1/29/65

Dispatched w/ 1-29-65 ltr to AEC
(Notice of Proposed Issuance)

HAZARDS ANALYSIS BY THE

DIVISION OF REACTOR LICENSING

IN THE MATTER OF

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

OPERATIONS UP TO ONE MEGAWATT THERMAL

DOCKET NO. 50-171 File Copy (Original)

I. INTRODUCTION

On February 25, 1964, the Philadelphia Electric Company filed an application for a provisional operating license for initial operation of the Peach Bottom Atomic Power Station at power levels up to one megawatt (thermal). Construction is nearly completed and it is anticipated that the plant will be ready for initial loading in early 1965.

The proposed initial operations consist of a planned series of experiments which are designed to verify the basic physics characteristics of the reactor. In essence, the program can be classed as a series of in-situ critical experiments for the purpose of determining plant and core characteristics. This program will be conducted at low power levels, and correspondingly low fission product inventories. As such, the degree of hazard presented by initial operations will be small, especially when it is considered that the program will be carried out in a facility designed to operate safely at power levels up to 115.5 megawatts thermal. As a result of our review, we have determined that the proposed operations up to one megawatt thermal can be carried out without undue risk to the health and safety of the public.

Since essentially all required information relative to full power operation has been provided by the applicant, we have considered full power operation of the plant in the safety evaluation presented herein. However, operations at power levels above one megawatt thermal will not be approved until the results of initial operations have been reviewed by the Commission.

The Advisory Committee on Reactor Safeguards (ACRS) considered the present application at its November 1964 meeting, and concluded in its letter of November 18, 1964, "It is the opinion of the Advisory Committee on Reactor Safeguards that the Peach-Bottom Atomic Power Station reactor can be operated at power levels of up to 1 Mw(th) without undue hazard to the health and safety of the public." A copy of the ACRS report is attached.

II. SITE

The Peach Bottom Atomic Power Plant is located in southeastern Pennsylvania on the westerly shore of Conowingo Pond at the mouth of Rock Run Creek, about three miles north of the Pennsylvania-Maryland state line. Conowingo Pond is

A-40

formed by the backwater of the Conowingo Dam on the Susquehanna River; the dam is located about nine miles downstream from the site.

The site is irregular in shape, consisting of 600 acres owned by the applicant, Philadelphia Electric Company. The reactor is located about 225 feet from the shoreline of the pond. The minimum exclusion distance is about 3000 feet to the west. The pond to the east is about 7600 feet wide.

The terrain immediately surrounding the site is moderately wooded and slopes toward the pond. Rough, wooded terrain persists to a distance of 12 to 15 miles from the site. The area immediately surrounding the site is predominantly in agricultural use or woodland, with some dairying.

There are no residents on the Peach Bottom site, and in general the region within several miles is characterized by a relatively low population density and scattered small communities. The population within various distances of the reactor can be summarized as follows:

Distance	Population
Within 3200 feet	4
Within 1 mile	35 to 70
Within 3 miles	1100
Within 5 miles	5700
Within 10 miles	68,600

Within a distance of ten miles, the only sizeable population concentration is located in the contiguous communities of Delta, Pennsylvania and Cardiff, Maryland about 4 miles to the southwest. These communities have a combined population of about 1570. Slate Hill, Pennsylvania (pop. 70) lies about 1.7 miles to the southwest; and Peach Bottom, Pennsylvania (pop. 70) lies about 2.3 miles to the east. The nearest large population center is Lancaster, Pennsylvania (pop. 61,000), 20 miles to the north.

From the standpoint of population distribution, the site is favorable for a nuclear reactor. As discussed in a later section of this analysis, the required low population zone boundary, as defined in the Commission's Regulations on Reactor Site Criteria (10 CFR Part 100) has been determined to be a maximum distance of three miles from the reactor by conservative analysis of the consequences of postulated accidents. The population distribution indicated above shows that the area within this distance is indeed a low population zone. As stated above, the distance to the nearest large population center is about 20 miles and is much greater than the guideline in 10 CFR 100.

In order to establish a basis upon which to predict the atmospheric dispersion characteristics in the vicinity of the site, the applicant established a weather station on the immediate site and an auxiliary station at a higher elevation nearby. These stations have been in operation since July 1959. Additional data on wind direction was obtained at the dams up and downstream from the site.

The data obtained from these stations shows that the local terrain has a marked influence on prevailing wind patterns at low elevations. In particular it has been found that the river valley produces a substantial channeling effect in the local winds. General down-channel winds (from N or NW) occur almost 50% of the time and up-channel winds (from S or SE) occur about 40% of the time. Calms are quite rare, occurring less than 3% of the time and winds at an appreciable angle to the channel occur about 7% of the time. By contrast, wind directions and speeds along the shore of the pond are substantially different from those for the channel as a whole. In the immediate vicinity of the site calms occur about 26% of the time and off-shore winds are comparatively frequent. On-shore winds remain infrequent, occurring only about 9% of the time. An evaluation of both the site and local channel meteorological records indicate that effluents released at the plant site would most likely be dispersed along the river channel rather than at significant angles to it.

