}Co	NM (65)	4 (34)[b]	880 (1,800)	3 (85)	18 (260)	230 (450)
^{60}Co	NM (33)	1 (46)	500 (1,350)	4 (180)	11 (1,400)	130 (600)
^{89}Sr	NM (140)	220 (320)	140 (170)	85 (89)	24 (34)	120 (150)
^{90}Sr[c]	NM (14)	12 (17)	30 (30)	8 (9)	9 (11)	15 (16)
^{91}Y	NM (66)	1 (26)	NM (< 1)	NM (1)	NM (< 1)	(19)
^{131}I	NM (5)	6 (6)	NM (45)	46 (49)	15 (15)	22 (24)
^{134}Cs	NM (9)	7 (10)	80 (90)	47 (50)	27 (34)	40 (39)
^{137}Cs	NM (29)	35 (35)	150 (170)	140 (160)	99 (103)	106 (99)
^{140}Ba[c]	NM (45)	40 (160)	65 (95)	22 (54)	35 (105)	41 (92)
^{144}Ce	NM (5)	NM (< 1)	NM (< 1)	NM (< 1)	8 (16)	(4)

[a]Concentration of radionuclides at indicated date of sampling; 1 pCi/ml = 1 x 10^{-6} μCi/ml.

[b]Values without parentheses are for filtered solution, values in parentheses for unfiltered solution.

[c]Solutions also contained ^{90}Y at concentrations equal to its ^{90}Sr parent; and ^{140}La at concentrations 1.15 times those of its ^{140}Ba parent.

TABLE III-3[a]

DRESDEN NUCLEAR POWER STATION UNIT 1

RADIONUCLIDE CONCENTRATIONS IN LIQUID LAUNDRY WASTE,[b] (pCi/ml)

Radionuclide	January 15, 1968	March 12, 1968
^{3}H	1.3	1.5
^{58}Co	2.2	1.2
^{60}Co	2.3	0.6
^{89}Sr	< 0.1[c]	0.9
^{90}Sr	0.1	< 0.1
^{131}I	< 0.1	< 0.1
^{134}Cs	0.6	< 0.1
^{137}Cs	3.4	0.8
^{140}Ba	< 0.1	< 0.1
^{141}Ce	< 0.1	< 0.1
gross beta	10	5
gross alpha	< 0.1	< 0.1

[a]Table taken from Reference 11.

[b]Radionuclide concentrations at sampling date; gross values approximately one week later.

[c] < values are 3 σ counting error.

TABLE III-4[a]

DRESDEN NUCLEAR POWER STATION UNIT 1

STACK RELEASES OF FISSION PRODUCT NOBLE GASES,[b] (μCi/sec)

Radionuclide	Sample Date and Time (Central Time) 11/15/67 (0600)	11/16/67 (0600)	1/17/68 (1000)	1/18/68 (0930)	1/31/68 (0900)	6/26/68 (1030)	6/27/68 (1630)	8/20/68 (1928)	8/21/68 (0547)
4.4 -hr ^{85m}Kr	---	580	340	280	---	---	---	---	---
10.7 -yr ^{85}Kr	---	---	---	---	---	0.024	---	---	0.25
76 -m ^{87}Kr	---	---	~ 540	~ 540	---	~ 1,300	---	---	---
2.8 -hr ^{88}Kr	---	~ 1,300	~ 780	~ 640	---	~ 260	---	---	---
2.3 -d ^{133m}Xe	---	14	11	11	---	8	5	31	20
5.3 -d ^{133}Xe	450	420	400	410	970	160	110	910	730
9.1 -hr ^{135}Xe	930	1,010	1,660	1,250	2,600	620	520	---	1,420
17 -m ^{138}Xe	---	---	~ 850	---	---	~ 2,800	~ 2,400	---	---
Gaseous fission product release rate	11,000	10,000	14,800	12,600	18,400	7,500	5,700	11,600	11,600

[a]Table taken from Reference 11.

[b]Based on sampling off-gas delay line; computed for release at top of stack after radioactive decay and dilution of 9.4 l/sec off-gas by exhaust air from containment and turbine buildings at flow rate of 21 m^3/sec.

--- indicates that radionuclide was not measured. In the case of the short-lived radionuclides, their absence in samples was due primarily to delays between sampling and analysis.

TABLE III-5[a]

DRESDEN NUCLEAR POWER STATION UNIT 1

SUMMARY OF STACK RELEASES

OF PARTICULATE RADIONUCLIDES AND GASEOUS IODINE-131

Radionuclide	Number of Measurements	pCi/sec	
		Mean	Range
^{58}C ^{58}Co	15	26	< 8 - 140
^{60}Co	16	25	2.5- 95
^{89}Sr	15	970	220 -2,300
^{90}Sr	15	5	2 - 25
^{137}Cs	16	35	13 - 55
^{140}Ba	17	430	70 - 710
^{131}I	17	920	200 -3,230
Gaseous fission product release rate, mCi/sec	17	11.7	5.7 - 53

TABLE III-6[a]

DRESDEN NUCLEAR POWER STATION UNIT 1

RELEASE RATE OF TRITIUM FROM STACK

Date 1968	Concentration in Primary Coolant, pCi/ml	Concentration in Delay Line, pCi/ml	Release Rate, μCi/sec	
			Measured	Estimated[b]
Jan. 18	---	1.5	1.4×10^{-2}	---
Feb. 1	1,660	---	---	5.0×10^{-3}
June 26	---	0.52 ± 0.3[c]	$4.8 \pm 0.3 \times 10^{-3}$	---
Aug. 22	1,300	---	--	3.9×10^{-3}

[a]Table taken from Reference 11.

[b]Based on equivalent water for hydrogen gas of 2.7 ml/sec and for water vapor of 0.32 ml/sec.

[c]Standard deviation of duplicate analyses; only one sample was analyzed on Jan. 18.

APPENDIX IV

HUMBOLDT BAY PLANT, UNIT 3

Humboldt Bay Nuclear Power Plant is located on Humboldt Bay near Eureka, California along with two fossil fuel plants. It utilizes a boiling water reactor with an authorized power level of 240 megawatts thermal, which corresponds to 68 net megawatts electrical. The plant has been in commercial operation since August 1963 and is operated by Pacific Gas and Electric Company.

Technical Specifications[19] limit average annual discharges of liquid wastes to levels published in Appendix B, Table II of 10CFR20. Gaseous radioactive waste discharges are limited as follows:

> "The annual average stack emission rate for noble and activated gases shall not exceed 0.05 curies per second; the instantaneous stack emission rate shall not exceed 0.5 curies per second.
>
> The annual average stack emission rate for halogens and particulate material based on the isotopes present on the sampling filters after 48-hour decay period shall not exceed the permissible air concentrations for unrestricted areas given in 10CFR20 multiplied by 6×10^8 cubic centimeters per second. Until the approximate composition of the mixtures sampled is indicated by monitoring history the permissible air concentration shall be assumed to be 3×10^{-10} μCi/cc for the halogen and particulate groups."

Based on the above guidance, the discharge rate limit for halogens and particulates would be 0.18 μCi/sec.

Figures IV-1 and IV-2 represent plots of monthly quantities of liquid gaseous waste discharged as reported in operation reports.[20] The power produced and refueling outages are also plotted for comparison. Operating reports include gross beta-gamma analyses less tritium for liquid waste. Gaseous wastes are reported as noble and activated gases separate from halogens and particulates. Noble and activated gases are reported in average monthly discharge rates whereas a range of discharge rates is reported for halogens and particulates as summarized in Table IV-1. Discharges of halogens and particulates are a smaller percentage of the established limit than is the case for noble and activated gases.

Between initial startup in 1963 and August 1965, the principal contributors to the radioactivity in the liquid radwaste discharge were

^{65}Zn, ^{54}Mn, and ^{60}Co. After equilibrium levels had been reached the average isotopic composition of typical discharges was 87% ^{65}Zn, 9% ^{54}Mn, and 4% ^{60}Co.

Beginning in May 1965, defects began occurring in the Type I stainless steel fuel. The effect of these failures did not affect liquid discharges until August 1965 due to hold-up in the radwaste system. From August 1965 until March 1966, the average isotopic composition of radwaste discharged was 46% ^{137}Co, 29% ^{65}Zn, 21% ^{134}Cs, 3% ^{54}Mn, and 1% ^{60}Co. Other fission and corrosion products are present but their concentrations are sufficiently low to be masked by the major contributors to the activity. ^{90}Sr and ^{89}Sr, not being gamma emitters, are not detected by this analysis; however, radiochemical analyses indicate that these nuclides comprise less than 1% of the gross beta activity.

The feedwater heater tube bundles were changed from Admiralty to stainless steel in November 1966 and present analyses reflect the expected decrease in ^{65}Zn activity. Typical analyses of liquid radwaste now show approximately 40% ^{137}Cs, 25% ^{60}Co, 10% ^{54}Mn, 10% ^{134}Cs, 10% ^{65}Zn, and 5% ^{131}I.

The following notes apply to Figures IV-1 and IV-2:

A Six-month average

C Initial core loading

F Commercial operation

R Refueling outage

TABLE IV-1

HUMBOLDT BAY NUCLEAR POWER PLANT

AVERAGE WEEKLY STACK DISCHARGE RATES FOR HALOGENS AND PARTICULATES [17]

