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HAZARDS ANALYSIS

by the

DIVISION OF LICENSING AND REGULATION

in the matter of

PHILADELPHIA ELECTRIC COMPANY
PEACH BOTTOM ATOMIC POWER STATION

Constr. Permit

INTRODUCTION AND BACKGROUND

Pursuant to the provisions of the Atomic Energy Act of 1954 and the Commission's regulations, the Philadelphia Electric Company of Philadelphia, Pennsylvania, has made application to construct and operate a nuclear power reactor facility, known as the Peach Bottom Atomic Power Station, in Peach Bottom Township, York County, Pennsylvania. The application, which is part of the public record, is available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. This application includes all of the information required for a safety evaluation of the proposed facility for purposes of issuance of a provisional construction permit. In particular it includes a hazards summary report containing a description of the facility and a discussion of the significant features and plans intended to safeguard the public against possible radiation hazards. This report has served as the basis for a comprehensive safety review by the staff of the AEC Division of Licensing and Regulation. The results of this safety review are summarized herein.

As required by the Atomic Energy Act, a public hearing will be held to consider the issuance of a construction permit for this facility to the Philadelphia Electric Company. A notice of this public hearing was published in the Federal Register on November 16, 1961. The hearing will commence at 10 A.M. on December 18, 1961 in the Auditorium of the AEC Headquarters at Germantown, Maryland. The principal technical issues to be considered at this hearing are as follows:

1. Whether there is sufficient information to provide reasonable assurance that a facility of the general type proposed in the application can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

2. Whether there is reasonable assurance that technical information omitted from the application will be supplied.

3. Whether the applicant and its principal contractors, Bechtel Corporation and General Atomics Division of General Dynamics Corporation, are technically qualified to design and construct the facility.

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The proposed Peach Bottom Atomic Power Station is to be constructed on a 600 acre site on the west bank of the Susquehanna River. The station will include an advanced type, high temperature, gas cooled, thermal neutron reactor, conventional steam operated generating equipment and necessary supporting facilities. The reactor is designed to produce 115 thermal megawatts (40 electrical megawatts) with gas conditions of 1380°F at 350 psi and steam conditions of 1000°F at 1450 psi. The reactor system incorporates several features and concepts not previously adapted to power reactors and will operate at higher temperatures than previously proposed for power reactors. All components of the reactor system which contain radioactive material will be enclosed in a high integrity, low leakage rate steel containment building.

Philadelphia Electric Company filed application for a construction permit on July 25, 1960. However, prior to filing the application the Company requested a preliminary informal evaluation of the site and submitted a comprehensive "Site Evaluation Report" in support of this request. The Commission staff reviewed this information and discussed the matter of site acceptability with the Commission's Advisory Committee on Reactor Safeguards (ACRS), a body composed of experts in various fields related to reactor safety. As a result of this review it was concluded that the Peach Bottom site was adequate for a reactor of the general type proposed. A copy of the ACRS report, dated March 14, 1960 is appended hereto. The site Evaluation Report referred to above is not included in the public record; however, all information contained in this report has been included in the present application.

The technical information submitted as part of the application was reviewed by the Commission staff and the ACRS in 1960. The report of the ACRS dated December 10, 1960 on this matter is appended. On the basis of this review in 1960 it was concluded that: (1) the proposed design was considered generally adequate, but several specific areas were identified wherein additional design effort appeared necessary, and (2) the application lacked sufficient information concerning the large supporting research and development program to justify certain design features. By letter dated April 10, 1961 the applicant was requested to furnish additional information to permit an adequate evaluation of the proposed facility.

On August 11, 1961, Philadelphia Electric Company filed Amendment No. 2 to the application which was intended to furnish the additional information requested by the AEC and to include more recent results of the research and development program. In particular this amendment included a revised Volume 1 - Part B, "Plant Description and Safeguards Analysis," which is a complete revision of that portion of the application relating to facility design and safeguards. In view of the considerable amount of additional information submitted, the Commission staff performed a complete re-evaluation of the facility design. The ACRS considered the revised application at its meeting of October 26-28, 1961, and reported its conclusions to the Commission on November 1, 1961, copy appended. Both the ACRS and the Staff have now concluded that sufficient information had been furnished to adequately support the application.

In evaluating the acceptability of constructing and operating an advanced gas-cooled type reactor such as proposed for the Peach Bottom Atomic Power Station, the AEC staff has given consideration to all aspects significant to a safety determination, specifically including the following:

1. The suitability of the proposed location with reference to the surrounding populace and the Conowingo reservoir on the Susquehanna River.

2. The proposed design of the facility including means for the collection, treatment, disposal and monitoring of radioactive material, with reference to the adequacy of design and procedural safeguards to assure compliance with the Commission's regulations during normal operation.

3. The adequacy of the proposed reactor design, particularly including the normal and emergency cooling system, the control and safety systems, and the inherent control and safety characteristics, with reference to the probability and consequences of potential hazards resulting from credible accidents to the facility.

4. The advanced nature of the proposed reactor design including several concepts not previously incorporated in power reactor application, viz, the extensive application of low permeability graphite as core structural material and for the fuel element components, unclad fuel elements including the related requirements for fission product purging and prevention of fission product leakage to the gas coolant, and fission product trapping involving the collection of radioactive material both within and external to the reactor system.

5. The extensive supporting research and development program with reference to the adequacy of this program to provide a basis for design features which are extensions of present technology.

6. The proposed containment building with reference to its ability to function as an ultimate confinement barrier against the release of radioactivity in the event of any foreseeable accident.

7. The technical qualifications of the applicant to design and construct the facility, including qualifications of the applicant's contractors.

At the forthcoming public hearing, the staff of the Division of Licensing and Regulation presently intends to recommend issuance of a provisional construction permit for this facility. The basis for this recommendation is summarized in the following AEC staff analysis. However, this evaluation and proposed recommendation are subject to modification in the light of any further information which may become available including the evidence to be introduced at the public hearing. Under the Commission's regulations, any person whose interest may be affected may file a petition to intervene and, if granted, may participate in the proceeding. The final decision of the Commission will be based upon the entire record in this proceeding, including the testimony to be developed at the public hearing.

STAFF HAZARDS ANALYSIS

I REACTOR SYSTEM

The reactor system consists principally of an all-graphite core located within a steel pressure vessel, with two helium circulation loops to transfer heat to the steam generators. The reactor core is essentially cylindrical in shape and includes an array of fuel elements, control rods, and safety shutdown rods surrounded by a graphite reflector. The core is supported by a steel support plate located near the bottom of the pressure vessel, which has dimensions of 14 feet inside diameter and 35 feet height. The fuel elements consist of a central core of compacts containing carbides of uranium and thorium dispersed in graphite and encased in a low permeability graphite sleeve. Each fuel element rests on a standoff pin in the core support plate. There is no similar fixed support for the top of the core; lateral support is provided by movable reflector pieces which exert a radial force on the core array.

Inlet gas at about 634°F enters the reactor vessel through two nozzles and flows downward in spaces between the core and vessel wall. The helium then flows upward through the core, passing over the outer graphite sleeves of the fuel elements. The helium exits the core at about 1358°F and discharges from the reactor pressure vessel through outlet pipes which are concentrically located within the two inlet pipes. This hot gas then enters the steam generators and generates high pressure superheated steam. The cooled helium is recirculated back to the reactor by means of two gas compressors.

Within the reactor a purge flow is drawn from the primary helium coolant flow through the fuel elements in order to remove fission products which escape from the fuel material. This purge gas is passed through internal traps located in the lower part of the fuel element assemblies and then to an external trapping system. The purified helium is returned to the main coolant stream as buffer helium at the compressors, control rod drives and pressure control points. The activity in the primary helium coolant at equilibrium operating conditions is calculated to be about 108 curies.

The control rods are designed for flexibility and pass through the core within graphite sleeves. These rods are hydraulically driven from below the reactor and scram in the upward direction. Nineteen emergency shutdown rods are also provided which are electrically driven from below the reactor, and which can be driven into the core with considerable force (10,000 lbs) if necessary. In addition, a minimum of 19 fuse-operated poison rods will be installed at the top of the reactor; these will be released if the coolant temperature becomes excessive and fall by gravity into the core.

The design life of the Peach Bottom reactor fuel elements is about 3 years with a core loading of approximately 0.19 excess k in the cold condition (80°F). The corresponding worth of the control rods is 0.26 k, and the shutdown rods 0.17 k. The fuse-operated rods will have sufficient worth to offset the reactivity increase resulting from xenon decay (about 0.03 k).

On the basis of the Staff's examination of the conceptual plans for the Peach Bottom reactor as outlined above, there appears to be no reason why a reactor of this general type cannot be developed to operate safely.

However, we have noted that the proposed reactor concept involves a number of features that are novel or represent extensions of current practice. These include a substantially all-graphite core, unclad fuel material, dependence on low permeability graphite as structural and fission product barrier material, purging of fission products from fuel elements, trapping and collection of fission products, reactor control and safety mechanisms, and high operating temperature. The Staff believes that a judgment of acceptability of this reactor depends to a large extent on the applicant's ability to provide sufficient information to provide reasonable assurance that such advanced concepts can be incorporated into a safe reactor design.

The applicant has undertaken an extensive research and development program in order to provide necessary technical information as a basis for the advanced features. This program, which has been in progress since August 27, 1959, is being performed by General Atomics Division of General Dynamics Corp., the reactor design contractor. A major portion of this effort is involved with investigations relating to the proposed applications of low permeability graphite.

In the following subsections the Staff deals with those areas of the Reactor System considered to require significant attention in evaluating the safety of this facility. In evaluating each area consideration has been given to the adequacy of the proposed design based on present knowledge and experience and the sufficiency of the supporting research and development program, including results reported to date and the scope of the continuing program.

A. Core and Reflector Design

The reactor core (fueled region) has the configuration of a right circular cylinder about 9.2 feet in diameter and the active fuel height is 7.5 feet. The core includes 804 fuel elements, 36 control rod positions and 19 emergency shutdown rod positions, arranged in a close packed array and surrounded by about 2 feet of graphite reflector material. Including the fuel element end fittings and internal traps and the radial reflector, the over-all dimensions are about 12 feet high and 12 feet in diameter. A steel thermal shield is spaced between the reflector and the pressure vessel; this component also serves to direct the cooler inlet gas over the vessel wall.

The core support plate is proposed to be made from a solid slab of carbon steel about 5 inches thick and 10 feet in diameter. The plate is supported at its periphery by a ring fastened to the lower head of the pressure vessel. In addition to the support plate, this ring also carries the weight of the reflectors and the thermal shield. Sleeves around the control rod guide tubes penetrating the lower vessel head extend between the lower surface of the support plate and vessel head to act as auxiliary support of the support plate in event it became hot enough to sag, as could occur following loss of cooling. The core support plate provides mountings for the fuel elements, control and emergency rod guide sleeves and the inside reflector pieces which are in contact with the outermost fuel elements. Hollow stainless steel standoff pins, 1 inch in diameter and 10 inches long, are threaded into the support plate to provide a fixed base for the fuel elements. The graphite rod guide tubes mount into the support plate with breech-lock type connections. Passages within the core support plate connect to the hollow standoff pins and serve as a manifold for collecting the purge gas from each fuel element.

The core is surrounded circumferentially by graphite reflector material about 2 feet thick. The inner 3.5 inches of the reflector consists of dummy hexagonal elements. The remainder consists of two segmented concentric graphite rings, each about 10 inches thick. Each segment of the inner 10-inch reflector ring is connected by rocker arms to the corresponding segment of the outer ring. On the upper surface of the inner reflector ring a circular ring of graphite about 2 inches square mates with a steel lip extending from the lower end of the upper steel plenum shroud to form a gas seal to prevent helium bypass.

The cool helium (634°F) returning from the steam generators divides inside the reactor vessel with a portion flowing downward between the thermal shield and vessel wall; the remainder flows through the segmented reflector pieces, which are interlocked to prevent gas leakage radially inward. The divided flows recombine in the lower plenum region and pass upward through the core to the upper plenum. Owing to the pressure drop of the gas, a pressure differential will exist across the reflector pieces resulting in a lateral force on the fuel elements. The action of the reflector segments is to tilt radially inward by pivoting on the rocker arm at the bottom. The purpose of this arrangement is to provide a firm (700 lbs force on each fuel element), yet thermally expansible restraint on the core. The ring of dummy reflector elements is located inside the inner tilting reflector and these elements are supported by the core support plate in a manner similar to the fuel elements. The inner faces of the 10 inch inner tilting reflector segments are machined to mate with the dummy hexagonal elements so as to maintain the triangular pitch arrangement of the core.

