October 25, 1961

ATOMIC ENERGY COMMISSION

DIVISION OF LICENSING AND REGULATION

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ON

PEACH BOTTOM ATOMIC POWER STATION

PHILADELPHIA ELECTRIC COMPANY

Note by Director, Division of Licensing and Regulation

The attached report has been prepared by the Staff of the Division of Licensing and Regulation for consideration by the Advisory Committee on Reactor Safeguards at its October 1961 meeting.

18 copies of this revised report forwarded to ACRS on 10-26-61 Some improvements [illegible] made but has [illegible] [illegible] judgment.

A-206

PEACH BOTTOM ATOMIC POWER STATION
PHILADELPHIA ELECTRIC COMPANY

The Peach Bottom Atomic Power Station has been reviewed by the ACRS and staff on two previous occasions: the site was considered at the March 1960 meeting and the initial preliminary design at the December 1960 meeting. As a result of the latter review it was concluded that: (1) the proposed design was considered generally adequate but the staff identified certain specific areas requiring additional design effort and (2) the application lacked sufficient information concerning the R & D program to support certain design features.

The Philadelphia Electric Company has now submitted Amendment No. 2 to their application for a construction permit. This amendment includes a revised Volume I - Part B, "Plant Description and Safeguards Analysis," which is a complete revision of the Preliminary Hazards Summary Report based on the latest results of the continued research and development program. New plant design features are proposed to resolve the previous deficiencies and also to take advantage of the more complete R & D program.

The DL&R Staff has reviewed the information submitted and has held numerous discussions with the applicant, including a visit to the General Atomic Laboratories. In this report it is intended primarily to discuss those plant features which were considered deficient previously or which have been modified to reflect more recent experimental results.

Fuel Elements

During previous reviews on fuel element design with the applicant it was concluded that additional information was needed regarding the effects of radiation during the long fuel cycles on structural integrity and dimensional stability, effective thermal conductivity, and the mobility of fuel and fission products in the compacts.

The latest amendment includes information on the research and development program to establish long term integrity of graphite components including low permeability graphite tubes, fuel compacts and control and reflector materials. The program involves the investigation of the effects of radiation, temperature, and chemical impurities on the mechanical and physical properties, as well as on the dimensional stability of the graphite materials.

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 defferent capsule tests at fast neutron exposures from 10^{19} to $2x10^{21}$ nvt and at temperatures from 326°C to 1400°C 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 326-1400°C. The resulting data indicate that the increase in permeability due to anticipated 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. Additional irradiated graphite samples are being analyzed.

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 1380-1430°C. 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.

HTGR graphite samples irradiated to 2×10^{21} nvt and at 1100-1400°C revealed increses in compressive strength of 100% and greater. These results are in good agreement with other referenced tests 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 1300°C for 1000 hours. No measurable changes were noted, indicating that only radiation-induced dimensional changes need be considered. Tensile strength tests on low permeability graphite at temperatures of 360-1480°C 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 been investigated. Samples at temperatures of 900, 1000, and 1200°C were exposed to circulating helium containing 3% water. Under these conditions a 1% loss in graphite and a factor of 10 increase in permeability were observed. This effect on permeability is considered equivalent to that 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 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 hrs; specimen temperatures were from 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 has also 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 $1\text{x}10^{-19}$ fissions/cc expected in the reactor, and at temperatures from 840 to 1480°C. The results of these tests indicate that particle size has a definite effect on the magnitude of dimensional change and that the minimum size fuel particles should be from 110 to 250 microns. 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 the specimens have received an exposure equivalent to about two years of reactor operation and will soon be removed for analysis. These coated particles have been subjected to 1000°F helium containing 26,000 ppm water for periods of 72 hours without apparent damage.