The applicant has compiled and analyzed local meteorological data to obtain correlations between local site wind conditions and those prevailing in the region as a whole. These correlations were used to establish the necessary parameters for equations which describe the diffusion of gaseous effluents released at the reactor site. From our review we are satisfied that the values derived for diffusion parameters are appropriate for the site and that their use in analyses of the dilution of normal integrated releases of radioactivity to the atmosphere and the consequences of postulated accidents lead to reasonably conservative results. Conditions for normal effluent release and the consequences of accidents are discussed in greater detail in later sections of this analysis.

The river below the Peach Bottom site is at present the source of water for the city of Havre de Grace, Maryland (population 8000, dist. 18 mi.), the Perry Point Veterans' Hospital and the Bainbridge Naval Training Station, which also supplies the town of Point Deposit, Maryland. To supplement its present water supply from other sources, the City of Baltimore has under construction a facility which will take water from a point about one-quarter mile upstream from the Conowingo Dam (about 9 miles downstream of the site). The flow patterns and dilution characteristics on Conowingo Pond have been investigated by the applicant. From the results of this investigation the applicant has determined, and we agree, that there is negligible likelihood that contamination levels at potable water intakes could exceed the allowable concentrations set forth in the Commission's Regulations either under conditions of normal effluent discharge or as a result of various postulated accidents. This is discussed in greater detail in later sections of this report.

In our previous review of the Peach Bottom site prior to issuance of a construction permit for this facility, we concluded that the site was acceptable for a reactor of this type and power level. In our current review of the site we have considered the detailed information now available, including the following matters: the characteristics of the facility, including the design, safeguards features, containment provisions, and the amount and nature of radioactive waste products expected to be generated; the meteorological, hydrological and geological characteristics of the site; the population distribution in the surrounding area; and the quantity and nature of radioactive releases resulting from credible accidents. From this review, we have

concluded that the site meets all the applicable requirements of the Commission's Regulations and is suitable from a safety standpoint for the proposed reactor facility.

III. GENERAL PLANT DESCRIPTION

The Peach Bottom Atomic Station is designed to produce 115.5 megawatts thermal power with a net electrical output of 40 megawatts. The reactor is fueled with approximately 236 Kg of enriched uranium and about 6 times that much thorium fertile material, both in the form of dicarbides dispersed in a graphite matrix. The reactor moderator and reflector, fuel cladding and core structural material with the exception of the core plate are all graphite. The gas coolant is helium at a pressure of about 350 psia, which enters the core at about 650°F and is heated to an average mixed temperature of about 1340°F. The hot helium gas flows to two steam generators in independent loops where superheated steam is produced at 1000°F and 1450 psig. The helium gas is circulated by a motor driven, single-stage centrifugal compressor in each primary coolant loop.

The superheated steam produced in the steam generators is utilized in a conventional steam turbine to generate electricity. The steam plant incorporates forced circulation steam generators, a single outdoor turbine-generator unit and river water for cooling the steam condenser.

The reactor, primary helium piping and compressors, steam generators and auxiliary systems are located in a steel containment building. Most of this containment building is filled with a depleted oxygen atmosphere during operation to eliminate the possibility of a graphite fire and the formation of an explosive mixture with combustible gases that might be generated by a steam-graphite reaction in the event of an accident. Access to this part of the containment is normally not permitted during operation. However, all those components and systems for which access is most likely to be necessary have been located within a relatively small air room space within the containment. This space has a separate air ventilating system and is accessible during operation by means of air locks in the walls of the containment building. The steam plant, control room and spent-fuel pit are all located in separate buildings outside the containment building.

The Peach Bottom Plant is designed to be started and operated manually with capability to automatically follow the turbine-generator loading at powers above about 30% of full load. Automatic control maintains constant steam conditions at the turbine inlet by varying the helium-coolant flow rate to hold steam pressure constant and reactor power to hold steam temperature constant. The reactor is normally controlled by 36 hydraulically driven, control rods which enter the bottom of the reactor vessel and move vertically in graphite guide tubes distributed throughout the core. These control rods can effect a reactor scram. In addition, two other reactor shutdown systems are provided; a group of 19 electrically driven emergency shutdown rods and a group of 55 thermally released gravity-drop neutron absorbers. The electric drives on the emergency shutdown rods are capable of driving these rods into the core even if a high force is required due to core disarrangement.

The uranium and thorium dicarbide particles are coated with pyrolytic carbon and are dispersed in annular graphite compacts. These compacts are stacked in a 3-1/2" O. D. sleeve of low-permeability graphite, which, along with the upper and lower graphite reflector assemblies and other components, forms a 12 ft. long fuel element. The core is formed by stacking these elements into a closely packed equilateral triangular lattice which is supported by a carbon steel core plate. A lateral restraining force is provided by a ring of graphite reflector blocks which tilt in radially toward the center of the core. The helium coolant gas flows upward in the spaces between the cylindrical fuel elements.

At the top of the fuel elements, a small fraction, about 0.2%, of the helium gas coolant flow passes into the fuel elements via an orifice and a porous plug. This small amount of purge gas flows downward in the fuel elements, between the fuel compacts and the low-permeability graphite sleeve, sweeping along any volatile fission products that escape from the compacts. It is expected that the release of fission products from the fuel compacts will be delayed by the pyrolytic carbon coatings on the fuel particles. The fission products which are released are swept along by the helium purge stream to the lower, cooler end of the fuel element to an internal fission product trap. These internal traps consist of activated charcoal and are expected to retain most of the condensible products which do escape the fuel compacts. The helium purge gas and the gaseous fission products which pass through the internal traps are collected by a header system in the core support plate and flow out of the reactor vessel to an external fission-product trapping system. This system consists of a number of heat exchangers, traps and delay beds which remove the fission products before the helium purge stream is returned to the primary cooling system.