In evaluating the proposed core and reflector design, as described above, the Staff has given consideration to various possible malfunctions which could interfere with safe reactor operation, including the effects of flow disturbances, lateral stability of the core array, interference with control rod motion and possible oscillatory movement. In order to study the behavior of the proposed arrangement during operation, General Atomics has constructed and operated a detailed one-half scale hydraulic model of the pressure vessel and internals. The objectives of the test program with this flow model are: (1) to determine the core restraining capability of the tilting reflector and the hydraulic stability of the core, (2) to determine the pressure drop characteristics of the internal flow paths, (3) to study the flow patterns in the coolant passages, (4) to measure the effectiveness of the inlet gas in cooling the pressure vessel wall and (5) to determine the effectiveness of the reflector gas seals.

The first phase of the hydraulic model test has been completed. This phase involved an investigation of the effects of flow distribution on the cooling of the vessel wall, core pressure drop and functioning of the core restraint with model flow rates in the range of 30-100% equivalent design flow. The data from the first phase is now being evaluated for possible design effects. Later test phases will be concerned with more specific design details, for example, design of the plenums, thermal shields, shrouds, reflectors and reflector seals. In addition, tests will be performed to simulate normal and emergency conditions and normal startup and shutdown conditions to obtain heat transfer data to assist in design of the pressure vessel.

In order to determine the dynamic stability of the fuel elements a cluster of 19 full-scale fuel elements was subjected to flow conditions similar to those in the reactor. In these tests the flow Reynolds Number was varied from 10 to 225% of the normal operating value expected in the reactor. Two types of fuel elements were studied: one included an aluminum section replacing the normal graphite sleeve and the other was a prototype graphite element, both tested without the fuel compacts installed. These configurations represented conservative models of the actual fuel elements since they had a lower mass and lacked the dynamic vibration absorption inherent with the loose fitting compacts. In addition to variations in flow conditions, the tests involved vibrations artificially induced in the fuel elements and pressure pulses impressed on the flow. The results of these tests indicated that none of the steady state flow conditions caused the fuel elements to vibrate and that disturbances forcibly impressed were not amplified by the gas flow.

Inasmuch as the coolant channels in the HTGR core are in the shape of tricuspids formed by the spaces of an equilateral array of fuel elements, and adjacent channels intercommunicate, the pressure drops affecting heat transfer rates and velocity profiles in the passages are different from those in ordinary ducts. Tests have been conducted to determine the characteristics of this arrangement. An array of seven simulated fuel elements was subjected to flow at isothermal conditions to determine local pressure conditions and flow velocity patterns throughout the coolant passage length. In addition, similar experiments were performed using internal electric heaters to simulate nuclear heat. By use of thermocouples inbedded at various points along the tube walls, heat transfer coefficients were measured for various flow conditions. This information will be used to calculate the physical design conditions for the fuel components and provide data for the safety evaluation of normal and emergency flow and temperature.

Tests have been performed on a prototype control rod and drive assembly to determine possible effects of core flexibility on control rod operation. In one series of tests the control rod was operated with the guide tube deflected 1½ inch at the upper end; the operability of the rod was not affected. In the second series of tests the drive unit was deflected 9 inches from centerline below the vessel entrance nozzle; only a slight reduction in rod drive speed was noted. These tests indicate that the control rod and drive should operate satisfactorily even under extreme core distortions.

The Staff considers the scope of the above described test program to be adequate to provide the technical information necessary for a safety determination of the core and reflector design. On the basis of results obtained to date it is concluded that the significant safety questions have been adequately resolved. Although the unfinished phases of the flow model tests are not expected to involve significant safety questions, the results will be reviewed for any possible hazard implications. It is the opinion of the Staff that sufficient information on the core and reflector design is now available to provide reasonable assurance that the proposed design is adequate from a safety standpoint.

B. Fuel Element Design

The fuel elements can be classed as a solid semi-homogeneous type in which graphite serves as the fuel matrix, moderator and cladding material. The fuel element assembly consists of an upper reflector section, a fuel bearing middle section and a bottom reflector section. A graphite sleeve of low permeability ($1x10^{-6}$ cm^2/sec) encases the entire fuel section and part of the lower reflector section. The lower reflector section includes an internal fission product trap which is within the sleeve. In this design the approach is to control the escape of fission products and to collect them in traps such that the activity of the primary helium coolant is maintained at a satisfactorily low level, rather than to attempt complete retention of fission products in the fuel elements.

The fuel element assembly is 3.5 inches in diameter and 144 inches long with a handling knob at the top and an opening in the lower end for mounting the element on the core support plate standoff pin. The fuel section consists of annular fuel compacts mounted on a central graphite spine. The compact dimensions are: length - 1.5 inches, outside diameter - 2.75 inches, and inside diameter - 1.75 inches. The compact material consists of a graphetized mixture of graphite, fissile material, fertile material and neutron poison material. The over-all length of the fuel section is 90 inches. The fissile and fertile material is in the form of particles of uranium-carbide and thorium carbide, respectively. These particles range in size from 100-400 microns and each particle is pyrolitically coated with a 50-60 micron thickness of dense carbon. The sleeve is about 10 feet long with a wall thickness of about 3/8 inch.

The internal fission product trap consists essentially of a graphite cylinder with slots on the outer surface which are packed with granular silver coated charcoal reagent material. During reactor operation a purge flow of about 1.1 lb/hr per fuel element is drawn through the upper reflector section and passes downward in spaces between the compacts and sleeve. The purge flow passes through the internal trap, then exits the fuel element through the standoff pin, passes through the external trapping system and is returned to the main coolant system. The performance of the traps is discussed in a later section.

During normal reactor operation, with a core outlet gas temperature of 1380°F, the average moderator (spine and sleeve material) will be about 1820°F, the average fuel compact temperature will be about 1985°F and the maximum fuel compact temperature will be about 2730°F.

The proposed fuel element design involves the application of graphite under conditions for which there is no previous experience in operating reactors. Particularly involved is the use of low permeability graphite in complex structural arrangements at conditions of temperature, temperature gradients and radiation exposures which are more severe than for present practice. Consequently the exact behavior of the fuel elements during operation is not completely predictable including the effects of radiation exposure, the structural integrity and dimensional stability, the effective thermal conductivity of the material, and the possible mobility of fuel, neutron poison and fission products within the fuel material. A significant portion of the supporting research and development program is directed towards resolution of these questions. This program involves the investigation of the effects of radiation, temperature and chemical impurities on the mechanical and physical properties and on the dimensional stability of the low permeability graphite. The important details of this program are discussed below.

Numerous in-pile and out-of-pile capsule tests have been conducted to investigate the integrity of the low permeability graphite specified for use in the Peach Bottom reactor core. Various grades of graphite have been irradiated in 58 different capsule tests at fast neutron exposures from 10^{19} to $2x10^{21}$ nvt and at temperatures from 618-2550°F in order to study dimensional stability at operating conditions. These tests have led to a better understanding of the amount of contraction of the graphite as a function of grade, temperature and neutron exposure, thus allowing more accurate specification of the dimensions of the core components. Additional tests will be conducted.

The effects of irradiation on the permeability of the graphite fuel element sleeve were investigated in a series of 13 capsule tests conducted on various graphites at neutron exposures of $2x10^{21}$ nvt and temperatures in the range 618-2550°F. The resulting data indicate that the permeability of graphite increases with neutron exposure, but that the increase due to anticipated reactor exposure conditions is less than a factor of 10. Although the permeability specification is $1x10^{-6}$ cm^2/sec, graphite of $2x10^{-10}$ cm^2/sec is available. Consequently, there appears to be no difficulty in meeting this specification.

Irradiation effects on the thermal conductivity of sleeve material were determined by means of capsule tests conducted at $2x10^{21}$ nvt (over 0.1 Mev) and 2510-2605°F. The results indicate that this neutron exposure causes a 15% reduction in thermal conductivity at operating temperatures. The minimum value of this parameter for irradiated graphite was found to be 20 BTU/hr-ft°F, in contrast to a value of 15 used as a basis for the design.

Graphite samples irradiated to $2x10^{21}$ nvt and at 2012-2550°F revealed increases in compressive strength of 100% and greater. These results are in good agreement with tests by others on graphite at lower exposure and temperatures which indicate strength increases from 50% to 125%.

In addition to the above mentioned in-pile tests, General Atomics has performed a number of thermal tests on sleeve material in order to separate radiation effects from thermal effects. The effect of temperature alone on dimensions of the graphite was studied by heating samples in helium to 2372°F for 1000 hours. No measurable changes were noted, indicating that the radiation-induced dimensional changes are predominant. Tensile strength tests on low permeability graphite at temperatures of 650-2710°F indicate that failure stresses will exceed the 2500 psi minimum specified for HTGR graphite at beginning of life.

Inasmuch as steam-graphite reactions could cause removal of graphite from the sleeve, thus affecting the permeability of the sleeve, such effects have also been investigated. Samples at temperatures of 1650, 1830, and 2190°F were exposed to circulating helium containing 3% moisture. Under these conditions a 1% loss in graphite and a factor of 10 increase in permeability were observed. The test conditions produced effects considered equivalent to those resulting from a continuous steam leak of 0.01 lb/hr over the entire fuel cycle, a leak rate that is readily detectable, as discussed later. From this information the calculated leakage of fission products through the fuel element sleeve is based on a permeability value of 10^{-5} instead of 10^{-6} cm^2/sec, in order to provide design conservatism.

During normal operations the temperature differential across the 3/8" thick fuel element sleeve will be about 300°F. Tests were conducted to ascertain whether more severe conditions would result in excessive thermal stresses. Sections of low permeability graphite sleeves were subjected to temperature differentials of 640°F for periods of 160 hrs and to 1000°F for several hours, with specimen temperatures in the range 1700-2000°F. No damage was detected, indicating that the sleeves should maintain their integrity against unexpected temperature differentials.

The effect of neutron radiation on the dimensional stability of fuel compacts also has been investigated. Fuel compacts with various ratios of thorium and uranium were irradiated; the fuel was in the form of uncoated particles varying in size from 4 to 500 microns. The compacts were irradiated to fuel burnup in the range $4.25\text{-}6.87 \times 10^{19}$ fissions/cc, compared to 1×10^{19} fissions/cc expected in the reactor, and at temperatures from 1550 to 2710°F. The results of these tests indicate that particle size has a definite effect on the magnitude of dimensional change of the graphite compact material and that the minimum size fuel particles should be from 110 to 250 microns in order to prevent excessive dimensional change. In addition, General Atomics is currently irradiating fuel compacts containing fuel particles which are coated with a 10-60 micron layer of dense carbon. To date these specimens have received an exposure equivalent to about two years of reactor operation and will soon be removed for analysis. These coated particles also have been subjected to 1000°F helium containing 26,000 ppm moisture for periods of 72 hours without apparent damage.

It is proposed that the uranium and thorium carbide particles (100-400 microns diameter) will be coated with a 50-60 micron thickness of dense carbon (more dense than 2.0 gr/cc). This coating serves the following purposes: prevents mobility of the particles in the graphite compacts, protects the carbide particles from oxidation during fabrication of the compacts, and increases the retention time of fission products within the compacts. This coating is applied by jettisoning the particles into a combustion chamber with a stream of high velocity gas which includes methane. The temperature in the chamber is 1830-2550°F. Pyrolytic carbon is deposited on the particle as it passes through the chamber; the final coating thickness is controlled by the dwell time in the chamber and the number of passes.

The fuel element design incorporates rhodium-103 (a stable isotope) in the fuel compacts for the purpose of providing a more extensive negative temperature coefficient throughout core life. It is anticipated that metallic particles of rhodium would be pyrolytically coated and mixed homogeneously in the compacts. The rhodium has the characteristic of a strongly increasing neutron absorption cross section as the temperature increases above 2240°F. Analysis indicates that at least 3 kg of rhodium are required in the core to provide the desired effect. It is proposed that about 5 kg will be included in the initial core in order to compensate for burnup losses. Rhodium has a melting point of 3570°F and boiling point of 8100°F; these characteristics in addition to the pyrolytic coating on the rhodium particles are expected to prevent migration of the poison from the fuel compacts. The temperature coefficient will be discussed later in this report.

A quarter length prototype fuel element is currently being tested in a loop at the General Electric Test Reactor at Vallecitos, Calif. This loop includes a complete fission product trapping system similar to that to be installed on the HTGR. This fuel element will be irradiated in a thermal neutron flux of 10^{13} neutrons/cm^2/sec in an atmosphere of pure helium at 350 psi. The test fuel element will generate 76 kw of power and the heat flux will be about 150,000 BTU/hr-ft^2. Thermal conditions in the test element are approximately equivalent to the design operating conditions for the reactor. This test is intended as a final verification of the proposed fuel element concept. Further discussion of this test is included later in the section on Fission Product Trapping.

Based on our review of the research and development program relating to fuel elements we have concluded that there is reasonable assurance that the impervious graphite diffusion specification of 1×10^{-6} cm^2/sec can be met, that the structural integrity against mechanical loads is adequate, that stress loads resulting from dimensional changes can be controlled by proper dimensional control, that the minimum specification for thermal conductivity of 15 BTU/hr/ft can be met, and that the use of pyrolitic coatings on the fuel particles adequately prevents the mobility and enhances the retention of fission products. Consequently, we now see no reason why the proposed fuel element cannot be developed to operate in a satisfactory manner.

