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OPERATING HISTORY REPORT FOR THE PEACH BOTTOM HTGR

VOLUME I REACTOR OPERATING HISTORY

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ABSTRACT

The Peach Bottom Unit No. 1 Atomic Power Station was the first installation of a High-Temperature Gas-Cooled Reactor (HTGR) in the United States. The plant was successfully operated between March 3, 1966 (first criticality, Core 1) and October 31, 1974 (end-of-life, Core-2) at which time the plant was shut down for decommissioning. The 40-MW(e) reactor demonstration plant operated very successfully and produced superheated steam at 1000°F and 1450 psig with an overall nuclear steam supply system availability of 88% and gross plant thermal efficiency of 37.2%. Over 1.2 million MW(e)-hr were produced for the Philadelphia Electric Company grid over a lifetime of 1349 equivalent full power days, with a gross plant capacity factor of 74%.

Subsequent to shutdown, the Peach Bottom End-of-Life Program, jointly sponsored by ERDA and EPRI, was initiated. The objective of the program is to validate specific HTGR design codes and predictions by comparison of observed and predicted physics, thermal, fission product, and materials behavior in Peach Bottom. This report was prepared under the Peach Bottom End-of-Life Program to provide a reference data base for future HTGR evaluations and design verifications.

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1. INTRODUCTION

The Peach Bottom Atomic Power Station Unit No. 1 was the first installation of a High-Temperature Gas-Cooled Reactor (HTGR) in the United States. Power operation began in January 1967 and commercial operation on June 1, 1967. The plant was operated successfully through October 31, 1974 when it was shut down for decommissioning. The Peach Bottom nuclear steam supply (NSS) system, designed and supplied by the General Atomic Company, generated more than 3.72 million MW(t)-hr and 1.38 million gross MW(e)-hr for an average gross plant thermal efficiency of 37.2%. The Peach Bottom NSS produced 1000°F superheated steam at a pressure of 1450 lb/in.², with an overall lifetime availability of 88%. The plant produced over 1.2 million MW(e)-hr for the Philadelphia Electric Company grid over a lifetime of 1349 equivalent full power days (EFPDs), with a gross plant capacity factor of 74%.

In addition to producing commercial power, Peach Bottom was a prototype nuclear power station. This status required that power changes, including shutdowns, be performed to accommodate testing of plant systems and components under the USAEC-sponsored postconstruction research and development program. Such surveillance programs to monitor core component performance, fission product release and plateout, circulating activity, coolant chemistry, and other important features of reactor operation were continued throughout reactor lifetime by General Atomic and Oak Ridge National Laboratory. In addition, during the operation of Core 2 more than 30 fuel test elements were installed and irradiated as part of a fuel testing program for advanced HTGRs.

The overall performance of the Peach Bottom HTGR, from a standpoint of both fuel and plant, was particularly gratifying. Although fuel problems were encountered in Core 1, Peach Bottom successfully demonstrated that

even with significant fuel failure present in the core, plant operation was not impaired and reactor operation could be continued. It was demonstrated that failed Core 1 elements could be identified and removed, the cause of the problem determined, and a new core fabricated to run to full design life without recurrence of the initial problem. Also, throughout Peach Bottom operation excellent agreement was found between predicted and actual core physics characteristics, thus verifying the methods used.

Subsequent to reactor shutdown, the Peach Bottom End-of-Life Program, cosponsored by ERDA and EPRI, was initiated. The prime objective of this program is to validate specific HTGR design codes and predictions by comparison of actual and predicted physics, thermal, fission product, and materials behavior in Peach Bottom. These design method verifications, to be completed in CY-77, utilize the data determined during three consecutive phases of the program together with the data determined in a complementary program of Peach Bottom driver fuel element PIEs at ORNL. The three phases are (1) nondestructive fuel and circuit gamma scanning at the Peach Bottom site, (2) removal of Peach Bottom steam generator and primary circuit components, (3) laboratory examinations of removed components. This report was prepared under the Peach Bottom End-of-Life Program to provide a reference data base for future HTGR evaluations and design verifications.

A brief description of the Peach Bottom HTGR and a brief summary of the plant operation are given in Section 2. The subsequent sections cover, year by year, the operating history of the reactor over its 8-year operational lifetime.

2. PLANT DESCRIPTION AND SUMMARY

The Peach Bottom Atomic Power Station Unit No. 1 is a 40-MW(e) HTGR owned and operated by the Philadelphia Electric Company. The plant is located about 80 miles southwest of Philadelphia on the west shore of the Susquehanna River (Fig. 2-1). The nuclear steam supply (NSS) system was designed, developed, and supplied by the General Atomic Company. The engineer-constructor was the Bechtel Corporation. Financing was provided by the High-Temperature Reactor Development Associates, a nonprofit organization composed of 53 investor-owned utilities from throughout the United States and by the USAEC as part of the power reactor demonstration program. The significant milestone dates for the Peach Bottom HTGR are shown in Table 2-1.

The heart of the Peach Bottom NSS system is a helium-cooled, graphite-moderated, 115-MW(t) reactor operating at high temperature on a thorium-uranium fuel cycle (Fig. 2-2). The fuel elements, which are 3-1/2 in. in diameter and 12 ft long, are oriented vertically in a closely packed circular pattern within the reactor vessel. The reactor core consists of 804 graphitic fuel elements, each containing 30 graphitized fuel compacts. The compacts contain pyrolytic carbon coated thorium - fully enriched uranium carbide fuel particles. Core 1 fuel particles were coated with a single layer of pyrolytic graphite solely to prevent hydrolysis during manufacture. The improved BISO coatings on the Core 2 fuel particles (Fig. 2-3) were designed to retain gaseous fission products during the life of Core 2. The compacts are arranged as annular rings aligned on graphite spines (Fig. 2-4), constituting an active fuel length of 90 in.

Radioactivity in the main coolant system is controlled by drawing a purge stream of main coolant helium over the fuel compacts inside the low-permeability graphite sleeves to the external fission product trapping

TABLE 2-1
PEACH BOTTOM UNIT NO. 1 HISTORY

August	1959	Contracts signed - USAEC, PE, GA
February	1962	Construction permit issued
January	1964	Vessel shipment to site
January	1965	Fuel shipment to site
January	1966	License for 1-MW(t) operation received
February	1966	Fuel loading began
March	1966	Initial criticality
April	1966	Core 1 loading completed
May	1966	Low-power testing completed
January	1967	Full-power license issued
May	1967	Full power reached
June	1967	Start of commercial operation for Core 1
October	1969	Shutdown for refueling
July	1970	Start of commercial operation for Core 2
October	1974	Core 2 end-of-life, plant shutdown for decommissioning

system. The fuel particle coatings and the graphitic fuel element structure are also significant contributors to the control of fission products. The fission product trapping system consists of packed beds of charcoal at successively lower temperatures that adsorb, delay, and permit decay of fission products in the purge stream. Sufficient delay time is provided to permit the decay of almost all fission products with the exception of Kr-85. A dehydrator, an oxidizer, and a liquid-nitrogen-cooled charcoal trap are provided to remove moisture, chemical impurities, and the Kr-85 from the main coolant system.

The side reflector, which extends the full height of the fuel elements, is approximately 2 ft thick and consists of three inner rings of hexagonal elements surrounded by tilting graphite blocks. Core clamping is provided by the differential pressure developed across the tilting reflector by helium flow through the vessel.

The nuclear heat is transferred to two parallel coolant loops (Fig. 2-5) in which helium circulates at approximately 335 psig at 220,000 lb/hr per loop. This flow is generated by motor-driven, variable-speed centrifugal compressors that force the reactor outlet gas at approximately 1300°F through steam generators, where it is cooled to about 650°F before it enters the circulators for return to the core. The primary pressure boundary is fabricated of carbon steel; the hot gas is contained inside concentric pipes or shrouds insulated with metallic thermal barrier to keep the steel temperatures within acceptable limits.

The steam generators are forced-recirculation drum-type boilers having pendant U-tube economizer, evaporator, and superheater sections. The superheater tubes are made of Incoloy 800 and the other sections are carbon steel. Each steam generator produces about 184,000 lb/hr of 1000°F steam at about 1450 psig. The two steam flows are combined at the throttle of a single tandem-compound, double-flow, 3600-rpm turbine generator rated at 45.9 MW(e) at 0.85 power factor. There are four stages of feedwater heating, including a deaerating heater located in the suction of the two half-capacity

boiler feed pumps, one turbine driven and the other electric-motor driven. The plant is designed to produce 40.0 MW(e) net maximum and can follow load automatically down to 30% at rates in excess of 3%/min. The plant auxiliary power requirements amount to about 4 MW. Gross thermal efficiency at the design operating conditions is approximately 39%, which represents the highest efficiency achieved by any reactor power plant operated in the United States to date.

Normal control of the reactor is accomplished by 36 hydraulically driven control rods which move in graphite guide tubes. In addition to their normal power regulating function, these rods can effect a scram. Two independent shutdown systems are provided in addition to the normal control rods: (1) a group of 19 electrically driven emergency shutdown rods, and (2) a group of 55 thermally released gravity-drop neutron absorbers.

The NSS system is housed in a cylindrical containment vessel that is designed to withstand 8 lb/in.² differential pressure. About 90% of the containment area is filled with a reduced-oxygen atmosphere (less than 4% O₂) during reactor power operation. Instrumentation, controls, and various small pumps and compressors are located in a small air-filled room that is accessible during power operation.

The Peach Bottom fuel cycle concept is based on continuous operation with a core of 804 fuel elements until the end-of-life (approximately 900 full-power days), at which time the entire core would be replaced with a new core. Peach Bottom operated with Core 1 until October of 1969, accumulating 451.5 equivalent full power days (EFPDs), at which time the plant was shut down for installation of a second core of 804 fuel elements. This premature installation of a second core was necessitated by the development of cracks in the graphite sleeve surrounding the fuel compacts on 90 of the fuel elements. The existence of cracked elements was detected by an increase in total primary system activity, which eventually reached a level of 270 Ci. While this activity level was well below the primary system design activity of 4225 Ci, it was considered prudent to replace the core

in view of the accelerating rate of fuel element cracking. It is important to note that the increase in primary system activity did not result in an increase in radiation levels throughout the plant, nor in an increase of gaseous and liquid radioactive waste activity released to the environment.

Core 1 operated with an overall plant availability of 58%. Two time-consuming fuel handling outages for the removal of broken elements accounted for 72% of the time that the plant was not available for operations. The length of these fuel handling outages was further extended by the time required to develop, test, and use special tools for the removal of the broken fuel. Also, little consideration was given to minimizing fuel handling time in the design of Peach Bottom, owing to the prototype nature of the plant. Excluding the time consumed by the removal of broken elements, Core 1 experienced a plant availability of 88%.

The Core 1 fuel contained fuel particles coated with a single layer of pyrolytic graphite. Fast-neutron-induced dimensional changes and damage due to fission product recoils resulted in cracking and distortion of the coatings on the fuel particles. The broken coatings, in the process of curling and changing dimensions, caused the compacts to distort and swell. The radial expansion produced in the compacts caused them to bind against the graphite sleeve, causing the sleeve to fracture. A total of 90 elements in Core 1 developed cracked sleeves.

During the first half of 1970, a new core containing improved coated fuel particles was inserted into the reactor. The design of this new coated fuel particle was characterized by (1) a greater retention of fission products, and (2) excellent stability of the fuel coating thus preventing fuel compact expansion. The new fuel particle (Fig. 2-3) was coated with a double layer of pyrolytic carbon (BISO) consisting of an inner low-density carbon coating and an outer high-density carbon coating. The primary circuit activity never exceeded 1 Ci during the entire Core 2 lifetime, thereby demonstrating the improved fission product retention characteristics of the Core 2 fuel particles. Core 2 was operated successfully for its entire design lifetime of 900 EFPDs. On October 31, 1974 the plant was shut down for decommissioning.

The decision to shut down and decommission the Peach Bottom HTGR was based upon several factors. First and foremost was the fact that the program for which the plant had been originally designed was completed; that is, the objective of demonstrating the technical feasibility and commercial operation of an HTGR had met with outstanding success. Second, the continued progress of the evolution of the HTGR is being carried on in the Fort St. Vrain plant which is currently in the startup phase. Third, the size of the Peach Bottom Unit 1 plant [40 MW(e)] makes it uneconomical in terms of operating costs or manpower as opposed to the large nuclear plants recently placed in operation by Philadelphia Electric (i.e., Peach Bottom Units 2 and 3). Finally, the changes that have been incorporated in the USAEC safety and licensing requirements since 1966 would undoubtedly necessitate major retrofitting if a permanent operating license were required for continued operation.

Faced with the above considerations, in 1972 General Atomic and Philadelphia Electric decided that continued operation after Core 2 end-of-life was not warranted, and a third core for Peach Bottom was not authorized.

The very good overall performance shown by Peach Bottom Unit 1 has been referred to previously in the introduction; however, major accomplishments also included the following:

1. The steam generator operated almost 8 years without tube leaking or plugging.
2. The reactor control system functioned exceptionally well and received commendation from Philadelphia Electric operators.
3. The performance of almost all reactor systems was without major problems, verifying the design philosophy of many areas used in Fort St. Vrain and larger HTGRs.
4. Philadelphia Electric operated the plant in a load-following manner during the majority of its 8-year lifetime, demonstrating the ability of the HTGR to function in this manner.

The major operational problems and their solutions were as follows:

1. High steam generator shell temperatures. Solved by adding baffles to steam generator interior and bottom head cooling unit to the steam generator exterior.
2. Fuel element sleeve failures - cracking and swelling of Core 1 pyrocarbon coatings resulting in fuel compact swelling which caused cracking of the fuel element sleeves. Solved by reloading reactor with an advanced core (Core 2) which operated the entire design life of 900 full-power days with excellent results.
3. Moisture monitor cell failures due to exposure to dry helium and impurities. Solved by adding at-power testing capability allowing replacement of cells prior to failure.
4. Purified helium compressor oil leak caused high (10 to 100 ppm) hydrocarbon impurity levels in the main loops. Solved by replacing the internal components of the oil separator section of the compressor and also the demistor and charcoal filters.

The overall station availability, excluding planned shutdowns for R&D programs during the reactor lifetime, was 88%. This compares very favorably with the best performances of fossil-fired and nuclear plants. A year-by-year listing of plant and NSS system availability, as well as gross plant capacity factors, is shown in Table 2-2.

The month-by-month power generation values for the thermal and gross electrical power generated at the station are shown for Core 1 and Core 2 in Tables 2-3 and 2-4, respectively. Also shown on a yearly basis are the equivalent full power days of operation and the net station electrical power generated.

Sections 3 through 12 discuss the operational accomplishments for each year of the 9 years of plant operation and present selected data graphs and tables of various operational parameters.

TABLE 2-2
PEACH BOTTOM PERFORMANCE^(a)

Year	Plant Availability (%)	NSS System Availability (%)	Gross Plant Capacity Factor (%)
1967	78	81	69
1968	88	88	82
1969	84	86	67
1970	95	95	88
1971	87	90	78
1972	71	71	58
1973	94	95	78
1974	95	96	70
Core 1 average	83	85	73
Core 2 average	88	89	74
Total lifetime station average	86	88	74

(a) Scheduled operation basis - because of the prototype nature of this plant, the time periods for technical specification and non-reactor related shutdowns were omitted when calculating these values.

TABLE 2-3
CORE 1 POWER GENERATION

Year and Month	MW(t)-hr	Gross MW(e)-hr	Net MW(e)-hr	EFPD
1967				
January	1,940	392		
February	0	0		
March	5,210	1,500		
April	0	0		
May	15,852	5,660		
June	76,901	29,240		
July	68,592	26,299		
August	74,799	28,517		
September	0	0		
October	49,173	18,364		
November	72,052	27,878		
December	72,433	27,747		
Total for 1967	436,951	165,597	147,598	158.3
1968				
January	29,129	10,975		
February(a)	0	0		
March	0	0		
April	0	0		
May	9,646	3,482		
June	76,177	28,827		
July	65,785	24,612		
August	69,911	26,069		
September	79,774	30,443		
October	60,112	23,050		
November(b)	0	0		
December	0	0		
Total for 1968	390,534	147,458	131,430	141.5
Total through 1968	827,485	313,055	279,028	299.8
1969				
January	0	0		
February	68	0		
March	56,677	21,452		
April	77,964	29,866		
May	76,320	28,460		
June	30,390	11,021		
July	73,445	27,308		
August	62,898	22,212		
September	38,059	13,045		
October	2,783	934		
November(c)	0	0		
December	0	0		
Total for 1969	418,604	154,298	137,528	151.7
Total through 1969	1,246,089	467,353	416,556	451.5

(a) Core 1 168-EFPD shutdown.

(b) Core 1 300-EFPD shutdown.

(c) Core 1 452-EFPD shutdown. End-of-life for Core 1.

TABLE 2-4
CORE 2 POWER GENERATION

Year and Month	MW(t)-hr	Gross MW(e)-hr	Net MW(e)-hr	EFPD
1970				
January	0	0		
February	0	0		
March	0	0		
April	0	0		
May	0	0		
June	0	0		
July	44,932	17,041		
August	81,278	30,337		
September	63,041	23,755		
October	73,299	27,899		
November	80,106	30,704		
December	81,300	31,086		
Total for 1970	423,956	160,822	142,907	153.6
Total through 1970	1,670,045	628,175	599,463	605.1
1971				
January	79,127	30,126		
February	70,958	27,177		
March	70,785	27,200		
April	51,835	19,873		
May (a)	0	0		
June	0	0		
July	44,385	15,918		
August	69,191	25,732		
September	66,999	25,353		
October	69,282	26,341		
November	22,193	8,654		
December	77,757	30,429		
Total for 1971	622,512	236,703	210,014	225.6
Total through 1971	2,292,557	864,878	769,477	830.7
1972				
January	17,311	6,777		
February (b)	0	0		
March	0	0		
April	0	0		
May	0	0		
June	4,273	1,205		
July	61,233	22,384		
August	56,743	19,893		
September	68,683	24,963		
October	74,946	27,515		
November	49,218	18,058		
December	0	0		
Total for 1972	332,406	120,794	105,093	120.4
Total through 1972	2,624,964	985,673	894,570	950.7

TABLE 2-4 (continued)

Year and Month	MW(t)-hr	Gross MW(e)-hr	Net MW(e)-hr	EFPD
1973				
January	45,831	17,891		
February	68,903	26,286		
March	70,732	27,089		
April	76,361	29,192		
May	43,576	16,299		
June	52,079	18,486		
July	82,237	28,821		
August	81,106	28,469		
September	35,688	12,520		
October(c)	0	0		
November	0	0		
December	0	0		
Total for 1973	556,513	205,053	181,120	201.7
Total through 1973	3,181,477	1,190,705	1,055,690	1152.8
1974				
January	65,866	25,679		
February	68,023	26,232		
March	68,999	26,255		
April	66,649	24,253		
May	53,770	20,386		
June	161	0		
July	40,255	12,767		
August	66,835	22,331		
September	53,783	18,169		
October	56,726	19,122		
November(d)	0	0		
December	0	0		
Total for 1974	541,066	195,194	169,193	196.1
Total through 1974	3,722,543	1,385,899	1,224,883	1348.9

(a) Core 2 252-EFPD shutdown.

(b) Core 2 385-EFPD shutdown.

(c) Core 2 701-EFPD shutdown.

(d) Core 2 897-EFPD shutdown. End-of-life for Core 2.

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Fig. 2-1. Peach Bottom atomic power station - unit 1

Fig. 2-2. Isometric view of reactor as installed in reactor cavity

Fig. 2-3. BISO coated fuel particle used in Core 2

Fig. 2-4. Peach Bottom fuel element

Fig. 2-5. Simplified process flow diagram for the main coolant system

3. 1966 OPERATING HISTORY

On January 24, 1966 the USAEC issued Provisional Operating License No. DPR-12 to the Philadelphia Electric Company authorizing operation of the Peach Bottom HTGR at power levels to 1 MW(t). On February 5, 1966, core loading was begun. On March 3, 1966 at 1:33 a.m., criticality was attained for the first core of the Peach Bottom HTGR at a loading of 682 elements. It was later determined that the unrodded core had a effective multiplication factor (k) of 1.00152. The predicted fuel loading for criticality was 689 ± 20 elements; thus, the actual loading was in close agreement with the expected results. The post-critical physics testing program was begun on March 5, 1966. On April 14, 1966 at 1:45 a.m., the 804th and final element was loaded.

Later in April, 24 type II elements were removed and replaced with type III elements to increase the shutdown margin from 4.1 to 5.8%. Type III elements contain boron lumped burnable poison spines, while type II elements do not. On April 29, 1966, the low-power physics testing program was completed. The results of the nuclear tests were in good agreement with predictions. Pressure, flow, and temperature coefficients were well within allowable limits. A value of $-4.3 \times 10^{-5} \Delta\rho/^{\circ}\text{F}$ was obtained for the temperature coefficient.

Because of difficulties in operating in an all-helium atmosphere, it was determined that the transfer machine and the viewing device required modification. During May 1966, they were sent to San Diego for modification and testing in a helium autoclave.

Work was also begun on in-situ replacement of all of the steam generator 304 stainless steel superheater section tubes in both steam generators with tubes made of Incoloy 800. This was required because of chloride stress corrosion cracking of the tubes at the tube sheet. The tube replacement

work was completed in December 1966. Following replacement, both steam generators and the Powdex system (condensate demineralizer) were chemically cleaned using Dow Vertan 690.

During June it was necessary to begin replacement of all tungsten carbide balls in the ball nuts of the linear actuators of the control rod drives. This action was required because of numerous ball failures over the previous several months. The ball replacement work was completed in September 1966.

It was determined through preliminary tests that the welded method of closure of the spent fuel cans was not acceptable. A modified canning machine utilizing a Magneform method of spent fuel closure was designed by General Atomic. In late September 1966, this machine was shipped to the Peach Bottom site. Checkout of the new system continued through September 1967, when it was shown that all systems operated satisfactorily except for leak-tight sealing of the spent fuel shipping can. The swaging coil and its associated hardware were removed and sent to General Atomic for further testing. Following return of these parts, the entire canning machine operation was successfully demonstrated in January 1968 using the Magneform swaged cap and double O-rings.