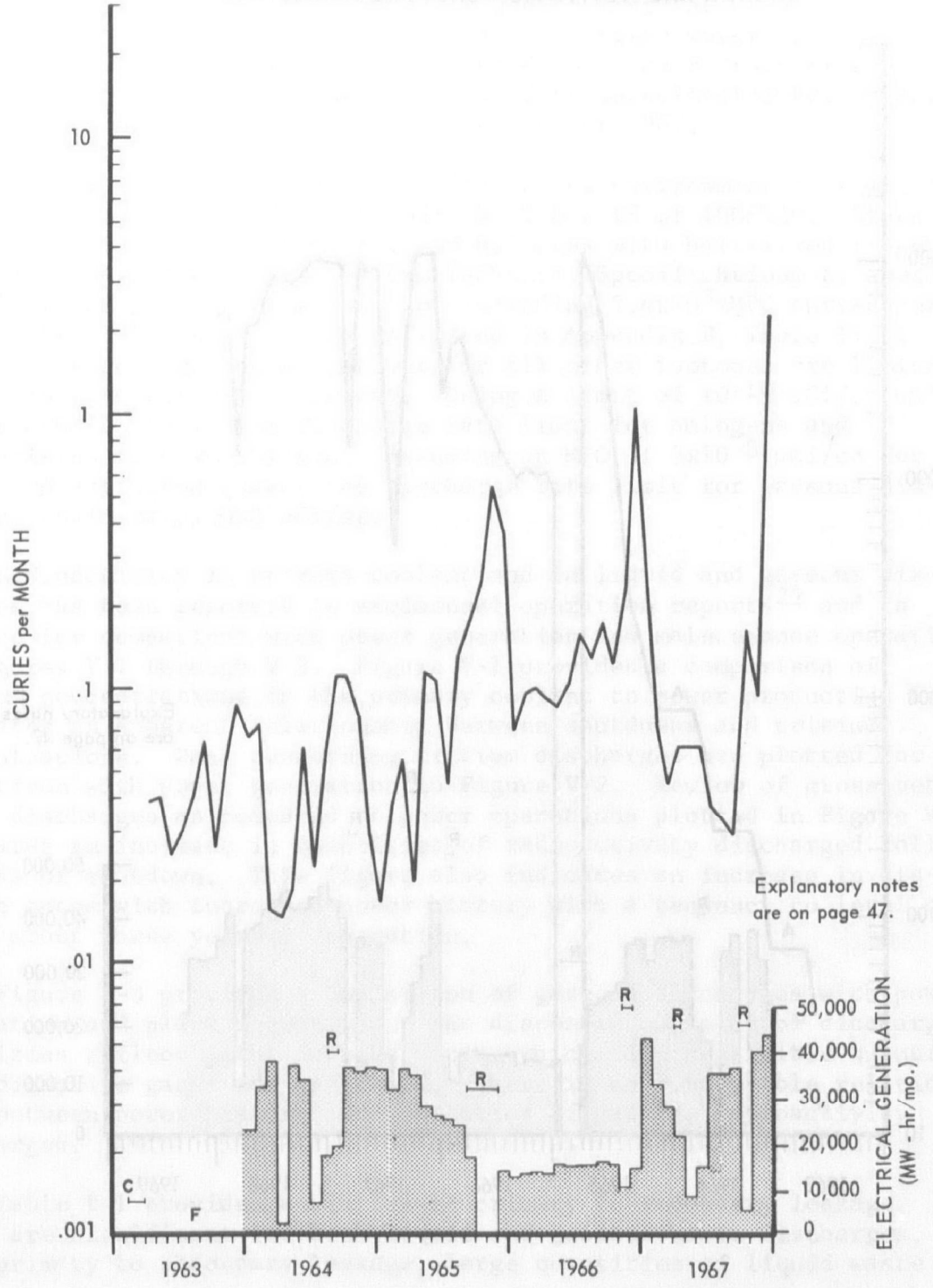
Period	Range of Discharge Rates μCi/sec [a]
2-16-63 to 8-15-63	6.0×10^{-6} to 2.0×10^{-3}
8-16-63 to 2-15-64	9.0×10^{-5} to 5.5×10^{-3}
2-16-64 to 8-15-64	2.4×10^{-6} to 9.3×10^{-6}
8-16-64 to 2-15-65	1.0×10^{-5} to 4.3×10^{-4}
8-16-65 to 2-15-66	3.2×10^{-5} to 2.8×10^{-2}
2-16-66 to 8-15-66	2.7×10^{-4} to 1.8×10^{-2}
8-16-66 to 2-15-67	1.0×10^{-4} to 7.1×10^{-2}
2-16-67 to 8-15-67	4.2×10^{-3} to 1.3×10^{-1}
8-16-67 to 2-15-68	9.0×10^{-3} to 9.2×10^{-2}
1-1-68 to 6-30-68	7.1×10^{-3} to 9.2×10^{-2}
7-1-68 to 12-31-68	3.0×10^{-3} to 3.3×10^{-2}

[a] Based on an MPC limit of 3×10^{-10} μCi/cc [17] the maximum permissible discharge rate would be 0.18 μCi/sec.

FIGURE IV - 1

HUMBOLDT BAY PLANT, UNIT 3

LIQUID WASTE DISCHARGED
(GROSS BETA-GAMMA LESS TRITIUM)

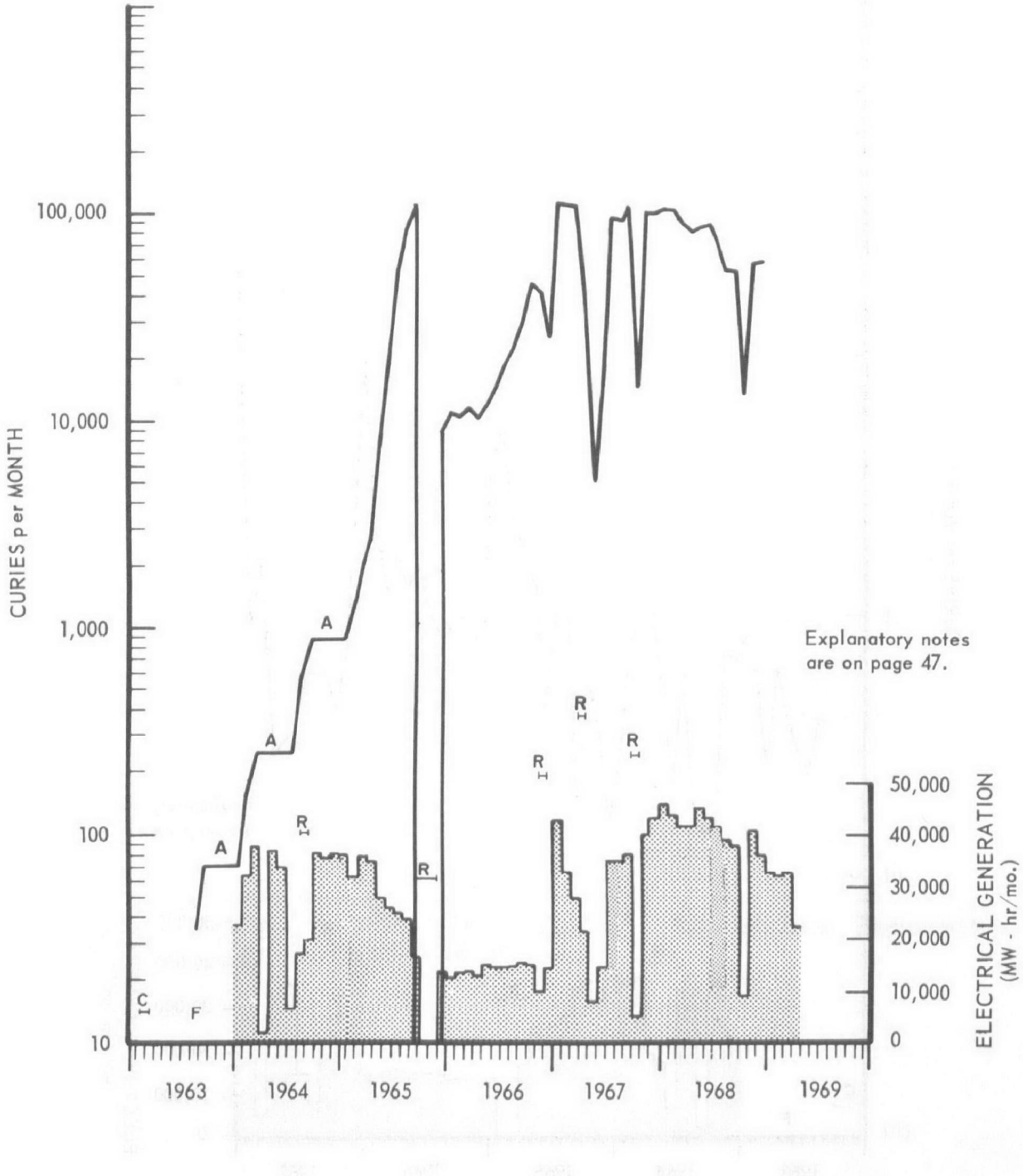


Explanatory notes
are on page 47.

FIGURE IV - 2

HUMBOLDT BAY PLANT, UNIT 3

GASEOUS WASTE DISCHARGED
(NOBLE AND ACTIVATION GASES LESS TRITIUM)



Explanatory notes
are on page 47.

APPENDIX V

INDIAN POINT STATION, UNIT 1

Indian Point-1 is located on the Hudson River about 25 miles north of New York City. It utilizes a pressurized water reactor of Babcock and Wilcox design and is licensed to a power level of 615 thermal megawatts. The plant is operated by Consolidated Edison and has been in commercial operation since October 1962.

Discharge of liquid radioactivity to the environment is limited to concentrations specified in Appendix B, Table II of 10CFR20. Stack discharges of particulate matter and halogens with half-lives longer than eight days are limited by the Technical Specifications to average annual discharge concentrations not exceeding $2.4 \times 10^{3} \times$MPC curies per second where MPC is in μCi/cc as listed in Appendix B, Table II of 10CFR20. Discharge concentrations for all other isotopes are limited to $1.7 \times 10^{6} \times$MPC curies per second. Using a limit of 10^{-10} μCi/cc which is the MPC for ^{131}I, the discharge rate limit for halogens and particulates is 0.24 μCi/sec. Assuming an MPC of 3×10^{-8} μCi/cc for noble and activated gases, the discharge rate limit for gaseous discharges would be 51,000 μCi/sec.

Radioactivity in primary coolant and in liquid and gaseous discharges has been reported in semiannual operation reports[22] and is plotted for comparison with power generation and maintenance operations in Figures V-1 through V-3. Figure V-1 provides a comparison of tritium concentrations in the primary coolant to power production and indicates an apparent relationship between shutdowns and tritium concentrations. Data concerning tritium discharges are plotted for comparison with power generation in Figure V-2. Review of gross beta-gamma discharges as compared to power operations plotted in Figure V-2 indicates an increase in quantities of radioactivity discharged following periods of shutdown. This figure also indicates an increase in discharge rates with increased power history with a tendency to level off after about three years of operation.

Figure V-3 provides a comparison of gaseous discharges with power generation and plant operations. The discontinuous plot of discharge quantities reflect plant outages during which time negligible quantities of radioactive gases are generated. There is no observable relationship between power history and quantities of gaseous radioactivity discharged.

Table V-1 provides a history of primary to secondary leakage. These are significant for both liquid and gaseous waste discharges. With primary to secondary leakage, large quantities of liquid waste

result from steam generator blowdown. Gases in the primary coolant may escape to the steam system where they are scrubbed out through the air ejectors and discharged to the environment.