As a result of the above discussed development effort on materials, the basic fuel element design has been modified, as described in the amended report. The fuel particles (100-400 microns diameter) will now 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 fuel particles in the graphite compacts, protects the carbide fuel particles from oxidation during fabrication of the compacts, and increases the retention time of fission products within the compacts. Coating of the fuel particles is accomplished by jettisoning the particles into a combustion chamber with a stream of high speed gas which includes methane. The temperature in the chamber is 1000-1400^{o}C. 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 modified fuel element design incorporates rhodium-103 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 absorption cross section as the temperature increases above 1500^{o}K. Analysis indicates that at least 3 kg of rhodium are required in the core to provide the desired effect. Consequently, about 5 kg will be included at beginning of life in order to compensate for burnup losses. Rhodium has a melting point of 3570^{o}F and boiling point of 8100^{o}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.

Several modifications have been made to the fuel element design, reflecting results of the research and development program. The previous design incorporated a double impervious graphite sleeve arrangement with purge flow entering the element near the grid plate stand-off pin, passing upward between the two sleeves, back down around the fuel compacts, through the internal fission product trap and out through the hollow stand-off pin. The new design provides a single impervious graphite sleeve; the purge flow is introduced through the porous upper and piece, passes down around the fuel compacts, through the internal trap, and out to the stand-off pin. As a result of a less complex design and better knowledge of graphite behavior under irradiation, the maximum fuel element operating temperature is calculated to be about 300^{o}C less than for the previous design.

A quarter length prototype fuel element is currently being tested in a loop at the GETR. 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} 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 1500,000 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 results to date of the R & D program and the further tests proposed we have concluded that there is now reasonable assurance that the impervious graphite diffusion specification of $1x10^{-6}$ 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^{o}F can be met, and that the use of pyrolitic coatings to the fuel particles adequately prevents the mobility and improves 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.

Fission Product Trapping

Based on the earlier submittals and previous reviews, the Staff had identified certain problems to be resolved concerning the proposed fission product trapping system relating to proportional distribution of products in the fuel and various traps, methods and materials to be used, the engineering arrangements to accommodate safe handling and storage, and the procedures for disposal of the trapped fission products.

The design presented in the amended report provides several features to maintain control over the fission product inventory. First, release of volatile fission products from the fuel compacts is delayed until the shortlived products have decayed. This is accomplished by the dense carbon coating on each fuel particle, as discussed in the previous section on Fuel Elements. Non-volatile fission products will be retained in the fuel compacts.

Second, 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 located 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 machined lengthwise on the outer surface. A graphite spine supports the cylinders in the fuel element assembly. The slots in the cylinders are filled with silver coated charcoal reagent material in granular form which holds 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 and 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. Purge helium leaves the core via the stand-off pins into two parallel connected manifolds. The manifolds operate about 3 psi below the main coolant pressure to insure any possible leakage 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 external trap system comprises the third feature for fission product control. In order to provide assurance that the capacity of the external traps is adequate their design is based on three times the expected escape of fission products from fuel compacts with uncoated particles and an iodine delay in the fuel compacts of seven days (32 days delay expected).

The purge flow from the fuel elements (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 35,300 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 cleansed 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.

The final feature intended to limit the fission products in the primary system is the outer sleeve of the fuel assembly which is fabricated from low permeability graphite. This sleeve houses the fuel compacts and internal traps and serves to restrict the diffusion of fission products from the fuel to the coolant, as discussed previously.

The amended report presents information on the experimental work 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 full HTGR burnup, and (d) investigation of the characteristics of the integrated trapping system in the GETR 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 HTGR 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 have 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 and as of May 1961 have accumulated 50% HTGR burnup. The results of these capsules to date indicate that the steady state release of xenon and krypton at 1700°C fuel temperature is less than 1% for the short-lived isotopes (less than 2.8h) 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 burnup 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 a burnup equivalent to end of life conditions, and then annealing them at 1700°C for 60 hours. This experiment showed degradation to 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 the fuel compacts with uncoated particles as a function of temperature. The method involves production of about 10^{13} fissions, followed by a decay period, and then to anneal 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 900-2000°C.