A 4% oxygen containment atmosphere was first successfully achieved on December 24, 1966.

4. 1967 OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1967 is shown in Fig. 4-1.

On January 12 the USAEC issued an operating license to the Peach Bottom facility authorizing operation of the reactor up to a power level of 115 MW(t). Beginning on January 20, the reactor was gradually brought to power. As the reactor power was raised, primary coolant activity increased. Since the low temperature delay beds were in service, no increase in primary coolant activity should have occurred. An investigation indicated that leakage from the tube side to the shell side in the first regeneration heat exchanger was causing bypassing of the low-temperature delay bed system. Upon bypassing the first regenerative heat exchanger, the loop activity began decreasing sharply within a few minutes. Electricity from nuclear power was first supplied to the Philadelphia Electric Company system at 1:25 p.m. on January 27. On January 30 the reactor reached 30% of full power. At this power level it became apparent that the steam generator outlet helium temperatures were higher than expected, which was caused by hot gas bypassing within the steam generator. The reactor was shut down to install new baffling in the steam generator. Work was started and completed on this task in February.

During the January hot run the helium cleanup continued. The purification system was in service throughout the run removing impurities. Results of main loop analyses are shown in Table 4-1.

Various main coolant gas samples were obtained during the power run, as shown in Table 4-2.

TABLE 4-1
MAIN LOOP IMPURITY LEVELS

Date	Time	Level (%)	Impurity Level (ppm by volume)								Reactor Inlet Temp (°F)	Reactor Outlet Temp (°F)	Reactor Pressure (psig)
			CO ₂	H ₂	Ne	O ₂ -Ar	N ₂	CH ₄	CO	H ₂ O			
1/1	1450	0	0	0	5	89	4660	0	0	55	440	440	230
1/7	0915	0	Trace	0	5	30	1650	0	0	11	400	400	200
1/20	2220	1	0	0	6	3	480	0	0	7	400	400	230
1/22	1540	1	0	0	5	0	10	0	0	5	400	400	210
1/25	1410	1	0	0	5	0	8	0	0	5	515	565	240
1/27	0720	1	0	0	5	0	30	0	0	7	450	490	280
1/28	0734	10	0	0	5	0	46	0	0	3	520	840	250
1/29	1000	20	1.7	0	5	0.5	44	2	2	5	620	1050	270
1/30	0958	30	0	0	5	0	42	Trace	3	2	625	1200	280
1/31	1400	0	0	0	8	0	44	0	0	2	<500	<500	220

TABLE 4-2
ACTIVITY IN MAIN COOLANT

Date (1967)	Time	Power (%)	Isotope	Activity (mCi/cm ³)	
				A301 Inlet (a)	Primary Coolant
1/24	1343	2	Kr-87	--	3.6×10^{-5}
1/26	0126	4.6	Xe-138	3.05×10^{-4}	--
	0508	4.6	Kr-85m	1.37×10^{-4}	--
			Kr-87	0.77×10^{-4}	--
			Xe-135	44.0×10^{-4}	--
1/29	0520	16	Kr-85m	2.75×10^{-3}	--
			Kr-87	8.1×10^{-3}	--
			Kr-88	9.3×10^{-3}	--
			Xe-135	8.4×10^{-3}	--
1/30	1322	26	Xe-138	1.08×10^{-2}	--
	1622	26	Kr-85m	--	1.35×10^{-5}
			Kr-87	--	1.35×10^{-5}
			Kr-88	--	0.48×10^{-5}
			Xe-135	--	0.33×10^{-5}

(a) A301 is trap for fuel element purge condensables.

During February the low-temperature delay bed cavity was opened, and tests were made to locate the suspected leak in the first regenerative heat exchanger. Since the nature of the failure was entirely unknown, it was decided to remove the defective section and cut a window in the shell to permit inspection of the tubes. When this was done, the rupture was found near the bottom of the "U"-bend. The failure appeared to have been caused by high internal pressure, and several other tubes also appeared to have been overstressed. It was found that ice plugs forming in the tubes could have caused the rupture.

A pressure test of the second regenerative exchanger was made successfully on February 8. On February 19, the exchanger was deiced in preparation for return of the system to service, but during this operation tube-to-shell leaks were detected. The exchanger was opened; four leaking tubes were located, plugged, and the heads rewelded. The exchanger was ready for service on February 28.

In March the plant was again started up and raised in planned steps to 59% power. During the power run it was determined that the modifications to the steam generator reduced the hot gas bypassing with the steam generators by about 50% but that further changes were required. The reactor was shut down and work was started on additional steam generator modifications to prevent hot gas bypassing. This work was completed in April.

On April 13 a fire in the low-temperature delay bed piping insulation caused considerable soot damage within the air-filled room and destroyed some insulation in this system. Smoke and soot accumulation was limited to the containment basement area and was cleaned up by the end of the month. The repair of the insulation and vapor barrier on the low-temperature delay bed piping was completed in May.

On May 7 reactor power was again increased and the turbine generator synchronized with the system. On May 10 reactor power was increased to 92% before reaching the maximum steam generator shell temperature. The design steam conditions of 1000°F, 1450 psig were first achieved on this date.

Between May 12 and 20, both steam generators were modified externally. Insulation was removed from the bottom head and an air-cooled funnel was installed. Through this system warm cavity atmosphere was circulated over the bottom head and then ducted to the cooling unit. The gas was then discharged to the steam generator cavity for recirculation. The maximum permitted steam generator shell temperature was 753°F.

During June the plant operated at essentially full power for the entire month. The plant officially went into commercial operation as of June 1.

Helium inventory was reported in June to show a comparatively constant loss of 7 to 10 lb/day.

The plant operated essentially at full power for July and August but was shutdown during all of September to repair a leak on a flange on the No. 1 steam generator, to permit repair of certain steam generator leaks, and to make heat cycle modifications which were awaiting a plant scheduled outage. During October, November, and December, the plant operated at essentially full power.

In November the fuel element purge stream activity was reported to be as follows:

<u>Isotope</u>	<u>R/B* for November 1967</u>
Kr-85m	3×10^{-3}
Kr-87	1.1×10^{-3}
Kr-88	1.3×10^{-3}
Kr-89	4×10^{-4}
Xe-133	5.5×10^{-3}
Xe-135	9×10^{-4}
Xe-138	2.6×10^{-4}

*R/B = release rate/birth rate.

The main loop radioactivity averaged 195 mCi as indicated by the Cary ion chamber in November.

The following data have been plotted as a function of reactor operating time for 1967. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 4-2
Primary circuit nitrogen	Fig. 4-3
Primary circuit oxygen and argon	Fig. 4-4
Primary circuit carbon monoxide	Fig. 4-5
Primary circuit methane	Fig. 4-6
Primary circuit total hydrocarbons	Fig. 4-7
Primary circuit flow	Fig. 4-8
Primary circuit core inlet and outlet temperatures	Fig. 4-9
Steam drum SiO ₂	Fig. 4-10
Steam drum Cl ion	Fig. 4-11
Steam drum pH	Fig. 4-12
Deaerator pH	Fig. 4-13
Deaerator hydrazine	Fig. 4-14
Deaerator oxygen	Fig. 4-15

The primary circuit helium pressure, although not recorded in the 1967 Unit Logs, was held at 335 psig during periods of reactor operation.

Typical operating data for 1967 are shown in Table 4-3. The significant 1967 plant operating data reported to the USAEC are shown in Table 4-4.

TABLE 4-3
1967 TYPICAL OPERATING DATA

	6/19	12/1
Reason for inclusion	Typical operation	Typical operation
Reactor power level, %	99.5	97.3
Reactor primary flow, lb/hr x 10 ⁻³		
Loop 1	235 ^(a)	231
Loop 2	247 ^(a)	239
Core inlet temperature, °F		
Loop 1	636	643
Loop 2	644	653
Core outlet temperature, °F		
Loop 1	1294	1303
Loop 2	1292	1302
Hot valve temperature, °F	NA ^(b)	NA
Core differential pressure, psi	NA	NA
Core purge flow, lb/hr	771	748
Total purification system flow (includes steam generator purge flow), lb/hr	980	960
Steam drum pressure, psig		
No. 1	1510	1520
No. 2	1510	1515
Boiler feed pump suction pressure, psig	137	132
Boiler feed pump discharge pressure, psig		
No. 1	1830	1820
No. 2	1840	1850
Boiler feed pump suction temperature, °F	344	341
Final feedwater temperature (from 4A and 4B heaters), °F		
4A	424	428
4B	423	423
Bleed steam pressure (to fourth heater), psig	350	332
Bleed steam temperature (to fourth heater), °F	684	683
Throttle pressure, psig	1400	1460
Throttle temperature, °F	996	1000

(a) Corrected to account for recalibration of reactor primary flow recorders on July 19.

(b) NA = not available.

TABLE 4-4
SIGNIFICANT PLANT DATA REPORTED BY THE
PHILADELPHIA ELECTRIC COMPANY TO THE USAEC IN 1967

	6/67	7/67	8/67	10/67	11/67	12/67
Gross generated power, MW(e)	43.5	43.6	44.2	42	42	42.5
Steam flow, lb/hr x 10 ⁻³	352	380	385	360	360	360
Throttle pressure, psig	1450	1450	1450	1450	1450	1450
Throttle temperature, °F	1000	1000	1000	1000	1000	1000
Condenser vacuum, in. Hg abs.	1.67	1.73	1.78	1.0	1.0	0.75
Reactor inlet temperature, °F	642	651	644	645	640	646
Reactor outlet temperature, °F	1294	1315	1313	1296	1293	1300
Total He flow, lb/hr x 10 ⁻³	476	473	470	460	461	457
Reactor pressure, psig	NR ^(a)	318	333	334	324	334
Reactor power, %	98	97.8	99	94.7	95.5	95.5
Reactor power, MW(t)	112.7	112.5	113.8	108.9	109.8	109.8
Hot spot fuel temperature (average), °F	2000	2000	No longer reported to USAEC			
Hot spot fuel temperature (maximum), °F	2200	NR				

(a) NR = not reported.

Fig. 4-1. Peach Bottom operating history - 1967

Fig. 4-2. Main loop activity - total inventory-1967

Fig. 4-3. Main loop N₂-1967

Fig. 4-4. Main loop O_2 Ar-1967

Fig. 4-5. Main loop CO-1967

Fig. 4-6. Main loop CH₄-1967

Fig. 4-7. Main loop hydrocarbons-1967

Fig. 4-8. Main loop flow-1967

Fig. 4-9. Main loop core inlet and outlet temperatures-1967

Fig. 4-10. No. 1 steam drum SiO_2 -1967

Fig. 4-11. No. 1 steam drum C1-1967

Fig. 4-12. No. 1 steam drum pH-1967

Fig. 4-13. Deaerator pH-1967

Fig. 4-14. Deaerator hydrazine-1967

Fig. 4-15, Deaerator O₂-1967

5. 1968 OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1968 is shown in Fig. 5-1.

Reactor operations continued at about 97% power through January 11 when the shutdown for the scheduled core element examination began. The reactor had successfully completed 168 EFPD when it was shut down. On January 9, a rapid rise in main coolant activity was noticed, increasing from approximately 330 mCi and reaching a total of 4.2 Ci. Radiochemical analyses of gas samples were made while tests were run on the purification system to determine the origin of the activity. It was concluded that insufficient purging of one or more fuel elements was causing the release. Possible causes of this condition were determined to be a crack in the fuel element sleeve or a restriction in the purge flow. Radiochemistry data (Table 5-1) indicated that the fuel was responsible for the increase in the main coolant activity. This was supported by the presence of short-lived Kr-89 (3.2 m) and Xe-138 (17 m) in the main loop grab samples and the increase of the longer-lived xenons and kryptons by a factor of 100.

During January the fuel element purge stream radioactivity averaged 39 Ci/lb He during full-power operation. This value represented no significant variance from previous values, indicating that the cause of the increase in main loop radioactivity had no effect on the fuel element purge stream. Release fractions as determined from fuel element purge stream analysis were as follows:

<u>Isotope</u>	<u>R/B for January 1968</u>
Kr-85m	2.3×10^{-3}
Kr-87	8.9×10^{-4}
Kr-88	1.2×10^{-3}

TABLE 5-1
CONCENTRATION OF KRYPTON AND XENON
($\mu\text{Ci}/\text{STD cm}^3$)

Sample	Date	Kr-85m	Kr-88	Kr-87	Kr-89	Xe-138	Xe-135	Remarks
Main loop	11/21/67	6.6×10^{-6}	7.7×10^{-6}	3.8×10^{-6}	--	--	3.1×10^{-6}	Typical analysis
Main loop	1/10/68	4.9×10^{-4}	3.2×10^{-4}	3.0×10^{-4}	7.6×10^{-4}	5.8×10^{-5}	2.7×10^{-4}	During excursion
Fuel element purge stream	12/27/67	0.35	0.75	0.87	11.0	3.4	0.23	Typical analysis
Fuel element purge stream	1/10/68	0.24	0.45	0.51	8.8	1.7	0.13	During excursion ^(a)

(a) This set of data taken during the excursion was not in variance with normal expected levels.

<u>Isotope</u>	<u>R/B for January 1968</u>
Kr-89	3.5×10^{-4}
Xe-133	2.4×10^{-3}
Xe-135	6.7×10^{-4}
Xe-138	3.0×10^{-4}

Because of the blasting schedule for the nearby site of Peach Bottom Units 2 and 3, reactor core inspection was delayed until February.

In mid-February, following the completion of blasting at the nearby site of Peach Bottom Units 2 and 3, the failed fuel element locator was used to measure the purge flow in each of the fuel elements. One element in location C05-05 was found to be purging improperly.

On February 20 two hexagonal reflector elements in locations A17-16 and B18-12 were visually inspected according to test JM-4; no abnormalities were detected.

On February 21 two fuel elements from locations B13-07 and B13-05 were visually inspected and also appeared to be in satisfactory condition. These two elements were gamma scanned on February 24 to measure the fuel compact stack length. These elements exhibited a total growth of slightly less than 0.6 in., which was expected. While returning the B13-05 element to the core, it was accidentally dropped from the charge machine grapple into the parking hole canister in the radial reflector. The upper reflector of the element broke free and bounced out of the canister, coming to rest on the solid reflector blocks to the east of the parking hole canister. Operations inside the reactor vessel were suspended to permit the necessary modifications to the fuel handling machine in order to move the damaged element from the reactor to the spent fuel pit. The dropped fuel element was removed on March 19, and a repeat gamma scan indicated that all of the fuel in this dropped element was recovered. The top reflector section of the element was shipped to San Diego for examination.

The tops of control rod guide tubes were visually inspected on February 24, and no abnormalities were detected. The thermal barrier in-vessel visual inspection was also performed during February; again, no deterioration was observed.

On April 1 the element from location C05-05 was removed from the reactor using the failed fuel element removal tool. Visual inspection of this element showed a circumferential separation of the element sleeve near the axial center of the element. Fourteen additional elements were gamma scanned and visually inspected between April 4 and 11. All of these elements appeared normal with the exception of the element from location E05-05, which showed a very small fuel compact stack growth and a surface marking, shown by use of television in the canning machine cavity, that was judged to possibly be a crack. This element was removed from the core, canned, and shipped to San Diego for closer examination in the hot cell. It was received at the hot cell on April 15. On April 19 it was again gamma scanned in San Diego with results similar to those obtained at the site. Subsequent examination in the General Atomic hot cell proved that no crack existed, but that the mark was due to light reflection from the graphite surface.

The failed fuel element (C05-05) was shipped to San Diego on April 29. New fuel elements were inserted into the core in locations B13-05 (the location of the dropped element), E05-05 (suspected abnormal element), and C05-05 (failed element). The elements temporarily removed from locations near C05-05 and the temporarily removed reflector elements were returned to their original location, and the Fort St. Vrain (FSV) test element (proof test element PTE-1) was inserted into location A11-11 in preparation for returning to power in early May.

On May 6 a step-wise rise to power was initiated. On May 8, the No. 1 steam generator shell reached the temperature limit of 753°F with the plant at 78% power. Reactor power was held at 78% for 2 days at design conditions while collecting data on: (1) main loop radioactive inventories, (2) xenon

buildup, and (3) low-level chemical impurities. At this time the main coolant radioactive inventory had equilibrated at about 900 mCi, or approximately three times the 300 to 350 mCi experienced prior to insertion of the FSV test element.

On May 10 the steam temperature and reactor outlet temperature were reduced to allow increasing the station power to 95% without elevating the No. 1 steam generator shell hot spot temperature above 753°F. During this power rise, the main coolant radioactive inventory increased to 3.5 Ci.

The reactor was shut down on May 11 to modify the steam generator cooling system and to modify welds in the helium purification systems. Because of the unexpectedly high main loop activity level, it was decided to remove the FSV PTE-1 from core location A11-11 and replace it with the original fuel element from this location.

On May 31 the plant was returned to power. With the reactor power at 40%, the fission product inventory of the main helium loop was comparable to that observed prior to the fuel element failure in January 1968. As power was increased from 40 to 48%, a sharp rise in the main helium loop activity to approximately 900 mCi was noted on the main loop activity monitor.

The plant was held at 58% power for approximately 3 hours until radiochemical analysis of grab samples verified that short-lived isotopes were responsible for the activity increase. Following this verification, the plant power was raised to 78% on May 31 to obtain data comparable to those gathered when the FSV proof test element was in the reactor. On June 1 station power was raised to approximately 95%, at which time the No. 1 steam generator shell temperature hot spot reached 753°F. During the rise in power from 78 to 95%, the primary system activity increased from 2.1 to 3.1 Ci. Data gathered during the approach to 78% power and during the sustained 78% power run indicated that the release to the main loop was unique to this startup. All information indicated that the 2.1 Ci main loop

activity at 78% power was a result of an unpurged fuel element. On June 2 an additional increase in primary system activity to approximately 6 Ci indicated the possibility of a second defective element.

With the exception of three minor outages, the plant operated at essentially full power until the scheduled shutdown on October 23 for the 300 EFPD core component surveillance tests.

The main loop activity level, as indicated by the Cary electrometer, experienced a number of step increases during this period. On July 29 the activity increased from 7.0 to 8.3 Ci. Following the return to power on August 14, an apparent increase in the main loop activity level to 9.3 Ci was indicated. On September 9, a loop activity increase to 11.3 Ci was indicated, followed by increases to 14.0 Ci on September 14 and to 19.4 Ci on September 17. On October 4 a main loop activity increase to 22.6 Ci was indicated, followed by an increase to 28 Ci on October 5. Two or three additional activity increases took place on October 22 and 23 just before the 300-EFPD shutdown. These increases brought the indicated main loop activity level to 36.2 Ci. The main loop activity level from the first activity increase in January through December is given in Fig. 5-2. The cumulative thermal energy output of the core is also given in this figure for reference.

Table 5-2 shows average values of specific isotope activities in the main loop plus R/B fractions in the purge stream for October.

Immediately following the October 23 shutdown, measurements were made for several postconstruction R&D tests. The material in the main loop dust separators was removed and shipped to San Diego at the end of October.

On November 6 the diffusion probe was removed and shipped to San Diego. The failed fuel element location test was performed November 8 through November 13. This test identified eleven improperly purged elements. The eleven unpurged fuel elements were in core locations A13-07, B12-02, B15-09, B12-12 (removed November 27), C14-07, C15-09, D06-01, D08-04, D10-03, E02-01, and E08-06.

TABLE 5-2
MAIN LOOP ACTIVITIES AND PURGE STREAM R/B

Isotope	Main Loop (mCi/cm ³)	Purge Stream R/B
Kr-85m	1.8×10^{-3}	3.5×10^{-3}
Kr-87	2.3×10^{-3}	1.2×10^{-3}
Kr-88	2.0×10^{-3}	1.7×10^{-3}
Kr-89	2.1×10^{-3}	2.5×10^{-3}
Xe-135	9.5×10^{-4}	1.4×10^{-3}
Xe-138	1.3×10^{-3}	3.4×10^{-4}

Within the reactor plenum, visual inspection was made of hexagonal reflector elements B18-12 and C17-03 on November 18. Additionally, fuel elements B13-07 and A16-15 were removed to the canning machine, visually inspected, gamma scanned, and returned to their core locations. Results of the gamma scanning indicated that the power distribution was as expected and that the elements exhibited a compact growth of 0.9 and 0.2 in., respectively. On November 17 and 18 the in-vessel thermal barrier inspection was performed using the reactor core viewing device. Results indicated that the thermal barrier was in excellent condition.

On November 20 and 21 five reflector elements were removed from the reactor and canned for shipment to San Diego for postirradiation examination. Four of these five reflector elements contained Charpy specimens of the reactor vessel material for testing in the General Atomic hot cell to determine the shift in the nil-ductility transition temperature.

On November 4 the No. 1 main helium loop was manually isolated to prepare for inspection of the water side of the No. 1 steam generator. The inspections of the tube sheets and the superheater outlet barrel dissimilar weld were done biennially in accordance with the technical specifications.

The first of the eleven improperly purged (failed) fuel elements (B12-12) was removed from the reactor on November 27, the second failed element (A13-07) was removed on December 6, and the third failed element (B15-09) was removed on December 9. In the process of removing B15-09, some force was required to insert the headless fuel element removal tool over the element. On December 16 the fourth failed element (C16-09) was removed. The removal of this element required even higher forces to place the removal tool over the element and then to remove the failed element from the core. Based on the experience in removing these two elements, it was decided to modify the removal tools before attempting to remove additional failed elements.

During November fuel element D06-01 was removed from the core and stored in the spent fuel pit awaiting shipment in the single element

(Hallam) cask to the General Atomic hot cells. This element was given a postirradiation examination as required by the technical specification for the reactor.

On December 30 the control rod guide sleeve from core location C08-01 was removed and canned for shipment to San Diego for postirradiation examination as required by the technical specifications.

The data listed below have been plotted as a function of reactor operating time for 1968. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 5-3
Primary circuit nitrogen	Fig. 5-4
Primary circuit oxygen and argon	Fig. 5-5
Primary circuit carbon monoxide	Fig. 5-6
Primary circuit methane	Fig. 5-7
Primary circuit flow	Fig. 5-8
Primary circuit core inlet and outlet temperatures	Fig. 5-9
Steam drum SiO ₂	Fig. 5-10
Steam drum Cl ion	Fig. 5-11
Steam drum pH	Fig. 5-12
Deaerator pH	Fig. 5-13
Deaerator hydrazine	Fig. 5-14
Deaerator oxygen	Fig. 5-15

The primary circuit helium pressure, although not recorded in the 1968 Unit Logs, was held at 335 psig during periods of reactor operation.