The following notes refer to Figures V-1, V-2, and V-3:

A Initial criticality

B First power generation

C Electrical generation based on a six-month average

D Shutdown for maintenance

E Three shutdowns for maintenance totaling 56 days

F Shutdown for refueling and maintenance

G Shutdown for refueling; core "B" installed

TABLE V-1

INDIAN POINT STATION UNIT 1

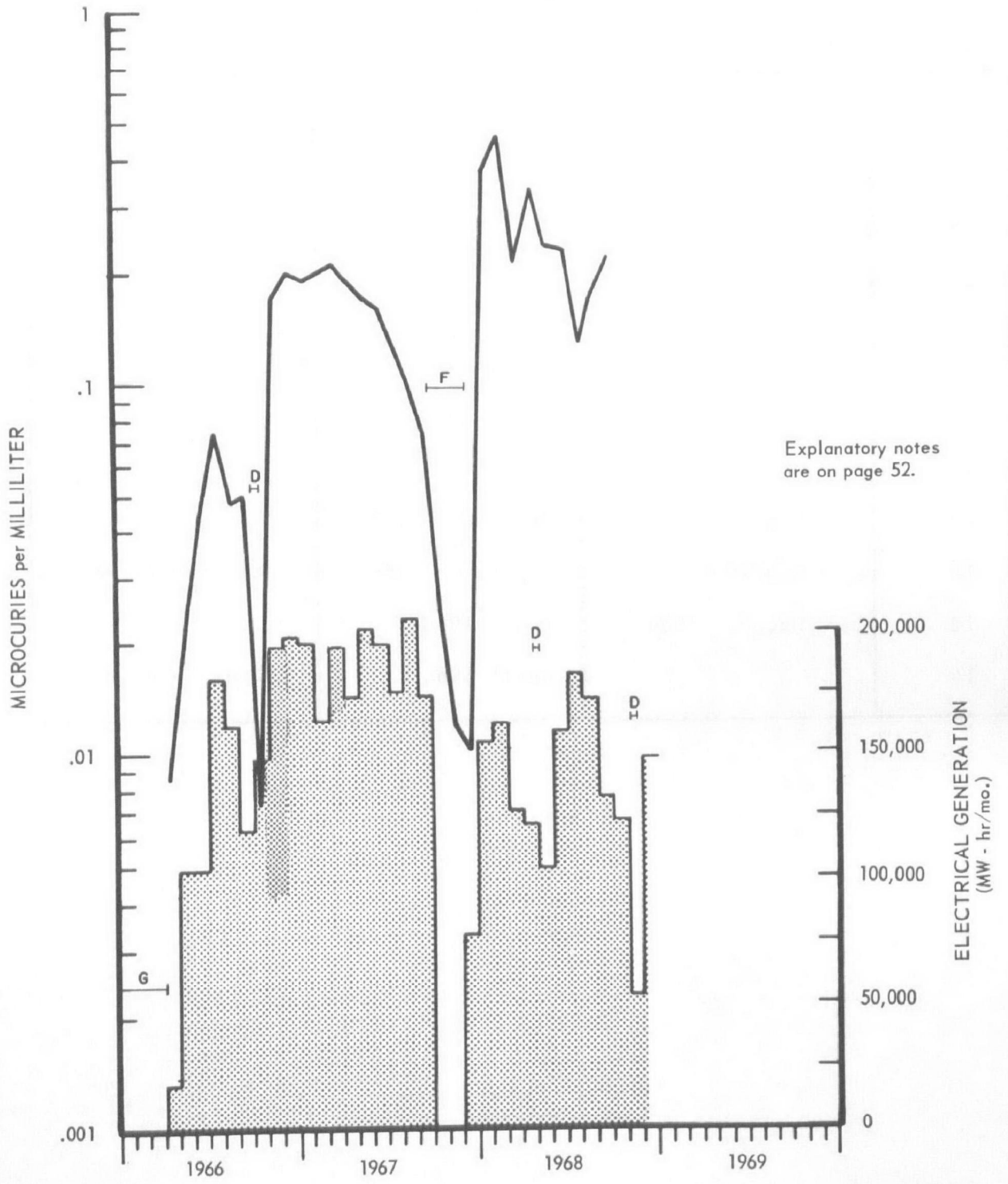
PRIMARY TO SECONDARY LEAKS

Boiler #	Date Leak Started	Date Fixed	Comments
14	June 3, 1963	October 1963	
11	Early 1964	October 1964	
14	August 1964	October 1964	
11	November 1964	March 1965	
11	April 1965	During refueling outage October 20-April 23, 1966	
14	April 1966	April 1966	
14	April 1966	May 1966	
12	May 1966	September 1966	
14	May 1966	September 1966	Two tubes leaking
12	October 1966	April 1968	
14		April 1968	Four tubes fixed

FIGURE V - 1

INDIAN POINT STATION, UNIT 1

PRIMARY COOLANT ACTIVITY
(TRITIUM)



Explanatory notes
are on page 52.

FIGURE V - 2

INDIAN POINT STATION, UNIT 1

LIQUID WASTE DISCHARGED

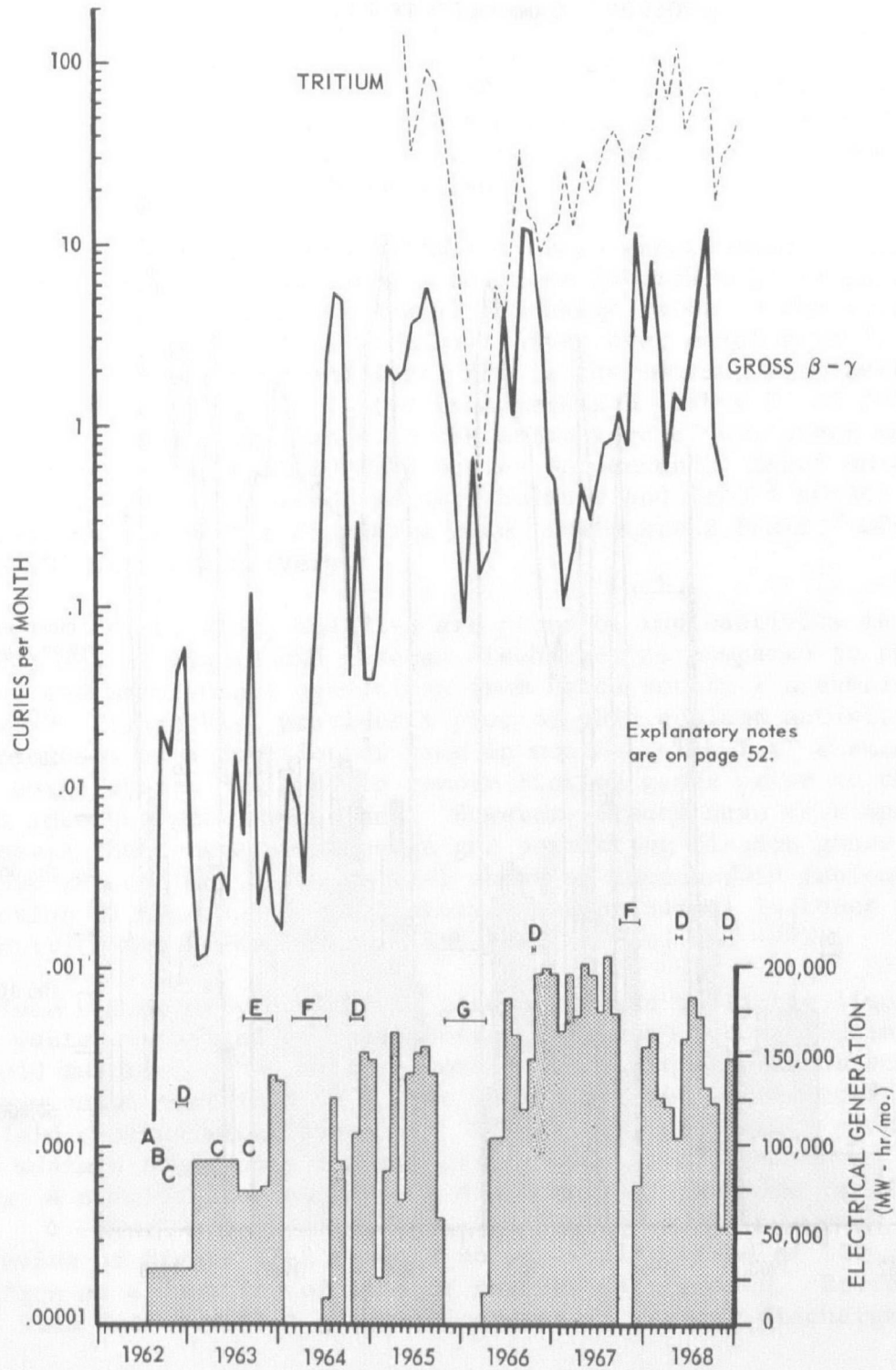
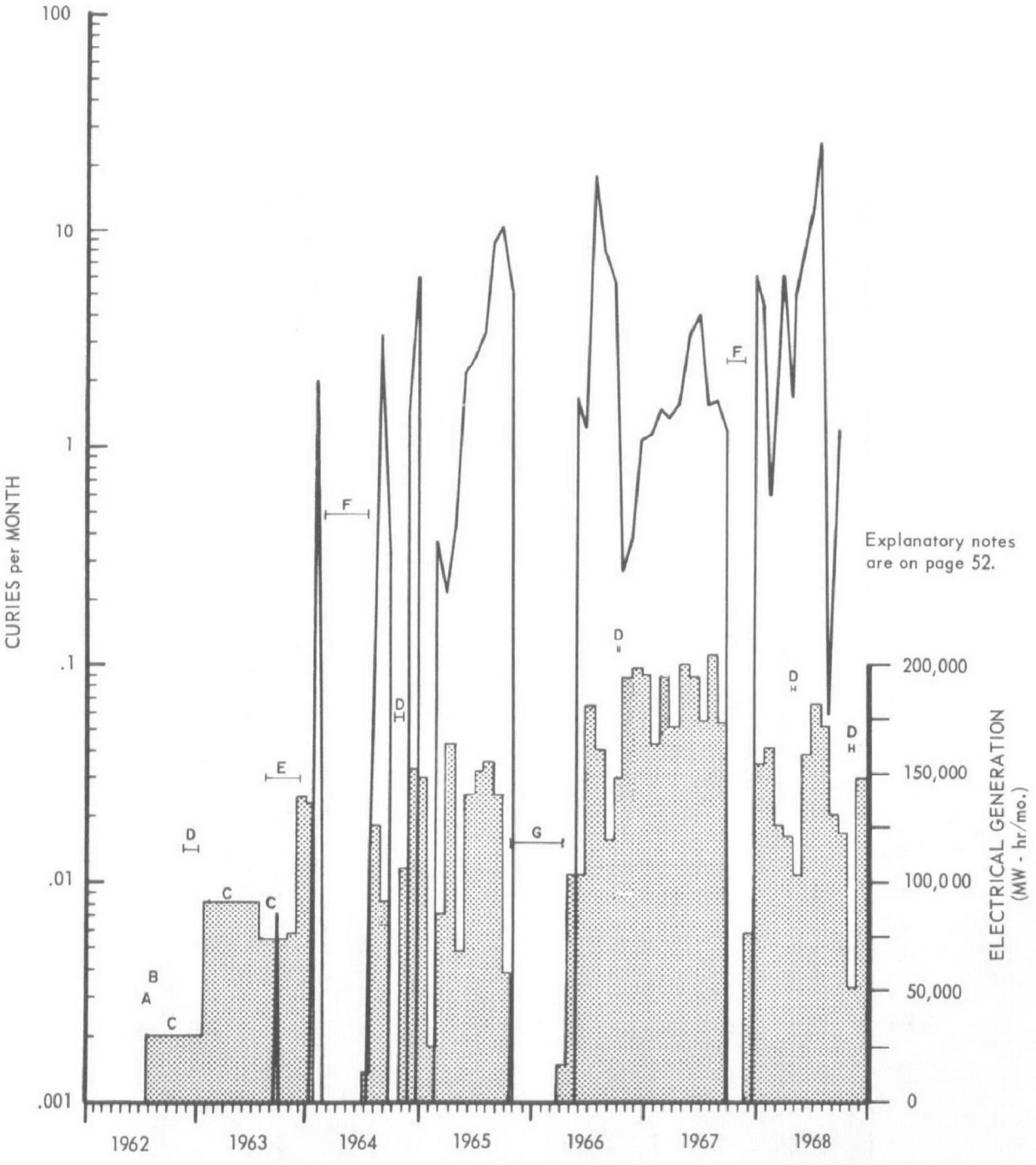


FIGURE V - 3

INDIAN POINT STATION, UNIT 1

GASEOUS WASTE DISCHARGED
(GROSS BETA-GAMMA LESS TRITIUM)



APPENDIX VI

PEACH BOTTOM ATOMIC POWER STATION

Peach Bottom Atomic Power Station is located on the Susquehanna River near the town of Peach Bottom, Pennsylvania. It utilizes a high temperature gas-cooled reactor of Gulf General Atomic design and is authorized to operate at a net power level of 40 MWe. The plant is operated by Philadelphia Electric Company and has been in commercial operation since January 1967.