Similar to the tests described above on uncoated particles, other experiments have been performed and are continuing with coated fuel particles. Many post annealing tests have been made on the Xe^{133} release from pyrolytic carbon coated particles at temperatures up to 2000°C. 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 1700°C 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 1400 to 2280°C in less than one minute, held at 2280°C for five minutes, and then reduced to 1850°C 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 2600°C (above the melting point of carbide fuel); within two hours the remaining Xe^{133} was released. Similar experiments were conducted with pyrolytic coated fuel particle compacts, but at more severe temperatures. The temperature of the compacts was raised from 1400 to 3000°C in several minutes and held at 3000°C for 30 minutes; during the period at 3000°C about 20% of the Xe^{133} was released. Following the 3000°C cycle the compacts were brought to room temperature, wherein all of the Xe^{133} was released, indicating cracks were produced by the rapid cooling from 3000°C 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 (900°F) of the internal trap renders it ineffective for rare gas adsorption, krypton and xenon were not included in the study. For studies involving cesium, activated charcoal was chosen as the trapping material and investigations were made of the specific sorption (gms Cs/gm/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 vapor pressure.

In order to study the adsorption of iodine and tellurium a dynamic experimental apparatus simulating the HTGR internal trap operating conditions was constructed. 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 vapor pressures similar to that 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 how long a particular fission product atom will remain in the trap. Results of these tests indicated that the RED value 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 bases of these results the present design of the internal trap including size and materials was established.

Tests 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 determination 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, the previously described fuel element test loop has been placed in operation at GETR to provide performance data under simulated operating conditions.

This loop incorporates a quarter length prototype fuel element consisting of 15 HTGR 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 maybe collected ahead of the internal trap or after the internal trap. During full power GETR 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 temp 2200°F, entrance gas 600°F, and exit gas 1400°F; these are substantially similar to conditions in HTGR.

The main components of the loop's 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 fuel element will be about 0.6 lbs/hr, compared to the HTGR 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_2CO, 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 products in the loop. Chemical and activity 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.

We believe that the amendment report and the data collected from the various tests indicate that the problem areas of the fission product trapping system are being resolved. From the data collected on experiments with coated fuel particles it has been calculated that the distribution of fission products now should be 98.3% retained in fuel compacts, 1.4% held in the internal traps and 0.3% in the external traps. From experiments on uncoated fuel compacts which were used to form the basis of design values, the comparative values are 90.45% in fuel, 7.17% in the internal traps and 2.38% in the external traps. The system is being designed out on the assumption that each trap must be removable, although the design of the external traps will be capable of operating for the entire plant life without replacement. Where necessary, shielding, vent lines and purging are provided for personnel access within five days following shutdown. We believe that handling and disposal of radioactive materials can be safely accomplished within the scope of the design proposed.

Control System

From previous information submitted by the applicant, uncertainties were identified in several areas of the control system. These related to the stability of control rod graphite under irradiation, possible effects of separation of the rod elements from their drives, and the requirement for a secondary shutdown system.

The amended information recently submitted indicates that extensive modifications of the control system have been accomplished and that several new design additions have been incorporated for the purpose of improving reliability of the system. The following paragraphs describe the major modifications.

A modification has been proposed in order to provide a signal of control rod separation. This consists of a high temperature metal clad conductor which will traverse the entire length of the control rod from the extreme upper tip of the poison section down through the rod and push rods, terminating at the juncture of the drive rod and the power unit. Any break in the conductor will result in an open circuit signal to the control room indicating that a structural failure or separation of the control rod has occurred.

Several important features have also been added to the control rod drive system. A mechanical holding lock is now provided for the ball nut actuator housing such that once the ball nut is in the position corresponding to rod insertion, a mechanical finger latch will prevent its retraction unless the operator manually releases the latch by activating a solenoid. This feature is intended to prevent rod fallout in the event of hydraulic motor pressure line rupture and failure of the backstop clutch.

The valve assembly controlling the hydraulic drive motor has been revised to incorporate dual scram valves in series. Each of these valves is position monitored on the console and allows the operator to routinely exercise the valves individually to determine proper functioning at any time.

In response to the concern regarding the lack of backup reactor shutdown, the applicant has provided two additional systems to compliment the 36 normal control rods. It is now proposed to include 19 electrically driven emergency shutdown rods uniformly placed among the control rods and under the manual control of the operator. In addition a number of thermally released (fused), gravity drop, absorbers would be placed in the upper end of the rod guide housings.