Typical operating data for 1968 are shown in Table 5-3. The significant 1968 plant operating data reported to the USAEC are shown in Table 5-4.

TABLE 5-3
1968 TYPICAL OPERATING DATA

	5/7	5/9	7/15	10/15
Reason for inclusion	Startup	Startup	Typical operation	Typical operation
Reactor power level, %	47.0	77.1	93.6	96.3
Reactor primary flow, lb/hr x 10 ⁻³				
Loop 1	123	186	233	249
Loop 2	121	189	234	248
Core inlet temperature, °F				
Loop 1	597	623	648	653
Loop 2	605	632	655	658
Core outlet temperature, °F				
Loop 1	1215	1280	1285	1267
Loop 2	1204	1276	1285	1270
Hot valve temperature, °F	NA ^(a)	NA	NA	NA
Core differential pressure, psi	NA	NA	NA	NA
Core purge flow, lb/hr	645	817	750	735
Total purification system flow (includes steam generator purge flow), lb/hr	870	1040	930	930
Steam drum pressure, psig				
No. 1	1420	1490	1505	1520
No. 2	1410	1480	1490	1500
Boiler feed pump suction pressure, psig	96	110	132	136
Boiler feed pump discharge pressure, psig				
No. 1	1880	1860	1850	1825
No. 2	1910	1900	1860	1850
Boiler feed pump suction temperature, °F	315	325	340	342
Final feedwater temperature (from 4A and 4B heaters), °F				
4A	363	410	428	430
4B	359	405	425	425
Bleed steam pressure (to fourth heater), psig	213	265	330	344
Bleed steam temperature (to fourth heater), °F	647	662	678	672
Throttle pressure, psig	1430	1460	1440	1450
Throttle temperature, °F	957	985	977	977

(a) NA = not available.

TABLE 5-4
SIGNIFICANT PLANT DATA REPORTED BY THE
PHILADELPHIA ELECTRIC COMPANY TO THE USAEC IN 1968

	1/68	5/68	6/68	7/68	8/68	9/68	10/68
Gross generated power, MW(e)	42.7	41.6	42.0	40.8	42.0	42.5	42.5
Steam flow, lb/hr x 10 ⁻³	350	360	367	350	368	370	370
Throttle pressure, psig	1450	1450	1440	1450	NR ^(a)	NR	NR
Throttle temperature, °F	1000	1000	990	980	967	980	977
Condenser vacuum, in. Hg abs.	0.50	1.4	1.7	1.4	1.6	1.4	1.2
Reactor inlet temperature, °F	646	645	653	650	655	651	655
Reactor outlet temperature, °F	1296	1296	1293	1255	1263	1268	1270
Total He flow, lb/hr x 10 ⁻³	474	470	478	490	499	496	494
Reactor pressure, psig	318	322	320	328	330	335	330
Reactor power, %	97.9	95.5	96.0	94	97	97.5	95
Reactor power, MW(t)	112.6	110.3	110.5	108.1	111.6	112.6	109.3

(a) NR = not reported.

Fig. 5-1. Peach Bottom operating history - 1968

Fig. 5-2. Main loop activity level from January through December 1968

Fig. 5-3. Main loop activity - total inventory-1968

Fig. 5-4. Main loop N₂-1968

Fig. 5-5. Main loop O_2 Ar-1968

Fig. 5-6. Main loop CO-1968

Fig. 5-7. Main loop CH₄-1968

Fig. 5-8. Main loop flow-1968

Fig. 5-9. Main loop core inlet and outlet temperatures-1968

Fig. 5-10. No. 1 steam drum SiO_2 -1968

Fig. 5-11. No. 1 steam drum CI-1968

Fig. 5-12. No. 1 steam drum pH-1968

Fig. 5-13. Deaerator pH-1968

Fig. 5-14. Deaerator hydrazine-1968

Fig. 5-15. Deaerator O₂ -1968

6. 1969 OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1969 is shown in Fig. 6-1.

On January 2 the control rod absorber from core location C08-01 was removed and canned for shipment to San Diego for postirradiation examination as required by the technical specifications. The control rod guide tube and the absorber section were both replaced with new components and control rod 4A was again operational.

On January 9 the fifth failed element (D06-01) was removed using a modified removal tool. The configuration of the modified removal tool allowed some viewing of the failed elements by means of a television system in the canning machine area. The sixth failed element (C14-07) was removed on January 18 and the seventh (D10-03) was removed on January 21. The eighth failed element (C08-04) was removed on January 27. This element was rechecked with the failed fuel element locator before removal; the locator reading had not changed significantly during the shutdown. Failed element E02-01 was removed on January 28, and E08-06 was removed on January 29. The eleventh and last failed element (B12-02) was successfully removed from the core on January 31. Element D06-01 was loaded into the Hallam shipping cask and shipped to the General Atomic hot cell facility in San Diego on January 21 for postirradiation examination. The fuel element in core location E10-01 was moved to D06-01 and Fort St. Vrain proof test element PTE-2 was placed in the vacated location.

New hexagonal reflector elements were installed in the reflector to replace the five removed as part of the 300-EFPD inspection, and all fuel removed during the outage was replaced with new fuel.

On February 13 two trays containing reactor vessel plate and weld specimens were removed from the access port shield plug. These Charpy specimens were sent to General Atomic for nil-ductility transition temperature testing.

On February 28 the reactor began the step-wise rise to power following the 300-EFPD testing and maintenance work.

The reactor was shut down from March 3 to 7 to repair a helium leak originating from a control rod drive unit in the subpile room. Except for a brief reduction in power on March 9 for maintenance on a helium transfer compressor, the reactor operated at essentially full power until March 24, when a shutdown was made to repair a leak at the flange of one of the steam drum safety valves. The reactor was returned to power on March 28, and was again shut down the following day to repair a check valve in the feedwater header. On March 30 the reactor was returned to essentially full-power operation which continued with no interruptions through May 18, when the reactor was shut down for 1 day to repair a generator field ground. The reactor was returned to power on May 19 and operated at 88% power until June 2. During the latter portion of this period, the reactor power level was increased to essentially full load during daily peak-load periods and reduced to 88% power during low-load periods in order to extend power operation through the peak summer load.

On June 2 the reactor was manually scrammed because of the inadvertent insertion of a single emergency rod (7E), and the reactor was not returned to power operation until June 19. During this period the reason for the inadvertent emergency rod insertion was investigated, and the emergency rod was returned to service. During the same period, modifications were made to the plant safety system to stop all fuel element purge flow should high loop moisture be indicated. The technical specification changes for these modifications in the safety system were also approved during this period. The reactor was shut down on June 20 to repair a leaking check valve in the feedwater line and returned to power on June 21.

Operation at 94% of full power continued until July 7, when the reactor was shut down for 2-1/2 days to repair a steam leak in the steam drum level instrumentation. Following the return to power on July 10, operation continued at 94% power until August 3. In order to extend operations through the summer load peak and to minimize fuel element cracking, the reactor power level was reduced to 77% on August 3, to 60% on August 20, and to 50% on August 28. With the exception of eight small power perturbations, operation at 50% power continued until the reactor was shut down on October 3 for the scheduled 450-EFPD core component examinations.

During and following the initial rise to power, February 28 through March 3, the maximum main loop activity was 0.17 Ci, indicating the successful removal of all failed elements during the 300-EFPD shutdown and satisfactory operation of the second FSV proof test element. During reactor startup on March 7, an activity increase to 6 Ci was indicated by the Cary electrometer. This increase was interpreted to be indicative of the failure of one or possibly two fuel element sleeves. Later in the same day, a second activity increase to 10 Ci was recorded, and on March 16 two additional increases to 12.5 and 15.5 Ci took place. On March 17 a fifth increase to about 20 Ci occurred. A sixth increase to about 30 Ci was observed on April 21.

On May 2 a 4-Ci increase in the indicated circulating activity took place. A 5-Ci increase occurred on May 14, and a 3-Ci increase occurred on May 28. The following day, the fourth increase (6 Ci) took place. When the reactor was returned to power on June 19, the main loop activity level indicated that a fuel element had lost purge capability between the June 2 scram and the June 19 startup. The activity level following the July 10 startup indicated that another element had lost purge flow. The following day another step increase of 5 Ci took place. A step increase of 3 Ci on July 17 indicated that another fuel element had lost purge. These increases in the main loop activity indicated that there were 14, or possibly 15, improperly purged elements present in the core as of July 31.

The main loop activity level for August, September, and October was quite erratic. Because of the differing power levels and the occurrence of multiple fuel element failures during a power level change during this period, it was extremely difficult to estimate the precise number of fuel element failures that took place. The use of the failed fuel element locator following the 450-EFPD shutdown to measure the purge flow in each fuel element identified 78 improperly purged elements. The maximum indicated loop activity at full power was 270 Ci. The resulting average value of 3.5 Ci per improperly purged element at full power is consistent with the value of 300 and 158 EFPD. The use of this value to interpret the number of fuel element failures was used to yield a reasonably accurate number.

On May 8 the control rod absorber section from core location C08-01 (4A) was shipped to General Atomic for hot cell examination as per the technical specifications.

The reflector element removed from core location B16-01 during the 300-EFPD shutdown was loaded into the single-element (Hallam) shipping cask at the site and shipped to the GA hot cell facility for postirradiation examination as required by the technical specifications. This reflector element was received at GA and unloaded on September 26. The cask was returned to the site, and the control rod guide tube that was removed from core location C08-01 at 300 EFPD was shipped on October 29 to the hot cell for post-irradiation examination in accordance with the technical specification.

The diffusion tube was removed from the hot valve following the October 3 shutdown and shipped to San Diego for examination and analysis.

The Peach Bottom reactor remained shutdown for the rest of the calendar year for the 450-EFPD core component examinations. The major operations at the reactor site involved removal of the majority of the 78 improperly purged fuel elements, which were identified following the 450-EFPD shutdown. As of December 31, 42 of the 78 improperly purged fuel elements had been removed.

Several circumstances during the movement of these fuel elements produced some delay in their removal. On December 18 while fuel element B16-12 was being transported over the core within a removal tool, it was inadvertently dropped into vacant core location B16-10. An investigation of the occurrence led to reworking of the remaining removal tools. The tools were shipped to San Diego for this reworking and then returned to the site. Special tools and viewing devices for removal of the dropped element were designed and fabricated. It was planned to begin removal of the dropped element in February 1970.

A meeting between Philadelphia Electric and General Atomic was held in Philadelphia on November 14 to review the results of the 450-EFPD core inspection and to determine the mode of future operations. It was concluded that Core 1 could likely continue operation at reduced power, but its replacement with Core 2 was preferable. The fabrication, shipment, licensing, and loading of Core 2 and installation of experimental test elements were also topics of discussion.

The data listed below have been plotted as a function of reactor operating time for 1969. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 6-2
Primary circuit nitrogen	Fig. 6-3
Primary circuit oxygen and argon	Fig. 6-4
Primary circuit carbon monoxide	Fig. 6-5
Primary circuit carbon dioxide	Fig. 6-6
Primary circuit methane	Fig. 6-7
Primary circuit flow	Fig. 6-8
Primary circuit core inlet and outlet temperatures	Fig. 6-9
Steam drum SiO ₂	Fig. 6-10
Steam drum Cl ion	Fig. 6-11
Steam drum pH	Fig. 6-12
Deaerator pH	Fig. 6-13

Deaerator hydrazine

Fig. 6-14

Deaerator oxygen

Fig. 6-15

The primary circuit helium pressure, although not recorded in the 1969 Unit Logs, was held at 335 psig during periods of reactor operation.

Typical operating data for 1969 are shown in Table 6-1. The significant plant operating data reported to the USAEC are shown in Table 6-2.

TABLE 6-1
1969 TYPICAL OPERATING DATA

	3/20	4/20	5/25	8/12	8/22	9/25
Reason for inclusion	Typical operation	Typical operation	Reduced power operation	Reduced power operation	Reduced power operation	Reduced power operation
Reactor power level, %	98.8	94.4	85.8	75.5	59.5	47.5
Reactor primary flow, lb/hr x 10 ⁻³						
Loop 1	256	243	221	177	139	102
Loop 2	257	252	225	186	152	134
Core inlet temperature, °F						
Loop 1	653	650	639	627	604	587
Loop 2	657	653	644	633	609	595
Core outlet temperature, °F						
Loop 1	1268	1260	1252	1287	1257	1232
Loop 2	1260	1249	1248	1288	1250	1225
Hot valve temperature, °F	1250	1238	1236	1247	1212	1170
Core differential pressure, psi	NA ^(a)	NA	NA	NA	NA	NA
Core purge flow, lb/hr	886	845	805	852	850	850
Total purification system flow (includes steam generator purge flow), lb/hr	1051	895	955	1002	1000	1000
Steam drum pressure, psig						
No. 1	1500	1500	1490	1485	1470	1460
No. 2	1495	1485	1470	1460	1450	1450
Boiler feed pump suction pressure, psig	134	130	120	109	87	72
Boiler feed pump discharge pressure, psig						
No. 1	1820	1850	1860	1890	1905	1910
No. 2	1860	1860	1885	1920	1930	1940
Boiler feed pump suction temperature, °F	343	340	332	323	205	290
Final feedwater temperature (from 4A and 4B heaters), °F						
4A	432	430	417	407	384	367
4B	425	421	415	401	377	360
Bleed steam pressure (to fourth heater), psig	345	331	298	255	193	146
Bleed steam temperature (to fourth heater), °F	670	663	651	663	628	611
Throttle pressure, psig	1450	1450	1450	1450	1450	1460
Throttle temperature, °F	975	960	980	1000	990	947

(a) NA = not available.

TABLE 6-2
SIGNIFICANT PLANT DATA REPORTED BY THE
PHILADELPHIA ELECTRIC COMPANY TO THE USAEC IN 1969

	3/69	4/69	5/69	6/69	7/69	8/69	9/69	10/69
Gross generated power, MW(e)	43.0	41.5	42.0	40	40.0	41	19	13
Steam flow, lb/hr x 10 ⁻³	372	356	363	352	345	360	164	118
Throttle temperature, °F	980	965	965	980	990	985	970	975
Condenser vacuum, in. Hg abs.	0.8	1.1	1.2	1.9	1.6	1.84	1.00	1.2
Reactor inlet temperature, °F	656	647	654	651	653	655	590	560
Reactor outlet temperature, °F	1263	1257	1260	1251	1274	1264	1227	1225
Total He flow, lb/hr x 10 ⁻³	511	488	499	496	472	501	237	166
Reactor pressure, psig	333	330	332	330	335	332	330	332
Reactor power, %	97	94	95	94	93	95	48	35
Reactor power, MW(t)	111.6	108.1	109.3	108.1	106.9	109.3	54.9	40.3

6-9

Fig. 6-1. Peach Bottom operating history - 1969

Fig. 6-2. Main loop activity - total inventory-1969

Fig. 6-3. Main loop N₂-1969

Fig. 6-4. Main loop O₂Ar-1969

Fig. 6-5. Main loop CO-1969

Fig. 6-6. Main loop CO₂-1969

Fig. 6-7. Main loop CH₄-1969

Fig. 6-8. Main loop flow-1969

Fig. 6-9. Main loop core inlet and outlet temperatures - 1969

Fig. 6-10. No. 1 steam drum SiO_2 -1969

Fig. 6-11. No. 1 steam drum C1-1969

Fig. 6-12. No. 1 steam drum pH-1969

Fig. 6-13. Deaerator pH-1969

Fig. 6-14. Deaerator hydrazine-1969

Fig. 6-15. Deaerator O₂-1969

7. 1970 OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1970 is shown in Fig. 7-1.

The reactor remained shutdown during the initial part of 1970 while the 450-EFPD core component examinations and reactor refueling were continued. In January failed element removal was continued until January 26 when the bottom weld of the spent fuel elevator failed, allowing an uncapped spent fuel canister containing a failed fuel element with its removal tool and steel liners to jam with the elevator in the fuel chute bend below the canning machine. The safe recovery was completed by February 2. Salvage operations for this mishap were completed by February 11, with placement of a salvage canister containing the standard canister and associated contents in the spent fuel pit.

The single-element Hallam shipping cask was shipped from San Diego to the reactor site on January 19. The reflector element from location B17-03 (removed from the vessel at 300 full-power days) was loaded in the cask on January 30 for shipment to the GA hot cell facility. This reflector contained Charpy specimens of the vessel parent material, which were tested in the GA hot cell for changes in nil-ductility transition temperature.

The fuel element (B16-12) which was inadvertently dropped into vacant core location B16-10 on December 18, 1969 was successfully removed during the first quarter of 1970. The upper two thirds was removed in March with a special probing tool, and the bottom one-third was removed on April 6 using a specially designed jacking tool. Failed element removal was completed on April 9.

Technical specification change No. 12, titled "Core 2 Loading and Low-Power Testing," and change No. 13, titled "Core 2 Design and Operational

Evaluation," were submitted to the USAEC on February 20. The Division of Reactor Licensing approval of the Core 2 loading document was received on March 24, and Core 2 loading began on April 10. By the end of April, 251 Core 2 elements had been loaded at a rate of about 15 elements per day. The reactor was taken critical on April 23 to verify the calculated shutdown margin and other physics calculations. The calculated criticality point was within 0.003 $\Delta\rho$ of the actual value.

Samples of particulate matter in the main loop dust collectors were transferred to a shipping cask and shipped to GA on March 31.

During April two reflector elements (D18-12 and E18-12) were cut at the Peach Bottom site and the bottom 15 in. of each (containing Charpy steel vessel irradiation surveillance specimens) was shipped to GA on April 28. These specimens, which were exposed for 452 EFPD, were used to further study the nil-ductility temperature shift. The results were reported in Refs. 1 and 2.

Refueling of the reactor was completed on June 22. A loading adjustment was made by changing eight type III fuel elements with eight type II fuel elements. (Type III elements contain lumped burnable poison while type II elements do not). This adjustment was requested by GA in order to change the reactor power distribution and thereby enhance test element performance. The reactor was taken critical following this change, and the shutdown margin was determined to be satisfactory at 0.728 $\Delta\rho$ at 80°F.

On June 26 the access nozzle shield plug was removed in order that two trays of Charpy samples could be removed for shipment to GA as part of the nil-ductility temperature surveillance program for the reactor vessel.

A bow in hexagonal reflector E18-12, apparently caused by irradiation in the reactor, was determined using a surveyor's transit on the refueling floor on June 8. This reflector had been removed from the reactor during the 450-EFPD outage. Manipulation of the hexagonal reflector to allow sights along a flat was difficult; therefore, sights were taken along the

intersection of two flats. The maximum deflection at the axial midpoint of the element was 0.0442 in. This deflection was considered small and presented no problems for the continued operation of reflector elements.

In order to obtain maximum fission product data from the test fuel elements being installed in the reactor, a test element purge gas sample system was installed. The system was installed in accordance with applicable codes and standards and was adapted to the acoustic thermometer installation associated with Core 1. The completion of this modification allowed sampling eight test element purge streams on an individual basis to the existing radiochemical sampling system XSM-310.

On May 28 official approval was received from the Division of Reactor Licensing to operate Core 2, including the 17 installed test elements (see Table 7-1).

At 12:45 a.m. on July 14, plant checkoff lists and low-power physics tests were completed, and the reactor power was increased to the heating range. The main generator was synchronized at 4:10 a.m. on July 14 and full power was reached at 7:00 p.m. Operation at essentially full power [40 MW(e)] continued through the remainder of the month. The primary system activity peaked at 0.55 Ci immediately after the rise to power but then decreased to less than 0.30 Ci as full-power operation continued.

During the rise to full power, it was noted that the difference in the reactor outlet temperature as measured by the loop 1 and loop 2 temperature detectors was increasing. When the plant reached full power, this difference was 40°F with a 15°F spread in the temperature of the steam leaving the two steam generators.

Further investigation disclosed that the lower outlet temperature was on the side of the reactor containing an additional type III element (E14-01). This element was very close to the loop 2 outlet line and was estimated to cause a flux depression in this area of approximately 4%.

TABLE 7-1
TEST ELEMENTS INSTALLED CORE 2

Element	Core Location	Description and Fuel Type
PTE-2	A07-07	Proof test element for FSV - rods
FBTE-1	A02-01	Fuel bed test element for LHTGRs - bonded rods
FBTE-2	D06-01	Fuel bed test element for LHTGRs - bonded rods
FBTE-3	B14-08	Fuel bed test element for LHTGRs - bonded rods
FBTE-4	F06-01	Fuel bed test element for LHTGRs - bonded rods
FBTE-5	D09-04	Fuel bed test element for LHTGRs - blended bed
FBTE-6	F14-08	Fuel bed test element for LHTGRs - blended bed
FTE-1	A14-08	Fuel test element for LHTGRs - blended bed
FTE-2	A11-11	Fuel test element for LHTGRs - blended bed
FTE-5	D14-08	Fuel test element for LHTGRs - bonded rods
RTE-2	F07-06	Recycle test element for ORNL - 1/2 beds, 1/2 rods
RTE-4	B10-06	Recycle test element for ORNL - 1/2 beds, 1/2 rods
RTE-5	C10-06	Recycle test element for ORNL - bonded rods
RTE-6	D10-06	Recycle test element for ORNL - bonded rods
RTE-7	E10-06	Recycle test element for ORNL - bonded rods
RTE-8	F10-06	Recycle test element for ORNL - 1/6 beds, 5/6 rods
FPTE-1	E14-08	Fuel pin test element for UKAEA - fuel pins

The nonsymmetrical loading and flux shift were planned and resulted from the loading adjustment made by changing eight type III fuel elements with eight type II fuel elements. To compensate for this shift, the rods in the group controlling the reactor power were offset by approximately 10 in. to limit the differential reactor outlet temperature to 25°F and the differential steam temperatures at the turbine to 10°F. The conditions were well within the turbine manufacturer's limit of 25°F differential steam temperature between the main steam leads.