Technical Specifications[23] limit average annual concentrations of radioactivity in liquid discharges to those published in Appendix B, Table II of 10CFR20. Average annual discharge rates to the atmosphere of halogens and particulates with half-lives over eight days is limited to 28.6xMPC curies per second where MPC is the maximum permissible concentration in μCi/cc as listed in Appendix B, Table II of 10CFR20. Annual average discharge rates to the atmosphere of all other radioisotopes are limited to 2×10^{4}xMPC curies per second. Based on an MPC of 10^{-10} μCi/cc for halogens and particulates and 2×10^{-8} μCi/cc for other radionuclides, the discharge rate limits are 2.86×10^{-3} μCi/sec and 400 μCi/sec respectively.

Figures VI-1, VI-2, and VI-3 are plots of radioactivity in the main loop and in liquid and gaseous discharges as compared to power produced and maintenance operations from Peach Bottom-1 operating reports.[24] Figure VI-1 provides a plot of main coolant activity and power produced as a function of time in months. The fuel elements have a purge system designed to remove fission gases prior to their leaking into the primary coolant. However, cracks have developed in the sleeves that contain the purge gas permitting fission gases to leak into the coolant. Figure VI-1 shows an increase in radioactivity as function of time and power history. The principal isotopes causing this activity are ^{85m}Kr, ^{88}Kr, ^{87}Kr, ^{89}Kr, ^{138}Xe, and ^{135}Xe.

Since the primary coolant is gaseous, there is little liquid radioactive waste discharged at Peach Bottom. The major contributing isotope in liquid wastes is ^{24}Na which occurs as a result of neutron activation of sodium salts contained in a rust inhibitor used in the treatment of the shield cooling water system.[25] Figure VI-2 provides a plot of liquid waste discharges and power generations as a function of time in months. A similar plot of gaseous discharges is provided in Figure VI-3.

Review of Figure VI-2 reveals no particular trend of liquid waste discharges as a function of time or reactor maintenance. Review of Figure VI-3 does reveal a general increase in gaseous discharges as a

function of time and power history. Essentially 100% of the gaseous waste released is ^{85}Kr.[25]

Footnotes for Figures VI-1, VI-2, and VI-3:

A Small amounts of fission products transported through purification system

B Breakthrough from Kr-85 trap caused an increase in main loop impurities

C Small amounts of chemical contaminants entered the primary coolant and low temperature delay bed system

D Feedwater leak into general containment caused a larger volume of radioactive waste

F First power supplied to Pennsylvania Electric Company

G Air in-leakage during fuel handling equipment operation

J Unpurged fuel element suspected

K Airborne activity in air room and stack

L Primary system activity increased by a factor of 13 due to fuel element leakage

R Refueling outage

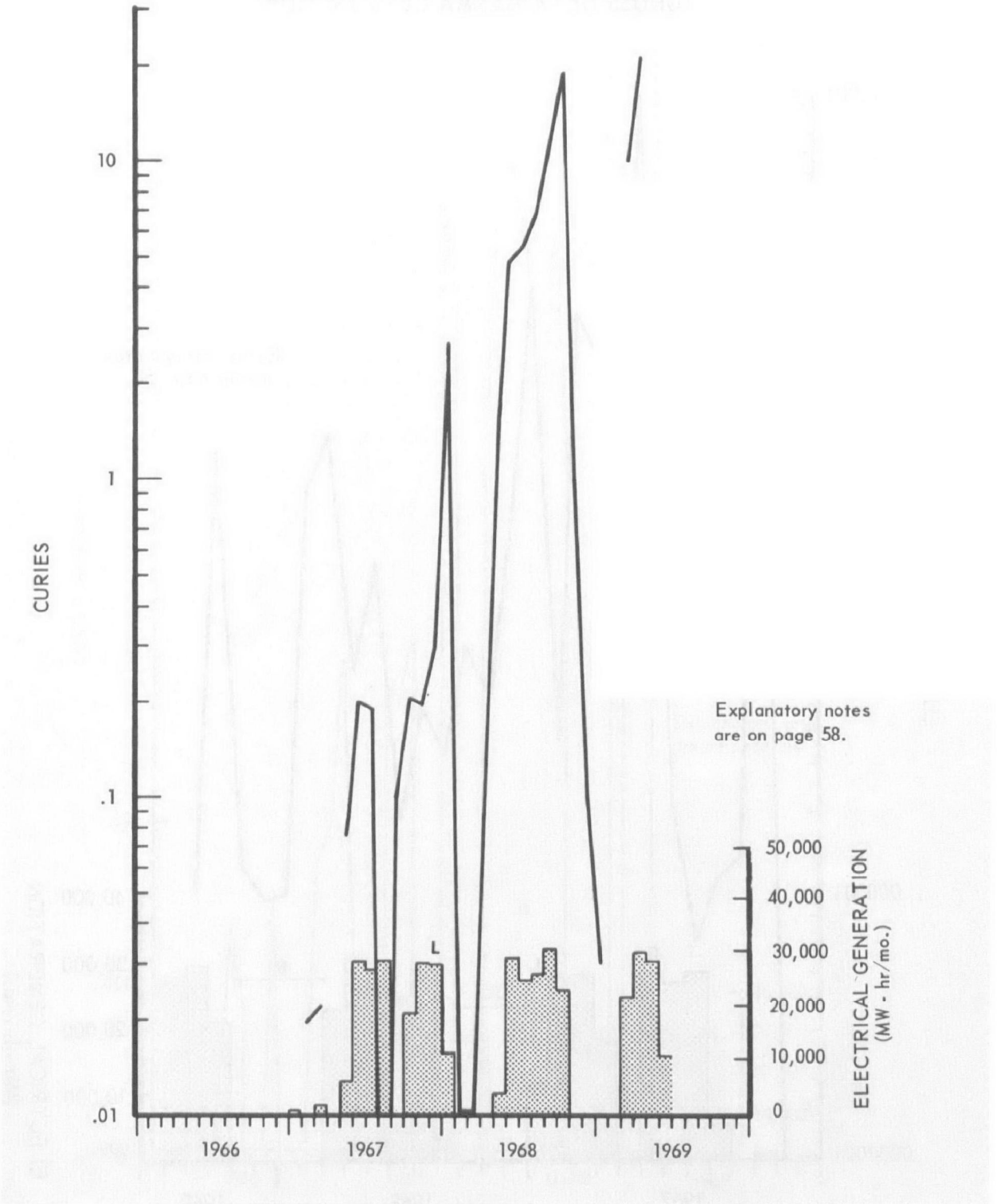
S Release of contaminated helium

T Leakage of helium from control rod drive housing

FIGURE VI - 1

PEACH BOTTOM ATOMIC POWER STATION, UNIT 1

MAIN LOOP ACTIVITY
(GROSS BETA GAMMA LESS TRITIUM)



Explanatory notes
are on page 58.

FIGURE VI - 2

PEACH BOTTOM ATOMIC POWER STATION, UNIT 1

LIQUID WASTE DISCHARGED
(GROSS BETA GAMMA LESS TRITIUM)