Two purposes exist for the electrically driven emergency shutdown rods. First, they provide additional shutdown capacity in the event that sufficient normal control rods fail to insert to provide cold shutdown; the nineteen backup rods are worth about 16% k_{eff} while the 36 normal rods are worth 26% k_{eff}. The second purpose is to enable shutdown in the remote event that some force would disarrange the core so that the normal rods could not function; the drives of the electric rods can each exert a drive force of 10,000 pounds, sufficient to force the rods through the core material. The main components of this unit are a graphite guide tube, a stiff metal tubular member filled with compacts of refactory poison material, a push rod, an acme drive screw, a gear reduction drive, an electric motor and a nickel-cadmium storage battery. Similar to the normal control rod, the backup rod is provided with a continuity circuit traversing its entire length to indicate rod separation. It is planned that extensive testing of a prototype shutdown rod and drive will be undertaken to prove the design.

The purpose of the thermally released absorber rods is to hold the core sub-critical in the event that insufficient control and emergency rods are inserted following xenon decay, or if the core should become hot enough to lose poison by evaporation. The minimum worth proposed for the thermally released absorbers will be sufficient to compensate for decay of xenon - about 3% k_{eff}. These rods consist of short absorber rods normally held in the top reflector by a metal tie bar and fusible link. Overtemperature of the core would cause the link to melt and release the absorber rod to fall into the midplane of the core. Tests will be run to establish the reliability of this mechanism.

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 control materials of B_4C in graphite. The irradiation tests are intended to determine dimensional stability; thus far a test conducted at 650°C (the maximum temperature to be encountered in operation), fast flux of 10^{19} nvt and thermal of 10^{20} nvt indicate dimensional changes of about $\pm$ 0.1%. 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 for potential temperature excursions. Samples of B_4C-graphite compacts which were warm pressed at 750°C and vacuum sintered at 2000°C were heated in a helium stream at 2140°C for 16 hr. 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.

The graphite guide tubes will be subjected to fast flux of about 10^{13} nv and the wall temperature will range from 660°F to 1400°F during HTGR operation. Tests have been conducted to examine the effects of these conditions. Recent out-of-pile tests to determine dimensional stability and physical characteristics indicated that after 1000 hours at 2372°F no measurable dimensional changes had occurred. 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 effect of irradiation, the material for the guide tube was subjected to a fast flux of about 10^{21} nv 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 are now being irradiated. The maximum load to which a control rod guide tube will be subjected is the 2500 lb force of the control rod either in tension or compression. 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 component 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 components temperature (maximum allowable - 1000°F), helium flow rate and helium pressure drop. The rod was subjected to various simulated core positions and various flow rates and the test data did confirm 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 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 program of testing 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 scrams as expected in the life of the rod. Initial testing in air at normal temperature for 2000 full strokes and 200 scrams 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.

Based on the amended information showing the extent of development of the control rod materials and results of the testing program, it is our opinion that the design of the control rod system is feasible. It also appears that suitable methods have been provided for indication of rod separation, and that the backup shutdown system would provide assurance that adequate shutdown capability is available in the event of abnormal conditions, or loss of the normal control rods. From a review of the research and development information and the tests which have been conducted thus far, it is our opinion that the control rod and drive unit proposed for HTGR appears to provide sufficient assurance that the capability requirements can be met.

Core Design

Following review of the earlier information submitted it was concluded that sufficient information was not available to adequately evaluate the design of the proposed core arrangement. In particular more complete data was needed on the effects of flow disturbances, lateral stability, interferences with control rod motion, and possible oscillatory movements.