Helium impurities varied during preparations for power operations. Peaks of 20 vpm O₂ and 200 vpm N₂ were observed. The maximum impurities during the approach to full power and for approximately 2 days after full-power operation was attained were as follows:

	<u>ppm by Volume</u>
H ₂ O	9
CO ₂	7.0
O ₂ -Ar	1.4
N ₂	9
CH ₄	2.4
CO	9.9

Subsequent to the rise to power, the impurity levels were all less than 2.0 vpm and H₂O was less than 1 vpm.

The main loop activity levels, as indicated by the Cary ion chamber, gradually increased with approach to full power to approximately 0.55 Ci and then gradually decreased to a value of approximately 0.24 Ci. Toward the end of July, the activity increased to approximately 0.29 Ci, the bulk of which was due to a slight increase in power. The fuel element purge activity was 1 to 2 Ci/lb. The R/B ratios were nominally 10⁻⁵ for short-lived isotopes and 10⁻⁴ for long-lived isotopes. These values are comparable to beginning-of-life of Core 1.

The reactor remained at essentially full power until September 18, when it was shut down to repair a leaking dump valve bonnet on the No. 2 steam drum. This scheduled shutdown terminated the longest sustained power operation of the reactor at 66 days. During this exposure of Core 2, the system activity remained at approximately 0.3 Ci, showing excellent fuel performance. The fuel element purge activity also remained constant at about 1 Ci/lb.

The average release fractions for the main loop for August were as follows:

<u>Isotope</u>	<u>R/B</u>
Kr-85m	2.7×10^{-4}
Kr-88	1.1×10^{-4}
Kr-89	6.7×10^{-6}
Xe-133	9.0×10^{-4}
Xe-135	1.8×10^{-4}
Xe-138	2.3×10^{-5}

After the plant was restarted on September 23, a high steam generator bottom head temperature was noted, and power operation was limited to about 78%. It was concluded that the bottom head cooling unit damper on the No. 1 steam generator was malfunctioning. On October 1 the plant was shut down to effect repairs on this unit and to correct an increasing helium leak that was attributed to an incompletely closed valve in the helium purification system. This maintenance was readily accomplished and the plant was returned to full power on October 4, where it remained until October 26.

During electrical testing on October 26, the 220-kV line breaker was inadvertently opened. The turbine-generator tripped on overspeed because of the rapid loss of load. The turbine-generator set went into the coast-down mode and house-load shedding and transfer occurred as programmed by the electrical protection system. Coastdown initiation indirectly produced

a reactor scram and containment isolation. The motor-driven boiler feed pump was shut down and both steam generators were supplied water from the turbine-driven boiler feed pump. However, because of the changeover and the drum level control action in the transient, the No. 2 loop automatically isolated on low drum level.

This incident provided an excellent example of the ability of the plant to withstand significant transients and to recover quickly without damage. Within 3 hr of the trip, the plant was resynchronized to the line and full-power operation was recommenced. The plant parameters were all well within normal range before and after the outage transients, indicating no unusual occurrences in the reactor system. In particular, the main coolant system activity stayed at or below 0.5 Ci, with Core 2 having operated for 182 EFPD. This clearly indicated the improved Core 2 fuel element performance compared to Core 1 at a similar point in its operating history.

Except for two brief outages, the reactor remained at full power through the end of the year. A brief power reduction was made on November 27, in anticipation of a reactor shutdown because of an apparent drop in the rod hydraulic reservoir level. It was found, however, that no hydraulic leak existed, and the reactor was returned to full power on November 28. In December a 19-hr shutdown was performed to permit testing of the main loop moisture monitors. Two of the hygrometer elements were replaced, and the plant continued normal power operation.

The data listed below have been plotted as a function of reactor operating time for 1970. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 7-2
Primary circuit nitrogen	Fig. 7-3
Primary circuit oxygen and argon	Fig. 7-4
Primary circuit carbon monoxide	Fig. 7-5
Primary circuit carbon dioxide	Fig. 7-6

Primary circuit methane	Fig. 7-7
Primary circuit core inlet and outlet temperatures	Fig. 7-8
Steam drum SiO_2	Fig. 7-10
Steam drum Cl ion	Fig. 7-11
Steam drum pH	Fig. 7-12
Deaerator pH	Fig. 7-13
Deaerator hydrazine	Fig. 7-14
Deaerator oxygen	Fig. 7-15

The primary circuit helium pressure, although not recorded on the 1970 Unit Logs, was held at 335 psig during period of reactor operation.

Typical operating data for 1970 are shown in Table 7-2. The significant 1970 operating data reported to the USAEC are shown in Table 7-3.

TABLE 7-2
1970 TYPICAL OPERATING DATA

	8/20	10/15	12/10
Reason for inclusion	Typical operation	Typical operation	Typical operation
Reactor power level, %	96.6	97	97
Reactor primary flow, lb/hr x 10 ⁻³			
Loop 1	256	251	244
Loop 2	263	259	241
Core inlet temperature, °F			
Loop 1	645	650	650
Loop 2	648	NA(a)	NA
Core outlet temperature, °F			
Loop 1	1233	1256	1285
Loop 2	1221	1248	1280
Hot valve temperature, °F	1238	1270	1308
Core differential pressure, psi	3.35	3.29	2.96
Core purge flow, lb/hr	839	844	781
Total purification system flow (includes steam generator purge flow), lb/hr	1050	1045	975
Steam drum pressure, psig			
No. 1	1530	1525	1540
No. 2	1530	1535	1550
Boiler feed pump suction pressure, psig	135	138	134
Boiler feed pump discharge pressure, psig			
No. 1	1840	1825	1840
No. 2	1870	1855	1870
Boiler feed pump suction temperature, °F	342	343	344
Final feedwater temperature (from 4A and 4B heaters), °F			
4A	421	423	421
4B	413	415	413
Bleed steam pressure, (to fourth heater), psig	342	350	349
Bleed steam temperature (to fourth heater), °F	656	664	678
Throttle pressure, psig	1450	1450	1450
Throttle temperature, °F	955	960	970

(a) NA = not available.

TABLE 7-3
SIGNIFICANT PLANT DATA REPORTED BY THE
PHILADELPHIA ELECTRIC COMPANY TO THE USAEC IN 1970

	7/70	8/70	9/70	10/70	11/70	12/70
Gross generated power, MW(e)	41	41	41	43	43.2	43.1
Steam flow, lb/hr x 10 ⁻³	360	361	358	380	375	373
Throttle pressure, psig	NR ^(a)	NR	1450	1450	1450	1450
Throttle temperature, °F	965	960	965	980	990	985
Condenser vacuum, in. Hg abs.	1.7	1.95	1.57	1.09	0.56	0.55
Reactor inlet temperature, °F	646	648	653	648	649	650
Reactor outlet temperature, °F	1226	1228	1230	1255	1289	1285
Total He flow, lb/hr x 10 ⁻³	510	515	511	512	485	485
Reactor pressure, psig	333	333	330	332	335	335
Reactor power, %	94	94	93.5	98	98	97.5
Reactor power, MW(t)	108.1	108.1	107.5	113	113	112

(a) NR = not reported.

Fig. 7-1. Peach Bottom operating history - 1970

Fig. 7-2. Main loop activity - total inventory-1970

Fig. 7-3. Main loop N₂-1970

Fig. 7-4. Main loop O_2 Ar-1970

Fig. 7-5. Main loop CO-1970

Fig. 7-6. Main loop CO₂-1970

Fig. 7-7. Main loop CH₄-1970

Fig. 7-8. Main loop flow-1970

Fig. 7-9. Main loop core inlet and outlet temperatures-1970

Fig. 7-10. No. 1 steam drum SiO_2 -1970

Fig. 7-11. No. 1 steam drum CI-1970

Fig. 7-12. No. 1 steam drum pH-1970

Fig. 7-13. Deaerator pH-1970

Fig. 7-14. Deaerator hydrazine-1970

Fig. 7-15. Deaerator O₂-1970

8. 1971 OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1971 is shown in Fig. 8-1.

The reactor operated at full power until January 18, when a shutdown occurred and the turbine-generator was removed from service to accommodate work on the main transmission lines and to perform safety tests on the moisture monitoring system. Four main loop moisture monitor cells were replaced as a preventative maintenance measure and to provide additional laboratory cells for test purposes.

On January 25 a full-load loss occurred, which was caused by transmission-line switching that involved the nearby Conowingo hydroelectric plant. With both plants separated from their load, frequency and voltage swings occurred that precipitated an overspeed trip of the main Peach Bottom turbine-generator set. A loop isolation and reactor setback occurred, and the operator manually scrambled the plant as an added precaution due to the loss of system power and frequency regulation. Within 5 hr the plant was resynchronized and put back into power operation.

Early on January 26 another trip occurred. During instrument maintenance work, an operator error caused the turbine-driven boiler feed pump to trip, which also resulted in a loop isolation and reactor setback. The plant was again operating at power within 7 hr. Power operation was continued until February 1, when the output was reduced to house load to allow work on the main transmission line. The reactor was returned to full power and operated at that point until February 12, when a boiler feed pump turbine exhaust valve solenoid failed. The station output was reduced to 19 MW(e) and the valve was replaced. The station was returned to full power on the same day. On February 25 the reactor was shut down to change

two moisture monitor hygrometer cells, which had indicated slower than normal response to routine injection of wet test gas. The reactor was returned to full power on the same day.

The reported main loop activity levels for February were as follows:

<u>Isotope</u>	<u>R/B</u>
Kr-85m	2.9×10^{-4}
Kr-87	8.0×10^{-5}
Kr-88	1.2×10^{-4}
Kr-89	5.5×10^{-6}
Xe-133	8.2×10^{-4}
Xe-135	2.0×10^{-4}
Xe-138	2.1×10^{-5}

The main loop radioactive inventory at full power varied between 0.36 and 0.5 Ci, depending on LN₂ trap operation.

The plant operated at near full power until March 24, at which time the plant protective system initiated a reactor scram due to control rod low hydraulic oil pressure. The scram was caused by failure of the high-pressure hydraulic hose to the 5A control rod. The failed hose was replaced and following checkout tests, the plant was started up on March 28.

The plant operated at near full power until April 7, at which time a shutdown was initiated to repair a water leak in the No. 1 feedwater bypass valve. In addition, two more hygrometer cells in the main loop moisture monitoring system were replaced due to slower than normal response to routine injection of wet test gas. Following repairs, the plant was returned to service on April 8.

On April 22 a plant shutdown was initiated to accommodate the biennial inspection of both steam generators and the Core 2 300-EFPD inspection of core components. The station had generated a total of 252 EFPD on Core 2 to this date.

Fuel handling operations commenced on May 17 with the inspection of in-core components. This included examination and photographing of fuel elements, reflector elements, control rod guide tubes, and the reactor plenum all-metal insulation. Five fuel elements were temporarily removed from the core for gamma scanning as part of the GA fuel development program. The results of all examinations indicated that the core was performing as designed, and no deterioration was observed.

The removal and canning of fuel elements started on May 24. Ten elements were removed (four standard fuel elements and six test elements). The test elements removed and installed during the 252-EFPD shutdown are listed in Table 8-1. A circumferential crack approximately 1/4 in. wide was noted in the bottom connector of PTE-2. This element was placed in the core in January of 1969 and had undergone 402 EFPDs of irradiation. It was later determined that the crack was caused by a lack of thermal expansion clearance for an internal metal part.

The tool for removal of broken fuel elements was used for the removal of FPTE-1 as a precautionary measure, due to the unique characteristics of this element. (The nonpurged nature of the element prevented checking its structural integrity). This element, which contained low enrichment oxide fuel, was irradiated 252 EFPD.

The reflector elements containing Charpy specimens were removed. One element (F18-08) was canned intact for the postirradiation examination required by the technical specifications. The other element (F17-16) was removed only to recover the Charpy specimens. In this case, the lower end was cut off and shipped in a special shipping cask to GA for nil-ductility-temperature testing. On June 10 two trays of Charpy specimens were removed from the access port plug and also shipped to GA for nil-ductility-temperature testing.

The required technical specification inspection of the No. 1 steam generator was completed during May, and the inspection of the No. 2 steam

TABLE 8-1
TEST ELEMENTS REMOVED AND INSTALLED DURING THE CORE 2 252-EFPD SHUTDOWN

Element	Core Position	Description	Action
PTE-2	A07-07	Proof test element for FSV	Removed
FBTE-4	F06-01	Fuel bed test element for LHTGRs	Removed
FBTE-6	F14-08	Fuel bed test element for LHTGRs	Removed
FTE-1	A14-08	Fuel test element for LHTGRs	Removed
RTE-7	E10-06	Recycle test element for ORNL	Removed
FPTE-1	E14-08	Fuel pin test element for UKAEA	Removed
FTE-3	A03-03	Fuel test element for LHTGRs	Installed
FTE-4	A14-08	Fuel test element for LHTGRs	Installed
FTE-6	C02-01	Fuel test element for LHTGRs	Installed
FTE-7	F14-08	Fuel test element for LHTGRs	Installed
FTE-8	D14-08	Fuel test element for LHTGRs	Installed
FTE-9	E02-01	Fuel test element for LHTGRs	Installed
FTE-10	B06-01	Proof test element for FSV	Installed
FTE-11	E10-06	Recycle test element for ORNL	Installed
FTE-12	F06-01	Fuel test element for LHTGRs	Installed
FPTE-3	E14-08	Fuel pin test element for UKAEA	Installed

generator was completed during June. The inspections required dye penetrant testing of the tube sheets and radiographic examination of the transition welds in the superheater outlet barrel sections. The steam generators were also hydrostatically tested. No defects that would cause rejection of the steam generators were noted.

During this shutdown, maintenance work was performed on eight of the emergency shutdown rod batteries, and eleven new batteries were installed. Also during the shutdown the fuel test element individual purge sample system was modified to eliminate cross flow between sample lines.

Prior to reactor startup, dust was removed from the No. 1 and No. 2 main coolant loop dust collectors. Samples were sent to ORNL for examination as part of their gas-cooled reactor study program.

Upon reactor startup on July 11, the reactor primary circuit activity was again established at approximately 0.3 Ci. The core at this time contained ten new fuel test elements. Except for five short shutdowns (≤ 87 hr each), the reactor operated fairly smoothly for the next 3.7 months. (The shutdowns and their causes are shown in Fig. 8-1.) The primary circuit activity remained at 0.5 Ci during this period. On October 31 the reactor was shut down to replace eight emergency shutdown batteries with new batteries. The reactor had accumulated 342 EFPD of operation at this time.

On November 21 the plant was returned to power. The main loop activity remained low at 0.5 Ci. The reactor continued to operate efficiently at full power through the end of the year, with the exception of a brief 1-day outage for testing of the main loop moisture detection safety system in accordance with technical specification requirements. During this shutdown on December 30, the moisture detectors in the plant protective system were tested for response time utilizing helium gas containing moisture. Because two of the six moisture detectors showed response times longer than 8 sec, all six detectors were replaced.

During the last half of 1971, a tritium survey of the Peach Bottom plant was conducted by GA personnel at the site. The main objectives of the survey were to determine (1) a material balance for tritium in the reactor system and (2) the principal source or sources of tritium in the waste disposal systems. Investigations performed to determine a material balance were as follows:

1. Fuel examinations at GA were made to determine tritium retention in the fuel.
2. Fuel element purge samples were analyzed for tritium to determine if gross amounts of tritium were being added to the main loop via this route.
3. Main coolant tritium concentrations were calculated assuming an annual input of 40 Ci via He-3 activation and a 200-lb/hr cleanup rate of the 1000-lb helium inventory.
4. The hydrazine concentrations in the secondary coolant system were varied to determine the effect on the possible tritium migration from the primary coolant through the steam generator tubes to the secondary coolant.

Tests performed to determine the principal sources of tritium in the waste disposal systems were as follows:

1. Calculations were made using maximum probable leak rates from the main coolant system and maximum probable specific tritium activities of the main coolant to determine input of tritium into the containment from this source.
2. A check was made of all piping associated with the waste disposal drain tank system which accepts tritium wastes from the auxiliary helium systems.

3. The oxidizer outlet sample was checked for tritium concentration to determine if a leak at this point would account for tritium in the containment.
4. The dehydrator bed was checked for any leaks which might communicate directly to the atmosphere and account for tritium in the containment.

During the survey, samples were collected from the following:

1. Main coolant system
2. Fuel element purge system
3. Sample points in purification system
4. Containment atmosphere
5. Air room atmosphere
6. Stack effluents
7. Outside air (to determine background)
8. Kr-85 and N₂-Ar holdup tanks
9. Steam system
10. Air ejector
11. Waste monitor tanks
12. Waste disposal drain tanks
13. Condensate system
14. Caustic and water scrubber

The following preliminary conclusions were made:

1. A direct leak from the main coolant to the containment would not account for the tritium concentration found in the containment.
2. The amount of tritium formed by He-3 activation could account for the total amount of tritium which was lost via waste disposal systems.

3. Tritium release to the waste disposal systems originated from areas of high tritium concentrations, e.g., waste disposal drain tank, dehydrator bed, water knockout pots, etc.
4. Tritium levels in the liquid waste disposal system, the secondary coolant, the main coolant, the containment atmosphere, and the air room atmosphere remained fairly constant; i.e., no sharp increases or decreases were noted.
5. The tritium concentration in the main coolant could be accounted for by He-3 activation.
6. No increase in tritium concentration of the fuel element purge over the tritium concentration of the main coolant was observed.

No conclusive results had been obtained from studies relating to retention of tritium by the fuel or studies relating tritium migration through steam generator tube walls to hydrazine concentrations in the secondary coolant.

The data listed below have been plotted as a function of reactor operating time for 1971. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 8-2
Primary circuit nitrogen	Fig. 8-3
Primary circuit oxygen and argon	Fig. 8-4
Primary circuit carbon monoxide	Fig. 8-5
Primary circuit carbon dioxide	Fig. 8-6
Primary circuit methane	Fig. 8-7
Primary circuit hydrogen	Fig. 8-8
Primary circuit flow and pressure	Fig. 8-9
Primary circuit core inlet and outlet temperatures	Fig. 8-10
Steam drum SiO ₂	Fig. 8-11

Steam drum Cl ion	Fig. 8-12
Steam drum pH	Fig. 8-13
Deaerator pH	Fig. 8-14
Deaerator hydrazine	Fig. 8-15
Deaerator oxygen	Fig. 8-16

Typical operating data for 1971 are shown in Table 8-2. The significant 1971 plant operating data reported to the USAEC through April are shown in Table 8-3. Reporting of these data to the USAEC beyond April 1971 was discontinued.

TABLE 8-2
1971 TYPICAL OPERATING DATA

	1/10	3/10	8/15	9/25	12/15
Reason for inclusion	Typical operation	Typical operation	Typical operation	Typical operation	Typical operation
Reactor power level, %	98	94.3	89.7	90.6	96.2
Reactor primary flow, lb/hr x 10 ⁻³					
Loop 1	259	255	251	249	249
Loop 2	254	251	248	249	251
Core inlet temperature, °F					
Loop 1	654	648	648	647	651
Loop 2	NA(a)	648	650	652	657
Core outlet temperature, °F					
Loop 1	1260	1237	1218	1225	1263
Loop 2	1259	1237	1217	1225	1262
Hot valve temperature, °F	1273	1251	1227	1230	1268
Core differential pressure, psi	3.25	3.18	3.03	3.06	3.08
Core purge flow, lb/hr	850	850	970	890	865
Total purification system flow (includes steam generator purge flow), lb/hr	1010	885	1010	1020	990
Steam drum pressure, psig					
No. 1	1525	1470	1495	1475	1495
No. 2	1515	1455	1465	1445	1500
Boiler feed pump suction pressure, psig	136	134	132	132	138
Boiler feed pump discharge pressure, psig					
No. 1	1850	1855	1900	1860	1820
No. 2	1865	1875	1860	1890	1875
Boiler feed pump suction temperature, °F	344	344	338	339	345
Final feedwater temperature (from 4A and 4B heaters), °F					
4A	427	423	422	421	426
4B	417	412	417	417	421
Bleed steam pressure (to fourth heater), psig	350	334	316	315	338
Bleed steam temperature (to fourth heater), °F	660	662	644	651	668
Throttle pressure, psig	1455	1405	1435	1410	1450
Throttle temperature, °F	955	945	927	932	955

(a) NA = not available.

TABLE 8-3
SIGNIFICANT PLANT DATA REPORTED BY THE
PHILADELPHIA ELECTRIC COMPANY TO THE USAEC IN 1971

	1/71	2/71	3/71	4/71 ^(a)
Gross generated power, MW(e)	43	43	41	42
Steam flow, lb/hr x 10 ⁻³	376	374	365	364
Throttle pressure, psig	1450	1450	1410	1410
Throttle temperature, °F	978	980	965	950
Condenser vacuum, in. Hg abs.	0.55	0.52	0.56	0.70
Reactor inlet temperature, °F	655	654	649	651
Reactor outlet temperature, °F	1255	1262	1235	1240
Total He flow, lb/hr x 10 ⁻³	510	505	501	512
Reactor pressure, psig	335	336	335	335
Reactor power, %	97	97	93	95
Reactor power, MW(t)	111.5	111.5	107	109.3

(a) Last month in which these data were reported to USAEC in Philadelphia Electric Company Monthly Operations Reports.

Fig. 8-1. Peach Bottom operating history - 1971

Fig. 8-2. Main loop activity - total inventory -1971

Fig. 8-3. Main loop N₂-1971

Fig. 8-4. Main loop O₂Ar-1971

Fig. 8-5. Main loop CO-1971

Fig. 8-6. Main loop CO₂-1971

Fig. 8-7. Main loop CH₄-1971

Fig. 8-8. Main loop H₂-1971

*2-9-71 NEW UNIT LOG
 FORMAT-REACTOR LOOP
 PRESSURE NOW RECORDED

Fig. 8-9. Main loop flow and pressure-1971

Fig. 8-10. Main loop core inlet and outlet temperatures-1971

Fig. 8-11. No. 1 steam drum SiO₂-1971

Fig. 8-12. No. 1 steam drum CI-1971

Fig. 8-13. No. 1 steam drum pH-1971

Fig. 8-14. Deaerator pH-1971

Fig. 8-15. Deaerator hydrazine-1971

Fig. 8-16. Deaerator O₂-1971

9. 1972 POWER OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1972 is shown in Fig. 9-1.