C. Reactor Control

The reactor control system includes a set of control rods for normal reactor control and safety functions, two separate groups of safety rods for emergency reactor shutdown, and appropriate nuclear and process instrumentation.

Normal control is accomplished by 36 boron-carbide control rods each driven by a separate hydraulic mechanism. The absorber portion of the assembly consists of cylindrical sections of poison bearing graphite supported on an axial metal support tube. A push rod connects the absorber section to the drive mechanism which is located below the reactor outside the primary biological shield in a region of moderate temperature (less than 200°F) and low radiation (1 r/hr during operation). The drive mechanism incorporates a hydraulic motor which is operated at different speeds for regulating and scram motions. The significant performance and duty requirements for the control rod assembly are as follows:

Total stroke	7' - 6"
Normal regulating velocity (maximum)	0.06 ft/sec
Scram velocity (maximum)	10 ft/sec
Scram acceleration (maximum)	100 ft/sec^2
Scram deceleration (maximum)	32 ft/sec^2
Duty life of motor and screw	10^6 revolutions
Duty life - number of starts	$5x10^6$
Duty life - number of scrams	$5x10^3$

The neutron absorber portion of the control rod assembly consists of 5 cylinders of graphite loaded with boron carbide with each cylinder 17 inches long, 2.25 inches outside diameter and 1.25 inches inside diameter. These sections are mounted on and supported by a 304 stainless steel axial tube. The sections are supported by collars brazed to the support tube and held separated approximately one inch by disc springs which provide for thermal expansion and shock effects. This control rod arrangement is intended to provide the necessary flexibility to allow the rod to function adequately in event of any distortion of the core or control rod guide tube. The support tube has sufficient strength to allow the assembly to support itself in a vertical position without assistance. The lower end of the support tube joins by means of a mechanical latch to the push rod which connects the absorber section to the drive mechanism. The upper end of the support tube terminates in the form of a lifting knob for fuel handling purposes. Axial motion is transmitted to the push rod by means of a leadscrew and ballnut mechanism.

The control rod operates within the graphite guide tube which is 12 feet long, 2.5 inches in outside diameter and 0.5 inch thick. Poison sections of the control rod do not contact the guide tube due to an annular clearance of 1/8 inch between these components. The rod is guided in the tube by split ceramic rings installed on the spacer collars between poison sections, with a radial clearance of about 0.005 inch between the ring and tube. Cooling of the rod is provided by drawing a small portion (3% for all rods) of the helium coolant from below the core support plate and passing this gas upward both inside and outside of the metal support tube. The design strength of the rod at operating conditions is about 4000 pounds yield and 11,000 pounds ultimate. The weight of the control rod will be approximately 45 pounds.

A screw actuator connects the push and absorber section to the drive unit. A leadscrew with a ball nut is splined directly to the drive motor causing the ball nut to translate axial motion to the push rod.

A rod separation monitor is installed on each control rod to detect possible separation of the flexible metal support tube. This device consists of a high temperature metal clad conductor which traverses the entire length of the assembly from the extreme upper end of the poison section down through the rod assembly and terminating at the drive mechanism. Any separation in the drive mechanism and control rod assembly will result in a break in the conductor and a signal to the control room indicating separation of the control rod assembly.

A mechanical latch is provided which will prevent a fully inserted control rod from dropping out of the core as a result of structural failure or malfunction of the mechanical, electrical or hydraulic components of the drive. This latch is located in the side of the drive actuator housing approximately 4 inches below the ball nut when the control rod is in the fully inserted position. The latch is designed such that the ball nut rides past the latch during rod insertion but it must be released by a solenoid to permit rod withdrawal. Consequently, a fully inserted rod is prevented from dropping more than 4 inches in event of drive failure.

The hydraulic drive mechanism is attached to the control rod drive pressure shell. This assembly consists of a hydraulic motor, a clutch brake assembly, position transmitter, actuating valves, deceleration valves, and a high pressure accumulator. The motor is of the axial piston type with the output shaft connected to the screw actuator of the drive assembly. Two modes of operation are provided by the hydraulic motor: low speed operation for normal rod velocity at 0.06 ft/sec and high speed operation for fast rod insertion at 10 ft/sec. Low speed operation is accomplished by valving low pressure (1000 psi) oil to the motor. High speed scram operation is accomplished by supplying high pressure (3000 psi) oil through larger delivery ports. An auxiliary dual-pressure hydraulic supply system furnishes oil at 1000 psi and 3000 psi. The high pressure supply is supplied by the individual accumulators to furnish the high flow rate required for scram action.

The scram mode is controlled by two scram valves in series arrangement. Two valves are provided to allow the operator to routinely exercise the valves individually during reactor operation to determine proper functioning. Each valve is position monitored in the control room.

There is no previous reactor experience with the application of this type of control rod system. The design incorporates a number of novel features including graphite guide tubes, graphite unclad poison sections, flexible control rod and the application of hydraulic motors as a power source for normal and scram action. However, many components are considered conventional with a high degree of reliability including the ball nut actuator, and the individual components of the hydraulic drive mechanism. In evaluating the safety aspects of the proposed design the Staff has given particular consideration to the safety aspects of the novel features including separation of the flexible rods, satisfactory operation under conditions of core and guide tube distortion, continued integrity of the graphite components, and proper functioning of the integrated mechanical components.

The application contains considerable information concerning the reliability and safety aspects of the proposed control rod system. In addition the applicant has performed a detailed analysis of the effects of possible malfunctions, viz, fracture of poison sections, fracture of push rod, failure of graphite guide tube, overtemperature of structural members, rod behavior due to primary coolant system rupture, jammed scram valve drive, motor seizure, ballnut screw seizure, push rod seizure, accumulator piston seizure, excessive friction in drive components, loss of pressurizer gas to accumulator, clutch malfunction, dirt in hydraulic fluid, leakage of fluid from drive motor, vibrations of ballnut screw actuator, excessive rate of wear of ballnut screw, deceleration valve seizure, regulating valve seizure, breakage of fluid supply lines, loss of system pressures, burst accumulator, foaming of oil, excessive oil leakage, fire in subpile room, lateral shift of vessel relative to biological shield, lateral shift of grid plate relative to reactor vessel, and lateral shift of core relative to the grid plate.

For each of the above possible malfunctions the applicant has furnished sufficient information to provide reasonable assurance that occurrence of the malfunction is not credible or that it would not result in an unsafe condition. Much of this information is based on the supporting research and development program, as discussed below.

With regard to the irradiation stability of graphite components, graphite-B_4C specimens have been irradiated to neutron exposures equal to or greater than those expected in the reactor and at temperatures which would equal the maximum expected. While the data from the tests conducted thus far indicate that dimensional and structural stability are adequate, a continuation of this program is planned.

Both irradiation and out-of-pile tests have been conducted and are continuing on graphite-B_4C control materials. The irradiation tests are intended to determine dimensional stability; thus far a test conducted at 1200°F (the maximum temperature to be encountered in operation), fast neutron exposure of 10^{19} nvt and thermal neutron exposure of 10^{20} nvt indicated that dimensional changes of only about $\pm$ 0.1% can be expected. Programs have been initiated to increase the exposures in order to obtain additional information. Other tests have been conducted and will be continued to determine the volatility and boron release rates from the control materials as a result of potential temperature excursions. Samples of B_4C-graphite compacts which were warm pressed at 1375°F and vacuum sintered at 3630°F were heated in a helium stream at 3885°F for 16 hrs. The amount of boron loss was about 0.5%. Specimens which were not vacuum sintered showed substantial boron losses due to impurities contained in the compacts, including oxides of boron which are considerably more volatile than boron carbide.

During operation the graphite guide tubes will be subjected to a fast flux of about 10^{13} and the wall temperature will range from 660°F to 1400°F. Tests have been conducted to examine the effects of these conditions. Tests to determine dimensional stability and physical characteristics indicated that no measurable dimensional changes had occurred during exposure of the tube material to 2372°F for 1000 hours. Creep experiments at 2250 psi tension and 1300°F indicated a creep rate of 10^{-9} in/in-sec for the first 300 hours with no change during the next 300 hours. Based on these results it is now calculated that a guide tube exposed to the worst conditions of pressure of adjacent fuel elements would suffer deformation of only 0.01 inch. To determine the dimensional effects of irradiation, the guide tube material was subjected to a fast neutron exposure of about 10^{21} at temperatures from 840°F to 1560°F. Contraction of the diameter was measured as 0.02% to 0.04% and contraction in length was 0.06% to 0.08%. Additional specimens will be irradiated. The maximum load to which a control rod guide tube will be subjected is 2500 lb due to the force applied by a binding control rod. The point of expected failure is the breach-type connection where the tube is anchored to the grid plate. Shear tests on this section indicate a factor of safety of 2.3 minimum and 2.43 maximum, which appears to be adequate.

Other tests on the guide tube included a series of environment tests on material combinations to determine wear and lubricating qualities.

A test was conducted to determine the dynamic behavior of various ball nut and screw configurations for determining stability, critical speeds, friction, and wear data. During these tests the prototype rod was subjected to 9000 full-stroke cycles and 800 scram cycles, in environments of both air and vacuum. The test results did not show any vibrational effects and the parts were virtually unworn.

A prototype control rod has been tested under simulated operating conditions to confirm calculations for metal component temperature (maximum allowable - 1000°F), helium flow rate and helium pressure drop. The rod was tested at various simulated core positions and various flow rates and the test data confirmed the calculations.

An over-all test of the prototype control rod and drive unit is presently being conducted. The rod is a full size prototype and the test system is provided with complete control and monitoring instrumentation such as proposed for the actual plant installation. Additional instruments are used to control and record test variables. The test is conducted at simulated reactor conditions including pressurized helium, expected temperatures, temperature gradients and various tube misalignments. The test program provides for subjecting the rod and drive assembly to at least 10^6 inches of random motion at normal speed and at least 5000 scram operations which is about five times as many cycles as expected in the life of the rod. Initial testing in air at room temperature for 2000 full strokes and 200 scram tests has revealed no design problems or wear effects. During these initial tests the simulated instrument response time was between 40 to 50 msec as compared to the design allowance of 75 msec.

On the basis of technical information furnished concerning the control rod system, including the results of the research and development program, there is good assurance that this system will function in an acceptable manner. A suitable method has been proposed for indication of control rod separation, it has been demonstrated that the materials should retain their integrity under various service conditions, and the satisfactory operation of the integrated assembly has been demonstrated.

Two independent emergency shutdown systems are provided to compliment the scram function of the normal control rods. These systems include a group 5
of 19 electrically driven shutdown rods and a minimum of 19 thermally released (fused) gravity-drop absorbers. They are considered a significant asset to the reliability of the shutdown capability of the reactor because of the lack of actual experience with the proposed type of control rod system.

The electrically driven emergency shutdown rods are intended to serve two purposes: (1) additional shutdown capability is provided in the event that sufficient normal control rods can not be inserted to provide shutdown in the cold condition; and (2) emergency shutdown capability is provided in the remote event that the graphite core could become disarranged such that the control rods could not function. The total worth of these emergency rods at operating temperature is calculated to be 16% k, compared to a worth of about 23% k for the normal rods. The drives are designed to insert the rods with a force of 10,000 pounds, sufficient to force the rods through the core material under any circumstances. The absorber section consists of a stiff metal tube filled with refractory neutron poison which passes within a graphite guide tube. The drive system includes an electric motor, gear reduction drive, acme drive screw and a nickel-cadimum battery as a power source. This safety rod is provided with a continuity circuit similar to that described for the control rod. It is planned that extensive testing of a prototype assembly will be undertaken.

A second emergency shutdown system is provided in the form of thermally released absorber rods. These rods will be released in event of excessive coolant temperature. The minimum worth of these rods will be sufficient to compensate for maximum xenon decay - about 3%k. The rods consist of short absorber sections normally held in the top reflector by a metal tie bar and fusible link. Overtemperature of the core will cause the link to melt and release the absorber rod to fall into the midplane of the core. Tests are planned to determine characteristics of this device.

The Staff believes that the safety and emergency shutdown capabilities proposed for the Peach Bottom reactor are adequate under any credible circumstances. In any foreseeable situation it appears that at least one of the three groups of rods will be available for emergency service. Although prototypes of the electrically driven rods and thermally released rods have not yet been tested, there appears to be no reason why the proposed mechanisms cannot be constructed to operate satisfactorily.

The reactor instrumentation system consists primarily of instrumentation to monitor neutron flux, condition of the control rod and safety rod systems, and cooling system parameters. All normal and emergency instrumentation for the entire plant is located in the control room which is located in an auxiliary building adjacent to the reactor building.