The Peach Bottom reactor has a design core life of 900 days of equivalent full power operation, corresponding to an average burnup of 60,000 MWD/tonne of uranium and thorium and 7-8% U-235 burnup. Batch loading of the core at the end of life will be performed with the reactor shut down, the primary coolant system depressurized and at reduced temperatures. Special fuel handling equipment is installed in the five nozzles provided for this purpose in the top of the reactor vessel. Fuel elements are removed one at a time and sealed in an aluminum canister before they are lowered in a spent-fuel elevator to the spent-fuel pit outside the containment building for storage under water.

A number of provisions are made for supplying reactor cooling. Each primary coolant compressor in addition to its normal drive motor may be driven by an auxiliary (pony) motor which operates at 1/10 speed. In the event the main motor is unavailable, the pony motors may be used to remove decay heat. At normal helium operating pressures, natural circulation in one or both loops can provide adequate gas flow to remove decay heat and cool the core without exceeding normal operating temperatures. In the unlikely event that all other means of cooling the core are lost, an emergency cooling system is provided to cool the reactor vessel and the core.

IV. REACTOR DESIGN

A. Fuel Element & Core Design. Structurally, the Peach Bottom fuel elements are made entirely of graphite and all have the same external geometry. There are, however, four types of fuel elements differing in their uranium, thorium, burnable poison (boron) and rhodium-103 content. The rhodium-103 is included to enhance the negative temperature coefficient of the reactor. Thirty-six of the 804 fuel elements contain instrumentation for measuring temperatures in the core.

Each fuel element assembly is 3.5 inches in diameter and 12 feet 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 element assembly consists of an upper reflector section, a fuel bearing middle section, a bottom reflector section and an internal fission product trap. The exterior portion of the fuel element is a sleeve of low permeability graphite which encases the entire fuel section, the internal fission product trap, and part of the lower reflector section. The fuel section consists of a 90-inch length of annular fuel compacts mounted on central graphite spines. Heat is removed by helium coolant flowing upward in the spacing between fuel elements.

The fissile and fertile material is in the form of particles of enriched uranium-carbide and thorium-carbide pyrolitically coated with dense carbon. These particles, as well as the rhodium-103, are uniformly distributed in the annular graphite fuel compacts. Burnable poison in the form of graphite compacts containing zirconium diboride is located in the center of selected fuel support spines.

The internal fission product trap consists of a 12-inch long graphite cylinder with longitudinal slots packed with activated charcoal. During reactor operation, a small flow of helium coolant gas is drawn through an orifice and porus plug in the upper reflector region to purge the space between the fuel compacts and the low permeability sleeve. The purge gas flow passes through the internal trap and exits through the standoff pin to the external trapping system. The performance of the internal and the external fission product traps is discussed in the section on The Fission Product Trapping System.

During normal full power reactor operation, with a nominal core outlet gas temperature of about 1340°F, the average core graphite temperature will be about 1700°F. The average fuel compact temperature will be about 1800°F, with a calculated maximum fuel compact temperature of about 2430°F.

The design involves the use of graphite under conditions for which there is little or no previous experience in operating reactors. Estimates of the probable behavior of the fuel elements during operation, therefore, are based primarily on the research and development program associated with this project. A large number of in-pile capsule tests were conducted under the R&D program to investigate the effects of irradiation, temperature, and chemical impurities on mechanical and physical properties, dimensional stability, and permeability of various types of graphite. From these data the behavior of

the graphite used for the Peach Bottom fuel element has been demonstrated reasonably well. However, final proof of the adequacy of the design will be obtained only under actual use in the reactor.

Our review indicates that appropriate consideration has been given to the use of graphite in the design of the fuel elements. The experimental data indicate that all types of graphite contract in size and gain somewhat in strength when irradiated at elevated temperature, and that the amount of contraction is a function of the irradiation temperature and the total fast neutron exposure. It appears that changes in graphite dimensions should have only very slight effects on flow distribution, heat transfer and reactivity. It also appears that structural members are unlikely to fracture due to stresses imposed by asymmetrical changes in dimensions.

It should be pointed out, however, that if the amount of graphite shrinkage were significantly greater than that predicted by the capsule tests, or if neutron fluxes and temperature gradients prove to be significantly greater than calculated, then the stresses attained might exceed the graphite strength limits and fracture the fuel element sleeves. We do not believe the consequences of such fracturing would be serious, since mechanical displacement of fractured parts of individual elements would be limited such that the effects on heat transfer and reactivity should be negligible. In the event that a large number of failures were to occur such that a gross mechanical disarrangement of the core could result, shutdown of the reactor is assured by the emergency shutdown rods which can be forced into the core. Fractured fuel element sleeves could lead to an increased radioactive inventory in the primary coolant system. This situation would not be serious, however, since means for cleanup of the system and replacement of the failed elements are available.