On January 7 the reactor was shut down to perform required maintenance on a control rod and to investigate the response characteristics of the primary system moisture monitors. The reactor had accumulated 384 EFPD of operation at that time.

In late February, a decision was made to extend the outage to accommodate recanning of a leaking spent fuel can stored in the containment vessel. Further delay was encountered when another spent fuel can developed a leak on March 25 and also required recanning. These operations proved to be time consuming and were not completed until April 21. However, since the first fuel unloading operations in early 1968, only three cans had developed leaks and thus required recanning.

On March 14 the containment vessel integrated leak rate test was completed. The measured leak rate was approximately one-half the maximum allowable leak rate permitted by the technical specifications.

Because of the extended outage, loading of six additional fuel test elements (phase 3) and unloading of three test elements were accomplished during the last half of April.

The test elements installed and removed at this time are listed in Table 9-1. Following insertion of FTE-13, -14, -15, -16, and -17, Peach Bottom Unit 1 contained a total of 24 test elements.

TABLE 9-1
TEST ELEMENTS REMOVED AND INSTALLED AT 384-EFPD SHUTDOWN

Element	Description	Action
FTE-13	Plutonium fuel test	Installed
FTE-14, FTE-15	Large HTGR fuel test	Installed
FTE-16, FTE-17	FSV fuel proof test	Installed
FTE-18	Hobeg/KFA molded fuel body test	Installed
FTE-2	Large HTGR fuel test	Removed
FTE-3	Large HTGR fuel test	Removed
FTE-4	Recycle fuel test	Removed

Core 2 fuel element E06-01 had been removed from the core to permit loading of FTE-18 during the 385-EFPD shutdown. At the request of ORNL, this element was shipped to the Oak Ridge hot cell for postirradiation examination as a part of the ORNL HTGR metallic fission product release study program.

Two diffusion probes were installed in the No. 1 main helium loop (one downstream of the No. 1 hot valve and one downstream of the No. 1 steam generator) during the last week in April. The purpose of the probes was to sample the main loop helium to acquire information on plateout activity in the primary system for a program performed in cooperation with ORNL.

Several representatives from ORNL conducted a gamma scan analysis of the No. 1 main helium piping during the week of April 24.

In connection with the main loop moisture monitor safety system, an investigation was conducted of cell response and system response. The manufacturer of the electrolytic hygrometer cells, Beckman Corporation, as well as GA concluded the cell response investigation by recommending the "short-path length" cell because of its consistently shorter response time.

The system response test program uncovered two important aspects. First, the detector used to monitor the wet test gas was shown to have an instability which caused inconsistent readings. Thus, prior testing relating to cell response time was inaccurate because of uncertainties related to test gas moisture content. Secondly, a differential pressure regulator installed downstream of each cell was operating erratically when inlet pressures varied slightly during the injection of wet test gas. The maloperation caused erratic cell flow resulting in slow and inconsistent response times. System modifications were made to correct the problems as follows:

1. The differential pressure controllers were removed from all eight sample stations. Tests were conducted which showed that the

controllers were not necessary and their elimination resulted in a substantial decrease in response times.

2. A fourth channel was installed on each main loop sampling system to allow testing at power.
3. Solenoid valves were installed on the calibration gas inlet lines of each of the eight sample stations. This change minimized the possibility of an operating error resulting in one channel trip of the safety system during testing.
4. The absolute filters and filter holders located upstream of the moisture cell were removed from the sample system to improve response times.

With the reactor still shutdown, increased cell response times and plugging of cells were experienced in testing between May 30 and June 3. These unsatisfactory responses were traced to high levels of hydrocarbons (113 ppm on June 3) in the main loop helium. The hydrocarbons were apparently released to the system from the low temperature delay beds following warmup of the beds. Cleanup of the helium was accomplished prior to installation of eight new moisture cells.

A flow test was performed on the eight test elements that were located in core positions from which the element purge flow could be individually sampled. The test was performed to determine if the selector switch designation was correct. Each element was lifted several inches by the transfer machine while being sampled. Perturbations in the sample flow meter were observed if the correct element had been selected. Several of the designations were found to be incorrect and were relabeled accordingly.

On June 17 the plant was started up. At approximately 60% power, an increase in main loop moisture and hydrocarbons was detected. The reactor was subsequently shut down to allow the replacement of moisture cells in the main loop moisture detection system. This precautionary action was

taken since past evidence had indicated cell performance deteriorated when exposed to hydrocarbons (resulting from suspected lubricating oil ingress into the primary circuit) and moisture simultaneously.

On June 19 the plant was again started up at a reduced rate to minimize the formation of hydrocarbons in the primary helium system. The plant remained at power until June 22, at which time a shutdown was initiated following forecasts of flooding of the Susquehanna River due to hurricane Agnes. Preparations were undertaken to protect the plant from possible high flood waters. The river crested at 114.3 ft on June 24. The grade elevation of the reactor building is 115.0 ft, and no flooding was experienced.

The shutdown caused by hurricane Agnes was extended through July 6 to investigate the cause of the higher than normal hydrocarbon concentration in the primary helium system. Following startup, the plant operated at 93% power through July 26. During this time the hydrocarbon concentrations in the primary system varied considerably. On this date, the reactor power was reduced to 72% and the reactor outlet temperature was decreased to 1100°F in order to comply with the technical specification, which limits methane concentrations to 2 ppm in the primary system when the reactor outlet temperature exceeds 1100°F.

Continuing investigation of this problem revealed that a reduction in the purification system flow decreased the methane concentration in the primary system. The purification system flow was subsequently reduced to approximately 80% with no noticeable increase in primary system activity. With the reduced purification system flow, the plant was returned to full power operation on several occasions to continue the investigation. Even with the reduced purification system flow, occasional spikes in the methane levels in the primary system occurred. Plant power levels and reactor outlet temperatures were again reduced while investigations continued. Subsequent investigations revealed traces of oil in the purification system piping downstream of the purified helium compressors. Several drain connections were installed, as well as a recirculation path for the oil back to the purified helium compressor suction. This minimized the leakage of

oil into the primary system and tests were again run at full-power levels. These tests showed some reduction in hydrocarbon concentrations but spikes continued to occur. The plant power was again reduced. Operation at the decreased power level and reactor outlet temperature continued during the remainder of the year.

The investigation of the pressure of hydrocarbons in the primary helium system indicated that the leakage of oil from the purification system may have been caused by deterioration of the oil removal separators and filters in the purification system. New internals for the various components were purchased. After receipt of these components the plant was shut down on November 25 to permit replacement of filter media.

Several outages occurred during the period from July 6 through November 25. A brief description of each is as follows:

1. On August 6 the plant was shut down to repair tube leaks on the 4B feedwater heater. Following repair, the plant was returned to power operation on August 9.
2. On August 26 the reactor was manually scrammed following a momentary loss of the 220-kV transmission line as a result of a lightning strike during an electrical storm. The turbine bypass and electromatic relief valves opened as designed in response to the steam pressure transient. When normal steam pressure was restored, two of the four electromatic relief valves failed to reseal. Following repairs to the relief valves, the plant was returned to power on August 27.
3. On September 27 a reactor scram from full power occurred. The scram was caused by a switching error while performing routine testing on the plant protective system. The plant was returned to full power later the same day.

4. A planned shutdown of the plant was initiated on November 25 for the purpose of performing maintenance on the purified helium compressor oil separator, demister, and oil removal filters.

During the November 25 shutdown, replacement of the filter media was accomplished. During this period additional problems were encountered with the control rod drives. First, three control rods failed to insert to rod lock during a turbine overspeed trip test, necessitating a report to the USAEC. This problem was traced to various accumulator malfunctions and repairs were made. During subsequent scram testing, it was noted that the scram time of control rod 3C was three times above average. Investigations revealed that the problem was somewhere within the linear accelerator section of the drive mechanism. Philadelphia Electric subsequently decided to run until 600 EFPD (the plant was currently at 500 EFPD) with this rod fully inserted and locked. Finally, during pressurization prior to startup in late December, ten oil leaks in the control rod drive hydraulic system were discovered when the housing level alarm detectors went off. Three of these proved to be significant leaks which were subsequently eliminated by Philadelphia Electric.

The plant was returned to power on January 12, 1973.

The data listed below have been plotted as a function of reactor operating time for 1972. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 9-2
Primary circuit nitrogen	Fig. 9-3
Primary circuit oxygen and argon	Fig. 9-4
Primary circuit carbon monoxide	Fig. 9-5
Primary circuit carbon dioxide	Fig. 9-6
Primary circuit methane	Fig. 9-7
Primary circuit hydrogen	Fig. 9-8
Primary circuit flow and pressure	Fig. 9-9

Primary circuit core inlet and outlet temperatures	Fig. 9-10
Steam drum SiO ₂	Fig. 9-11
Steam drum Cl ion	Fig. 9-12
Steam drum pH	Fig. 9-13
Deaerator pH	Fig. 9-14
Deaerator hydrazine	Fig. 9-15
Deaerator oxygen	Fig. 9-16

Typical operating data for 1972 are shown in Table 9-2. The reporting of typical plant operating data to the USAEC was discontinued.

TABLE 9-2
1972 TYPICAL OPERATING DATA

	7/9	7/18	7/30	9/4	11/12
Reason for inclusion	Startup	Typical operation	Typical operation	Typical operation	Typical operation
Reactor power level, %	88.8	92.7	73.9	79.3	72.7
Reactor primary flow, lb/hr x 10 ⁻³					
Loop 1	241	251	252	214	246
Loop 2	240	259	254	212	249
Core inlet temperature, °F					
Loop 1	646	650	630	632	627
Loop 2	651	657	635	637	634
Core outlet temperature, °F					
Loop 1	1232	1228	1095	1223	1093
Loop 2	1232	1229	1094	1223	1097
Hot valve temperature, °F	NA ^(a)	NA	NA	1228	<1100
Core differential pressure, psi	NA	NA	NA	2.63	2.71
Core purge flow, lb/hr	745	850	733	585	640
Total purification system flow (includes steam generator purge flow), lb/hr	1020	1030	1020	800	865
Steam drum pressure, psig					
No. 1	1475	1480	1465	1475	1470
No. 2	1435	1440	1420	1420	1425
Boiler feed pump suction pressure, psig	128	135	110	116	110
Boiler feed pump discharge pressure, psig					
No. 1	1875	1810	1830	1840	1850
No. 2	1875	1850	1830	1900	1850
Boiler feed pump suction temperature, °F	338	342	373	327	323
Final feedwater temperature (from 4A and 4B heaters), °F					
4A	420	425	404	409	399
4B	415	418	397	408	397
Bleed steam pressure (to fourth heater), psig	309	328	252	270	243
Bleed steam temperature (to fourth heater), °F	650	654	582	636	586
Throttle pressure, psig	1440	1440	1430	1440	1445
Throttle temperature, °F	937	935	880	935	887

(a) NA = not available.

Fig. 9-1. Peach Bottom operating history - 1972

Fig. 9-2. Main loop activity - total inventory-1972

Fig. 9-3. Main loop N₂-1972

Fig. 9-4. Main loop O_2 Ar-1972

Fig. 9-5. Main loop CO-1972

Fig. 9-6. Main loop CO₂-1972

Fig. 9-7. Main loop CH₄-1972

Fig. 9-8. Main loop H₂-1972

Fig. 9-9. Main loop flow and pressure-1972

Fig. 9-10. Main loop core inlet and outlet temperatures-1972

Fig. 9-11. No. 1 steam drum SiO_2 -1972

Fig. 9-12. No. 1 steam drum CI-1972

Fig. 9-13. No. 1 steam drum pH-1972

Fig. 9-14. Deaerator pH-1972

Fig. 9-15. Deaerator hydrazine-1972

Fig. 9-16. Deaerator O₂-1972

10. 1973 OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1973 is shown in Fig. 10-1.

Power operations were resumed on January 12. During the next 4 months, a power level of approximately 93% was maintained with the exception of the following shutdowns and load reductions.

1. On January 12 several hours after startup, the reactor was manually scrammed following a trip of the main generator breaker. Following repairs to the relay, the plant was returned to power on the same day.
2. A reactor setback and loop isolation were initiated by a low No. 1 steam generator drum level condition on February 10. The low drum level was the result of a reduction in feedwater delivery caused by an electrical malfunction of the feedwater valve controller. The plant was returned to power on February 10, following replacement of the defective component.
3. On February 26 the generator output was reduced to house load to accommodate maintenance on the outside transmission line. Full power was restored the next day.
4. On February 28 the reactor scrammed as a result of a switching error during performance of routine testing on the plant protective system. The operator inadvertently tripped the scram breakers on the "B" safety system logic while the "A" logic breakers were in the tripped condition, thus resulting in a reactor scram. The plant was returned to power after a 5-hr shutdown.

5. A scheduled shutdown was initiated on March 5 to repair a steam leak in the No. 2 feedwater bypass valve breech plug. The steam generators were cooled down to accommodate repair work. While the plant was shutdown, an oil leak developed on the 8C control rod drive hydraulic hose connection. The containment was subsequently de-inerted and the hose replaced. On March 7 the reactor was returned to power.
6. On March 17 the intermediate compressor motor bearing on the "B" low-temperature delay bed (LTDB) refrigeration unit failed, necessitating a shutdown of the unit. A normal reactor shutdown was performed to minimize the increase in primary system activity and chemical impurities. The startup of the "A" refrigeration unit was delayed until the bearings in the low-pressure compressor motor could be replaced. On March 18 the plant was returned to full power following restoration of normal LTDB temperatures.

The activity levels in the primary helium system were consistent with previous radiochemistry history. The primary system gaseous activity remained below 0.55 Ci during power operations and below 0.15 Ci while the reactor was shutdown. Total fuel element purge stream activity averaged 3 Ci/lb helium during power operations. This represented a slight increase from the previous levels; however, it was well below the design level of 10,000 Ci/lb. An increase in fuel element purge activity was expected as core end-of-life was approached.

Full-power operations continued until May 17 at which time power was varied over a 2-day period to accommodate control rod reactivity measurements.

On May 18 the plant was shut down for a scheduled 3-week outage to allow reswaging of six spent fuel cans in the storage pool. The shutdown was necessary because access to the canning machine in the containment vessel was required. During the outage, the six spent fuel cans were moved to the canning machine and reswaged. This outage also provided an opportunity to perform routine maintenance and surveillance testing. The primary

system was depressurized to accommodate maintenance on the control rod drives and liquid nitrogen traps, removal of the diffusion probes, and inspection of the No. 2 main helium compressor drives.

Following return to power on June 9, higher than normal hydrocarbon concentrations were detected in the primary system. It had been determined that the hydrocarbons were formed by the decomposition of oil carried over from the purified helium compressor into the primary system. On June 11 the "B" purified helium oil filter was placed in service, and the "A" filter was taken off-stream, since the latter filter appeared to be saturated. Within minutes after valving in the "B" filter, an increase in the primary system moisture was observed. An accelerated plant shutdown was immediately initiated, and the "B" oil filter was valved out of service. An investigation showed that the moisture originated from the "B" oil filter. The charcoal had been replaced in December 1972, and apparently the subsequent purging operation had not been of sufficient duration to remove all residual moisture from the charcoal. During the shutdown the "B" filter was purged and the primary system helium was cleaned up. The plant was returned to power, after a 21-hr shutdown, late on June 11.

On June 12 the reactor power was reduced to 75% in response to an increase in the primary system hydrocarbon concentration. The plant was returned to full power the following day. The brief hydrocarbon transients experienced shortly after the extended May-June outage appeared to result from oil accumulation in the primary and auxiliary helium systems during the shutdown while in the depressurized condition.

The plant remained at essentially full power for the remainder of June except for a brief power reduction on June 21. On this date, the generator output was reduced to house load to allow removal of the outside 220-kV transmission line from service for repair of its protective relaying control cables. The reactor power was maintained at 30% of full power during this period.

From June 21 until September 14, the plant operated continuously at approximately 94% power. The plant was shut down on September 14 for a scheduled outage after a continuous power run of 2278 hr. This exceeded the previous longest power run of 1602 hr. The outage was scheduled for the biennial inspection of both steam generators and the Core 2 600-EFPD inspection of reactor internal components as required by the technical specifications. When the reactor was shut down, a total of 701 EFPD had been accumulated on Core 2.

Core 2 fuel elements A16-15 and E11-07 were removed from the core and gamma scanned. Element A16-15 was returned to the same location. Element E11-07 was stored in the spent fuel pit. This element was ultimately sent to ORNL for a postirradiation examination in compliance with the technical specifications. The results of the gamma scanning indicated an increase in fuel compact axial length which was consistent with the predicted Core 2 compact behavior.

Two fuel elements (E11-07 and A16-15), two hexagonal reflectors (C17-03 and B18-08), the 55 control rod guide tubes, and the internal plenum shroud were inspected and photographed with the reactor viewing device. These components were found to be structurally sound, and no unusual surface indications were observed.

The following test elements were removed from core during the shutdown for examination at the GA or ORNL hot cell:

<u>Element</u>	<u>Description</u>
RTE-2	Recycle fuel test
FTE-4	Large HTGR fuel test
FTE-14	Large HTGR fuel test

The biennial inspection of the steam generators was also accomplished during the shutdown. The tube sheet area and the dissimilar welds in the superheater outlet barrel on both steam generators were inspected to comply with the requirements of the technical specifications. The dissimilar

welds were inspected by the dye penetrant technique and radiography. The superheater tube sheet was inspected by the dye penetrant method. No minute cracking or indications of rejectionable conditions were noted by either radiography or dye penetrant inspection on either steam generator. In addition, all of the studs which were removed from the channel head were inspected for fatigue cracks by Magnaflux Corporation. All tests showed satisfactory results.

Prior to sealing the steam generator, all U-tubes were flushed by blowing sponges through each tube using treated water. A satisfactory hydrostatic test was conducted at 1500 lb/in.² on each steam generator before returning them to service.

Preparations for returning the plant to power were in progress at the end of the year.

The last shipment of Core 1 spent fuel was sent to the USAEC storage facility at Idaho Falls, Idaho on July 9. On the same date, four hexagonal reflectors, seven Core 2 fuel elements, and one test element were removed from the spent fuel pool and loaded into a shipping cask for temporary storage. These components were transferred to the cask to accommodate draining and repairs to the spent fuel pit. All water was removed from the pool and released in accordance with established radioactive waste procedures and controls. The pool was refilled during October 1973, following repairs to the inner wall coating and storage rack. Subsequent leakage from the spent fuel pit was negligible.

The data listed below have been plotted as a function of reactor operating time for 1973. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 10-2
Primary circuit nitrogen	Fig. 10-3
Primary circuit oxygen and argon	Fig. 10-4

Primary circuit carbon monoxide	Fig. 10-5
Primary circuit carbon dioxide	Fig. 10-6
Primary circuit methane	Fig. 10-7
Primary circuit hydrogen	Fig. 10-8
Primary circuit flow and pressure	Fig. 10-9
Primary circuit core inlet and outlet temperatures	Fig. 10-10
Steam drum SiO_2	Fig. 10-11
Steam drum Cl ion	Fig. 10-12
Steam drum pH	Fig. 10-13
Deaerator pH	Fig. 10-14
Deaerator hydrazine	Fig. 10-15
Deaerator oxygen	Fig. 10-16

Typical operating data for 1973 are shown in Table 10-1.

TABLE 10-1
1973 TYPICAL OPERATING DATA

	2/15	8/18
Reason for inclusion	Typical operation	Typical operation
Reactor power level, %	94.6	93.9
Reactor primary flow, lb/hr x 10 ⁻³		
Loop 1	258	252
Loop 2	254	258
Core inlet temperature, °F		
Loop 1	651	636
Loop 2	656	643
Core outlet temperature, °F		
Loop 1	1238	1221
Loop 2	1238	1223
Hot valve temperature, °F	1240	1227
Core differential pressure, psi	3.11	2.88
Core purge flow, lb/hr	700	645
Total purification system flow (includes steam generator purge flow), lb/hr	800	805
Steam drum pressure, psig		
No. 1	1500	1375
No. 2	1465	1315
Boiler feed pump suction pressure, psig	136	136
Boiler feed pump discharge pressure, psig		
No. 1	1800	1830
No. 2	1870	1880
Boiler feed pump suction temperature, °F	345	340
Final feedwater temperature (from 4A and 4B heaters), °F		
4A	423	415
4B	421	411
Bleed steam pressure (to fourth heater), psig	330	330
Bleed steam temperature (to fourth heater), °F	661	666
Throttle pressure, psig	1455	1310
Throttle temperature, °F	950	937

Fig. 10-1. Peach Bottom operating history - 1973

Fig. 10-2. Main loop activity - total inventory-1973

10-10

Fig. 10-3. Main loop N₂-1973

Fig. 10-4. Main loop O_2 Ar-1973

Fig. 10-5. Main loop CO-1973

Fig. 10-6. Main loop CO₂-1973

Fig. 10-7. Main loop CH₄-1973

Fig. 10-8. Main loop H₂-1973

Fig. 10-9. Main loop flow and pressure-1973

Fig. 10-10. Main loop core inlet and outlet temperatures-1973

Fig. 10-11. No. 1 steam drum SiO_2 -1973

Fig. 10-12. No. 1 steam drum CI-1973

Fig. 10-13. No. 1 steam drum pH-1973

Fig. 10-14. Deaerator pH-1973

Fig. 10-15. Deaerator hydrazine-1973

Fig. 10-16. Deaerator O₂-1973

11. 1974 OPERATING HISTORY

A histogram of reactor power and accumulated total reactor energy for 1974 is shown in Fig. 11-1.

Power operations were resumed on January 5 following an outage for the biennial inspection of both steam generators and the 600-EFPD inspection of reactor internal components. The plant operated at 93% power until a forced shutdown on February 25. During the startup later the same day, attempts to close the No. 2 main helium compressor bypass valve were unsuccessful. The plant operated during the next 3 months at reduced power (80%) as a result of the lower main coolant helium flow conditions caused by the failure of the bypass valve control system. The following shutdown and load reductions were experienced during this period.