Neutron flux is monitored by nine channels of instrumentation including three low level count rate channels for use during startup, two intermediate level channels which also furnish period information, and four power level channels which monitor the flux in the range 1-100% of full power. Fail safe criteria have been specified for all nuclear instrumentation components and all duplicate channels are cross-compared with significant deviations annunciated. It is proposed that the electronic components will be of solid state design to enhance reliability. Multiple circuits will be arranged to permit functional testing of a single channel without interfering with the safety function of the remaining channels. The neutron detectors are installed in water cooled thimbles located in the concrete shield adjacent to the reactor vessel.

The control rod system is instrumented for the purpose of detecting and indicating the position of significant components and possible malfunction conditions. Position monitors are provided for the control rods, scram valves, scram accumulator pistons and rod holding locks. The pressure in the low and high pressure hydraulic supply systems is monitored continuously with scram protection provided if the high pressure supply decreases to 85% of the normal value. Other control rod system monitors include those for clogging of filters and strainers, fluid leaks in the rod drive housings and past the scram accumulator pistons, low flow for buffer helium and pressure, temperature and level conditions in the hydraulic storage tank.

The electrically driven shutdown rods are actuated manually by the reactor operator. A single switch actuates all rods. Limit switches provide indication of extreme rod positions and the condition of the storage batteries is monitored continuously.

Instrumentation in the helium coolant loop provides for detection of the loop pressure, flow rate, temperature, activity level, gas leakage, moisture content and gas analysis at various locations in the system. Because of the damaging effect of moisture on the hot graphite core materials, provision is made for rapid detection of moisture in the helium coolant. Three detector located at each steam generator continuously monitor the moisture content of the helium. Two trip levels are provided for each detector output - an alarm for low moisture content and a scram for high moisture content. These detecto will respond to a high moisture level within 5 seconds and are capable of detecting moisture leaks into the helium as low as 0.01 lb/hr. In addition the helium coolant is to be sampled intermittently for trace impurities of CO, CO_2, O_2, N_2, CH_4, and argon.

The Peach Bottom Atomic Power Station is primarily a manually loaded generatin station with capability to follow automatically the turbine load. Constant steam conditions are maintained automatically at the turbine inlet. Automatic control loops vary helium coolant flow rate to maintain the steam pressure and vary reactor power to maintain the steam temperature. Automatic rod control is provided by 3 cascaded controllers: the steam temperature controller actuates the reactor helium outlet temperature controller (controls helium flow rate), which in turn actuates the neutron flux controller (controls rod position). Manual operation of all automatic control loops is available as a backup for controller failure. Although no details of the automatic control system have been provided, the Staff considers the design relatively straightforward and finds no reason why the proposed system cannot be developed.

The helium outlet temperature and neutron flux controllers are provided with high and low setpoint limits such that malfunctions or deviations from normal limits cause an alarm or reactor scram. Manual control only is available during reactor startup and low power operation. Rod operation will be limited to simultaneous movement of 3 rods in the outer two rings and two in the inner rings, including the automatic rod. It is proposed that no more than 3 rods will be in intermediate positions during power operation with the remainder either fully inserted or fully withdrawn as conditions require. This operating limitation would prevent more than 3 rods from dropping out of the core as a result of any malfunction.

Helium loop control variables include the pressure, temperature and flow rate. Helium loop pressure is maintained at 335 psig by balancing several inflows (such as to the buffer seals) with the fuel element purge outflow. Overpressure protection for the helium coolant system is provided by both dump and relief valves. At a loop pressure of 390 psig the dump valves open to relieve helium to the dump tanks; these valves can also be operated manually. Further overpressure protection is provided by a reactor relief valve to be set at 437 psig and a relief valve on each steam generator to be set at

450 psig. Each relief valve is provided with a full capacity spare. The entire primary system is designed for 450 psig at operating temperature.

The plant protective system includes 3 types of logic to provide trip signals: single trip with backup protection, two-out-of-three logic for trip from most measured variables and two-out-of-four logic for highly critical variables such as reactor flux. In the latter case four channels monitor the same variable with pairs of channels fed from each of two power supplies; hence scram results on simultaneous trip of two channels fed from different power supplies. Failure of one power supply would not cause scram action but provides one leg of the required signal. Conditions causing reactor scram inserting all 36 control rods at the maximum rate include: a high flux excursion initiating two-out-of-four power channel trips, high reactor outlet temperature measured by two channels in each reactor loop using two-out-of-four logic, high moisture detection from three detectors in each loop with two-out-of-three logic in either loop, low helium pressure from three detectors in main coolant system using two-out-of-three logic, low pressure in the control drive high pressure header, high rate of change in neutron flux in both channels, and isolation of either helium loop will effect scram.

Provisions are included for isolation of each helium loop under the following conditions: loss of helium compressor, loss of boiler feedwater, tube rupture in steam generator as detected by the moisture detector and rupture of the main coolant piping as indicated by radioactive gas detectors. The accompanying reactor scram is intended to minimize temperature transients. Interlocks permit only one loop to be isolated at a time in order that core cooling is continued. Simultaneous with helium loop isolation, the corresponding steam loop is also isolated automatically.

The Staff has examined the conceptual plans submitted for the reactor instrument system, including the related protective functions, and considers the proposed system adequate. It appears that the design includes due consideration for requirements of reliability, monitoring of significant parameters and protection against foreseeable malfunctions. It is anticipated that this system will be reviewed in greater detail when the design and component testing is completed and additional system characteristics are known.

D. Nuclear Design

The nuclear design of the reactor involves the analysis of reactor materials and arrangements for control effectiveness, power distribution, temperature coefficients and reactor kinetics. The following tabulation includes the pertinent characteristics of the proposed design:

Reactor power (output of fuel elements)	112.5 Mw
Nuclear heating of reactor internals	2.8 Mw
	115.3 Mw(th)
Effective core diameter	9.16 ft
Active core height	7.5 ft
Number of fuel elements	804
Number of control rods	36
Number of emergency shutdown rods	19

Initial fuel loading	173.3 kg U-235
Initial thorium loading	1987 kg
Initial boron burnable poison	0.950 kg
Initial rhodium loading	5 kg
C/Th/U atom ratio	
696 inner fuel elements	2126/9.57/1
108 outer fuel elements	3511/24.46/1
Average moderator temperature	1820 F
Average fuel compact temperature	1985 F
Maximum fuel compact temperature	2732 F
Fuel life at full power	900 days
Excess reactivity of core at 1700°F (clean)	8% k_{eff}
" " " " " " (equilibrium poison)	5% k_{eff}
Worth of 36 control rods (1700°F)	23% k_{eff}
Worth of 19 emergency rods (1700°F)	16% k_{eff}
Maximum peak-to-average power ratio	1.5

The following objectives have been established for purposes of reactor control:

1. The core loading will include fuel, thorium, burnable poison and rhodium poison such that the maximum k_{eff} at operating temperature will be 1.08.

2. Only three control rods will be in a partially inserted position at one time, in order to obtain good axial power distribution.

3. The k_{eff} with all control rods inserted at shutdown temperature (80°F) will always be less than 0.95.

4. The k_{eff} with any one control rod withdrawn will always be less than 0.97.

5. Boron burnable poison will be used to achieve a uniform reactivity variation throughout core life.

6. Rhodium poison will be added to the core to enhance a strong negative temperature coefficient.

The above basic core nuclear objectives appear to represent an adequate degree of design conservatism. In particular, it is noted that the relatively low available excess reactivity and adequate shutdown margins should enhance safe reactor operation.

Reactivities for the various reactor shutdown conditions have been calculated and the values for control rod effectiveness have been checked by critical assembly measurements. The more significant results of this program are tabulated below:

	k_{eff} (80°F)	k_{eff}(1700°F)
All control rods out All shutdown rods out No Xe or Sm	1.19	1.08
All control rods out All shutdown rods out Equilibrium Xe and Sm	1.16	1.05

36 control rods in All shutdown rods out No Xe or Sm	less than 0.93	less than 0.85
No control rods in 19 shutdown rods in No Xe or Sm	1.02	0.92

In order to assure that the indicated shutdown margins can be accomplished, additional fuel elements containing burnable poison will be provided to permit final adjustment of the reactivity.

It is proposed to maintain a relatively flat radial power distribution by diluting the fuel concentration and increasing the thorium concentration in the outermost ring of 108 elements. This feature will reduce the power peak at the core-reflector interface due to thermal neutrons returned to the core by the reflector. Power peaking in the axial direction will be minimized by maintaining the maximum number of rods either fully inserted or withdrawn, by incorporating burnable poison in the top reflector and by the use of partial length burnable poison in the core. As a result of these features and the proper programming of control rods, it will be possible to maintain a maximum peak-to-average power ratio in the core of 1.5 throughout life. now 1.57 This low value of the power peaking ratio should contribute considerably to better core performance characteristics, including low temperature ratios, low burnout ratios, low thermal stresses and increased Doppler coefficient.

The temperature coefficient of reactivity has been studied extensively in the research and development program. Relative to reactivity coefficients, it is noted that this type reactor would not normally be expected to exhibit void coefficients; consequently dependence for inherent shutdown characteristics must be placed on temperature coefficients. In the Peach Bottom reactor the most significant temperature effects arise from the temperature dependence of the neutron thermal utilization and the Doppler broadening of the thorium resonances (Doppler effect). A strong Doppler effect is important inasmuch as it serves to rapidly reduce the core reactivity during conditions of increasing fuel temperature. The over-all temperature coefficient performs a similar function, but with a considerably delayed effect (about 3 minutes).

The Doppler coefficient was determined by calculational methods and the results checked experimentally. In the experimental program the reactivity of hot and cold fuel elements, both with and without thorium, was determined by pile oscillator methods. On the basis of these experiments it is believed that the calculated values have uncertainties no greater than 10%.

In order to enhance a strong negative over-all temperature coefficient, it is proposed to include about 5 kg of rhodium in the graphite fuel element spines. Rhodium exhibits an absorption peak of about 5000 barns for 1.25 ev neutrons. The result of this feature is a strong contribution to the over-all temperature coefficient and some contribution to the prompt coefficient.

The reactivity coefficients at normal operating temperatures are given below:

	Beginning of Life ($10^{-5}/^{o}C$)	End of Life ($10^{-5}/^{o}C$)
Prompt temp. coeff. (fuel compact - 1985°F)		
No xenon	-6.21	-3.65
Equilibrium xenon	-4.91	-2.35
Over-all temp. coeff. (fuel element - 1817°F)		
No xenon	-6.46	-4.50
Equilibrium xenon	-5.16	-3.35

The kinetics of the core have been studied by means of a digital computer program in order to determine the effects of various reactivity transients. In this program the peak fuel element temperature was determined for various conditions of reactivity insertion and scram delay time. The results indicate that long intervals of time are available to scram the reactor before excessive fuel element temperatures develop. The most severe condition considered involved the addition of 2% delta k at the normal rod withdrawal rate (.06 ft/sec) and scram delay times of 0.075 sec, 30 sec, and infinity; the resulting average fuel temperatures were 1976°F, 2372°F and 2912°F, respectively. The maximum fuel temperature in the hottest fuel element was calculated to be 4228°F, which is below the fuel carbide melting temperature (4532°F) and the vaporization temperature of graphite (7200°F). Consequently, it appears that the kinetic behavior of the reactor is adequate to prevent damage from core overheating as a result of credible reactivity incidents.

In reviewing the information on nuclear design presented by the applicant, the Staff notes that core parameters appear to be conservative, strong negative temperature coefficients of reactivity are expected and reactor kinetic calculations indicate satisfactory behavior during reactivity transients. Consequently, it is concluded that the reactor should demonstrate satisfactory operating characteristics.

E. Reactor Cooling Systems

Three separate systems are provided for cooling of the reactor under various conditions including the primary cooling system, the afterheat cooling system and the emergency cooling system. The primary system is designed to function during power operation and for periods up to one hour following shutdown from power operation; the afterheat cooling system is designed to remove decay heat generated subsequent to one hour after shutdown; and the emergency system functions when the normal means of cooling are denied.

The primary cooling system includes two identical loops each with a steam generator, helium compressor and associated piping and auxiliaries. Helium flows from the reactor to the steam generator through the inner passage of a concentric pipe; the annular passage between the two pipes serves for return of the cooler helium from the compressor to the reactor. The inner pipe is 30 inches and the outer pipe 42 inches in diameter. Both pipes are constructed of carbon steel. The inner pipe is designed to withstand interna or external loss of pressure and will be thermally insulated on the inside surface to maintain the wall temperature below 700°F.