The effect of high temperature and neutron irradiation on the dimensional stability of fuel compacts has also been investigated. Fuel compacts with various ratios of thorium and uranium and varying fuel particle sizes were tested. The results obtained indicate that particle size has an effect on the magnitude of dimensional changes, and that the minimum fuel particle size should be in the range of 110 to 250 microns in diameter in order to minimize dimensional changes. The range of fuel particle size specified for the Peach Bottom reactor is 210 to 595 microns.

Fission products which diffuse out of the fuel compacts enter the helium purge stream within the elements and are prevented from entering the main coolant stream by the low permeability fuel element sleeves. For this reason the resistance of fuel material to release of fission products and the permeability of the sleeves are directly related to coolant contamination. Tests have indicated that release of gaseous fission products from fuel compacts is retarded very significantly by the pyrolytic carbon coating on fuel particles. Capsule irradiation data indicate that the rate of release of fission products from the compacts rises sharply in early fuel life but that it levels off quickly and rises only slightly toward end of life. Fission products that are released from the fuel compacts must permeate the fuel element graphite sleeve in order to enter the primary coolant system. The effective permeability of the graphite sleeve for this outward leakage of fission products is even less than the already low permeability of this material to gases because the helium purge flow produces a lower pressure on the inside of the element. The pressure differential provides a small inflow of helium through the pores of the graphite sleeve which substantially reduces the outflow of fission products. Experimental results indicate that the effective permeability of fuel element graphite to fission product gases will limit fission product migration through the sleeve and into the primary coolant to about two parts in 10,000 of those present in the purge stream. Experimental data also indicates that permeability is not expected to be significantly affected by radiation exposure. Calculations of primary coolant activity levels based on release data from coated fuel particle compacts and the effective sleeve permeability indicate that the activity level should be only about one-fortieth the value used for design purposes. We believe that the fission product barriers in the fuel element are sufficient to limit migration of fission products to the primary coolant to an acceptable level.

Steam generator leakage into the primary system could lead to removal of fuel element sleeve graphite through a steam-graphite reaction. Graphite samples at temperatures of 1650, 1830, and 2190°F were exposed to circulating helium containing moisture. Under accelerated test conditions representing a continuous steam generator leak of 0.01 lb/hr over an entire fuel cycle, effective permeability was increased by less than a factor of two. Such a leak rate would be detected by the moisture monitoring equipment in the primary system, as discussed later. The applicant therefore believes, and we concur, that reductions in graphite permeability due to steam-graphite reactions should be relatively unimportant.

In-pile capsule tests indicate that exposure of the graphite to fast neutrons can cause a significant reduction in its thermal conductivity at operating temperatures and thus adversely affect heat transfer and increase temperatures. The minimum measured values of thermal conductivity for samples exposed to integrated fluxes greater than those expected in the reactor were found to fall in the range of 18 to 24 BTU/hr-ft-°F. A conservative value of 15 BTU/hr-ft-°F, was used in the design. This value was reduced

another 10% at the beginning of core life and 20% at the end of core life for purposes of hot spot analyses.

Inasmuch as the coolant channels in the core intercommunicate, core flow patterns are difficult to predict analytically. Accordingly, 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 lengths of the coolant passages. Similar experiments were performed using internal electric heaters to simulate nuclear heat, and heat transfer coefficients were measured at a number of points for various flow conditions. The results from these tests showed that available correlations may be used to predict average heat-transfer coefficients and pressure losses and that the local heat-transfer coefficient around the circumference of the elements should vary by no more than a factor of about three.

We have reviewed the core hot-channel and hot-spot analysis and believe that it is a conservative analysis. If the very conservative assumption is made that all secondary effects which raise core temperatures occur simultaneously at the hottest point of the hottest fuel element, the calculated normal maximum fuel temperature of about 2430°F would be increased to about 2690°F. The only anticipated consequence would be a localized, moderate increase in fission product diffusion and release. Since only a small part of the core would be involved, the effect on the radioactive inventory of the primary system and the fission product trapping system should be insignificant.

Additional assurance that the Peach Bottom fuel element design should perform satisfactorily may be obtained from the results of in-pile loop testing of two prototype elements. Exposures equivalent to 19% and 93% respectively of the power generated in a corresponding section of an average Peach Bottom element for its projected lifetime were attained. Coolant temperatures and pressure were about the same as the Peach Bottom design values; fuel compact and sleeve temperatures were somewhat higher than maximum design values; and heat fluxes and power densities were significantly higher. Fission product activity in the coolant purge streams were substantially lower than Peach Bottom design values and comparable to the calculated values actually expected in the reactor. These results demonstrate that it should be possible to maintain a low activity level in the primary coolant system using a fuel element and a fission product trapping system such as those in the Peach Bottom design.

Post-irradiation examination of the first prototype element showed no irradiation damage of any type. The second element was inadvertently exposed to water and dropped during removal from the test loop, making the results of the final examination difficult to interpret. However, the general appearance of internal portions of the fuel compacts indicated that irradiation damage was very slight. The graphite sleeve was in good condition. The fact that fission product activity in coolant and purge gases showed no marked change to the end of the test is a further indication that the element had experienced no serious deterioration.