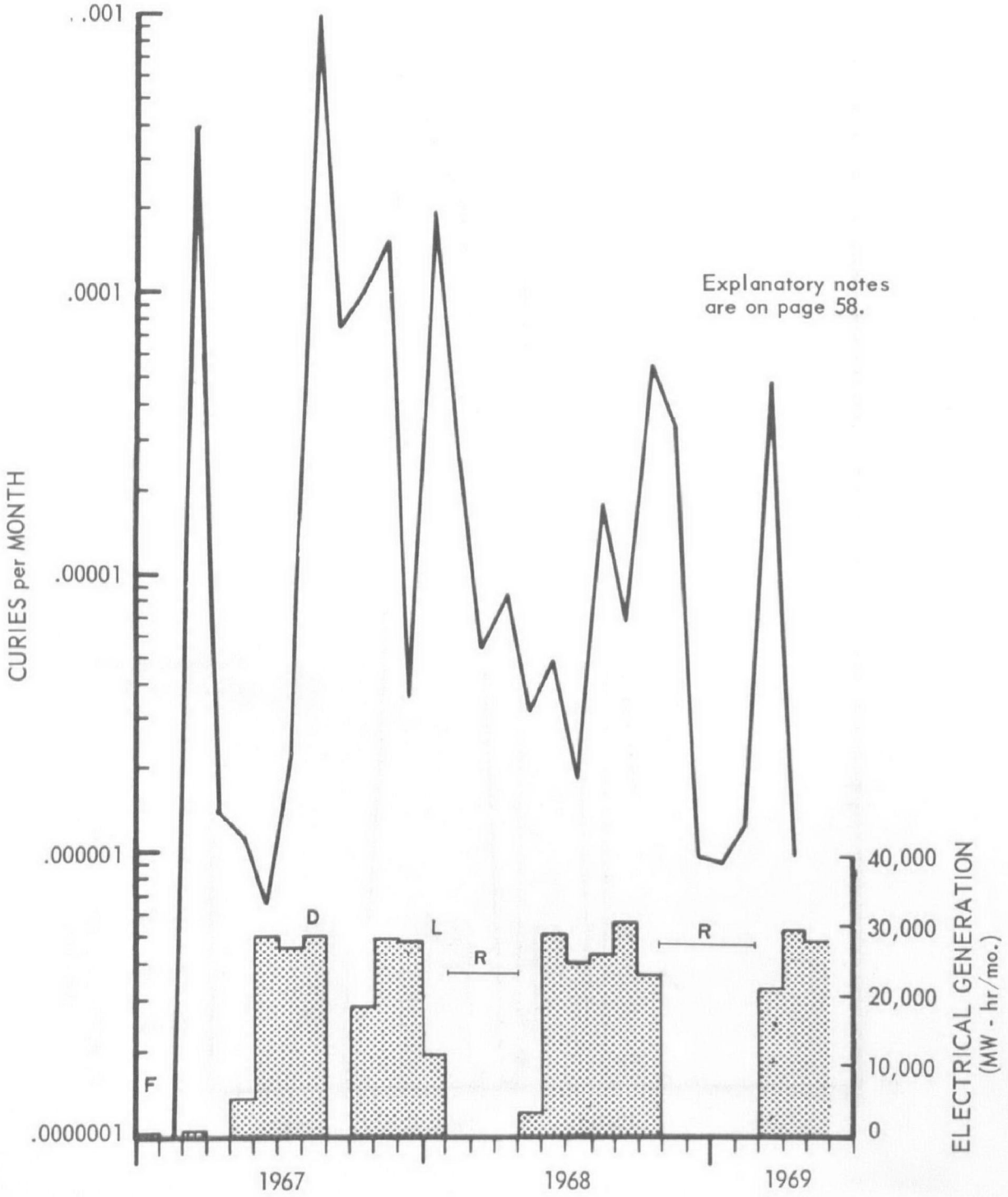
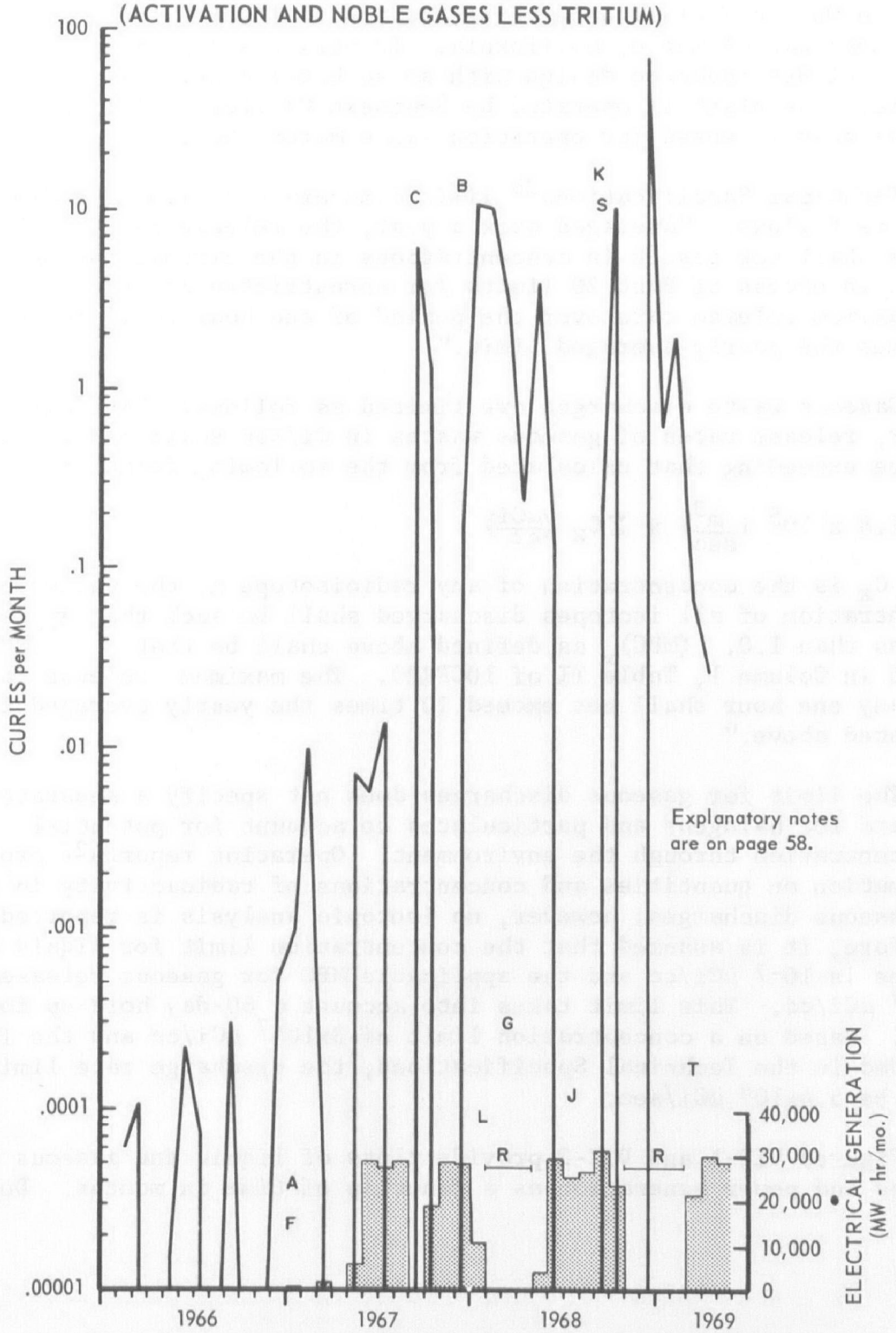


FIGURE VI - 3

PEACH BOTTOM ATOMIC POWER STATION, UNIT 1

GASEOUS WASTE DISCHARGED
(ACTIVATION AND NOBLE GASES LESS TRITIUM)



APPENDIX VII

SAN ONOFRE NUCLEAR GENERATING STATION

San Onofre Nuclear Generating Station is located on the Pacific coast near San Clemente, California. It utilizes a pressurized water reactor of Westinghouse design with an authorized net power level of 430 MWe. The plant is operated by Southern California Edison Company and has been licensed for operation since March 1967.

Technical Specifications[26] limit discharges of liquid radioactive waste as follows: "Averaged over a year, the release rates of liquid wastes shall not result in concentrations in the circulating water discharge in excess of Part 20 limits for unrestricted areas, except that the maximum release rate over the period of one hour shall not exceed 10 times the yearly averaged limit."

Gaseous waste discharges are limited as follows: "Averaged over a year, release rates of gaseous wastes in Ci/sec shall not result in a value exceeding that calculated from the following formula:

$$1.8 \times 10^5 \left(\frac{m^3}{sec}\right) \times \Sigma C_x \left(\frac{\mu Ci}{cc}\right)$$

Where C_x is the concentration of any radioisotope x, the values of the concentration of all isotopes discharged shall be such that $\Sigma \frac{C_x}{(MPC)_x}$ is less than 1.0. $(MPC)_x$ as defined above shall be that stated in Column 1, Table II of 10CFR20. The maximum release rate over any one hour shall not exceed 10 times the yearly averaged limit as stated above."

The limit for gaseous discharges does not specify a separate standard for halogens and particulates to account for potential reconcentration through the environment. Operating reports[27] provide information on quantities and concentrations of radioactivity in liquid and gaseous discharges; however, no isotopic analysis is reported. Therefore, it is assumed that the concentration limit for liquid discharges is 10^{-7} μCi/cc and the applicable MPC for gaseous releases is 3×10^{-7} μCi/cc. This limit takes into account a 60-day hold-up for decay. Based on a concentration limit of 3×10^{-7} μCi/cc and the formula provided in the Technical Specifications, the discharge rate limit would be 5.4×10^4 μCi/sec.

Figures VII-1 and VII-2 provide plots of liquid and gaseous discharges and power generation as a function of time in months. Due to

the limited period of operation, trends in discharges of radioactivity are not yet discernible. The following notes refer to Figures VII-1 and VII-2:

A Initial criticality

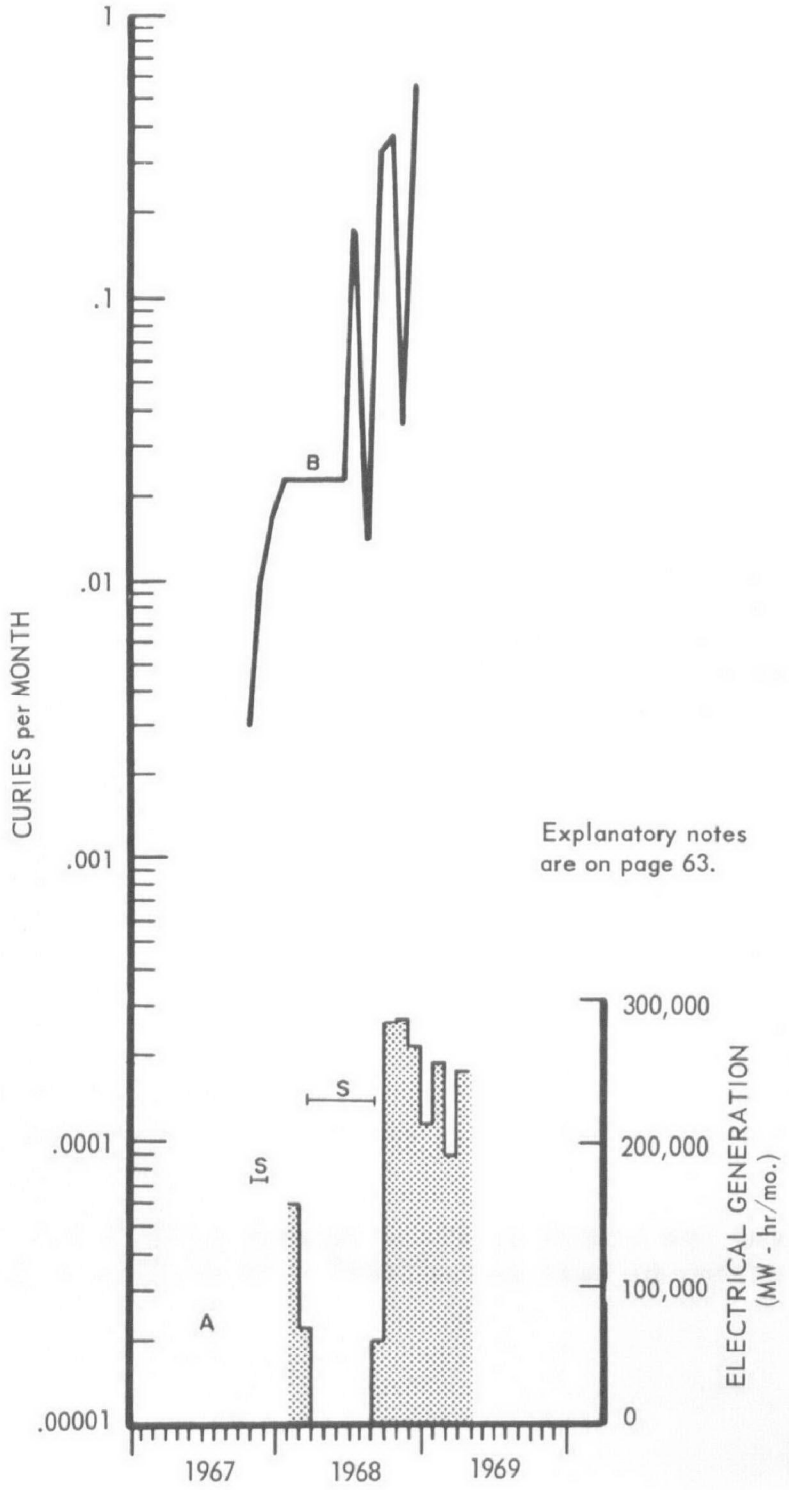
B Six-month average

S Shutdown

FIGURE VII - 1

SAN ONOFRE NUCLEAR GENERATING STATION

LIQUID WASTE DISCHARGED
(GROSS BETA GAMMA LESS TRITIUM)