Isolation valves are included in the reactor outlet pipe and at the inlet and discharge of the compressor. The reactor outlet valve is of the gate type and is installed in the concentric piping such that the valve disc close: only the inner pipe. The cooler return gas flows around and cools the valve body. This valve is actuated by a helium actuator which can close the valve in 3 seconds. During operation the maximum temperature at the seat, disc and barrel will be less than 900°F, and the inner and outer bodies less than 700°F The maximum temperature under any abnormal circumstance is expected to be less than 1000°F. The hot leg isolation valves serve the following functions: to isolate a rupture in the helium loop, to isolate a leak in a steam generator, to permit maintenance on a loop and to permit isolation for testing purposes.

Each steam generator is a vertical shell and tube type unit containing separate economizer, evaporator and superheater sections. Hot helium enters the steam generator and consequently the superheater section at 1352°F, entering the evaporator section at 1129°F, entering the economizer section at 749°F and exits the steam generator at 620°F. On the steam side, evaporation occurs at 604°F and 1578 psig and the discharge from the superheater is at 1005°F and 1544 psig. Materials of construction for the steam generator are as follows: shell - SA-212-grade B carbon steel, superheater tubes - SA-313 type H stainless steel, evaporator tubes - SA-192 carbon steel and economizer tubes - SA-179 carbon steel. Helium flow within the steam generator is arranged to pass the cooler exit helium over the shell wall.

As previously discussed, moisture would have a deleterious effect on the core materials. The most probable source of water in-leakage to the helium coolant is leakage past the tube sheet welds. In order to detect and remove any such moisture a baffle is provided immediately below the tubesheet; 100 lb/hr of helium is purged from the space between the baffle and tubesheet and monitored for moisture content. In addition, 3 moisture detectors are provided for each helium loop which are capable of detecting a water leak of 0.01 lb/hr.

The helium piping which connects the steam generator outlet to the compressor and the compressor to the outer concentric reactor inlet line is 30 inches in diameter and of carbon steel. Gate type isolation valves are provided in both inlet and discharge pipes of the compressor. These valves are rapid closing, actuated by a helium actuator similar to that for the reactor outlet valve.

Each helium loop includes a horizontal single stage compressor to circulate the gas. The design rating is 33,800 cfm at 325 psig and 628°F, with a shaft speed of 3500 rpm. Inasmuch as the compressor is a vital component of the circulation loop, the machine is provided with features to monitor its performance and enhance continuous operation. For each compressor, pressures, temperatures, flows, motor current and vibration will be monitored continuously. The rotating shaft is sealed by both a mechanical and a helium buffered, oil flooded, floating bushing seal. A separate seal-oil supply system is provided for each compressor. Each seal-oil system includes two electric motor driven pumps. One pump serves as a standby and is started automatically on loss of oil pressure; in addition the standby is operated from an emergency power supply. The pressure lubrication system for the compressor and motor bearings is also equipped with dual electric motor driven pumps and emergency standby power. A variable speed fluid coupling connects the compressor shaft to the compressor drive motor. Each coupling is provided with an oil supply system including a shaft driven pump and an electric motor driven pump which starts automatically on loss of supply pressure. The compressor is driven by an electric motor; a backup motor is provided which operates from the emergency power supply system to maintain cooling in event normal power is disrupted.

Each helium loop is equipped with a cyclone type filter assembly to remove any graphite dust or particulate matter from the helium. Bypass flow through these filters is one per cent of the loop flow. The filter is designed to remove 92% of particles larger than 5 microns. Material collected by the filters will be monitored for activity. Shielded containers will be provided for disposal of any activated material.

The primary helium loops are intended to remove reactor heat during normal power operation and for the first hour following shutdown. The afterheat removal system is intended to cool the reactor during subsequent periods. Afterheat removal involves normal circulation of the helium; however, the helium is now cooled by sub-cooled water passing through the coils of the evaporator. This sub-cooled water transfers heat to the plant service water in a separate heat exchanger. The capacity of the system is 7 million BTU/hr, equivalent to about 2 Mw.

The emergency cooling system operates on the concept of transferring heat by radiation and convection from the reactor vessel to the water-cooled liner of the reactor cavity. The liner consists of a steel plate which totally covers the cavity walls, except where reactor nozzles penetrate. A series of one inch diameter water cooling coils is welded to the outside of the steel plate. Insulation separates the liner from the adjacent structural concrete which contains imbedded cooling coils to remove heat from the concrete. The steel cavity liner is separated from the reactor vessel by 21 inches on the sides and 12 inches at the top and bottom. Conventional plant service water is passed through the cooling coils to remove the heat.

The application includes an analysis of the performance of emergency cooling system. System temperatures were calculated on the basis of the following assumptions: the reactor has been operating continuously for three years at 124 Mw, all normal cooling is lost and the reactor is shutdown. The results indicate that the maximum temperature of the reactor vessel would be about 867°F. Other results indicate that the maximum core temperature would be 3600°F occurring about 30 hours after shutdown. Some damage may occur to the metal components of the core and to the core support plate which reaches a temperature of 1300°F. However, the reactor vessel would not be damaged.

The Staff has reviewed the design criteria for the various reactor cooling systems and considers the proposed systems adequate to provide the necessary cooling. In addition, it is noted that the various system components will be designed and tested in accordance with applicable codes. The isolation valves in the primary loops are capable of closing rapidly to minimize helium leakage as a result of loop rupture and to prevent moisture admission to the core. Design of the emergency cooling system appears to be conservative, providing assurance that the emergency cooling is adequate. Consequently, the Staff has concluded that the proposed designs are adequate.

F. Pressure Vessel

The reactor pressure vessel is a vertical cylinder with elliptical heads; principal dimensions are inside diameter - 14 feet and over-all height - 35 feet-6½ inches including the heads. The upper head is flanged and removable for insertion of the reactor internal structure. During normal operation this flange will be seal-welded to minimize helium leakage. The vessel will be made of ASTM A-212 Grade B carbon steel and will be nominally 2½ inches thick except for a 4½ inch thick band in the region of the main coolant nozzles. No internal cladding is provided. The vessel is supported in a steel lined cavity of the biological concrete shield by a support skirt near the same plane as the main coolant nozzles. The design pressure and temperature for the vessel are 450 psig and 725°F.

Five vertical nozzle penetrations are installed in the upper head to provide access ports for fuel handling operations. These ports are arranged with one 20-inch diameter port located on the vessel centerline, and three 10-inch and one 6-inch diameter ports located 59 inches from the center. These five nozzles extend upward from the vessel head, through a thick concrete shield to the operating floor level. During reactor operation these ports contain shield plugs and are closed with blind flanges. The function of these nozzles will be discussed in the section on Fuel Handling.

Fifty-nine nozzles having an inside diameter of 4 inches penetrate the bottom head of the vessel. These nozzles will accommodate the 36 control rods, the 19 emergency shutdown rods, and 4 fission product purge lines and instrumentation conduits.

Two 42 inch nozzle penetrations are provided in the sidewall of the vessel; these are located just below the upper flange in the 4½ inch thick wall section

Design of the vessel will conform to appropriate ASME and Pennsylvania State codes, including the latest nuclear case rulings. Detailed engineering will be guided by appropriate standards set forth for reactor vessel design by the Department of U. S. Navy, Bureau of Ships. Prior to fabrication of the vessel, all materials will be subjected to physical tests and ultrasonic inspection in accordance with appropriate ASME and Military Standards. During fabrication, quality control will be maintained by both the designer and the fabricator. The vessel will be inspected by means of magnetic particle, liquid penetrant, and radiographic methods in accordance with the applicable codes. Hydrostatic pressure tests and helium pressure and vacuum tests will be conducted following completion of the pressure vessel.

During reactor operation, two hundred specimens of the vessel materials will be placed in radiation fields similar to that of the reactor vessel. Specimens will be extracted periodically for examination to provide a continuing check on the integrity of the vessel materials.

The detailed one-half scale hydraulic model test program which was previously discussed in the section on Core and Reflector Design, will contribute data to be used in the detailed analysis of the vessel design.

It is the opinion of the Staff that the proposed design critieria for the reactor pressure vessel are compatible with the operating requirements. In particular it is noted that accepted practices will be followed during construction and testing.

G. Fission Product Control

As discussed previously the fuel and fertile material are dispersed in bare graphite compacts and encased in a graphite sleeve. Inasmuch as this graphite has a finite, although very low permeability, it is necessary to provide means to prevent the diffusion of fission products into the helium coolant. A large amount of activity in the coolant would create operational problems and constitute a hazard if released from the system. Four features are incorporated to limit the level of contaminatic of the helium coolant including the pyrolytic coating of the fuel particles internal traps, external trap system and the low permeability graphite sleeves.

Release of the volatile fission products from the fuel compacts is delayed until the short-lived isotopes have decayed. This is accomplished by the dense carbon coating on the fuel particles; as discussed in the section on Fuel Elements. Non-volatile fission products will be retained in the fuel compacts. Most of the condensible products which do escape the fuel compacts are removed in an internal trap located in the cold end of the fuel element. This trap is immediately below the fuel compacts, within the low permeability outer sleeve, in the relatively cool lower graphite reflector region. The internal trap consists of a 12 inch long, 2.75 inch diameter, hollow cylinder with slots lengthwise on the outer surface. The slots are filled with silver coated charcoal reagent material in granular form which retains fission products by means of condensation, adsorption, and reaction with silver. When fission products are released by the fuel compacts they are carried by the purge helium flow to the internal trap. The tellurium, cesium, barium, strontium, antimony, and rubidium fission products which reach the internal trap are retained completely; the iodine and bromine fission products are delayed for at least 32 days. Krypton and xenon, along with delayed iodine and bromine, are carried on to the external trapping system. The purge helium exits the core via the fuel element stand-off pins into two parallel connected manifolds. The manifolds operate at about 3 psi below the main coolant pressure to insure that any possible leakage is into the manifold and not into the core region. The purge line which connects the manifolds to the external traps is a colinear double pipe with uncontaminated helium in the outer pipe and the purge gas in the inner pipe. The outer pipe is kept at slightly higher pressure to insure in-leakage.

The purge flow from the core (1000 lbs/hr) is first cooled to room temperature. It then flows in series through a water cooled charcoal bed, for halogen removal; through two room temperature and five freon-cooled delay beds having a total charcoal inventory of 16,200 lbs, providing a krypton delay of 189 hours and a xenon delay of 34,300 hours; and finally a portion (100 lbs/hr) is passed through two nitrogen cooled traps, for removal of krypton and contaminants escaping the previous traps. The decontaminated helium is returned to the primary cooling system. This trapping system is designed to maintain a primary helium purity of less than 50 curies of krypton-85. The trapped Kr-85 is removed periodically in appropriate shipping containers for off-site disposal.

As described previously the function of the low permeability graphite sleeve is to restrict the diffusion of fission products to the main coolant.

An extensive experimental program has been performed to provide quantitative information relating to the above four means of limiting fission products in the primary system. Experiments on fuel compacts to measure the release of individual fission products at elevated temperatures and various degrees of burnup include the following: (a) tracer experiments in which lightly irradiated compacts were heated and the fission product release was measured as a function of time, (b) purge capsules in which the release of fission products was measured directly during irradiation, (c) measurement of the change in release rate of volatile fission products following accelerated irradiation of compacts to expected reactor burnup, and (d) investigation of the characteristics of the integrated trapping system in a prototype loop. As a result of this program the volatility characteristics of the 35 predominant fission product elements have been determined. Seventeen of the 35 are non-volatile at reactor operating temperatures and are expected to remain fixed in the fuel compact; in fact, 12 of these form stable carbides. The remaining 18 are either gases or have appreciable vapor pressures at operating temperatures and must be retained in the trapping system. In addition to this study information on the trapping of iodine, krypton and xenon at Hanford, Oak Ridge, NRTS Chemical Processing Plant, and Great Britain has been reviewed.

Experimental fission product release data has been obtained using both coated and uncoated fuel particles. In one program fuel compacts containing ThC_2-UC_2 particles 50-60 microns in diameter and with 23 microns of pryolytic carbon coating are being irradiated in capsules which are purged continuously to monitor the activity as an indication of holdup capability. These have been under irradiation since November 1960. The results to date indicate that the steady state release of xenon and krypton at 3092°F fuel temperature is less than 1% for the short-lived isotopes (shorter than 2.8 hr) and up to about 3% for the longer-lived species, not including Kr^{85} which indicated a release of 20% relative to its rate of production. Additional capsules of this type are now being irradiated.

The effect of irradiation and fission product accumulation on the integrity of pyrolytic carbon coatings has been investigated by irradiating graphite matrices containing fuel particles with 25 micron thick coatings to equivalent reactor burnup conditions. The matrices were then annealed at 3092°F for 60 hours to enhance the escape of fission products. This experiment showed failure of about 18% of the pyrolytic coatings while 82% maintained their retentive properties. These particles had coatings of only one-half the thickness (25 microns) of that for the anticipated design (50-60 microns). This program is being continued in order to obtain more definitive information about the integrity of the particle coating.