1. On February 25 a reactor setback and loop No. 2 isolation occurred as a result of a blocking error while attempting to repair a leaking fitting on a pressure switch that feeds the reactor protection system. All systems functioned as expected during the shutdown. Duration of the outage was 5 hr.

Following the isolation of loop No. 2, the bypass valve on the No. 2 main helium compressor was opened in accordance with normal procedures. During the startup later the same day, attempts at closing the bypass valve were unsuccessful. This resulted in a recycle flow through the bypass of approximately 15% of the compressor flow. The loop flow was therefore reduced by the same amount, limiting plant operation to approximately 80% of full power.

2. On April 5 the No. 2 boiler feed pump tripped. The pump was immediately restarted. The plant was reduced to 40% power for

several hours to remove the pump from service to investigate the cause of the tripping. The pump motor and its control system were found to be in good conditions, and the pump was returned to service.

3. On April 19 a scheduled power reduction was initiated to repair a leak on the No. 2 boiler feed pump recirculation valve. Following a return to 80% power on the same day, the main generator tripped due to actuation of the generator backup overcurrent relay. The reactor was subsequently scrammed. An investigation revealed that the relay actuation was due to a malfunction within the relay. The relay was replaced and the plant was returned to power late on April 19. The duration of the outage was 10 hr.
4. A scheduled shutdown was initiated on May 14 to repair a leak in the No. 1 feedwater control valve bypass pipe. The steam generators were cooled down to accommodate repair work. The leak was caused by a pin-hole defect in a welded connection. The reactor was returned to power, after a 38-hr shutdown, on May 15.
5. The reactor scrammed on May 28 due to low control rod oil pressure. An investigation following de-inertment of the containment revealed a leaking oil hose associated with the 3B control rod drive unit. All plant control systems functioned properly during the shutdown with the exception of the 2C control rod, which stopped 3 in. short of the fully inserted position. The 35 other control rods were fully inserted. The failure of this rod to insert the last 3 in. resulted in a loss of only 0.07% Δp in the reactor shutdown margin. An investigation showed the 2C rod failed to fully insert as a result of an oil leak across its accumulator piston, which filled the gas side with oil. Repairs were completed on the 2C accumulator and the rod was satisfactorily scram tested.

The outage was extended to accommodate routine maintenance and surveillance testing required by the technical specifications. The reactor was returned to power on June 29; however, a ground on the main generator exciter rotor prevented synchronization of the unit. The outage was therefore further extended until July 6 to correct the electrical problem with the generator exciter.

The gaseous activity in the primary helium system remained at very low levels (approximately 0.6 Ci) during power operations. The total fuel element purge stream activity remained unchanged at approximately 3 Ci/lb of helium flow.

The only deviation from previous radiochemistry conditions was an increase in the primary system plateout activity observed for June 1974 in comparison with that for the shutdown covering the fourth quarter of 1973. Radiation fields measured in the cavity housing the steam generators and the main coolant piping were generally higher than those measured in September 1973 by factors of 3 to 5. The general area was approximately 75 mR/hr and on contact with the exterior of the piping, readings ranged between 50 and 200 mR/hr. Analysis by gamma ray spectrometry showed the additional activity to be almost exclusively Cs-134 and Cs-137.

An investigation by GA indicated that this phenomenon was due to an increase in the diffusion rate of cesium through the BISO coating surrounding the fuel kernel. The diffusion rate is primarily a function of fuel temperature. During the power run prior to the June shutdown, fuel temperatures had been rising, with peak temperatures approaching 2700°F. It was believed that most of the cesium came from the fuel elements in the center region of the core where the fuel temperatures were the highest. The cesium diffused into the fuel element purge stream, jumped the gap between the compact and sleeve, and diffused through the sleeve into the main coolant system, where it plated out on the internal surfaces. Plateout activity levels through the end-of-life were well below design activity levels and, therefore, this condition did not compromise the

nuclear safety of the plant. Additional radiation detectors were installed to monitor cesium activity levels during power operations.

Power operations were resumed on July 6 following an outage for routine maintenance, surveillance testing, and maintenance of the main generator exciter rotor. With the exception of a brief shutdown to perform reactivity measurements in September, the plant operated between 55 and 82% power until it was shut down for decommissioning on October 31. Reduced power operations were conducted in July, August, and September in an effort to minimize cesium plateout in the primary system and to conserve fuel in order to extend the operating life. By early October, all control rods were fully withdrawn and power slowly decayed as the steam temperature was maintained at 905°F. Following plant shutdown on October 31, 1974 and check-out of the fuel handling machine, off-loading of the core commenced on November 22. Core 2 had accumulated 897.4 EFPDs when the plant was shut down, and the Unit 1 station had accumulated a total of 1349 EFPDs.

Radiation levels measured in the cavity housing the steam generator and the main coolant piping in November after final plant shutdown were approximately 20% higher than the level measured in June 1974. An increase had been expected due to the increased cesium plateout on the primary system internal surfaces. The increased plateout was a result of an increase in cesium diffusion through the BISO coating surrounding the fuel kernel and was expected as a result of the increased fuel temperatures experienced during the last 200 EFPD of Core 2. The general area was approximately 100 mR/hr and on contact with the exterior of the piping, the readings ranged between 50 and 300 mR/hr. Analysis by gamma ray spectrometry showed the activity to be almost exclusively Cs-134 and Cs-137.

A total of 107 fuel elements were removed before the end of the year (total core was 804 fuel elements); of these, 44 were gamma scanned for GA prior to the canning operation. The gamma scan provided data on power and fuel burnup distribution suitable for code verification in commercial HTGR cores.

The data listed below have been plotted as a function of reactor operating time for 1974. For quick reference, the reactor power level has also been plotted in each figure.

Primary circuit activity (Cary meter)	Fig. 11-2
Primary circuit nitrogen	Fig. 11-3
Primary circuit oxygen and argon	Fig. 11-4
Primary circuit carbon monoxide	Fig. 11-5
Primary circuit carbon dioxide	Fig. 11-6
Primary circuit methane	Fig. 11-7
Primary circuit hydrogen	Fig. 11-8
Primary circuit flow and pressure	Fig. 11-9
Primary circuit core inlet and outlet temperatures	Fig. 11-10
Steam drum SiO_2	Fig. 11-11
Steam drum Cl ion	Fig. 11-12
Steam drum pH	Fig. 11-13
Deaerator pH	Fig. 11-14
Deaerator hydrazine	Fig. 11-15
Deaerator oxygen	Fig. 11-16

Typical operating data for 1974 are shown in Table 11-1.

TABLE 11-1
1974 TYPICAL OPERATING DATA

	1/13	5/20	7/22	8/22	10/15	10/31
Reason for inclusion	Typical operation	Typical operation	Low power operation	Typical operation	Coast-down	Coast-down
Reactor power level, %	94.1	79.1	55.2	81.5	67.2	59.7
Reactor primary flow, lb/hr x 10 ⁻³						
Loop 1	249	219	177	234	235	205
Loop 2	251	204	180	244	237	206
Core inlet temperature, °F						
Loop 1	640	633	604	634	621	612
Loop 2	644	638	607	640	633	615
Core outlet temperature, °F						
Loop 1	1237	1225	1094	1176	1074	1069
Loop 2	1237	1229	1094	1177	1081	1077
Hot valve temperature, °F	1248	1230	NA ^(a)	NA	NA	NA
Core differential pressure, psi	2.83	2.00	NA	NA	NA	NA
Core purge flow, lb/hr	628	585	655	690	685	670
Total purification system flow (includes steam generator purge flow), lb/hr	810	795	815	805	800	800
Steam drum pressure, psig						
No. 1	1480	1440	1445	1470	1455	1445
No. 2	1475	1410	1415	1440	1420	1415
Boiler feed pump suction pressure, psig	134	118	86	122	103	92
Boiler feed pump discharge pressure, psig						
No. 1	1810	1880	1920	1900	1925	1950
No. 2	1900	1943	1960	1910	1945	1960
Boiler feed pump suction temperature, °F	340	328	300	328	315	305
Final feedwater temperature (from 4A and 4B heaters), °F						
4A	412	404	371	405	388	376
4B	409	402	370	404	386	375
Bleed steam pressure (to fourth heater), psig	329	287	180	282	230	198
Bleed steam temperature (to fourth heater), °F	660	640	557	616	576	556
Throttle pressure, psig	1450	1410	1440	1440	1445	1445
Throttle temperature, °F	950	950	887	917	880	880

(a) NA = not available.

Fig. 11-1. Peach Bottom operating history - 1974

Fig. 11-2. Main loop activity - total inventory-1974

Fig. 11-3. Main loop N₂-1974

Fig. 11-4. Main loop O_2 Ar-1974

Fig. 11-5. Main loop CO-1974

Fig. 11-6. Main loop CO₂-1974

Fig. 11-7. Main loop CH₄-1974

Fig. 11-8. Main loop H₂-1974

Fig. 11-9. Main loop flow and pressure-1974

Fig. 11-10. Main loop core inlet and outlet temperatures-1974

Fig. 11-11. No. 1 steam drum SiO_2 -1974

Fig. 11-12. No. 1 steam drum CI-1974

Fig. 11-13. No. 1 steam drum pH-1974

Fig. 11-14. Deaerator pH-1974

Fig. 11-15. Deaerator hydrazine-1974

Fig. 11-16. Deaerator O₂-1974

12. 1975 OPERATING HISTORY

Defueling of the reactor, which had commenced on November 22, 1974, was completed on June 11, 1975. All 804 fuel elements were canned and stored in the spent fuel pool. During this operation, 74 fuel elements were gamma scanned by GA personnel immediately prior to canning. The data derived will provide power and fuel burnup distribution information suitable for verification of commercial HTGR core codes.

Three hexagonal reflector elements and one control rod guide sleeve and absorber were removed, gamma scanned, and canned. Several of these components will be examined at a hot cell facility as part of the GA end-of-life program.

Five elements were shipped in the single element Hallam shipping cask (two to General Atomic and three to ORNL) for examination in the hot cell facilities. The first Core 2 shipment in the 18-element cask left the site on June 24 for long-term storage in Idaho. Spent fuel shipping will continue using the two 18-element casks until 1977, the expected completion date.

Decommissioning of the facility involving the removal of contaminated equipment was started in late 1975. The initial decommissioning involved site work associated with the Peach Bottom End-of-Life Program, including:

1. External and internal gamma scanning of the primary circuit ducting.
2. Gamma scanning inside steam generator tubes.
3. Removal of main coolant ducting samples (by trepanning).

4. Removal of steam generator tubing samples (by side grinding).
5. Macroexamination of the loop 1 steam generator internals.

The objective of the gamma scanning was to map the distribution of cesium plateant activity around the primary circuit for subsequent comparison with PAD code predictions. The objective of the component removal was to provide samples for subsequent metallurgical, radiochemical, and tritium permeation tests under the End-of-Life Program and for blowdown tests sponsored by General Atomic to be conducted by CEA in France.

It is intended that data compilation reports, final analysis reports, and technical papers will be prepared through December 1977 to document all the work performed under the Peach Bottom End-of-Life Program.

13. GENERAL CHEMISTRY DATA

As part of a continuing R&D surveillance program and a requirement of the Peach Bottom Unit No. 1 Technical Operating Specifications (Ref. 3), the chemical and radiochemical impurities in both the primary and secondary coolant were monitored throughout the life of Core 1 and 2. The reactor coolant systems were composed of two loops consisting of steam generator, circulator, and interconnecting piping. Both loops were monitored for impurities during the life of the reactor.

The secondary coolant loops were sampled to provide data concerning chemical impurities such as Na, Si, Fe, Cu, Cl^- , total solids, and hydrazine. These data, given in Sections 4 through 11, were used to evaluate the steam generator material performance and possible corrosion of the water-side components. Additionally, the tritium levels in the secondary coolant were tested to determine release rates to the environment and to assure these levels were within the maximum permissible concentration (MPC) as set forth in 10CFR20.

The primary coolant system was monitored weekly for radiochemical impurities such as the noble gases krypton and xenon and periodically for metallic fission products such as cesium and iodine. The gaseous radio-nuclides were sampled and analyzed utilizing a gamma counting technique developed for this purpose (Ref. 4). The krypton isotopes Kr-85m, -87, -88, and -89 and xenon isotopes Xe-133, -135, and -138 were determined weekly. The specific activity of these isotopes was then used to verify that main coolant loop gross activity was maintained within the technical specification for operation of the plant. Metallic fission product plateout activity within the primary loop was determined by examination of plateout diffusion probes, designed for removal and analysis during periods of

reactor shutdown. Additional data were obtained for the chemical impurities H_2O , CO , CO_2 , O_2 , H_2 , and N_2 and for hydrocarbons such as methane. The concentration of these impurities was maintained below specified levels to prohibit interaction and corrosion of the reactor fuel, graphite, and metallic components.

In order to maintain the impurity levels, both chemical and radiochemical, below specified values, an external purification system was incorporated in the reactor coolant loop. The fuel elements were designed so that a fraction of the coolant gas flowing across the core was diverted internal to the element. Fission products were then purged through a purification train for delay or removal of the various impurities. The outlet of the purification system was sampled weekly to determine the efficiency and breakthrough condition of the train. Various sections of the system were capable of being regenerated in the event of a breakthrough.

R/B data for Core 1 operation are shown in Figs. 13-1 through 13-7. The main coolant system activity for the first 7 months of commercial operation rose slowly from about 0.1 to 0.4 Ci total gaseous fission products. This value was extremely low compared to a beginning-of-life prediction of 108 Ci. The low value was attributed to the effectiveness of the fuel element purging action since the release fractions were initially about as high as anticipated. However, just before the scheduled shutdown for the core element examination in January 1968, an increase in main loop activity to approximately 4 Ci was noted. Sample gas analysis found the short-lived fission product gases Kr-89 and Xe-138 in the main coolant system, where they had previously not been detected. It was concluded that one of the fuel elements was not purging properly and that these gases were thus leaking into the primary coolant.

The R/B data for Core 2 operation (Figs. 13-8 through 13-14) show a fairly constant release up to approximately 400 EFPDs. A general increase in R/B is evident from 400 EFPD to core end-of-life at 897 EFPDs. A

comparison of Core 1 and 2 R/B data indicates that the Core 2 release remained lower by more than an order of magnitude over the same time period of operation. At end-of-life for Core 2, the R/B was approaching 1×10^{-3} , still a factor of 5 below Core 1 release. However, the increase in R/B would indicate some fuel deterioration with burnup.

Near the end of Core 2 operation, several tests were performed to evaluate (1) the effect of power level on R/B and (2) the effect of power transients on R/B. Figures 13-15 through 13-19, show the effect of power on R/B and Figs. 13-20 through 13-24 show the effect of transients on R/B.

The krypton isotopes (Figs. 13-15 through 13-19) show a generally increasing R/B with increasing power, as would be expected since fuel temperatures are also increasing. However, the xenon data (Figs. 13-18 and 13-19) are relatively flat, indicating contribution at low power from another source.

The transient tests were conducted by increasing the reactor power from 30 to 75% as rapidly as possible. The data obtained for Kr-85m (Fig. 13-20) show a slight peaking just after reaching 73% power, with a gradual decrease to a steady-state R/B value approximately 25% lower than the peak value. A similar overrun was observed at Dragon; however, the Dragon data indicated much higher peaking.

The tritium behavior in the primary circuit was also determined during the latter part of reactor operation. Plant tritium surveys were conducted periodically to determine the source and operating concentration of tritium in the helium coolant (Ref. 5). These data were used to model tritium release from the fuel into the secondary coolant loops and ultimately release to the atmosphere via water loss from the plant (Ref. 6).

13.1. FISSION GAS RELEASE

Fission gas release R/B measurements were made throughout the operating life of the Peach Bottom HTGR. Fuel element purge gas samples were obtained weekly using a grab sample technique (Refs. 7, 8). Samples of purge gas were taken from the inlet of delay bed A-301 by backfilling a previously evacuated 2.7-cm³ glass vial and gamma counting. The gamma counting equipment consisted of a Na(I) crystal and 400-channel spectrometer. The data were then recorded on punched paper tape and forwarded to GA for processing on the SIGMA II computer-analyzer. The Peach Bottom operating staff was required to report only specific activities for the gaseous radionuclides; thus a computer program, Peach Bottom R/B, was written to process the data tapes at GA and to calculate R/B values.

Computer input information needed for the R/B calculations included sample date and time, reactor power level, helium flow, sample volume and pressure, count time, decay time, detector counting efficiency, isotope half-life, and various conversion factors. The data thus obtained are shown in Figs. 13-1 through 13-7 for reactor operations during the life of Core 1 and Figs. 13-8 through 13-14 for the life of Core 2. The isotopes monitored were Kr-85m, -87, -88, and -89 and Xe-133, -135, and -138.

The Core 1 Kr-85m R/B fission gas release values, given in Fig. 13-1, show a rapid change in a logarithmic fashion and then level off to a slightly increasing linear function versus effective full power days (EFPDs) of operation. A sharp increase in release occurred at approximately 50 EFPDs of operation. This increase continued for an additional 50 EFPDs, after which a gradual increase was apparent up to 200 to 250 EFPDs where the R/B appeared to have reached equilibrium release. The R/B observed is representative of totally failed fuel coatings as determined in GA laboratory fission gas release tests. Postirradiation examination of fuel compacts indicated 90 to 100% fuel failure. The other gaseous radionuclides, shown in Figs. 13-2 through 13-7 show a similar trend in release as a function of EFPD.

13.2. CHEMICAL IMPURITIES

13.2.1. General Data

The typical average operating ranges of the various measured chemistry parameters are shown in Table 13-1. The following chemistry data have been plotted for each year of reactor operation and are included in yearly operation plots (Sections 4 through 11) for all of the operating years of the Peach Bottom HTGR:

Main loop N ₂	1967-1974
Main loop O ₂ -Ar (oxygen and argon could not normally be chromatographically separated)	1967-1974
Main loop CO	1967-1974
Main loop CO ₂	1969-1974
Main loop CH ₄	
Main loop H ₂	1971-1974
Main loop hydrocarbons	1967
No. 1 steam drum SiO ₂	1967-1974
No. 1 steam drum Cl ion	1967-1974
Deaerator pH	1967-1974
Deaerator hydrazine (N ₂ H ₄)	1967-1974
Deaerator O ₂	1967-1974

On each of the plots, the typical causes are given for minor changes in each of the plotted parameters.

During the course of normal operations at the Peach Bottom HTGR, recurring perturbations in chemical impurities were observed by plant staff. The following is a brief description of the general causes, classed by the system affected:

1. Feedwater. This system consisted of the condenser, Powdex polishing unit, feed pumps, heater, and deaerator. Chemical

TABLE 13-1
TYPICAL CHEMISTRY PARAMETERS FOR THE PEACH BOTTOM HTGR

	Core 1	Core 2
Main loop		
Total gross activity	0.2 - 270 Ci	0.2 - 0.6 Ci
O ₂ -Ar	1 - 3 ppm	<0.1 ppm
N ₂	1.0 - 7.0 ppm	0.5 - 3.0 ppm
CH ₄	0.5 - 1.5 ppm	0.5 - 1.5 ppm
CO	0.5 - 1.5 ppm	0.5 - 1.0 ppm
CO ₂	<0.5 ppm	<0.5 ppm
H ₂	Not routinely monitored	5 - 20 ppm
Inlet temperature	650°F	635°F
Outlet temperature	1280°F	1235°F
Flow per loop	250,000 lb/hr	250,000 lb/hr
Pressure	335 psig	335 psig
Steam drum		
SiO ₂ ion	100 - 500 ppb	250 - 500 ppb
Cl ion	100 - 400 ppb	200 - 600 ppb
pH	8.8 - 9.2	8.8 - 9.2
Deaerator		
pH	9.4	9.4
Hydrazine	50 - 150 ppb	50 - 200 ppb
Dissolved O ₂	0 ppb	0 ppb
Chemical cleanup system		
Oxidizer exit CO	<0.1 ppm	<0.1 ppm
Oxidizer exit CO ₂	0.5 - 1.5 ppm	0.5 - 1.0 ppm
Oxidizer exit H ₂	Not routinely monitored	<0.1 ppm
Oxidizer exit H ₂ O	--	5 - 20 ppm
Dehydrator exit CO ₂	<0.1 ppm	<0.1 ppm
Dehydrator exit H ₂ O	<0.1 ppm	<0.1 ppm
LN ₂ purification system (all impurities except H ₂)	<0.1 ppm	<0.1 ppm

treatment consisted of hydrazine injection at the Powdex discharge in order to maintain a pH of 9.40 ± 0.05 in the deaerator. The hydrazine injection pump failed periodically, causing a reduction in deaerator pH. This in turn preceded a drop in steam drum pH and conductivity. Since the drum blowdown was manually controlled according to drum conductivity read in the control room, one result was an increase in SiO_2 and Cl because of a reduction in conductivity due to hydrazine.

2. Main steam drums. Due to the concentrating effect in the steam drum, a loss of the Powdex condensate polishing unit would show an increase in drum impurities (i.e., SiO_2 -Cl and conductivity). In addition, when maximum generating capacity was called for during peak load periods, the drum blowdowns were interrupted or cut off completely with a resultant increase in impurity concentrations.
3. Main loop chemical impurities. The steady-state concentration of chemical impurities in the main loop helium were affected by the loss of one of the cleanup systems due to regeneration or malfunction. The loss of the LN_2 system, which removed all impurities except H_2 , affected all impurities measured. The loss of the oxidizer bed in the chemical cleanup system resulted in increases in H_2 . At some time prior to their complete replacement (November 1972), the oil adsorber and removal systems downstream of the oil-lubricated purified helium compressors were saturated with oil. This oil vapor contamination of the main loop was most apparent during a startup following a shutdown of sufficient length to allow oil accumulation. Equilibrium H_2 and CH_4 were reduced after the replacement of oil adsorbers and filter elements.

4. Main loop activity. Changes in primary coolant activity were usually minor and due to a loss of the LN_2 cleanup system. Gross changes during Core 1 operations (June 1967 to October 1969) were due to failed sleeves on 78 fuel elements. The cracked sleeves allowed some of the purge helium (1 lb/hr/element) to find its way into the primary coolant helium, causing an eventual build up to 270 Ci gross activity.