Surveillance of the condition and performance of the fuel elements following startup of the reactor will be carried out as part of a post-construction research and development program. The concentrations of gaseous fission products in the main coolant helium and in the fuel element purge stream should serve as sensitive indicators of fuel element condition and performance. If either of these streams indicate significant fuel deterioration, the reactor will be shut down for identification and removal of defective elements. Such elements would be subjected to detailed hot-cell examination. After the equivalent of 150 days of full-power operation, two fuel elements will be subjected to an in-core visual examination and a gamma scan. Similar examinations will be repeated on two fuel elements every 150 days of equivalent full-power operation up to the end of core life. After the equivalent of 300 days full-power operation, two fuel elements will be removed from the core and subjected to detailed hot-cell examinations. Another two elements will be removed and examined after every additional 300 days of equivalent full-power operation up to the end of core life. In addition to fuel elements, five of the graphite side reflector elements will be removed after the first 300 days of equivalent full-power operation for hot-cell examination. Another reflector element will be removed for examination after every additional 300 days of equivalent full-power operation up to the end of core life. We believe that this surveillance program will provide the necessary information on the condition and performance of the fuel and reflector elements to assure that continued operation will be safe.

Based on our review of the design, the stringent material and manufacturing specifications, and the results of the research and development and testing program relating to the fuel elements, we have concluded that there is reasonable assurance that the Peach Bottom fuel elements should perform in a satisfactory manner. We also believe that the fuel surveillance program is adequate to assure that the reactor will not be operated with seriously defective fuel.

B. Core and Reflector Support. 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 in all directions. The inner portion of the radial reflector is 10.5 inches thick, consisting of three rows of solid hexagonal graphite elements. The remaining 13 inches consist of a segmented ring of 30 graphite blocks which are hinged at their bottom to the core support plate in such a manner that they can tilt in the radial direction.

Individual fuel elements are supported by hollow standoff pins screwed into the core support plate. Graphite sleeves for the control and emergency shutdown rods pass through the support plate. The fuel

elements are restricted from moving vertically by their weight, and laterally by restraint of the standoff pins and the reflector segments. There is no upper grid plate or other mechanical supporting structure for the core. The core is prevented from twisting or tilting circumferentially by the tilting side reflectors which are contoured to engage the full length of the hexagonal reflector elements. During operation, the reflector blocks are caused to tilt inward and apply a compressive force to the core by differential coolant pressure. When coolant flow is stopped, the blocks tilt outward, eliminating the compressive force on the fuel elements.

In order to study the behavior of this core configuration, the reactor designer constructed and operated a detailed half-scale flow model. Results obtained from testing with this flow model (1) verified that the coolant gas pressure drop through the core was conservatively calculated, (2) demonstrated that there should be no areas in the vessel with a deficiency of coolant flow, (3) verified the conservatism of film coefficients used in heat transfer calculations for the inner surfaces of the pressure vessel, and (4) demonstrated that the restraining effect of the tilting reflector blocks on the core is sufficient to hold the core in a stable position, with no detectable vibration during any mode of operation. To further 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 addition to variations in flow conditions, these tests involved vibrations artificially induced in the fuel elements and pressure pulses impressed on the flow. The results indicated that none of the various flow conditions caused the fuel elements to vibrate and that disturbances forcibly impressed were not amplified by the gas flow.

From our review of the core design and the results of the model tests that have been performed, we believe that the core assembly will be mechanically stable under all anticipated conditions.

C. Control Rods and Drives. Reactivity control is provided by two completely independent bottom driven control rod systems. Thirty-six normal control rods powered by independent hydraulic motors with self-contained hydraulic fluid accumulators provide normal regulating and safety control. Nineteen emergency shutdown rods, driven by independent battery powered electric motors provide backup shutdown capability. Additional shutdown capability is provided by 55 top entry, thermally-fused absorbers which would automatically be released in the event of an abnormally high reactor outlet coolant temperature. In all cases, the absorber material is graphite loaded with boron carbide.

The absorbing portion of the normal control rods is comprised of six 17-inch long annular poison segments. These segments are placed on a stainless steel support tube and separated by a set of pre-loaded discs. The segmenting allows the control rods some flexibility to be inserted in a bent guide tube as well as providing

a means of compensating for any irradiation and thermally induced dimensional changes. Each control rod has a loop of metal wire running its entire length which serves as an electrical continuity monitor to indicate any separation of control rod segments or of a control rod from its drive. The entire control rod rides within a graphite guide sleeve with the only rubbing contact being provided by graphite rider rings located between each segment. Control rod cooling is provided by directing a small portion of the reactor inlet helium through each of the control rod guide tubes.

The 36 normal control rods have a total hot reactivity worth of 25% and the most reactive rod is worth about 1%Δ p Maximum core excess reactivity will be about 15% at cold, clean conditions so that the minimum shutdown margin with the 36 rods fully inserted will be about 10%. Maximum rod speed is limited to 0.72 inches per second, corresponding to a control reactivity rate of less than 3.0×10^{-4} $\Delta\rho$/sec. During normal operation the applicant plans to program rods in such a manner that all but three control rods are either fully withdrawn or fully inserted, with only one rod actually moving at a time at power levels above 10% of full power. The consequences of reactivity transients induced by control rod motion are discussed in the Accident Analysis Section.