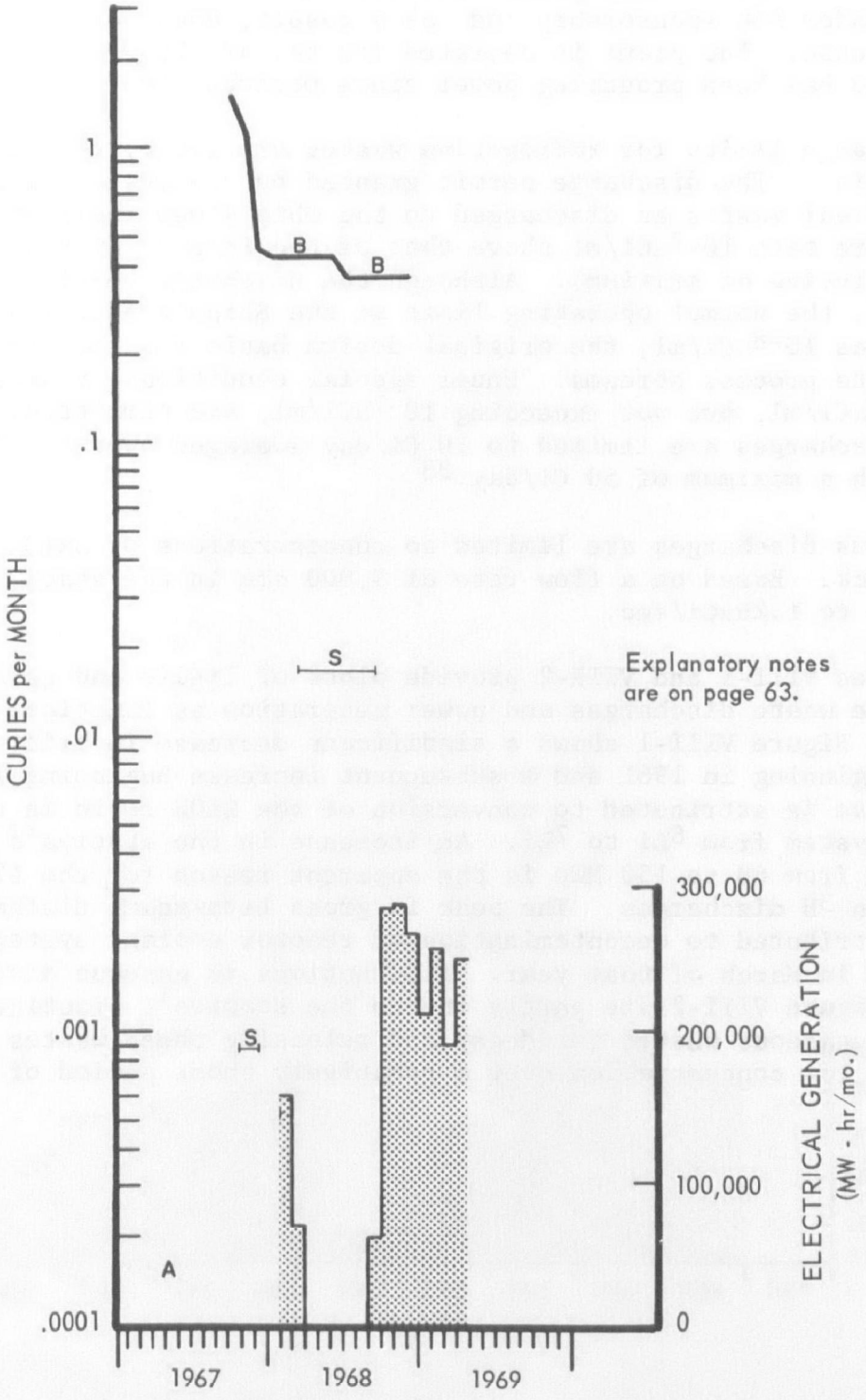


Explanatory notes
are on page 63.

FIGURE VII - 2

SAN ONOFRE NUCLEAR GENERATING STATION

GASEOUS WASTE DISCHARGED
(GROSS BETA GAMMA LESS TRITIUM)



Explanatory notes
are on page 63.

APPENDIX VIII

SHIPPINGPORT NUCLEAR POWER STATION

Shippingport Nuclear Power Station is located on the Ohio River about 25 miles northwest of Pittsburgh, Pennsylvania. It utilizes a pressurized water reactor of Westinghouse design with a power level equivalent to 150 MWe, and a gross electrical output of 100 megawatts. Shippingport, unlike other power reactors, has been developed and operated under AEC sponsorship and, as a result, does not operate under an AEC license. The plant is operated for the AEC by Duquesne Light Company and has been producing power since December 1957.

Discharge limits for radioactive wastes are set by the State of Pennsylvania. "The discharge permit granted by the State stipulated that the final wastes as discharged to the Ohio River shall at no time contain more than $10^{-7}\mu$Ci/ml above that of the intake water of the Ohio River (exclusive of tritium). Although the discharge permit allows $10^{-7}\mu$Ci/ml, the normal operating limit at the Shippingport station is specified as $10^{-8}\mu$Ci/ml, the original design basis for the various liquid waste process streams. Under special conditions, discharges above $10^{-8}\mu$Ci/ml, but not exceeding $10^{-7}\mu$Ci/ml, are permitted."[7] Tritium discharges are limited to 10 Ci/day averaged over any 365 day period with a maximum of 50 Ci/day.[33]

Gaseous discharges are limited to concentrations of $3\times10^{-7}\mu$Ci/ml in the stack. Based on a flow rate of 9,000 cfm in the stack, this is equivalent to 1.26μCi/sec.

Figures VIII-1 and VIII-2 provide plots of liquid and gaseous radioactive waste discharges and power generation as functions of time in years. Figure VIII-1 shows a significant decrease in tritium discharges beginning in 1961 and a subsequent increase beginning in 1965. The decrease is attributed to conversion of the LiOH resin in the purification system from ^{6}Li to ^{7}Li. An increase in the station's power capability from 68 to 150 MWe is the apparent reason for the 1965 increase in ^{3}H discharges. The peak in gross beta-gamma discharges in 1964 is attributed to decontamination of reactor coolant systems that took place in March of that year. Fluctuations in gaseous discharges shown in Figure VIII-2 are partly due to the station's practice of holding up gaseous wastes for decay and releasing these wastes at a controlled low concentration over a relatively short period of time.

FIGURE VIII - 1

SHIPPINGPORT ATOMIC POWER STATION

LIQUID WASTE DISCHARGED

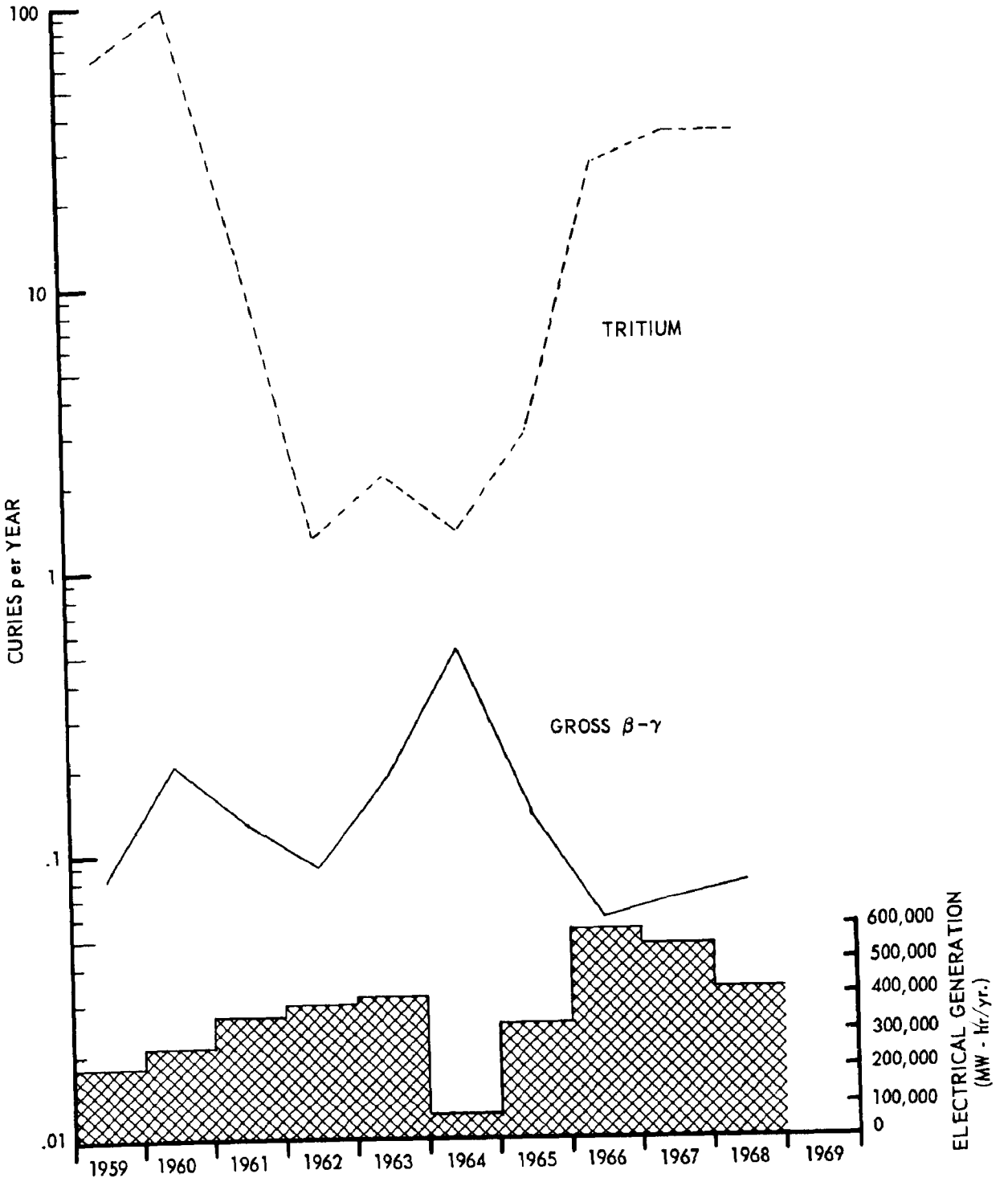
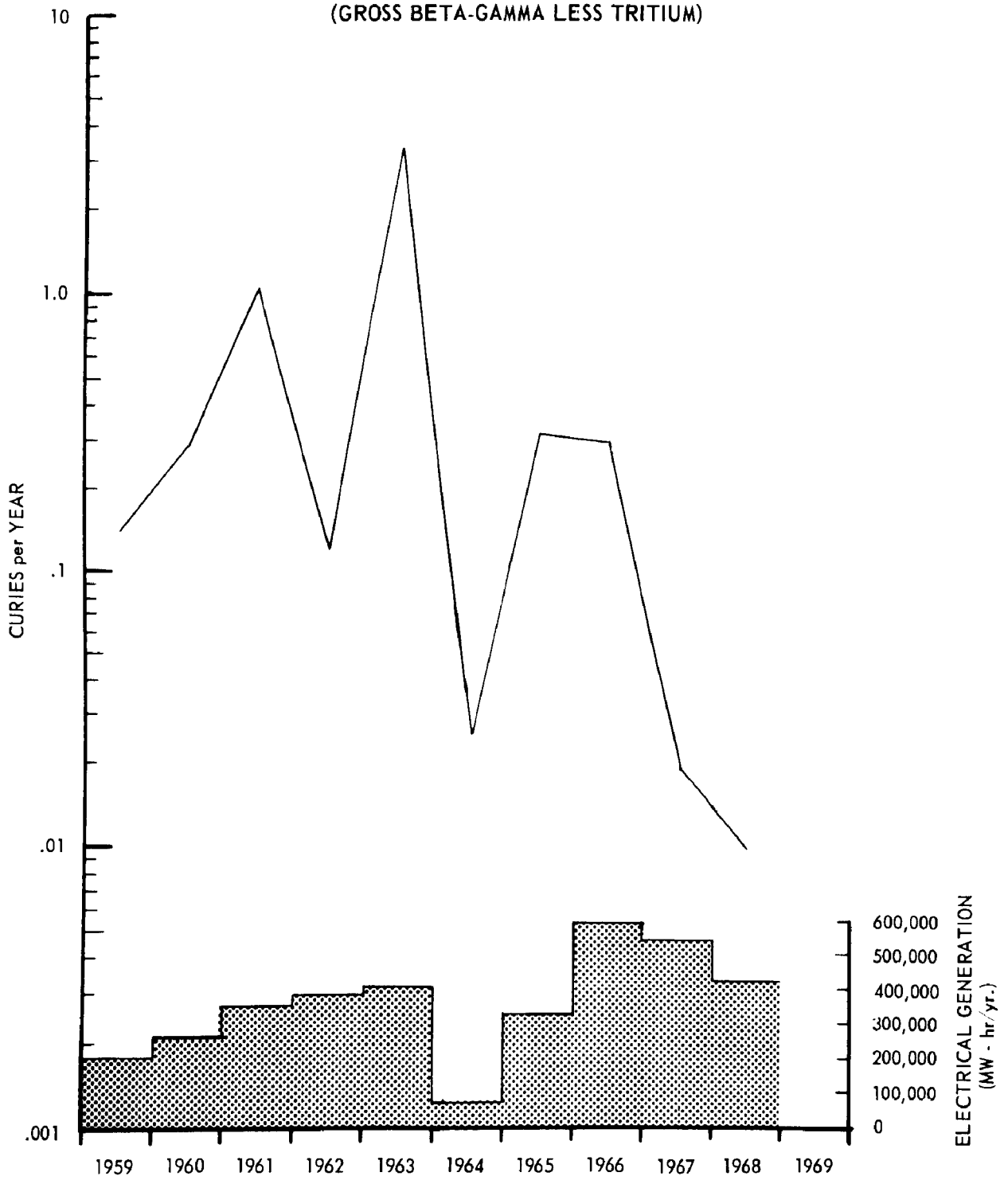