The information now submitted provides data from test programs recently conducted to resolve these questions. A detailed one-half scale hydraulic model of the entire pressure vessel and core was constructed and operated to determine the feasibility of the hydraulic design. The objectives of the test are: (a) to check the core restraining capabilities of the tilting reflector and the stability of the core; (b) to determine the pressure drop characteristics of the internal flow paths; (c) to study the flow patterns in the coolant passages; (d) to measure how effectively the inlet gas cools the vessel wall; and (e) to determine the effectiveness of the core reflector seals. The first phase of the model test is complete. This phase involved an investigation of the over-all effects of flow distribution, pressure drop, and the general functioning of the core restraint. The data from this phase is now being evaluated for possible design effects. Later phases will be concerned with more specific core flow problems. In addition, tests will be run to simulate reactor emergency operating conditions and normal shutdowns to gather heat transfer data for reactor vessel design. Other specific tests will include: pressure drop, flow rates, heat transfer film coefficients, leak rates and loads for regions of the core such as thermal shields, plenum shroud, exit nozzles, side reflector, and the core support plate. The information obtained from this program will provide substantial basic design data for the HTGR core structure.

In order to determine the dynamic stability of the fuel elements a cluster of 19 full-scale fuel elements was subjected to flow conditions equivalent to those in the reactor. These tests were conducted at maximum flow conditions equivalent to 225% of the maximum Reynolds number existing in the core. Two types of elements were tested: one included an aluminum section replacing the graphite sleeve and the other was a graphite prototype fuel element, except that fuel compacts were omitted. It is believed that these represented conservative models of the actual elements since they had a lower mass and lacked the dynamic vibration absorption inherent with the loose-fitting fuel compacts. Six types of test runs were made consisting of various modes of flow and placement of externally induced vibrations. Flow conditions were monitored for vibrations and excited vibrations were intentionally induced to determine damping characteristics. The sensitivity of measurement was such that a half-amplitude deflection of 0.0001 inch could be detected. The findings from the tests indicate that under no flow condition did the fuel element vibrate naturally and vibrations forcibly imposed are not amplified or reinforced 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 imbedded at various points along the tube walls heat transfer coefficients were measured for various flow consitions. This information will be used to calculate the physical design conditions for the fuel components and will enable good 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 test the control rod was operated with the guide tube deflected 1¼ inch at the upper end; the operability of the rod was not affected. In the second test 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 will probably operate satisfactorily under extreme core distortions.

It is our opinion that sufficient information is now available to provide assurance that uncertainties previously identified with respect to the core design can be resolved.

Facility Design

Three features of the proposed reactor system were previously identified as requiring additional study in determining the adequacy of safeguards. These were: a) provision of an emergency cooling system, b) precautions to prevent water in-leakage to the core, and c) safeguards against accidents which would allow air to enter the primary system.

In the amended report, the applicant presents several new design studies and discussions relating to the above three items. To protect against damaging results of core overtemperature due to decay heat an emergency cooling system is now provided. The reactor vessel is not insulated as in the early design; however, the reactor cavity is lined with a steel plate shroud which has cooling coils attached to the outside surface. The shroud is then insulated to prevent heat loss to concrete shield. In the event of failure of the coolant circulators the heat from the reactor core would be conducted from the core internals to the reactor vessel and then by radiation and convection to the water cooled shroud. The analysis of this arrangement shows that the maximum core temperature following the accident would be about 3600°F, occurring at 30 hrs following scram and the maximum vessel temperature would be about 867°F. The reactor vessel would be depressurized to remove most of the membrane stresses. To avoid excessive temperatures in the lower grid plate the applicant has stated that heavy heat conducting sleeves will be installed around each control rod drive to extend from the lower surface of the grid plate to the bottom head of the vessel.

In addition to the information contained in the amended report on the reactor pressure vessel, the applicant has been requested to provide more information regarding the design specifications and fabrication and operational limits of the pressure vessel. This has now been submitted. On the basis of review of this information we believe that the integrity of the reactor vessel will be acceptable.

Because of the effect of moisture on the graphite core materials provisions are now incorporated in the primary loop for rapid moisture detection and loop isolation. Three detectors are provided at each boiler which are able to detect a tube failure within 5 seconds by means of electrolytic hygrometers. In order to detect very small leaks which would produce carbon monoxide from the steam-graphite reaction, an infrared analyzer continuously monitors the coolant. The steam generators are now designed with a baffled plenum near the header plate whereby 100 lbs/hr of helium purge flow is removed and continuously monitored for moisture content; by this means a leak of 0.01 lb/hr can be detected. Main coolant helium is sampled and measured for impurities such as CO, O_2, N_2, CO_2, CO_4 and argon. Moisture detectors are likewise installed on each of the three helium handling system transfer compressors since these compressors have water cooled heat exchangers.