13.2.2. Dehydrator and LN_2 Trap Regeneration

The dehydrators were monitored periodically at the effluent of the upstream bed for CO_2 ; a positive indication (0.4 ppm) dictated a regeneration. After regeneration the new bed was placed back in service as the downstream bed. A positive indication of H_2 (3 ppm) at the oxidizer bed effluent would require its regeneration. The upstream LN_2 trap (Kr-85 trap) effluent was monitored periodically, and a positive indication of N_2 (0.5 ppm) indicated when a regeneration was necessary. Normally when one LN_2 trap required regeneration, both traps were regenerated.

13.2.3. Hydrogen Detection and Hydrocarbons

In 1969 the more sensitive Loenco gas chromatograph replaced the less sensitive and less dependable Beckman chromatograph as the primary analytical instrument. In 1969 H_2 was routinely monitored with the Loenco chromatograph at various points. One result of monitoring the main loop for H_2 was the discovery of a high equilibrium H_2 concentration. In an attempt to connect the H_2 source to hydrocarbons and oil ingress, it was discovered that the Beckman hydrocarbon analyzer had not been responding for an indeterminate length of time. After its replacement, the hydrocarbon analyzer indicated spikes to 90 ppm during startups. Straight and branch chain hydrocarbons from methane to hexene were found, consistent with the oil ingress theory. H_2 analysis in conjunction with an operating hydrocarbon analyzer led to an inspection of the oil removal filters and

mist eliminator internals downstream of the oil-lubricated purified helium compressors. They were found saturated with oil and were replaced in November 1972. It is not known how long this problem existed prior to the installation of the Loenco chromatograph and a new hydrocarbon analyser.

13.2.4. Hydrogen Detection and Moisture Monitor Deterioration

The primary coolant high equilibrium H_2 concentration is suspected, but not conclusively proven, to be connected to the short life expectancy of Beckman P_2O_5 moisture detection cells used in the plant protective system. As a result of the discovery in 1972 that the cells had a finite and relatively short service time, a comprehensive response test program was developed at the plant with technical assistance from GA and Beckman. This test program created a large amount of data on the cell characteristics during power operations and it became possible to predict the cell service life according to amended plant technical specifications (cell response in 8 sec to an indicated 200 ppm H_2O in helium with 4000 ppm H_2O in helium as the test gas). After the test program had been established in excess of 1 year, the resulting data indicated a service life of 4 to 6 weeks during power operations with an equilibrium H_2 concentration of 5 to 10 ppm (after replacement of oil removal components). When GA conducted an unrelated H_2 injection to primary coolant test in October 1974 (Ref. 9), the primary coolant H_2 concentration was taken to the 120 ppm range and held there for several hours. Of interest is the fact that immediately following the test, there was an unanticipated (not predicted by the response test program) failure of two out of the six primary coolant moisture cells.

Technical input from Beckman indicated they felt that a trace level of H_2 in very dry (<1 ppm H_2O) helium dehydrated their electrolytic $P_2O_5 - 1.5 H_2O$ cell to the completely anhydrous state, thus changing cell response characteristics. The cell failures during the GA H_2 injection test appear to give some validity to their analysis of the Peach Bottom moisture cell problem.

13.2.5. PO₄ Injection to Steam Drums

Another mechanical failure with resulting chemical impurity ramifications was a small condenser tube leak in August 1973. This failure required a temporary PO₄ injection to the steam drums for chemical treatment while the plant was shutdown and the leak repaired. The PO₄ level was kept in the 2 to 10 ppm range.

13.3. TRITIUM DATA

Monitoring of tritium (H-3) levels in the Peach Bottom reactor was initiated in late 1970. A tritium survey was performed on the secondary coolant circuit to determine the concentration of tritium released through the secondary coolant water losses. Tritium concentrations were determined in the steam drum, superheat condensate, and feedwater makeup systems. Results indicated that about 2% of the calculated gas-borne tritium, produced by ternary fission and neutron activation of He-3, Li-6, Li-7, and B-10, in the primary coolant helium was released with the secondary coolant water losses. After verifying the existence of tritium in the secondary coolant, GA and Philadelphia Electric Company entered into a cooperative agreement to monitor the levels of tritium in both the primary and secondary coolant systems of the Peach Bottom HTGR.

Due to the nature of tritium decay (very low energy beta emission), special techniques for obtaining and analyzing solutions for tritium were developed by GA. Liquid solutions containing tritium were beta counted using a liquid scintillation detection system. The secondary coolant (water) was sampled and beta counted directly. However, the primary coolant (helium gas) had to be sampled using an oxidizer-sampler to convert the tritiated hydrogen gas to water, which was in turn liquified and scintillation counted. Tritium was found to exist as tritiated hydrogen (HT) and tritiated water (HTO). The moisture levels were extremely low in the primary circuit (<<1 ppm); therefore, it was found that the tritiated water levels were several orders of magnitude lower than the tritiated

hydrogen levels. In fact, calculations made in May 1974 by equating the ratios of H_2/H_2O and HT/HTO resulted in a water concentration in the primary circuit of 0.006 ppm.

The main objectives for conducting periodic tritium surveys at Peach Bottom were to determine (1) a material balance for tritium in the reactor system, and (2) the principal source or sources of tritium in the waste disposal systems. Investigations performed to determine a material balance were as follows:

1. Fuel examinations at GA were made to determine tritium retention in the fuel.
2. Fuel element purge samples were analyzed for tritium to determine if gross amounts of tritium were being added to the main loop via this route.
3. Main coolant tritium concentrations were calculated assuming an annual input of 40 Ci via He-3 activation and a 200 lb/hr cleanup rate of the 1000-lb helium inventory.
4. The hydrazine concentrations in the secondary coolant system were varied to determine the effect upon the possible tritium migration from the primary coolant through the steam generator tubes to the secondary coolant.

Tests performed to determine the principal sources of tritium in the waste disposal systems were as follows:

1. Calculations were made using maximum probable leak rates from the main coolant system and maximum probable specific tritium activities of the main coolant to determine input of tritium into containment from this source.

2. All piping associated with the waste disposal drain tank system which accepts tritium wastes from the auxiliary helium systems was checked.
3. The oxidizer outlet sample was checked for tritium concentration to determine if a leak at this point would account for the tritium in containment.
4. The dehydrator bed was checked for any leaks which might communicate directly to the atmosphere and account for the tritium in containment.

During the survey, samples were collected from the following systems:

1. Main coolant
2. Fuel element purge
3. Sample points in purification system
4. Containment atmosphere
5. Air room atmosphere
6. Stack effluents
7. Outside air (to determine background)
8. Kr-85 and N₂ Ar holdup tanks
9. Steam system
10. Air ejector
11. Waste monitor tanks
12. Waste disposal drain tanks
13. Condensate system
14. Caustic and water scrubber

Typical values found during tritium surveys are given in Table 13-2.

TABLE 13-2
TYPICAL LEVELS FOUND DURING TRITIUM SURVEYS AT PEACH BOTTOM

Sample Location	HT + HTO ($\mu\text{Ci}/\text{cm}^3$)	HTO ($\mu\text{Ci}/\text{ml}$)
Fuel element purge	5×10^{-5}	
Containment atmosphere	5×10^{-7}	
Air room atmosphere	ND ^(a)	
Outside air	ND	
Kr-85 holdup tanks	1×10^{-2}	
N ₂ -Ar holdup tanks	6×10^{-3}	
Deaerator		2×10^{-4}
Condensate		2×10^{-4}
Steam drum No. 1		2×10^{-4}
Steam drum No. 2		2×10^{-4}
Air ejector	2×10^{-7}	
Waste disposal drain tank		400
Main coolant loop No. 1	2×10^{-5}	
Main coolant loop No. 2	2×10^{-5}	

(a) ND = none detected; levels were at or below background for the instrumentation.

Plant tritium surveys were performed annually during the time period 1970 to 1974. The following conclusions were made concerning the source and levels of tritium in the reactor during this time period:

1. A direct leak from the main coolant to containment did not account for the tritium concentration found in the containment.
2. The amount of tritium formed by He-3 activation can account for nearly all of the tritium which was lost via waste disposal systems.
3. The method of tritium release to the waste disposal systems originated from an area of high tritium concentration, e.g., waste disposal drain tank, dehydrator bed, water knock-out pots, etc.
4. The tritium levels in the liquid waste disposal system, the secondary coolant, the main coolant, the containment atmosphere, and air room atmosphere remained fairly constant; i.e., no sharp increases or decreases were noted.
5. The tritium concentration the main coolant can be accounted for by He-3 activation.
6. No increase in tritium concentration of the fuel element purge over the tritium concentration of main coolant was observed.

In addition to the tritium surveys conducted annually, several attempts were made to determine the effect of hydrazine concentrations on hydrogen levels and ultimately the tritium levels in the Peach Bottom reactor primary coolant gas. Hydrazine was used in the Peach Bottom reactor secondary coolant water to scavenge occluded oxygen. The hydrazine is normally maintained at approximately 150 ppb, which is a

factor of ten higher than needed to remove the oxygen present in the system. Decomposition of the excess hydrazine was considered a possible source of the higher than expected steady-state levels of hydrogen in the primary coolant helium. Therefore, the hydrazine concentration was reduced a factor of ten (15 ppb) and maintained at this level for several days, during which time the helium impurity levels (H_2 , O_2 , N_2 , CH_4 , CO, and tritium) were continuously monitored.

The tritium and hydrogen data are plotted versus time in Fig. 13-25. The data show only a slight reduction in the hydrogen and tritium levels (approximately 20%) during the period of reduced hydrazine concentration. However, during a concurrent unscheduled reduction in reactor power to 40% of maximum power, the tritium and hydrogen levels were noticeably reduced. Subsequent to returning to normal reactor power conditions, the main coolant hydrogen and tritium levels returned to the levels observed prior to the reduction in power. The O_2 , N_2 , CH_4 , and CO impurity levels were constant throughout the test period and essentially unchanged from the initial values obtained prior to hydrazine reduction. Thus, the data gathered at reduced hydrazine levels indicate that the decomposition of excess hydrazine does not appear to be a major source of hydrogen contamination in the main loop coolant gas. It is also evident from Fig. 13-25 that the secondary coolant water tritium levels are essentially unaffected by change in impurity concentrations or reactor power levels for short periods of time.

The tritium levels in the primary and secondary coolant were monitored from August 1971 to October 1974 when the reactor was shut down for decommissioning. Figures 13-26 through 13-28 show the tritium levels observed during 1971, 1973, and 1974, respectively. The data obtained during 1972 were of little value because of sampling difficulties experienced during that time period.

The data shown in Figs. 13-26 and 13-27 were obtained by manual sampling of both primary and secondary coolants. The primary coolant was monitored by converting the tritiated hydrogen to the form of tritiated water. The apparatus for accomplishing this consisted of a copper oxide bed operating at approximately 775°K for conversion to water and a liquid nitrogen coolant glass U-tube for trapping the water formed. The trapped tritiated water was then beta counted using a scintillation detector. The gaseous tritium data plotted in Figs. 13-26 through 13-28 obtained by the manual method are the sum of tritiated hydrogen and water, the water contribution being much lower than the hydrogen levels.

In late 1973, a continuous tritiated hydrogen monitor was developed and installed in the primary coolant of Peach Bottom. The system consisted of a liquid-nitrogen-cooled charcoal trap for delay of all radioactive species in the coolant except for tritium. The off-gas then passed through a Cary ionization chamber previously calibrated with known concentrations of tritiated hydrogen gas. The output of the electrometer gave a continuous tritium level in the primary circuit. The data thus obtained are plotted in Fig. 13-28 for the year 1974.

The data obtained during operation of the reactor indicate approximately a factor of 10 lower tritium concentration in the primary circuit. Also, the secondary circuit concentration appeared to be relatively stable regardless of the reactor power level. The primary coolant tritium levels closely followed the reactor power level, as demonstrated by the reduced tritium activity in July 1974 when the reactor power level was 50%. Upon returning to 80% power in August 1974, the tritium activity returned to the level observed earlier in the year (approximately $5 \times 10^{-4} \mu\text{Ci}/\text{cm}^3$).

13.4. DIFFUSION PROBE DATA

The Peach Bottom diffusion probe experiments provided information on the behavior of condensable fission products in the primary circuit.

Fission products and induced activities in the loop were identified, plate-out concentrations and partial pressures were determined, and some information regarding chemical form was obtained. Partial pressure values are a measure of release from the Peach Bottom core and, consequently, are of particular value for use in testing methods (e.g., FIPER) for calculating fission product release from HTGR-type fuel elements.

The experiments and data discussed below relate specifically to Core 1 which was monitored in this fashion by GA. The equivalent work performed by ORNL for Core 2 has been summarized and is to be presented in a topical report to be issued in the near future (Ref. 10).

13.4.1. Procedure

The diffusion probe system and procedures are described in detail in Refs. 11 and 12. Briefly, the diffusion probe system consists of (1) a plateout sleeve, (2) three diffusion tubes contained in an outer tube (the probe body), (3) a membrane filter, (4) an activated charcoal trap, and (5) a flow control unit. Each diffusion tube is a small-diameter tube through which a portion of the coolant stream passes in laminar flow. The majority of the condensable fission products diffuse to and react with the inside surface of the tube. After being in the reactor for a sufficient time, the probe, membrane filter, and charcoal trap are removed from the reactor loop and shipped to GA for examination and analysis. A new probe is installed when the exposed probe is removed. The Peach Bottom diffusion probe is located in a thermowell near the hot valve in loop 1 and samples the hot coolant gas.

In the laboratory, the diffusion tubes are removed from the probe and sectioned into 1-cm lengths. The sections are gamma counted, and selected sections are subjected to radiochemical analysis for the determination of nuclides, such as Sr-89 and Sr-90, which cannot be determined by gamma counting. The resulting data are in the form of activity profiles for each

nuclide detected. The shape of an activity profile is indicative of the form (atoms, molecules, or particulates) in which condensable fission product nuclides are transported in the helium coolant. In the ideal case, diffusion coefficient values can be determined from the profile data.

Partial pressures are calculated from data on the total deposition of particular nuclides in the diffusion tube, total helium flow through the tube, and decay constants. Release rates can then be derived from the partial pressure data. Release fractions are calculated by taking into account production rates and release rates. Plateout data are determined by analysis of surfaces of the plateout sleeve and probe body. Concentrations in the helium of nuclides associated with particulate matter are determined by analyzing deposits on the membrane filter.

13.4.2. Core 1 Experiments

Three diffusion probe sets, each containing an Incoloy 800 plateout sleeve, stainless steel probe body, three diffusion tubes, a 0.45- μ m membrane filter, and an activated charcoal trap, were exposed during Core 1 operation. The first probe was removed following the January 1968 shutdown (168 EFPD), the second following the October 1968 shutdown (300 EFPD), and the third following the October 1969 shutdown (452 EFPD). The emphasis was on the study of condensable fission products present in the helium coolant in atomic or molecular form. All probes were exposed in the same position between the core outlet and steam generator inlet.

13.4.3. Core 1 Diffusion Probe Results

From diffusion tube data, partial pressures for the nuclides detected and release fractions for fission products were calculated. Partial pressures were in the range of 10^{-16} to 10^{-14} atm. Fractions of fission products released from the core to the main loop were in the range of 10^{-7} to 10^{-4} . Values for specific nuclides are given in Tables 13-3 and 13-4.

TABLE 13-3
CONCENTRATIONS OF NUCLIDES IN PRIMARY COOLANT EXPRESSED AS
PARTIAL PRESSURES
(FROM DIFFUSION TUBE DATA)

Nuclide	Average p (atm)	
	0 to 168 EFPD	168 to 300 EFPD
Cr-51	2×10^{-16}	2×10^{-16}
Co-60	5×10^{-16}	6×10^{-16}
Sr-89	$<3 \times 10^{-17}$	4×10^{-16}
Sr-90	$<5 \times 10^{-17}$	5×10^{-16}
Zr-95	5×10^{-17}	2×10^{-16}
I-131	9×10^{-17}	5×10^{-16}
Cs-137	1×10^{-14}	4×10^{-14}
Ba-140	$\sim 4 \times 10^{-16}$	2×10^{-16}
Ce-141	$\sim 2 \times 10^{-17}$	1×10^{-16}
Ce-144	$\sim 9 \times 10^{-17}$	5×10^{-16}

TABLE 13-4
FRACTIONS OF FISSION PRODUCTS RELEASED
FROM CORE TO MAIN LOOP
(FROM DIFFUSION TUBE DATA)

Nuclide	0 to 168 EFPD	168 to 300 EFPD
Sr-89	$<1 \times 10^{-7}$	1×10^{-6}
Sr-90	$<1 \times 10^{-7}$	1×10^{-6}
Zr-95	9×10^{-8}	4×10^{-7} (a)
Ru-103	1×10^{-7}	4×10^{-7} (a)
I-131	4×10^{-7}	2×10^{-6}
Cs-137	2×10^{-5}	7×10^{-5}
Ba-140	$\sim 7 \times 10^{-7}$	4×10^{-7}
Ce-141	$\sim 3 \times 10^{-8}$	3×10^{-7}
Ce-144	$\sim 2 \times 10^{-7}$	9×10^{-7}

(a) Corresponds to ~ 3 g fuel compact dust containing
0.6 g Th-232 and <0.1 g U-235.

The nuclides detected on diffusion probe 3 included Mn-54, Fe-59, Co-60, Zr-95 - Nb-95, Mo-99 - Tc-99, Ru-103, I-131, Te-132 - I-132, Cs-134, Cs-137, Ba-140 - La-140, Ce-141, Ce-144, and Pa-233. Surface activity concentrations were on the order of 10^{-3} to 10^{-1} $\mu\text{Ci}/\text{cm}^2$ on exposed surfaces. Concentrations on the stainless steel mandrel upon which the Incoloy plateout sleeve was fitted averaged 22% (range = 4 to 50%) of concentrations on outside exposed surfaces of the plateout sleeve.

Concentrations of selected nuclides on surfaces of the three Incoloy 800 plateout sleeves exposed during Core 1 operations are given in Table 13-5. Between 168 and 300 EFPD, concentrations on outside plateout sleeve surfaces remained the same for Cs-137; doubled for I-131, Cs-134, and Ba-140; tripled for Zr-95; and increased by factors of 7 and 11, respectively, for Ce-141 and Pa-233. The largest increases were exhibited by nuclides not normally expected in the primary circuit, but likely to be associated with fuel material released from failed elements. Between 300 and 452 EFPD, surface concentrations doubled for Ba-140, Ce-141, and Pa-233; tripled for Zr-95 and Cs-137; quadrupled for Cs-134; and increased by factors of 5 and 7, respectively, for Ce-144 and I-131. The relatively large increase in iodine concentration may have been the result of a slight reduction in plateout sleeve temperature during reduced power operation just before the 452-EFPD shutdown.

Fission product concentrations on the side of the Incoloy plateout sleeve facing the oncoming hot coolant gas were about twice as high as those on the opposite side. A study (Ref. 13) indicated that the release of Sr-90 at 168 EFPD, either by way of its precursor Kr-90 or as strontium vapor, was probably much lower than indicated by the diffusion tube deposits. The greatest contribution was probably from the fission products in fuel particle dust or other fissile coolant steam contaminants.

The fractional release of Zr-95 and Ru-103 determined from diffusion tube data indicates that during the operational period from 168 to 300

TABLE 13-5
FISSION PRODUCT CONCENTRATIONS ON SURFACES OF
INCOLOY PLATEOUT SPECIMENS (a)
($\mu\text{Ci}/\text{cm}^2$)

	Probe 1, 168 EFPD Shutdown, 1 Failed Element	Probe 2, 300 EFPD Shutdown, 11 Failed Elements	Probe 3, 452 EFPD Shutdown, 78 Failed Elements
Zr-95	2.8×10^{-3}	8.2×10^{-3}	2.6×10^{-2}
I-131	4.8×10^{-4}	7.0×10^{-4}	4.8×10^{-3}
Cs-134	6.9×10^{-3}	1.3×10^{-2}	4.9×10^{-2}
Cs-137	2.3×10^{-2}	2.4×10^{-2}	6.5×10^{-2}
Ba-140	5.7×10^{-3}	1.0×10^{-1}	1.9×10^{-1}
Ce-141	2.4×10^{-3}	1.7×10^{-2}	4.1×10^{-2}
Ce-144	ND ^(b)	1.1×10^{-2}	5.0×10^{-2}
Pa-233	3.1×10^{-3}	3.4×10^{-2}	8.0×10^{-2}

(a) Specimens positioned at top of diffusion probe inserted into loop 1 hot valve located between reactor vessel outlet and steam generator inlet. Probes were removed and replaced at each of the scheduled shutdowns, and thus individually experienced 168, 132, and 152 EFPD of exposure.

(b) ND = not detected.

EFPD, approximately 4×10^{-7} fraction of the fuel was released to the primary coolant, presumably from failed fuel elements (Ref. 14). The fraction 4×10^{-7} corresponds to about 3 g of fuel-carbon dust containing 0.6 g of Th-232 and less than 0.1 g U-235. A strontium release fraction of 1×10^{-6} was calculated from diffusion tube data.

The analysis of isotope activity ratios (e.g., Sr-89/Sr-90) after 300 days indicated further that the major contributor to the metallic fission product activity in the main loop of the reactor was fuel released from cracked fuel elements rather than diffusional release or decay of gaseous precursors (Ref. 14).

13.5. ADDITIONAL GENERAL ATOMIC REPORTS ON PEACH BOTTOM CHEMISTRY INFORMATION

The following is a list of the General Atomic reports pertaining to chemistry operational tests performed at the Peach Bottom HTGR:

<u>Report Title</u>	<u>Report No.</u>	<u>Ref.</u>
Fission Product Control and Fuel Element Release at the Peach Bottom HTGR	GAMD-7904	8
Chemical Impurities in the Helium Coolant at the Peach Bottom HTGR	Gulf-GA-A10809	15
Particulate Matter in the Peach Bottom HTGR Primary Coolant System	Gulf-GA-B10034	16
Peach Bottom Helium Purification System Performance	GAMD-7902	17
Peach Bottom Hydrogen Injection Test Results	--	9
A Review of Tritium Behavior in HTGR Systems	GA-A13461	6

The Peach Bottom postconstruction R&D quarterly reports contain the results of the postconstruction R&D operational chemistry tests performed at Peach Bottom. A listing of these tests and the reports in which the results can be found is given in Table 13-6.