Each control rod is monitored by position indicators and limit switches to aid in proper positioning of the rods. Instrumentation is provided to indicate possible failures which could result in a loss of the rods control or scram capability. The only control rod system failure which we can identify which could ultimately result in a loss of scram capability for all normal control rods is a loss of the hydraulic header which provides the makeup oil to the individual accumulators for each control rod. These individual accumulators have adequate reserve to provide fluid for the scram of a control rod without makeup for an appreciable time even if the main header were to fail. As a precaution, however, the reactor is automatically scrammed on indication of low hydraulic header pressure. In order to prevent an inserted control rod from falling out of the core, a mechanical latch positively locks the rod into place until it is energized to free the rod.

The applicant has conducted extensive testing on all portions of the control rod and drive systems. A prototype control rod and hydraulic motor assembly was tested in accelerated life tests to more than ten times expected service with no malfunctions or undue signs of wear. Abnormal conditions were simulated by injecting sand into the hydraulic system and graphite chips into the guide tube. The rod performed normally even under these severe conditions. Guide tube misalignment was simulated by bending the tube as much as seven inches off center and the articulated rods and drive continued to operate normally. Tests also indicate that control rod and guide tube temperature and wear characteristics will be satisfactory. From these tests it appears that the control rods have a high probability

of remaining functional even if the guide tubes were distorted much more severely than is anticipated. Examination of the design and the clearances provided also indicate that a fractured guide tube would be unlikely to be displaced enough to cause significant interference. It appears possible that a fractured guide tube could be levitated somewhat if the reactor were suddenly depressurized. However, under this circumstance the control rod itself would remain in place since it is firmly latched to the drive mechanism and all parts of the assembly have more than adequate strength to resist upward movement. The continuity monitor provides high assurance that if a rod should part or become delatched, it would not unknowingly be left in the core on withdrawal of a rod drive mechanism. If a short in the monitor should occur, the circumstance would not necessarily be detected, however, and a rod could drop out during operation. This accident is discussed in the Accident Analysis section. From our review of the design of the control rods and the tests that have been performed, we believe they will be sufficiently reliable to assure adequate and safe control of the reactor.

The electrically driven emergency shutdown rods are a single full length stainless steel tube filled with graphite loaded with boron carbide poison material. These rods ride within graphite guide tubes similar to those used by the normal control rods. Continuity monitors are provided to indicate any separation of a rod from its drive. The significant feature of these rods, as well as their drives, is that they are capable of being inserted into the core against a resistive force of up to 10,000 pounds, which appears adequate to allow insertion regardless of any possible degree of core disarrangement. The 19 emergency rods are worth about 13% in reactivity. They are interlocked such that they must all be fully withdrawn before the normal control rods can be withdrawn. These rods will be used only in the emergency situation where insufficient normal control rods are available to shut down the reactor, due to simultaneous failures of the control rod mechanisms or mechanical disarrangement of the core. They will be tested periodically to verify their operability. From our review of the design of these rods and their actuators and power supplies, we are satisfied that they will function properly under any circumstances where they would be needed.

The thermally released absorbers are 12-inch lengths of graphite loaded with boron carbide suspended above the core within each of the 55 guide tubes of the bottom entry rods. Total worth is about 10% Δp. They are held in place by a fusible link designed to melt and release the rod at a temperature of about 1756°F, some 700°F above normal temperatures in this area. Upon release, these absorbers fall to predetermined locations in the core established by mechanical stops machined inside the control rod guide tubes. From the design and test work done on these absorbers we believe they would perform this intended function satisfactorily.

It is our belief that the normal control system and the emergency shutdown systems are adequately designed and sufficiently reliable to afford safe reactor control under all operating conditions.

D. Reactor Vessel. The reactor pressure vessel is a vertical cylinder with elliptical heads, having a nominal inside diameter of 14 feet and an over-all height of about 35-1/2 feet. The vessel is fabricated of unclad carbon steel (ASTM A 212 - Gr. B) and is designed for 450 psig pressure at 725°F temperature in accordance with the 1959 ASME Boiler and Pressure Vessel Code, Section VIII, including the nuclear case rulings through 1960. The nominal operating pressure of the primary system is 335 psig.

The pressure vessel is nominally 2-1/2 inches thick, except for a 4-1/2 inch thick reinforcing band in the region between the bolting flange for the top head and the mid-plane of the vessel. Two 42 inch nozzle penetrations are provided on opposite sides of the vessel below the top head flange and in the 4-1/2 inch thick wall section. These nozzles accommodate the concentric coolant gas inlet and outlet pipes from each of the primary coolant loops. The vessel is supported by an inverted skirt attached to the reinforced section between these coolant nozzles and the top head flange.

The top head is permanently bolted in place and seal welded. Five nozzle penetrations are provided in the head to serve as access ports for fuel handling equipment. These five nozzles extend upward approximately 8 feet from the vessel head, through a concrete shield to the operating floor level. During reactor operation these ports contain shield plugs and are closed with bolted blind flanges with provisions for seal welding.

Fifty-five nozzles with a nominal inside diameter of 4.3 inches each penetrate the bottom head of the vessel. These accommodate the 36 control rods and the 19 emergency shutdown rods. Three 5-inch nozzles are provided for instrument leads and two 4-inch nozzles are provided for the fission product purge gas lines.

The core support plate is a solid slab of carbon-molybdenum steel, 4-1/2 inches thick and 10 feet in diameter. The plate is supported at its periphery by a welded support assembly which transmits the weight of the core to the bottom head of the reactor vessel. A system of steel T-beams is provided to prevent significant lateral motion of the core plate relative to the reactor vessel.