FIGURE VIII - 2

SHIPPINGPORT ATOMIC POWER STATION

GASEOUS WASTE DISCHARGED
(GROSS BETA-GAMMA LESS TRITIUM)



APPENDIX IX

YANKEE ATOMIC POWER STATION

Yankee Atomic Power Station is located on the Deerfield River in Rowe, Massachusetts. It utilizes a pressurized water reactor of Westinghouse design with an authorized net power level of 175 MWe. Yankee is operated by Yankee Atomic Electric Company and has been in operation since August 1960.

Technical Specifications[29] limit liquid discharges to concentrations listed in Appendix B, Table II of 10CFR20. Gaseous discharges are limited as follows: "As determined at the point of discharge from the primary vent stack and averaged over a period not exceeding one year, the concentration of radioactive gaseous wastes discharged shall not be in excess of 1,000 times the limits specified in Appendix B, Table II, 10CFR20." Based on a stack exhaust rate of 15,000 CFM, and a concentration of 1,000 times MPC* in the stack, the limiting discharge rate for noble gases is 212 μCi/sec or 6,700 Ci/yr. This does not agree with the limit of 226 μCi/sec used in Table 10 for calculating percent of limit. The limit of 226 μCi/sec is based on analysis of radionuclides in 1968 discharges as reported in Reference 8.

Figures IX-1 through IX-4 provide plots of primary coolant activity and waste discharges as a function of time and also as compared to power produced. These data are taken from operating reports.[30] Some of the data were not available for Figure IX-1; however, sufficient data are plotted to indicate that gross beta-gamma activities in the primary coolant are maintained generally in the vicinity of 0.1 μCi/ml with rather large reductions during periods of shutdown. Since data in Figure IX-3 do not show corresponding increases in discharge during shutdown, it is assumed that the reduction in primary coolant concentrations during these periods resulted primarily from decay of short half-lived radionuclides. The increase in tritium concentrations in 1968 follows a refueling and boron addition to the primary coolant. The increase would result in increased interaction of neutrons with ^{10}B to produce tritium.

Figure IX-2 indicates a wide fluctuation in monthly quantities of liquid waste discharged with peaks occurring either during or immediately following refueling outages.

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*The MPC used is 3×10^{-8} μCi/ml which is based on a typical noble gas mixture with less than two hours hold-up for decay.

The following notes refer to Figures IX-1 through IX-4. The notes in general confirm that the operations listed had little or no effect on waste discharges.

A NH_4OH added to primary coolant

B Boron added to primary coolant

C High carryover from waste disposal evaporator causing an increase in liquid waste

D Corrosion of control rods increase in primary coolant activity

E Steam generator blowdown to waste disposal system

F Radiochemistry samples released to gaseous waste system

G KOH added to primary coolant

J Crud burst

K During blowdown of surge tank cover gas escaped

L Primary drain collection tank purged

M Outage to repair steam generator leak

N Inadvertent gas release

P Primary to secondary leak

R Refueling outage

S Initial criticality August 19, 1960

W NH_4OH removed

X Boron removed

FIGURE IX - 1

YANKEE NUCLEAR POWER STATION

PRIMARY COOLANT ACTIVITY

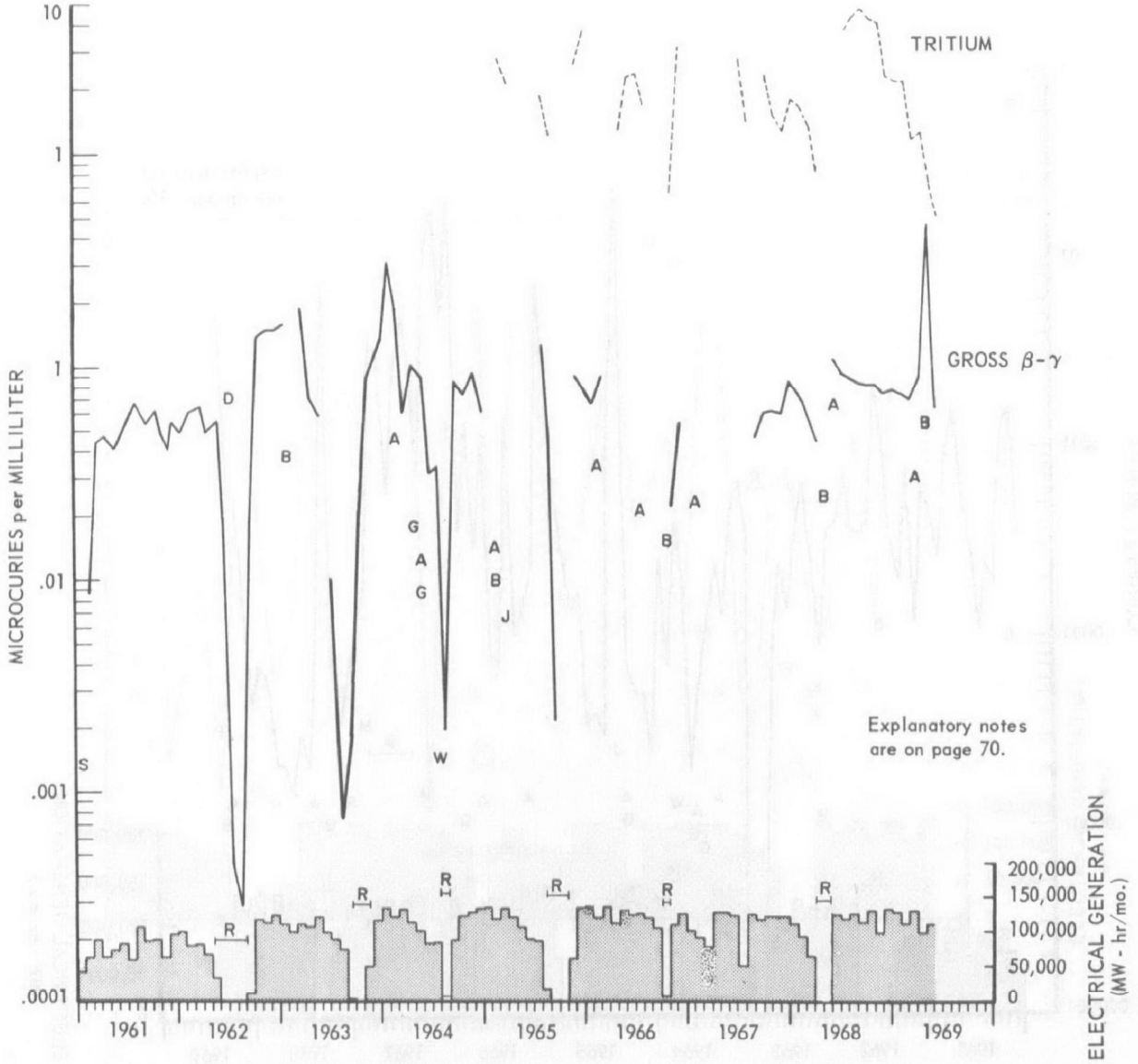


FIGURE IX - 2

YANKEE NUCLEAR POWER STATION

LIQUID WASTE DISCHARGED
(GROSS BETA-GAMMA LESS TRITIUM)