If an abnormal amount of moisture is detected the reactor is scrammed and the following action occurs automatically:

1. Within one second after the moisture is detected the generator dump valves will begin discharge of steam and terminate when steam pressure is 50 psi above helium system.
2. Within three seconds of moisture detection the hot and cold valves on the failed loop will close isolating that steam generator.
3. Within three seconds the steam generator feed water valves will close.

In order to prevent the possibility of rapid oxidation of graphite in the event of a primary coolant system rupture, the entire containment vessel, except for an isolated air room, is now designed so that it is filled with a depleted oxygen atmosphere rather than air. The air room is provided for equipment requiring frequent maintenance such as control rod drive auxiliaries. The depleted atmosphere is provided by an oxygen burning system capable of producing nitrogen gas containing 0.5 per cent oxygen by volume. Containment atmosphere will be maintained and monitored to oxygen contents less than 5% by volume.

It now appears that the applicant has provided sufficient assurance that the problems of emergency cooling, water leaks and air entry have been resolved.

Safety Analysis

In response to the previous comments that the applicant should direct more study toward the identification of failures which could lead to on-site or off-site hazards, the applicant has provided additional information in the amended report based on the latest system designs.

Contained in the safety analysis are studies on the following:

1. Incidents involving the reactor
   a. Reactivity accidents
   b. Loss of fission product barriers
   c. Loss of both main loops following rupture.
2. Incidents involving the Fission Product Trapping System
   a. Loss of full cooling capacity
   b. Loss of system integrity
   c. Change in purge environment.

3. Safety of fuel handling

   a. Escape of fission products
   b. Stuck element in charge machine.

4. Plant behavior under abnormal conditions arising external to the plant

   a. Loss of power
   b. Earthquake
   c. Floods
   d. Landslides
   e. Fire
   f. Severe weather.

5. Environmental consequences of accidents

   a. Summary of accidents releasing activity to the containment
   b. Assumptions for dose calculations
   c. Discussion of consequences.

The most severe accident postulated in the report would be caused by simultaneous multiple failures. These failures included a primary system rupture, simultaneous failure of both coolant loops and failure of the purge line check valve to close. As the core heats up to peak temperature as a result of reduced heat removal, additional fission products would be released from the fuel compacts. Combining these conditions with inversion conditions the dose for the first 24 hours at the site boundary would be: whole body gamma - 0.4 rem, thyroid - 100 rem, and bone - 20 rem.

The most severe accident involving a single failure is a rupture in the primary loop which would allow back flow from the fuel element purge gas to the containment. This accident produces a whole body dosage at the site boundary after 24 hours during inversion conditions of about 0.005 rem, and a thyroid dose of 0.05 rem.

Associated with the safety analysis of the reactor the applicant has completed more extensive work to verify the Doppler coefficient and over-all temperature coefficient. The Doppler coefficient has been recomputed using more refined calculational methods and has been verified by experimental work using the linear accelerator as a source of pulsed neutrons. This work was also compared with measurements on the Zenith critical assembly in England and measurements with the

HTGR critical. As a result of adjustments to the loading of thorium in the fuel, the lower fuel compact temperatures and better calculations, the over-all temperature coefficient is shown to be negative throughout core life. To provide added assurance of negative temperature coefficient, 5 kg of rhodium-103 will be added to the core.

We believe that the safety analysis conducted by the applicant, including the effects of new design features, indicates that an adequate evaluation of incidents has been conducted.

Conclusions:

The Staff believes that the design deficiencies identified during the previous review of the Peach Bottom reactor have been adequately resolved by appropriate design modifications. In addition, several other design nodifications made as a result of more recent experimental results appear to be desirable from safety considerations.

It is our opinion that the results to date of the supporting experimental program provide substantial assurance that the novel features of this reactor can be developed to meet the required specifications.

We therefore conclude that the design now proposed for the Peach Bottom reactor provides reasonable assurance that the health and safety of the public will be adequately protected.