Heat conducting copper collars are provided around 36 control rod guide sleeves between the core plate and the bottom head of the reactor vessel. A stainless steel collar surrounds the central emergency shutdown rod guide sleeve. These collars are provided to help cool and support the core plate in the event of a complete loss of cooling accident.

Because helium gas is used as the coolant, severe transient thermal stress conditions in the reactor vessel are unlikely to arise. However, levels of steady state thermal stresses depend on maintenance of a relatively symmetric temperature distribution. A steel thermal shield is spaced between the radial reactor reflector and the pressure vessel such that a bulk of the helium returning from the steam generators flows downward along the internal wall of the vessel to the inlet plenum below the core support plate. A portion of the return coolant passes upward along the vessel wall in the annulus formed by the insulated shroud which forms the core exit plenum in the upper portion of the vessel before it passes to the inlet plenum at the bottom of the vessel. With this arrangement no portion of the vessel is exposed to hot exit coolant.

Results from reactor vessel flow model studies indicate that there should be no significant stagnant gas areas and that heat transfer coefficients at the wall will be high enough so that the vessel wall temperatures will be close to the return gas temperatures (630°F). Consequently, high local temperatures and steep gradients are not expected. We have noted in our review that if flow patterns are significantly modified or if internal insulation is not as effective as anticipated, temperatures and stresses could be higher. However, we believe that adequate thermocouples have been installed to detect such circumstances.

The highest 30 year integrated fast neutron flux (3.2×10^{18} nvt at 1.0 Mev) was calculated to occur at the bottom head of the reactor vessel. Assuming continuous operation during this exposure at a lower limit operating temperature range of 500°F, the shift in the nil ductility temperature (NDT) of the steel is estimated to be in the range of 35-75°F. Correlation of Charpy V-notch test results for vessel steel plate using the current ASME Code Section III criteria indicates a minimum NDT of about -30°F. Adding the 75°F maximum estimated value of shift in NDT and the 60°F safety margin required by Section III of the ASME Code to the -30°F NDT value results in a critical operating temperature of 105°F at the end of vessel design life. Tests on the forged steel material used in the vessel flanges and nozzles indicate a minimum NDT value of about 0°F. Using the 75°F maximum estimated shift in NDT value due to exposure and the 60°F safety margin results in a critical operating temperature of 135°F for these flanges and nozzles at the end of vessel life. In both cases there is a substantial margin of safety between the critical operating temperature and the lowest service temperature of 500°F for the vessel. Although it does not appear that NDT will be a problem in this vessel, specimens of vessel steel will be exposed in the reactor to permit surveillance. We have examined the details of the surveillance program proposed and believe that it will provide the necessary information to ascertain the continuing safe operating condition of the reactor vessel.

The normal full load operating pressure of the primary system is 335 psig but it may, on occasions, be somewhat greater. The pressure should, however, never rise above 390 psig which is the setting of the helium dump valves to the dump tanks. These dump valves are backed-up by ASME code safety valves set at 437 and 450 psig. From our review of the pressure relief system we are satisfied that settings and capacities are such that the primary system is unlikely to be over-pressurized under any circumstances.

It is our opinion that the reactor vessel design is compatible with its operating requirements. We have noted that quality control during fabrication was stringent. We therefore believe that the vessel is adequate to meet design service requirements.

V. NUCLEAR PHYSICS

The analytical techniques used in core physics design were verified by a series of physics experiments. Consequently there is reason to believe that physics predictions are reasonably accurate. A lattice-cell critical assembly containing a central 19 fuel element mockup provided basic information on the resonance-absorption characteristics and the doppler coefficient of thorium. The measured values were in good agreement with those calculated from theoretical considerations. Additional experimental verification was obtained from a one-sixth-size critical assembly of the Peach Bottom core. The results from this critical assembly indicate that over-all reactivity can be calculated within an uncertainty of 0.01 $\Delta\rho$ and the over-all temperature coefficient with an accuracy of about $\pm$ 10%. Results of the theoretical calculations will be further verified by physics experiments which will be performed during initial operations.

In order to enhance the effectiveness of the control and burnable poisons and to provide a more uniform power distribution throughout the core, the rhodium, boron, thorium, and uranium are distributed in three axial and three radial zones. Power peaking will also be minimized by control rod programming, so that it is expected that the maximum peak-to-average power ratio in the core will be about 1.59 throughout life.

At the beginning of core life the clean excess reactivity with all control rods removed is estimated to be about 15% at room temperature and about 9% at operating temperature. With all control rods inserted the minimum cold shutdown reactivity margin is estimated to be 10%. Since reactivity does not increase with burnup, it appears that more than ample shutdown margin will always be provided by the normal control system. Since the maximum worth of an individual control rod is less than 1% $\Delta\rho$, more than 10 rods would have to be withdrawn to make the cold clean core go critical. Thus there is considerable margin available for the possibility of stuck rods.

In the unlikely event that the normal control rods are not able to shut down the reactor, we believe that the negative reactivity available in the emergency shutdown rods and the thermally released absorbers is sufficient to assure a complete shutdown under any credible circumstances.