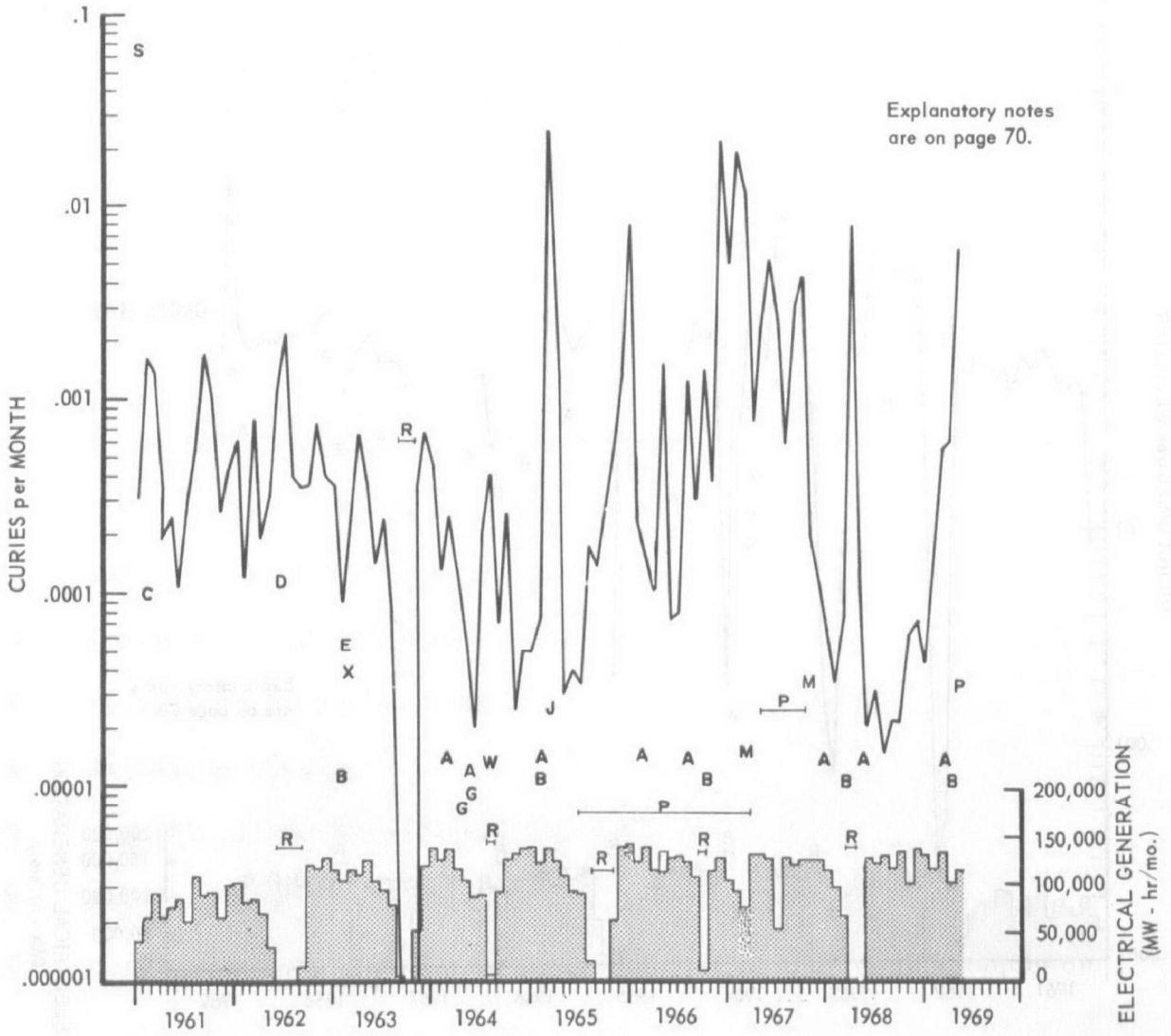


FIGURE IX - 3

YANKEE NUCLEAR POWER STATION

LIQUID WASTE DISCHARGED
(TRITIUM)

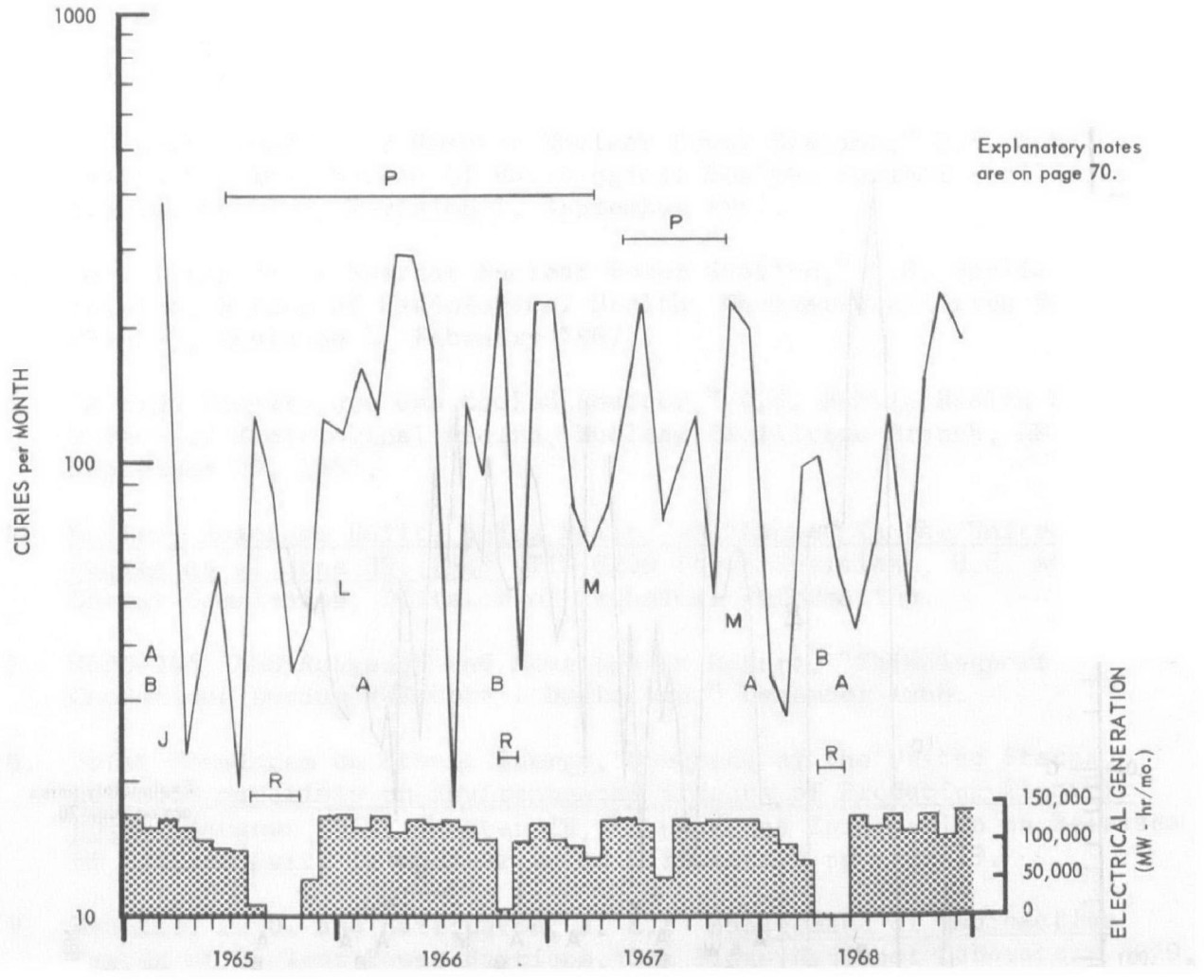
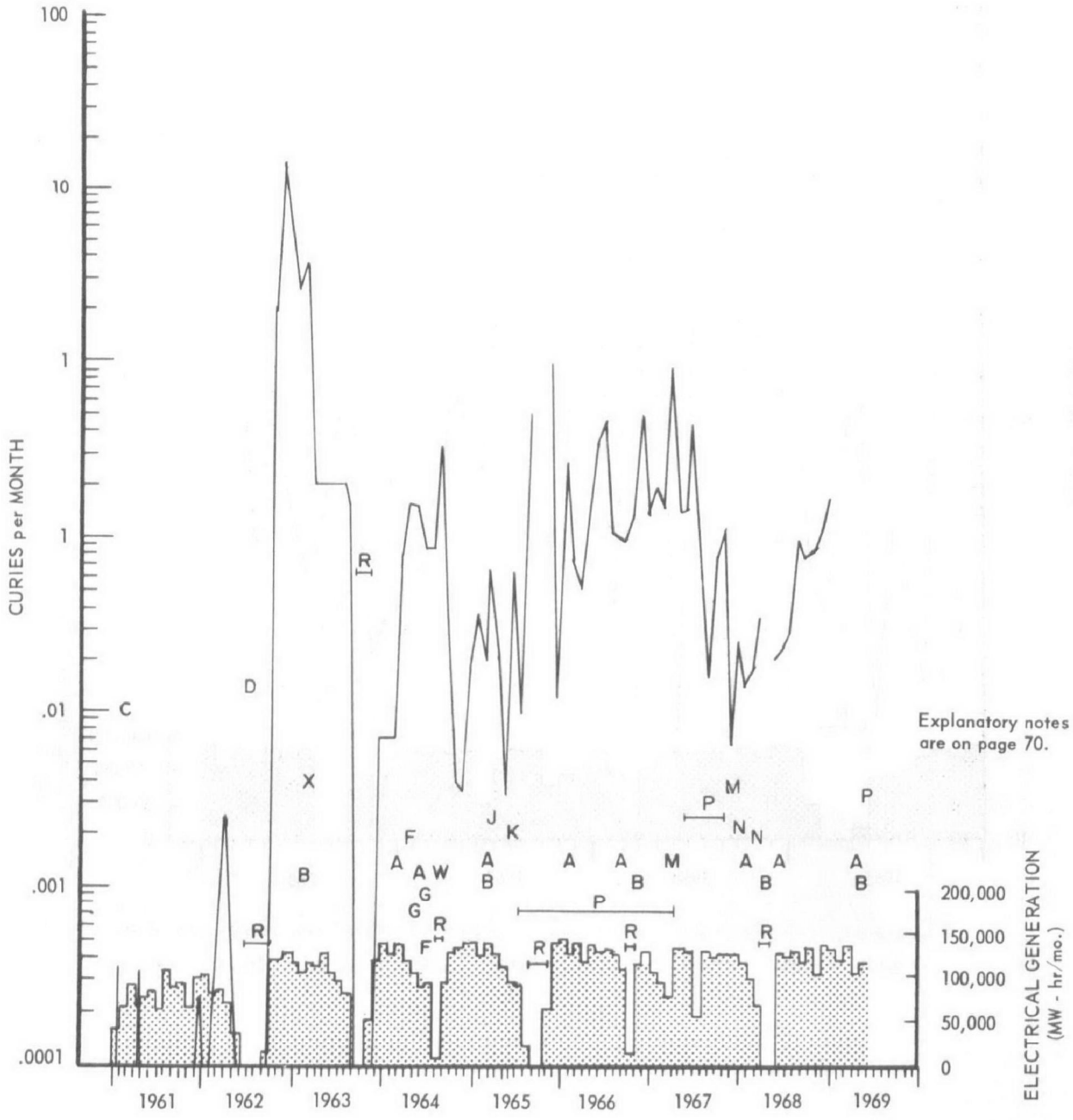


FIGURE IX - 4

YANKEE NUCLEAR POWER STATION

GASEOUS WASTE DISCHARGED
(GROSS BETA-GAMMA LESS TRITIUM)



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