Experiments have been performed to determine the diffusion time of fission gases from fuel compacts with uncoated particles as a function of temperature. The method involves exposure of the compacts to neutrons until about 10^{13} fissions occur, followed by a decay period, and then annealing for 24 to 48 hours during which the Xe^{133} in the purge gas is monitored as a function of time. In addition, a series of steady state release-rate experiments on Kr and Xe has been conducted using a linear accelerator as a neutron source. In these experiments continuous release rates were determined with the fuel compacts at temperatures of 1652-3632°F.

Similar to the tests described above on uncoated particles, experiments have been performed and are continuing with coated fuel particles. Many post-annealing tests have been made to determine the Xe^{133} release from coated particles at temperatures up to 3632°F. Only a very small release was noted, and this release is believed due to incompletely-coated particles. In addition, possible long term reactions between the carbide compact material and the coating were studied. Tests conducted at 3092°F for 1500 hours showed insignificant reaction.

To simulate the effects of fission product release due to temperature excursions in the reactor, post-irradiation annealing type experiments have been carried out on both coated and uncoated fuel particle compacts. In the case of uncoated fuel particle compacts the post-irradiation temperature was increased from 2552 to 4136°F in less than one minute, held at 4136°F for five minutes, and then reduced to 3362°F for 16 hours. At the end of 16 hours 46% of the Xe^{133} was still retained in the graphite body. Following this the compacts were raised to 4712°F (above the melting point of carbide fuel); within two hours the remaining Xe^{133} was released. Similar experiments were conducted with coated fuel particle compacts, but at more severe temperatures. The temperature of the compacts was raised from 2552 to 5432°F in several minutes and held at 5432°F for 30 minutes; during this period about 20% of the Xe^{133} was released. The compacts were then brought to room temperature, wherein all of the Xe^{133} was released, indicating that cracks were produced by the rapid cooling from 5432°F to room temperature.

As a result of the above described experiments on fission product release characteristics of fuel particles and compacts, it is believed that sufficient information is now available to serve as an adequate basis for the requirements of the trapping system.

The performance of the internal trap located in the lower section of the fuel element was evaluated by studying its capability to remove iodine, cesium and tellurium. Since the temperature of the internal trap (900°F) renders it ineffective for rare gas adsorption, krypton and xenon were not included in this study. For the studies involving cesium, activated charcoal was chosen as the trapping material and investigations were made of the specific sorption (gms Cs/gms C) of various charcoals as a function of temperature and cesium vapor pressure. A cesium isotope of known activity was passed over the charcoal and the sorption rate on the charcoal was continuously monitored. From this work a mathematical expression for specific sorption was obtained and verified by tests. This procedure permits calculation of the specific sorption of the cesium for various conditions of temperature and pressure.

In order to study the adsorption of iodine and tellurium, a dynamic experimental apparatus was constructed which simulated actual internal trap operating conditions. Investigation of the characteristics of different metals and combinations of metals and charcoal resulted in the determination that migration of the iodides was significantly suppressed by the presence of charcoal. About fifty tests were conducted at iodine partial pressures similar to those expected in the actual purge stream (5×10^{-6} atm upper limit). In these tests the capacity of various trap materials to delay the iodine and tellurium under various conditions was measured in terms of "Reactor Equivalent Delay" (in days). The "RED" values give an indication of the time that a particular fission product atom is expected to remain in the trap. Results of these tests indicate that the RED value for iodine is 97. With respect to tellurium, the tests indicated that essentially none of this element could be detected downstream of the experimental traps. On the basis of these results the present design of the internal trap including size and materials was established.

Tests also have been conducted to determine the performance of charcoal as a rare gas adsorber. Krypton and xenon adsorption data have been obtained from dynamic adsorption experiments under conditions applicable to an external trapping system. These experiments involved the determinatio of the dynamic adsorption coefficient (cc gas adsorbed/gm) for various grades of charcoal and at conditions of temperature and pressure similar to those expected in reactor operation. The results of these tests have permitted the establishment of firm design criteria for the external traps.

The experimental programs described above were intended to provide basic design data for trapping systems. In addition, a fuel test loop incorporati a section of the design fuel element and a complete prototype trapping system has been placed in operation to provide performance data under simulated operating conditions. This loop incorporates a quarter-length prototype fuel element consisting of 15 fuel compacts containing pyrolytic coated fuel particles, with a total active length of 22.5 inches. Below the fuel section is the internal trap which contains the silver coated charcoal reagent discussed previously. Two purge lines connect the fuel element to the external trapping system such that the element purge flow may be collected ahead of the internal trap or after the internal trap. During loop operation the element will receive a thermal neutron flux of about 10^{13} nv and will generate 76 Kw of fission and gamma power with a heat flux of about 150,000 BTU/hr-ft^2. Although the loop is designed to operate over a range of temperatures, flow rates and pressures, the typical operating conditions are: fuel element temperature-3060°F, sleeve temperature-2200°F, entrance gas-600°F, and exit gas-1400°F; these are substantially similar to conditions in the Peach Bottom reactor.

The main components of the test loop external trapping system include a water cooled charcoal trap to remove volatile fission products other than the noble gases, a freon cooled charcoal trap operating at -40°F to provide a holdup period for the noble gases, a copper oxide bed to convert CO to CO_2 and H_2 to H_2O, a liquid nitrogen cooled trap to remove the CO_2 and H_2O formed in the oxide bed, a charcoal trap cooled by liquid nitrogen to -320°F to retain the noble gases delayed in the -40°F trap, and two emergency back-up charcoal traps. The purge gas flow rate through the external trap system from the test fuel element will be about 0.6 lbs/hr, compared to the expected reactor fuel element purge of 1.1 lbs/hr. Operation of the loop will provide fission product data as follows:

a) The purge line will be monitored continuously upstream and downstream of the first water cooled charcoal trap to provide data on the release of activity from the fuel element as a function of irradiation and data on the over-all effectiveness of the trap.

b) By taking periodic samples from various points throughout the loop and analyzing for radioactive isotopes and non-radioactive impurities such as O_2, H_2, CO, CO_2, and CH_4, the effectiveness of the various components can be evaluated.

c) Removable sections of the loop are included to allow determination of the plate-out of fission products in the loop. Chemical and isotopic analysis will be performed on plated material.

Because the long term burn-up experiments duplicating end-of-life conditions have not in all cases been completed, the design requirements of the traps have been conservatively based on a factor equivalent to three times the activity and heat load values corresponding to uncoated fuel particles. Data collected to date on coated fuel particles at a burnup equivalent to 50% of the end-of-life value indicate that the coating on the particles decreases the amount of fission products released from the fuel compacts by a factor of 2.5. On this basis the present design of the fission product traps should have a design conservatism factor of about 7.5.

The main helium system is expected to contain some chemical contaminants which must be removed in order to maintain minimum loop activation and deterioration of the core graphite. The primary chemical contaminants are expected to be N_2, A, H_2, CO_2 and water vapor. Sources of these contaminants include residual air unpurged from the system during initial filling, air entering the system during maintenance operations and moisture in-leakage from the steam generators. In order to remove these chemical contaminants a chemical cleanup system is incorporated into the helium purification system. This system consists of an oxidizer to convert CO and H_2 to CO_2 and H_2O, a condenser to remove most of the H_2O, two molecular-sieve beds to remove CO_2 and H_2O and a plate-out bed to remove any remaining condensible material. The effluent from the chemical system is then passed through the fission product trapping system to remove any residual fission product activity.

On the basis of the data collected to date from experiments it has been calculated that the distribution of fission products should be as follows: 98.3% retained in the fuel elements, 1.4% held inthe internal traps and 0.3% in the external traps. For comparison, the corresponding values used for design of the trapping systems (designed on the basis of uncoated fuel) are 90.45% in fuel, 7.17% in the internal traps and 2.38% in the external traps. The Staff considers this to be a conservative approach to the design of these systems, particularly with respect to the external traps.

The external trapping system is being designed on the concept that each trap must be removable, although the system is being designed to operate for the entire life of the reactor without replacement. Where necessary, shielding, vent lines and purging are provided such that personnel access would be permitted within 5 days following shutdown to perform any required maintenance.

Although the concept of fission product purging and trapping has not previously been employed, the principles have been applied in other operations and can be considered sound. On the basis of technical information contained in the application, the design requirements for the system appear to be reasonable. Consequently, the Staff believes that an acceptable trapping system can be designed and constructed for the Peach Bottom reactor.

H. Fuel Handling

It is proposed that the entire fuel inventory will be replaced at about three-year intervals; consequently, fuel handling activities normally will be performed infrequently. The fuel handling system includes a fuel transfer machine, a charge machine, a fuel element canning machine, a pickup cell and a viewing system. A water-filled storage pit and associated equipment are provided outside of the containment building for storage of spent fuel.

The function of the transfer machine is to transfer core elements (fuel elements, control rods and reflector elements) between the reactor core and several parking holes provided in the reflector. The charge machine serves to transfer irradiated core elements from the parking holes to the canning machine, or to transfer new elements from the pickup cell to the parking holes. The pickup cell serves to store an element preparatory to loading it into the reactor. The canning machine is used to hermetically seal the elements in a metal canister by remote means. Access to the core is gained by means of five ports in the top of the pressure vessel: a 20-inch diameter central port through which the transfer machine is installed; a 6-inch diameter fuel charging access port through which the charge machine operates; and three 10-inch diameter ports for viewing system and salvage equipment.

The transfer machine includes a grappling head on a rotating radius arm and necessary mechanisms for remotely raising, lowering and rotating the head. When the grappler head is within 1-1/2 inches of the proper handling knob on a fuel element, it is aligned automatically by a sensing mechanism in the grappler head. The machine includes equipment for indicating the force being applied to an element.

The charge machine operates on rails between the fuel charging access port, pickup cell and canning machine. The machine incorporates a grappling device and winch to pick up elements from the reactor or pickup cell. A pneumatically operated piston and seal device provides sealing between the machine and the components which are serviced.

The canning machine is located within the reactor biological shield. Following placement of an element in a 13-foot long canister, the canning machine remotely brazes a lid on the canister. Before the canister is removed from the machine it is tested for leaks. The canned element is then transported by an elevator through a tube and is deposited in the spent fuel storage pit.

During all operations involving handling of core elements the machines are initially filled with purified helium; the gas evacuated from the machines is transferred to a storage tank. This arrangement is intended to retain about 99% of the activity released to the machines during a complete core change. The activity escaping to the containment building is estimated to be about 0.16 curies for the fuel change or about 0.01 curies/day during the operation. This activity is discharged through the waste gas system to the stack.

The fuel handling system includes a Failed Fuel Element Locator (FFEL) which is used for detecting elements that are releasing an excessive amount of fission products. The machine consists of a sniffing device for sampling helium cooling gas, positioning mechanism for the sniffer, sample monitor and control console. When abnormal activity occurs in the helium cooling system the reactor is shut down, the FFEL installed through the central access port, the reactor power is raised to a low level, helium samples are taken from coolant channels adjacent to the fuel elements and monitored for activity. The control console is located outside the reduced oxygen containment building. When the defective fuel element is located it is removed by the normal fuel handling procedures.

On the basis of the conceptual information provided, it is concluded that the proposed system should function satisfactorily for this reactor. Considerable effort will be made to minimize helium leakage from the reactor during fuel handling. It is noted that the amount of activity released during these operations can be controlled by prior purification of the helium system.

I. Auxiliary and Emergency Power

The Peach Bottom Atomic Power Station will be a part of a regional power pool with a total capacity of 15,000 megawatts, of which the Philadelphia Electric Company represents about 3500 megawatts. For startup and during periods when the Peach Bottom plant may be inoperative, station auxiliary power will be supplied from feeder lines connected to the power pool. Power for plant auxiliaries will be supplied by one of three auxiliary transformers: one is connected to the plant generator, one to the main feeder line and the third transformer to a reserve feeder line.

Equipment essential for emergencies and orderly plant shutdown will be connected to bus sections arranged for power supply from the feeder systems, with automatic transfer to an emergency diesel-generator power source under abnormal conditions. The diesel-generator will start automatically and run independent of station auxiliary sources when the emergency bus voltage is reduced to an abnormal low value.

The system is designed such that failure of the main transformer or loss of the main line supply during operation will leave the station power intact, being supplied by the turbine generator. If in addition the turbine should trip, there is sufficient energy in the rotating mass of the turbine-generator to supply power for initial reactor cooling. Further cooling can be accomplished by the helium compressors operated on their auxiliary motors supplied either by the reserve feeder line or the diesel-generator.

In addition to the power sources described above, a battery supply is available to power the control and instrumentation system and critical equipment. A separate battery supply is provided for excitation of the main turbine-generator during the coastdown period if bus power is not available.

It appears that adequate precautions have been taken to assure that reliable power will be available during all possible circumstances.

J. Research and Development

The research and development program, which has been in progress under AEC contract since the Fall of 1959, was undertaken to provide the necessary technical information in support of the proposed advanced reactor design. The major areas involved in this program include: (1) graphite component integrity; (2) fission product control; (3) flow and heat transfer experiments; (4) control rod and drive system, and (5) emergency shutdown systems.