TABLE 13-6
CHEMISTRY TESTS AND QUARTERLY REPORTS CONTAINING RESULTS

	GA-8500 (Ref. 18)	GA-8679 (Ref. 19)	GA-8816 (Ref. 20)	GA-9080 (Ref. 21)	GA-9237 (Ref. 22)	GA-9360 (Ref. 13)	GA-9494 (Ref. 14)	GA-9797 (Ref. 23)	GA-9898 (Ref. 24)	GA-10099 (Ref. 25)	GA-10300 (Ref. 1)	GA-10400 (Ref. 26)	GA-10502 (Ref. 27)
Peach Bottom Chemistry Tests	Core 1										Core 2		
Activity increase due to Ar-41	X												
Reduced purge flow experiment	X												
Rapid activity increase	X												
Iodine release	X				X			X					
Diffusion probe experiment	X	X	X	X	X	X	X	X	X	X	X		
Main loop particulate matter	X	X	X	X	X		X	X		X	X	X	X
Chemical impurities in the main coolant	X	X	X	X			X	X		X	X	X	X
Fission product plateout survey	X				X								
Effect of power change on loop activity			X										
Delay times for Xe and Kr to reach main loop			X										
Fuel element E05-05 corings				X									
Fuel element C05-05 corings				X									
Broken particle fraction in core 1				X									
Distribution of fission products in fuel elements					X	X	X	X	X	X	X	X	X
Aerosol monitor experiment					X								
Strontium transport						X							
Activity ratios of isotope pairs							X						
Calculation of equilibrium fission product activities in a nonreleasing core							X						
Calculation of diffusion release of metallic elements							X

Fig. 13-1. R/B values for release of Kr-85m into the purge stream

Fig. 13-2. R/B values for release of Kr-87 into the purge stream

Fig. 13-3. R/B values for release of Kr-88 into the purge stream

Fig. 13-4. R/B values for release of Kr^{89} into the purge stream

Fig. 13-5. R/B values for release of Xe-133 into the purge stream

Fig. 13-6, R/B values for release of Xe^{135} into the purge stream

Fig. 13-7. R/B values for release of Xe-138 into the purge stream

Fig. 13-8. Peach Bottom Core 2 Kr-85m fission gas release R/B data

Fig. 13-9. Peach Bottom Core 2 Kr-87 fission gas release R/B data

Fig. 13-10. Peach Bottom Core 2 Kr-88 fission gas release R/B data

Fig. 13-11. Peach Bottom Core 2 Kr-89 fission gas release R/B data

Fig. 13-12. Peach Bottom Core 2 Xe-133 fission gas release R/B data

Fig. 13-13. Peach Bottom Core 2 Xe-135 fission gas release R/B data

Fig. 13-14. Peach Bottom Core 2 Xe-138 fission gas release R/B data

Fig. 13-15. Peach Bottom fuel element purge R/B as a function of power, Kr-85m

Fig. 13-16. Peach Bottom fuel element purge R/B as a function of power, Kr-87

Fig. 13-17. Peach Bottom fuel element purge R/B as a function of power, Kr-88

Fig. 13-18. Peach Bottom fuel element purge R/B as a function of power, Xe-133

Fig. 13-19. Peach Bottom fuel element purge R/B as a function of power, Xe-135

Fig. 13-20. Peach Bottom fuel element purge fission gas release data, Kr-85m

Fig. 13-21. Peach Bottom fuel element purge fission gas release data, Kr-87

Fig. 13-22. Peach Bottom fuel element purge fission gas release data, Kr-88

Fig. 13-23. Peach Bottom fuel element purge fission gas release data, Xe-133

Fig. 13-24. Peach Bottom fuel element purge fission gas release data, Xe-135

Fig. 13-25. Tritium and hydrogen levels before, during, and after reduction of N_2H_4 levels in the Peach Bottom reactor

Fig. 13-26. Tritium data for 1971

Fig. 13-27. Tritium data for 1973

Fig. 13-28. Tritium data for 1974

14. GENERAL PHYSICS DATA

A listing of the General Atomic reports pertaining to physics operation tests performed on both Core 1 and Core 2 is given below.

<u>Report Title</u>	<u>Core</u>	<u>Report No.</u>	<u>Ref.</u>
Progress of Zero Power Experiments in the Peach Bottom HTGR	1	GA-7121	28
Physics Tests During the Initial Operation of the Peach Bottom HTGR	1	GA-8025	29
Physics Performance of the Peach Bottom HTGR	1	GA-8117	30
Peach Bottom Initial Loading to Criticality	1	GA-7351	31
Pulsed Source Measurement of Control Rod Worth in Peach Bottom	1	GA-7352	32
Interpretation of Pulsed-Source Experiments in the Peach Bottom HTGR	1	GA-8231	33
Flux Distribution Measurements in Peach Bottom	1	GA-7353	34
Differential Rod Worth and Loading to 804 Elements	1	GA-7354	35
Reactivity Coefficient Measurement in Peach Bottom	1	GA-7355	36
Reactivity Effect of Helium Mass and Flow in Peach Bottom	1	GA-7356	37
Temperature Coefficient Calculations for Peach Bottom	1	GA-7357	38
Isothermal Temperature Coefficient Measurement in Peach Bottom	1	GA-7358	39
Zero Power Noise Analysis and Rod Scram Transient Measurements in Peach Bottom	1	GA-7359	40
Use of an On-Line Reactivity Computer to Calibrate Control Rods at Power	1	GA-8104	41
Xenon Poisoning During Rise to Power in Peach Bottom	1	GAMD-7910	42

<u>Report Title</u>	<u>Core</u>	<u>Report No.</u>	<u>Ref.</u>
Noise Analysis Measurement During Rise to Power in Peach Bottom	1	GAMD-7908	43
Temperature and Power Coefficient Measurement During Rise to Power in Peach Bottom	1	GAMD-7909	44
Control Rod Calibrations During Initial Rise to Power in Peach Bottom	1	GAMD-7912	45
Summary of Peach Bottom Core-2 Startup Physics Testing	2	--	46

In addition, the Peach Bottom postconstruction R&D quarterly reports contain the results of the postconstruction R&D operational physics tests performed on Core 1 and Core 2. A listing of these tests and the reports in which the results can be found are given in Table 14-1.

TABLE 14-1
PHYSICS TESTS AND QUARTERLY REPORTS CONTAINING RESULTS

	GA-8500 (Ref. 18)	GA-8679 (Ref. 19)	GA-8816 (Ref. 20)	GA-9080 (Ref. 21)	GA-9237 (Ref. 22)	GA-9360 (Ref. 13)	GA-9494 (Ref. 14)	GA-9797 (Ref. 23)	GA-9898 (Ref. 24)	GA-10099 (Ref. 25)	GA-10300 (Ref. 1)	GA-10400 (Ref. 26)	GA-10502 (Ref. 27)
Peach Bottom Physics Tests	Core 1										Core 2		
Plant thermal efficiency	X												
Xenon decay and buildup measurements	X		X	X	X			X				X	
Short-term reactivity changes	X		X			X							
Long-term reactivity changes with burnup	X	X	X	X		X	X	X					X
Reactivity handbook		X					X						
Plant control confirmation			X										
Control rod worths			X			X					X	X	
Temperature coefficient measurement			X			X					X	X	
Helium flow coefficient			X								X		
Predicted control rod worth changes with exposure					X								
Fuel element replacement analysis				X	X	X							
Failed element locations, type, and exposure									X				
Core 2 loading and physics testing										X	X		X
Helium pressure (mass) coefficient											X		
Core 2 temperature correlations												X	

15. LOCATION OF SENSING ELEMENTS IN THE PRIMARY HELIUM CIRCUIT

The principal sensing elements in the Peach Bottom primary helium circuits are as follows:

<u>Parameter</u>	<u>Loop</u>	<u>Element/Transmitter</u>
Helium flow	1	FE-201
Helium flow	2	FE-204
Helium pressure	1	PT-201
Helium pressure	2	PT-228
Reactor inlet temperature	1	TE-210
Reactor inlet temperature	2	TE-218
Reactor outlet temperature	1	TE-204
Reactor outlet temperature	2	TE-212
Primary circuit activity	1	Cary meter

The approximate location of the primary sensing elements for loop 1 are shown in Fig. 15-1. Both the major components and the sensing elements for loop 2 are located according to a mirror image of loop 1.

Additional primary circuit sensing elements and their locations are described below.

<u>Parameter</u>	<u>Loop</u>	<u>Element/ Transmitter</u>	<u>Location</u>
Helium pressure	1	PT-202	Just upstream of main loop 1 flow element
Helium pressure	2	PT-229	Just downstream of helium compressor outlet
Reactor outlet temperature	1	TE-203	Just upstream of TE-204

<u>Parameter</u>	<u>Loop</u>	<u>Element/ Transmitter</u>	<u>Location</u>
Reactor outlet temperature	1	TE-205	Just downstream of hot valve HV-201 (The "hot valve" temperature reported under typical data for Sections 4 through 11 came from TE-205.)
Reactor outlet temperature	2	TE-213	Just downstream of hot valve HV-202
Reactor outlet temperature	2	TE-211	Just upstream of TE-212
Reactor outlet temperature	1	TE-85 through -94	Just outside the reactor vessel
Reactor outlet temperature	2	TE-110 through -119	Just outside the reactor vessel

The primary piping and instrumentation drawings for the Peach Bottom HTGR are listed below. All of these drawings are Bechtel drawings. Each Bechtel drawing indicates that the body of the Bechtel drawing was reproduced from a GA drawing bearing the same last eleven significant figures as the Bechtel drawing number (Bechtel drawing number minus 3105).

	<u>Drawing No.</u>
Fuel handling purge and auxiliary systems	3105-800-100-PI001
Control rod drive system	3105-800-100-IC002
Emergency shutdown system	3105-800-100-IC003
Main coolant system helium circuit - loop 1	3105-800-200-PI001
Main coolant system helium circuit - loop 2	3105-800-200-PI002
Main coolant system steam circuit - loop 1	3105-800-200-PI003
Main coolant system steam circuit - loop 2	3105-800-200-PI004
Main coolant system P&I details	3105-800-200-PI005
Helium compressor C-201	3105-800-200-PI006
Helium compressor C-202	3105-800-200-PI007
Seal oil system for helium compressor C-201	3105-800-200-PI008

	<u>Drawing No.</u>
Seal oil system for helium compressor C-202	3105-800-200-PI009
Main coolant valves - loop 1	3105-800-200-PI010
Main coolant valves - loop 2	3105-800-200-PI011
Fuel element purge purification system	3105-800-300-PI001
Oxidation system S-304	3105-800-300-PI002
Kr and Xe low-temperature delay system	3105-800-300-PI003
Purified helium compressor and oil removal system	3105-800-300-PI004
Liquid-nitrogen-cooled trap system S-302	3105-800-300-PI005
Freon refrigeration and brine system S-301	3105-800-300-PI006
Dehydration system (S-303) and aerosol removal filter (F-302)	3105-800-300-PI007
Helium handling system - nonpurified	3105-800-400-PI001
Helium handling system - purified	3105-800-400-PI002
Transfer compressors	3105-800-400-PI003

Fig. 15-1. Location of primary sensing elements within the No. 1 Peach Bottom primary helium circuit

16. POSTIRRADIATION EXAMINATION AND TESTING OF REACTOR COMPONENTS

16.1. FUEL ELEMENTS

The following Peach Bottom fuel elements have been or will be given a postirradiation examination:

<u>Element</u>	<u>Total EFPD</u>	<u>Core</u>	<u>PIE at</u>	<u>Results Reported in</u>	<u>Ref.</u>
E05-05	168	1	GA	GAMD-8743	47
C05-05				GA-8816	48
				GA-9080	49
D06-01	300	1	GA	GA-9360	50
				GA-9797	51
				GA-9898	52
				GA-10099	53
D13-05	452	1	GA	Gulf-GA-A1249	54
C11-07	252	2	GA	GA-A12460	55
E06-01	385	2	ORNL	ORNL-TM-4705	56
E11-07	701	2	ORNL	To be reported when completed by ORNL	
E14-01	897	2	ORNL		
F03-01	897	2	ORNL		
E01-01	897	2	ORNL		
F05-05	897	2	ORNL		
B11-03	897	2	ORNL		

The following Peach Bottom fuel elements were examined within the reactor vessel using the reactor viewing device and were also gamma scanned using the charge machine as required by the technical specifications:

<u>Element</u>	<u>Total EFPD</u>	<u>Core</u>	<u>Results Reported in</u>	<u>Ref.</u>
B13-07	168	1	GA-8679	57
			GAMD-8703	58
B13-05	168	1	GA-8679	57
			GAMD-8703	58
B13-07	300	1	GA-9237	59
A16-15	300	1	GA-9237	59
B13-07	452	1	GA-10099	53
A16-15	452	1	GA-10099	53
C11-07	252	2	Gulf-GA-A10798	60
A16-15	252	2	Gulf-GA-A10798	60
E11-07	701	2	--	61
A16-15	701	2	--	61

16.2. REFLECTOR ELEMENTS

The following Peach Bottom reflector elements have been or will be given a postirradiation examination:

<u>Element</u>	<u>Station EFPD</u>	<u>PIE at</u>	<u>Results Reported in</u>	<u>Ref.</u>
B16-01	300	GA	GAMD-9911	62
			GA-9797	51
			GA-9898	52
F08-08	704	GA	--	53
A18-08	1349	ORNL	To be reported when completed by ORNL	

16. POSTIRRADIATION EXAMINATION AND TESTING OF REACTOR COMPONENTS

16.1. FUEL ELEMENTS

The following Peach Bottom fuel elements have been or will be given a postirradiation examination:

<u>Element</u>	<u>Total EFPD</u>	<u>Core</u>	<u>PIE at</u>	<u>Results Reported in</u>	<u>Ref.</u>
E05-05	168	1	GA	GAMD-8743	47
C05-05				GA-8816	48
				GA-9080	49
D06-01	300	1	GA	GA-9360	50
				GA-9797	51
				GA-9898	52
				GA-10099	53
D13-05	452	1	GA	Gulf-GA-A1249	54
C11-07	252	2	GA	GA-A12460	55
E06-01	385	2	ORNL	ORNL-TM-4705	56
E11-07	701	2	ORNL	To be reported when completed by ORNL	
E14-01	897	2	ORNL		
F03-01	897	2	ORNL		
E01-01	897	2	ORNL		
F05-05	897	2	ORNL		
B11-03	897	2	ORNL		

The following Peach Bottom fuel elements were examined within the reactor vessel using the reactor viewing device and were also gamma scanned using the charge machine as required by the technical specifications:

<u>Element</u>	<u>Total EFPD</u>	<u>Core</u>	<u>Results Reported in</u>	<u>Ref.</u>
B13-07	168	1	GA-8679	57
			GAMD-8703	58
B13-05	168	1	GA-8679	57
			GAMD-8703	58
B13-07	300	1	GA-9237	59
A16-15	300	1	GA-9237	59
B13-07	452	1	GA-10099	53
A16-15	452	1	GA-10099	53
C11-07	252	2	Gulf-GA-A10798	60
A16-15	252	2	Gulf-GA-A10798	60
E11-07	701	2	--	61
A16-15	701	2	--	61

16.2. REFLECTOR ELEMENTS

The following Peach Bottom reflector elements have been or will be given a postirradiation examination:

<u>Element</u>	<u>Station EFPD</u>	<u>PIE at</u>	<u>Results Reported in</u>	<u>Ref.</u>
B16-01	300	GA	GAMD-9911	62
			GA-9797	51
			GA-9898	52
F08-08	704	GA	--	53
A18-08	1349	ORNL	To be reported when completed by ORNL	

The following Peach Bottom reflector elements were examined within the reactor vessel using the reactor viewing device:

<u>Element</u>	<u>Station EFPD</u>	<u>Results Reported in</u>	<u>Ref.</u>
A17-16	168	GA-8679	57
		GAMD-8703	58
B18-12	168	GA-8679	57
		GAMD-8703	58
C17-03	300	GA-9237	59
B18-12	300	GA-9237	59
C17-03	452	GA-10099	53
B18-12	452	GA-10099	53
C17-03	704	Gulf-GA-A10798	60
B18-12	704	Gulf-GA-A10798	60
C17-03	1153	--	61
B18-08	1153	--	61

16.3. CONTROL RODS AND CONTROL ROD GUIDE TUBES

The following Peach Bottom control rods have been or will be given a postirradiation examination:

<u>Control Rod</u>	<u>Station EFPD</u>	<u>PIE at</u>	<u>Results Reported in</u>	<u>Ref.</u>
C08-01	300	GA	GA-10196	64
			GA-9797	51
E08-01	1349	GA	To be reported when completed by GA	

In addition, the C08-01 Peach Bottom control rod guide tube and thermally released shutdown rod were given a postirradiation examination after 300 EFPD and were reported in GA-10196 (Ref. 64) and GA-10099 (Ref. 53).

All of the control rod guide tubes were also regularly given a visual examination using the reactor viewing device. The guide tubes were examined according to the following schedule:

<u>Date of Examination</u>	<u>Station EFPD</u>	<u>Results Reported in</u>	<u>Ref.</u>
Feb. 24, 1968	168	GA-8679	57
		GAMD-8703	58
Nov. 21, 1968	300	GA-9237	59
April 1, 1970	452	GA-10099	53
May 17-18, 1971	704	Gulf-GA-A10798	60
Nov. 28, 1973	1153	--	61

16.4. THERMAL BARRIER

The Peach Bottom thermal barrier was regularly given a visual examination using the reactor viewing device. The thermal barrier was examined according to the following schedule:

<u>Date of Examination</u>	<u>Station EFPD</u>	<u>Results Reported in</u>	<u>Ref.</u>
Feb. 20-24, 1968	168	GA-8679	57
		--	65
		GAMD-8212	66
Nov. 17-18, 1968	300	GA-9237	59
		GAMD-8212	66
April 1-2, 1970	452	GA-10099	53
		--	67
May 17-19, 1971	704	Gulf-GA-A10798	60
		--	68
Nov. 28, 1973	1153	--	69

16.5. REACTOR VESSEL CHARPY IMPACT TESTS

The fast neutron irradiation damage to the Peach Bottom reactor vessel was monitored closely throughout the operating lifetime of the reactor through the use of Charpy impact tests. The change in nil-ductility transition temperature was monitored periodically according to the following schedule:

<u>Date of Charpy Sample Removal</u>	<u>Charpy Samples Removed From</u>	<u>Station EFPD</u>	<u>Results Reported in</u>	<u>Ref.</u>
Nov. 20-21, 1968	Reflector element B16-01	300	GAMD-9393	70
	Reflector element B17-03		--	2
	Reflector element B18-08		GA-9797	51
	Reflector element A17-16		GA-10099	53
	Two samples from nozzle access plug trays			
April 1970	Reflector element D18-12	452	--	71
	Reflector element E18-12			
	Four samples from nozzle access plug trays		GA-10300	1
June 10, 1971	Reflector element F17-16	704	--	72
	Reflector element F18-08			
	Two samples from nozzle access plug trays			

17. ADDITIONAL REPORTS ON THE OPERATING HISTORY OF PEACH BOTTOM

Additional General Atomic reports pertaining to the operating history of the Peach Bottom Atomic Power Station are listed in the following sections.

17.1. REPORTS PERTINENT TO ALL PEACH BOTTOM OPERATIONS

<u>Report Title</u>	<u>Report No.</u>	<u>Ref.</u>
Recent Operating Experience at the Peach Bottom Atomic Power Station	GA-9047	73
Significant Operating Experiences of the Peach Bottom Atomic Power Station	GA-9013	74
Engineering Tests During the Initial Operation of the Peach Bottom HTGR	GA-8040	75
Design and Operation Experience With Control and Instrumentation Systems, Peach Bottom Station Unit 1	Gulf-GA-A10876	76
The Peach Bottom HTGR	GA-7636	77
Core Heat Transfer Measurements During Rise to Power in Peach Bottom	GAMD-7911	78
Peach Bottom Shielding Surveys	GAMD-7901	79
Peach Bottom Initial Power Plant Performance	GAMD-7903	80
Peach Bottom Atomic Power Station, Report on Reactor Plant Components from January 1, 1967 to October 3, 1969	GAMD-10341	81
Peach Bottom Control Performance and Dynamic Behavior	GAMD-7900	82
Operational Data on Decay Heat and Shutdown Cooling for the Peach Bottom Reactor	GAMD-5632	83

17.2. REPORTS PERTINENT TO CORE 1 OPERATIONS

<u>Report Title</u>	<u>Report No.</u>	<u>Ref.</u>
HTGR Fuel Element Performance in Peach Bottom Reactor	GA-8113	84
40-MW(e) Prototype High-Temperature Gas-Cooled Reactor Postconstruction Research and Development Program Summary Progress Report for the Period Ending:		
July 31, 1964	GA-5700	85
October 31, 1964	GA-5924	86
January 31, 1965	GA-6134	87
April 30, 1965	GA-6406	88
July 31, 1965	GA-6646	89
October 31, 1965	GA-6829	90
January 31, 1966	GA-6995	91
May 31, 1966	GA-7232	92
July 31, 1966	GA-7426	93
October 31, 1966	GA-7595	94
January 31, 1967	GA-7830	95
May 31, 1967	GA-7988	96
July 31, 1967	GA-8153	97
October 31, 1967	GA-8370	98
Metallurgical Examination of Diffusion Probes Removed from Peach Bottom Atomic Power Station	GAMD-10626	99

17.3. REPORTS PERTINENT TO CORE 2 OPERATIONS

<u>Report Title</u>	<u>Report No.</u>	<u>Ref.</u>
Peach Bottom Core 2 Semiannual Progress Report No. 1 for the Period Ending June 30, 1971	Gulf-GA-A10798	100
Peach Bottom Core 2 Progress Report No. 2, Annual Progress Report for the Period Ending June 30, 1972	GA-A13093	5
HTGR Fuel Performance in the Peach Bottom Reactor	Gulf-GA-A12675	101
Irradiation Experience with HTGR Fuels in the Peach Bottom Reactor	GA-A12897	102

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