In each of the above sections of the Staff Hazards Analysis for the Reactor System the related research and development program has been discussed, including the scope and significant results to date. The Staff has determined that the scope of each phase of the program is sufficient to provide for an adequate investigation of the problems involved in the design of the proposed reactor. The program is presently in an advanced stage of completion; results to date have provided good substantiation of the proposed design concepts. The unfinished portion of the program will serve primarily to provide more refined data as a basis for design of specific components. On the basis of the information presently available on the research and development program, the continuing phases are not expected to involve significant safety questions. However, these results will be reviewed for possible hazards implications.

It is concluded that the results to date of the supporting research and development program, and the scope of the continuing program, provide reasonable assurance that the features of this reactor can be developed to meet the design requirements for safe reactor operation.

II. CONTAINMENT

The containment building is a vertical structure consisting of a cylindrical steel shell with dished top and bottom heads. The diameter is 100 feet and the overall height is 162 feet. This building will contain the entire reactor system and the steam generators; the reactor control room and conventional plant equipment are located outside the containment in conventional structures.

The design specifications for the containment are as follows:

Material	A-201 carbon steel made to A-300 specification
Design internal pressure	8.0 psig
Design negative pressure	0.2 psig
Design temperature	150°F
Leakage rate (at 8.0 psig)	0.2% per day of contained volume
Snow load	30 lb/sq ft.
Seismic load	0.05 gravity
Wind load	100 mph

Detailed design and testing for the containment will be in accordance with the applicable codes of the State of Pennsylvania and ASME, including the design of leak-tight penetrations for piping and conduit access. All penetrations which are normally open during reactor operation such as steam lines, feedwater lines and ventilation ducts will have provisions for closing by either automatic or remote manual means to insure that the integrity of the building is maintained if a penetrating line or duct is ruptured.

Personnel and equipment access doors of the autoclave type are provided. These will be fully sealed and interlocked to prevent simultaneous opening of both doors.

The internal pressure specification for the building was established on the basis of simultaneous occurrence of the following failures: rupture of the primary system, rupture of one steam generator tube, failure of the helium loop isolation valves to operate, and loss of forced circulation cooling of the core. In order to calculate the overpressure consequences of these failures, the following assumptions have been applied:

(1) The helium inventory of 926 pounds is released to the containment at 790°F.

(2) 2200 pounds of water and steam is released from the ruptured steam generator tube. Half of this moisture is assumed to react immediately with the core graphite with the resulting CO and H_2 released to the containment. The remainder is released through the ruptured primary system to the containment as steam at 660°F.

(3) The water, steam, CO and oxygen in the containment building is assumed to react with the core graphite at a rate controlled by natural connection of the containment gas mixture through the reactor vessel.

On the basis of the above assumptions it is calculated that the postulated accident results in a peak internal pressure of 8.0 psig at 150°F; this valve has been selected as the overpressure specification. Results of calculations have been presented to indicate that a release of 2200 pounds of water and steam into the primary system following a tube rupture represents the maximum release under the circumstances. It is noted that the highest containment overpressure resulting from a single failure occurs as a consequence of rupture of the primary system. The calculated equilibrium conditions for this accident are 3.8 psig at 155°F. The staff believes that the proposed pressure specification represents a conservative value for this reactor system.

During normal reactor operation the containment building will contain a depleted-oxygen atmosphere, except for an isolated air room for critical control instruments requiring frequent attention. The depleted atmosphere will be maintained at less than 5% oxygen by volume, which is sufficiently inert to prevent combustion of the core graphite in event the containment atmosphere enters the reactor and combustion of CO and H_2 gases released as a result of the reaction between water and graphite. The depleted atmosphere is supplied by a conventional oxygen burning system which is capable of reducing the oxygen content of normal air to about 0.5%.

The containment ventilation system provides for internal recirculation of the contained atmosphere, purge and makeup of the inert atmosphere and ventilation for the air room. During normal operation the building is maintained at a slight negative pressure by an exhaust fan. The amount of gas discharged by this fan is equivalent to the inleakage due to the building pressure differential and the depleted makeup required to maintain the oxygen content below 5%. The exhausted gas is passed through an absolute filter and is monitored for radioactivity. In event of high stack discharge activity the building isolation valves will be closed automatically and the containment atmosphere can be recirculated through a 2000 cfm absolute filter for removal of particulate activity. This recirculation filtering is intended to reduce the amount of particulate material available to escape the containment building due to normal leakage (0.2%/day) when the isolation valves are closed.

In reviewing the conceptual plans for the containment building the Staff has observed the following: the building will be constructed in accordance with accepted standards; the building overpressure specification has been conservatively determined; the proposed leak rate represents an acceptable low value; the depleted building atmosphere will eliminate the possibility of combustion of core materials or released gases; and the ventilation system is adequate to provide proper control of gaseous effluents. It is concluded that the proposed containment including the associated ventilation system is acceptable.

III. WASTE HANDLING

During normal operation of the Peach Bottom facility the gas, liquid and solid effluents will be collected and/or processed separately in the waste handling system.

Possible sources of radioactive waste gases as a result of reactor operation and maintenance activities include the following:

1. Leakage from the primary coolant system to the containment atmosphere. This gas is eventually discharged from the stack due to the continuous filtered bleed flow required to maintain the containment negative pressure and reduced oxygen atmosphere.

2. Leakage from the helium purification system to the containment atmosphere.

3. Activated gases from the equipment cavities in the containment.

4. Purging and flushing gases from maintenance and refueling operations.

The leakage of activity from the primary system is based on a system fractional leak rate of 10^{-4} per day and an average coolant activity over an annual period of 1800 curies. On the basis of this release and assuming 80% removal by the filters of the gross activity for all escaping gases except the halogens and rare gases, the release from the stack is calculated to be about 0.124 microcuries/sec. With respect to the release of activity from the primary system, it is noted that the system will be fabricated to be essentially leak tight and efforts will be made to maintain loop activity at the lowest feasible value. The Staff believes that the above values for estimated release are conservative.

Leakage from the external fission product trapping system will be minimized by incorporating doubly contained pipes and vessels in that portion of the system which includes the water cooled traps and delay beds. Double containment consists of an external barrier surrounding the subject pipe or vessel with higher pressure purified helium within the outer barrier to assure inleakage. The design objective for leakage from the external traps is less than 500 microcuries per hour during reactor operation. Following decay and filtration the stack release from this source is calculated to be less than 0.026 microcuries per second.

Neutron activation of the environment surrounding the reactor cavity will produce radioactive argon-41 and carbon-14. This cavity which is steel lined will be provided with gas flow restrictions at all penetrations in order to reduce the free convection circulation of the cavity atmosphere to the containment. It is estimated that a 100% per day interchange of cavity gas and the containment atmosphere will occur. On this basis the activity discharged from the stack is calculated to be about 0.046 microcuries/sec of A-41 and 0.0064 microcuries/sec of C-14.

As previously noted, the refueling operations will involve the flushing and purging of fuel handling equipment. It is estimated that these operations will release about 0.16 curies for each reloading operation, which is equivalent to an annual average of about 0.005 microcuries/sec from the stack. In addition, it is estimated that other purging operations will result in a maximum stack release of about 2.5 curies/year, or an annual average of about 0.08 microcuries/sec.

The above releases of gaseous activity from the stack can be represented in terms of exposure at the nearest site boundary. Assuming a stack height of 150 feet and an annual average dilution from top of stack to site boundary, the fraction of the maximum permissible concentration (MPC) contributed by each source is determined to be as follows:

Area	Fraction of MPC (10CFR20)
Primary loop leakage	0.062
Purification system leakage	0.013
A^{41} effluent	0.006
C^{14} effluent	0.003
Refueling operation	0.003
Equipment maintenance	0.040
Total	0.127

The containment ventilation and gaseous waste system includes equipment for filtering and monitoring all gas exhausted to the stack. A continuous flow sample of stack gas is collected and passed through a filter which can be removed and monitored for particulate contamination. The sample stream is monitored with a dual channel gamma spectrometer capable of recording gross activity and a single energy or band of energies. The stack monitor will provide signals for alarms and automatic closure of the containment isolation valves if the stack activity exceeds predetermined limits.

The Staff considers the applicant's estimates of routine gaseous effluents to provide a reasonable basis for calculation of the average exposure from these sources. The conceptual plans for disposing of gaseous effluents appear feasible. The maximum estimated annual exposure at the site boundary is about one-eight of allowable limits.

The Peach Bottom reactor is expected to generate a minimum of liquid wastes. Sources of liquids for routine disposal include laundering activities, water from chemical impurities removal system, sump drains and the spent fuel pit.

The liquid waste system provides for collection, processing, holdup, monitoring, and controlled discharge. Equipment consists of two liquid receiver holdup tanks, a separate laundry waste holdup tank, filters, demineralizer and two waste monitoring tanks.

Waste solutions resulting from the decontamination of refueling equipment are pre-processed in a separate decontamination system consisting of decontaminant chemical mixing tanks, piping to the fuel machines, drain tanks, and filters. Sludge collected in the tanks will be shipped off-site for disposal and liquids will be processed through the liquid waste system.

All waste liquid will be processed or stored until its activity permits mixing with the discharge of condenser cooling water to the Conowingo pond (50,000 gpm), without exceeding tolerance values. A continuous sampling system will monitor the total activity released.

The applicant has attempted to estimate the total quantity of liquid waste produced. It is conservatively estimated that one load of laundry will be washed per hour containing 19 microcuries (10^{-4} microcuries/cc). The biological shield cooling system is a closed system containing about 1000 gallons of water in which the activity level may be as high as 10^{-4} microcuries/cc (378 microcuries total). Leaks from this system will be collected and processed. Spent fuel pit water is normally uncontaminated since the spent fuel canisters are sealed and tested prior to storage in the pit. Nevertheless, provisions are included for processing of the pit water in event of a canister leak. Any leakage of cooling water for the external fission product trapping system will be collected and processed. All liquid effluent from the chemical impurity removal system of the helium purification system will be processed through the liquid waste system.

It is the opinion of the Staff that the small amount of liquid wastes do not represent a significant problem and that the proposed liquid waste handling system is adequate for proper processing of such wastes.

All solid radioactive wastes will be shipped off-site for disposal. Typical solid wastes consist of filters, dust collected in the primary loop filters, spent resins from the demineralizers and miscellaneous material collected from maintenance operations and laboratories.

The Staff believes that the applicant has provided a reasonable estimate of the quantities of radioactive wastes and has included adequate plans for safe disposal.

IV. FACILITY SAFETY ANALYSIS

The applicant has performed an analysis of all hypothetical accidents that conceivably could occur to the Peach Bottom reactor as a result of operational errors and failure of various components and systems. The various postulated failures include those involving the reactor, the fission product trapping system, fuel handling system and conditions arising external to the plant. In each case it was demonstrated that either through the inherent safety characteristics of the reactor and its systems, or by following proper corrective action, these accidents would not result in undue public hazard. The Staff believes that all mechanisms by which significant accidents might occur have been evaluated, and is in agreement with the conclusions reached. As discussed in the following analysis, the consequences of the worst accident postulated would not result in serious exposure to the public.

An outline of the specific accidents analyzed by the applicant is included below:

A. Incidents Involving the Reactor

1. Reactivity accidents

a. Excessive removal of control poison
   (1) rapid control rod removal during operation
   (2) control rod fallout ← now not considered credible
   (3) startup accident
b. Loss of fission product poisons
c. Rearrangement of core components
d. Effect of steam in core voids
e. Sudden cooling of reactor core

2. Accidents to fission product barriers

a. Fuel element malfunction
b. Effect of moisture on core graphite
c. Primary system rupture
d. Simultaneous steam leak and primary system rupture
e. Simultaneous loss of cooling and primary system rupture

B. Incidents Involving the Fission Product Trapping System

1. Loss of cooling capability

a. Loss of purge flow
b. Loss of purification compressors
c. Primary system depressurization
d. Loss of cooling water
e. Loss of refrigeration

2. Rupture of trapping system

3. Change in purge flow contents

a. Increase in fission product concentration
b. Increase in water content
c. Increase in chemical impurities

C. Fuel Handling Accidents

1. Ruptured spent fuel canister
2. Stuck fuel element in charge machine

D. Plant Behavior Due to External Conditions

1. Loss of outside power
2. Earthquakes
3. Floods
4. Landslides
5. Fire
6. Severe Weather

The applicant has been able to demonstrate that certain of the above types of accidents should not involve any release of radioactivity, for example, for each of the reactivity accidents indicated above. For those accidents which might release radioactivity to the containment building, the amount of activity released has been calculated. On the basis of these calculations, the consequences of each of the following circumstances have been determined: (1) the most severe accident involving a single failure, (2) the most severe accident involving two component failures and (3) "the worst conceivable accident." The results of these accidents are summarized below.

In computing the environmental consequences of the assumed releases, the following assumptions have been made which the Staff considers to be conservative:

1. The specified containment leakage of 0.2%/day at 8 psig overpressure is assumed to occur throughout the duration of the release.

2. The post-incident containment filter system removes 50% of the particulate activity in the 2000 cfm flow through the filters.

3. The atmospheric dilution factor at the site boundary is 3×10^{-4} sec/m^3 for inversion conditions and 1×10^{-5} sec/m^3 for unstable conditions.

4. Whole body gamma doses are calculated on the basis of immersion in a semi-infinite cloud.

5. Direct radiation from fission products inside the containment building is based on a 50% self shielding factor.

The most severe accident involving a single failure in the reactor system is a primary system rupture of sufficient magnitude to allow back flow of the fuel element purge gas to the containment. It is calculated that 0.84 megacuries would be released to the containment including the coolant activity, the activity released by purge gas back flow through the fuel elements, activity in the purge line and the release resulting from increased activity of the external trap effluent. This accident results in the following exposures at the site boundary for the first 24 hours and with inversion conditions: whole body-0.005 rem and thyroid-0.05 rem. The Staff considers this accident credible, although improbable. However, the consequences to the public would be insignificant.

The most severe accident involving two component failures in the reactor system is the rupture of the doubly contained fission product purge line. (Although this accident requires failure of two pipes, it is noted that a single event could cause the accident.) The incident results in a total of 2.26 megacuries of activity released to the containment. The resulting exposures at the site boundary for the first 24 hours and with inversion conditions are: whole body-0.01 rem and thyroid-6 rem. The Staff considers this accident to be the probable "maximum credible accident" and concludes that the results would not cause undue hazard to the public, since the calculated doses are not significant.

The most severe accident postulated by the applicant involves several simultaneous failures: primary system rupture, loss of circulation and failure of the purge line check valve to function. This accident would result in an immediate release of the activity in the primary system and purge line, followed by the release of additional fission products from the compacts as the core increases in temperature. The maximum activity present in the containment at any time is calculated to be 5.2 megacuries, although activity continues to be released. The maximum value of 5.2 megacuries occurs at 16 hours following the incident. The resulting exposures at the site boundary for the first 24 hours and with inversion conditions are: whole body-0.4 rem, thyroid-100 rem and bone-20 rem. Owing to the multiplicity of failures required the Staff does not consider this accident credible. Furthermore, it is noted that the calculated exposures are less severe than the exposure criteria contained in the Commission's proposed Reactor Site Criteria (proposed 10 CFR 100) which are 25 rem to the whole body and 300 rem to the thyroid. Consequently, it is concluded that no credible accident will result in undue hazard to the public.

In reviewing the applicant's reactor safety analysis the Staff has found that (1) the scope of the analysis includes all accidents considered credible, (2) the assumptions on which the releases and consequences are based appear to be sufficiently conservative and (3) the worst postulated releases do not result in unacceptable exposures at the site boundary.

V. SITE

The proposed plant will be located on the westerly shore of Conowingo Pond, in Peach Bottom Township about 3 miles north of the Pennsylvania-Maryland State Line. Conowingo Pond is formed by the backwater of the Conowingo Dam on the Susquehanna River; the dam is located about six miles south of the Pennsylvania-Maryland border. The pond extends upstream about 15 miles to the Holtwood Dam. It varies in width from 0.6 to 1.5 miles and contains, when full, 240,000 acre feet or 80 billion gallons of water. The top ten feet (80,000 acre feet) is used at pondage to generate power at the 252 Mw Conowingo hydroelectric plant.

The site consists of 600 acres owned by the Philadelphia Electric Company and situated about nine miles upstream from the Conowingo Dam. The reactor will be located about 225 feet from the shoreline. The minimum exclusion distance on the west side of the pond is 3000 feet. The pond to the east is about 7600 feet wide. The terrain immediately surrounding the plant location is moderately wooded and slopes toward the pond. Within a mile steep hills rise to an elevation about 300 feet above plant grade level.

The population distribution in the vicinity of the Peach Bottom site is summarized as follows:

	Pop.	Census Date
Within 3000 feet	0	1959
Within 1 mile	120	1959
Within 5 miles	5,700	1959
Within 10 miles	25,000	1959

Significant population centers within a 10 mile radius of the site include the following:

	Pop.	Dist.	Direction
Slate Hill	50	1.7 miles	SW
Peach Bottom	70	2.3	E
Delta, Cardiff	1590	4	SW

At present the river below Peach Bottom is the sole source of water for the City of Havre de Grace (pop. 8000, distance 18 miles), the Perry Point Veterans Hospital, and the Bainbridge Naval Training Station, which supplies the town of Point Deposit. A total of 856 million gallons of water were withdrawn below Conowingo Dam in 1958. In order to supplement its present water supply, the City of Baltimore is currently constructing a pipeline that will intake from a point about one quarter of a mile upstream from the Conowingo Dam. This system is scheduled for completion in the mid 1960's.

The applicant proposes to maintain positive control over all liquid effluents produced during operation to assure that the concentration of radioactivity at the point of release to the Conowingo Pond will not exceed the value permitted by the AEC regulations. However, in the unlikely event of an accident involving a radioactivity release to the pond, it would be useful to know the probable disposition of the material in the pond. The applicant is performing studies to facilitate predictions of the concentration of materials released to the pond under various conditions of river flow and hydroelectric plant operation.

In the general area of Peach Bottom the prevailing winds are from the northwest in the winter months and southwest in the summer months. The most frequent wind direction for the region is west, occurring about 17% of the time; the least frequent is southeast, occurring about 8%. Calms occur about 7% of the time, primarily during the night and early morning hours. In the immediate area of the site the river valley produces a significant channeling influence on the local winds at low elevation. Throughout the year down-channel winds occur 47% of the time and up-channel winds 43% of the time. Winds at an appreciable angle to the channel are found to occur only 7% of the time.

The application includes a preliminary evaluation of the dilution characteristics of the atmosphere in the area of the site. In addition, a program has been initiated to measure the characteristics at the site and to relate them with the meteorological conditions of the general region. The applicant's evaluation of the implications of the meteorological characteristics relative to the consequences of a radioactive release to the atmosphere are considered reasonable by the Staff.

In evaluating the acceptability of the Peach Bottom site the Staff has considered the following aspects: the characteristics of the proposed reactor including the design, safeguards features, containment capability and nature of waste products generated; the meteorological, hydrological and geological characteristics of the site; the present and future population distribution in the surrounding area; and the quantity and nature of radioactive releases resulting from any credible accidents. In addition, the Staff has evaluated the site on the basis of the Commission's proposed Reactor Site Criteria, (proposed 10 CFR 100) as published in the Federal Register on February 11, 1961.

With respect to the proposed AEC Reactor Site Criteria, site acceptability is based on derived distances for the exclusion area boundary, low population zone boundary and nearest population center. The basic premises of the criteria are such that no person would be expected to receive a radiation exposure in excess of 25 rem to the whole body or 300 rem to the thyroid. For the power level and leak rate conditions of the Peach Bottom reactor, the Staff has calculated an exclusion zone distance of 0.2 mile, a low population zone distance of 3.8 miles, and a population center distance of 5.1 miles. The nearest site boundary is 0.6 mile and the area low population density extends beyond 10 miles, which compares favorably with the proposed site criteria.

No postulated accident would result in a direct release of radioactivity to the Conowingo Pond. However, if an accident involving the release of airborne material should occur during a period of precipitation, activity would be deposited in the water by rainout action. The resulting concentration of activity in the water would depend on the quantity deposited, which is a function of the intensity and duration of rainfall and the wind conditions, and on the dispersion of radioactivity in the pond. The Staff has conservatively estimated that a continuous hard rainfall of at least 4 to 6 hour duration would have to occur simultaneously with the postulated worst conceivable accident (described in the previous section) before the concentration of radioactivity in the pond would approach values permitted by Part 20 of the AEC Regulations for public consumption. The Staff is of the opinion that even if the postulated accident did occur there is essentially no possibility that usage of the pond as a source of domestic water supply would need to be restricted, even temporarily. However, the applicant has indicated that the water downstream of the reactor will be monitored continuously in order that appropriate action can be taken, if required, to restrict consumption of the water.

In reviewing the proposed Peach Bottom site the Staff has observed the following: the adequacy of the exclusion zone; the low population concentration in the surrounding area; the lack of any population centers in the area; the radiation exposure at the site boundary as a consequence of the worst accident is not unacceptable; and although excessive contamination of the pond water is not expected, if it occurs measures can be taken to minimize exposure from this source. Consequently, it is concluded that the proposed site is acceptable for a reactor of this general type.

VI. TECHNICAL QUALIFICATIONS

Philadelphia Electric Company has applied for a construction permit and class 104(b) license for the Peach Bottom Atomic Power Station. The facility will be designed and constructed for the applicant by Bechtel Corporation under contract to Philadelphia Electric and High Temperature Reactor Development Associates, Inc. This latter group is a nonprofit corporation financed by 53 utility companies interested in acquiring technical information, experience and training in the design, construction and operation of the plant. The nuclear steam system will be designed and manufactured by General Atomic Division of General Dynamic Corporation under a subcontract with Bechtel Corp. In addition, General Atomic is performing the supporting research and development program under contract with the Atomic Energy Commission. The facility will be operated by Philadelphia Electric Co.

Philadelphia Electric Company is a public utility engaged in supplying electric and gas services. The Company and its subsidiaries presently own and operate eight steam-electric plants and one hydroelectric station with total generating capacity of over 3000 megawatts. Since 1952 Philadelphia Electric personnel have participated in other reactor projects for purposes of training and experience in reactor design, construction and operation. The Company presently has about 9100 employees, of whom about 260 are professional engineers engaged in power plant design and engineering.

Bechtel Corporation has been engaged in construction activities since 1898 and in nuclear activities since 1949. The Corporation has constructed over 40 conventional power plants ranging in size from 1.6 to 625 megawatts. In addition, Bechtel has participated in the following significant nuclear power facilities: Dresden Nuclear Power Station - conventional engineering, Vallecitos Boiling Water Reactor - engineering and construction, General Electric Test Reactor - preliminary design, SM-1A, Fort Greely, Alaska - nuclear system design and containment requirements, PG&E Humboldt Bay Power Plant - prime contractor, Hallam Nuclear Power Facility - architect-engineer, and Big Rock Point Nuclear Plant - prime contractor and engineer-constructor.

General Atomic is located in San Diego, California, including laboratories, accelerator, critical assemblies, two research reactors and engineering facilities. General Atomic has manufactured and sold about 20 Triga research reactors. Its staff numbers about 1200 including 750 scientists and engineers. Efforts relating to the design of the Peach Bottom include materials studies involving physical and mechanical properties at high temperatures, corrosion effects and radiation effects. General Atomic has devoted a considerable effort to the development of graphite for application in the Peach Bottom reactor.

The AEC Staff considers Philadelphia Electric and its principal contractors, Bechtel and General Atomic, well qualified to design, construct and operate the Peach Bottom Atomic Power Station.

VIII. SUMMARY AND CONCLUSIONS

The Philadelphia Electric Company has applied for a construction permit for an advanced type gas-cooled power reactor of 115 megawatts (thermal) capacity to be located in Peach Bottom Township on the westerly shore of Conowingo reservoir. The proposed reactor includes certain features not previously incorporated in power reactors. A supporting research and development program of substantial magnitude is in progress to furnish basic information for the proposed design. Principal contractors include Bechtel Corporation, prime contractor, and General Atomic, subcontractor for the nuclear steam system.

The Staff of the AEC Division of Licensing and Regulation has reviewed all technical information submitted by the applicant. As indicated in the previous Staff Hazards Analysis, it is concluded that sufficient information is available, relative to each aspect of the facility, to provide reasonable assurance that the Peach Bottom reactor will not cause undue hazard to the health and safety of the public.

Specifically the Staff notes that the proposed site possesses generally good characteristics, although care must be exercised to protect the reservoir water from contamination; the conceptual reactor design including the advanced features can be developed to function as described; the reactor design incorporates adequate provisions to cope with possible emergency situations including inherent safety characteristics; the supporting research and development program is adequate to provide the necessary technical information concerning the design; the proposed containment building will provide an adequate ultimate barrier against an accidental release of radioactivity; acceptable plans have been submitted for the collection, handling and disposal of any radioactive effluents; no credible accident is expected to cause undue hazard to the surrounding environment; the applicant and the applicant's principal contractors are considered well qualified to construct and operate the proposed facility; and the Commission's Advisory Committee on Reactor Safeguards has concluded that the site and design for the proposed facility are acceptable.

Based on the foregoing analysis and findings, the Staff has concluded that a reactor of the general type proposed can be constructed and operated at the Peach Bottom site without creating undue risk to the health and safety of the public.

December 1, 1961

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