The core has been designed for a full power reactivity lifetime of 3 years with an 80% load factor, corresponding to an average burnup of 60,000 mw-day/tonne of initial thorium and uranium (7-8% burnup of the U-235 loading). Calculations indicate that the ratio of maximum to average fuel burnup in the core will be less than 1.4. More of the fission product poisons, notably the xenons, cesiums, and samariums, may diffuse out of the fuel and be carried out of the core than was assumed for the reactivity calculations. For this reason it is possible that excess reactivity may still be available at the end of the design lifetime and core life may be extended if conditions permit. We see no significant increase in hazards associated with a reasonable extension of the burnup, provided that the results from the fuel and control rod surveillance programs have been favorable.

Xenon in the Peach Bottom reactor is of little consequence. The reactor has a relatively hard neutron spectrum due to its high moderator temperature (1200°K) and as a result, the absorption rate per atom of xenon is relatively small. The peak xenon poisoning (hot) is estimated to be only 1.12 times the equilibrium poisoning and will occur about 5 hours after shutdown. There should also be no problem with respect to xenon oscillations, since the relatively small, closely coupled core should not sustain fundamental mode xenon oscillations.

The temperature coefficient of reactivity has been computed in considerable detail for both the beginning-of-life and end-of-life conditions. The values calculated at operating conditions are as follows:

TABLE I

Temperature Coefficients

Beginning-of-life:	
Prompt:	-2.8×10^{-5} $\Delta p/^{o}C$
Over-all:	-4.5×10^{-5} $\Delta p/^{o}C$
End-of-life:	
Prompt:	-2.0×10^{-5} $\Delta p/^{o}C$
Over-all:	-2.3×10^{-5} $\Delta p/^{o}C$

The calculated values do not change significantly at higher temperatures. At lower temperatures they are even larger in magnitude. Both the prompt and the over-all coefficients at beginning-of-life include a negative contribution from the rhodium-103 which is added to take advantage of the neutron spectrum shift toward the very strong absorption resonance of this isotope at 1.26 ev as temperature increases. The magnitude of this contribution at operating temperatures is -1.1×10^{-5} $\Delta\rho/^{o}C$ which is appreciable. However, we believe that the magnitude of the prompt coefficient is large enough so that even if the rhodium contribution were completely neglected, reactor behavior would not be appreciably affected. The rhodium initially added is burned up during operation but is replaced by fission product Rh-103, so that the negative temperature contribution of rhodium should not vary much with core life.

The power coefficient of the reactor was evaluated as a function of power for both beginning and end-of-life conditions. At full power, the value is -1.1×10^{-4} and -0.7×10^{-4} $\Delta\rho/mw(t)$ for beginning and end-of-life respectively. These correspond to about 1.5 and 1.0 inches of average control rod motion per megawatt of power change. Because of the relatively large heat capacity of the core, the large magnitude of temperature coefficients, and the relatively long neutron lifetime ($\sim 2 \times 10^{-4}$ sec.), dynamic response of the core to reactivity disturbances is expected to be comparatively sluggish.

The helium coolant has a very small reactivity worth. The presence of helium at full pressure results in an increase in reactivity of less than 0.001 $\Delta\rho$ over the condition with no helium, so that any reactivity effects due to changes in helium density resulting from temperature or pressure transients are negligible.

As previously stated, the core structure is expected to be mechanically stable and reactivity changes associated with core looseness and possible skewing or tilting should be negligible under all normal operating conditions. Aside from reactivity changes induced by the control system we can identify no other mechanisms that are likely to affect short term stability.

It is our belief that physics characteristics of the reactor have been predicted with reasonable accuracy. The shutdown control margins are adequate relative to the excessive reactivity in the core. Strong negative temperature coefficients of reactivity are expected and reactor kinetic calculations indicate satisfactory behavior during reactivity transients. We therefore believe that there is no reason for concern about the nuclear physics characteristics of this reactor.

VI. REACTOR COOLING SYSTEMS

A. Primary Coolant System. The primary coolant system includes two separate loops to circulate helium coolant gas from the reactor outlet through the steam generators and back to the reactor vessel inlet. Under full power conditions, nominal inlet and outlet temperatures are about 635°F and 1340°F respectively. Quick closing stop valves are provided in each loop at both the suction and discharge of each main helium circulator and in the hot reactor outlet line to the steam generator.

Between the reactor vessel outlet and the steam generator inlet, the hot and cold leg piping is concentric. Hot coolant flows in the central pipe to the steam generator and cooler inlet helium returns to the reactor vessel in the outer pipe. The working pressure on the high temperature pipe is limited to the pressure drop across the reactor vessel of less than 5 psi, and stainless steel reflective insulation on the inner surface of the high temperature pipe limits its temperatures to about 700°F. All piping, including the inner concentric pipe, is carbon steel designed for 450 psig to Section VIII of the 1959 ASME Boiler and Pressure Vessel Code, including applicable nuclear case interpretations, or to the ASA Code for Pressure Piping, including applicable nuclear case interpretations. The piping is installed with large expansion loops and minimum restraint. In addition, the inner concentric pipes contain bellows expansion joints to allow for differential expansion between them and the outer pipes. Satisfactory performance of the hot-leg pipe depends directly on the insulation provided. We have reviewed the properties of the insulation and the manner in which it is attached in some detail and found that it should perform adequately and should not be readily subject to damage or deterioration under operating conditions.

The main drives for the helium circulators are 3600 rpm, 2300 volt induction motors coupled to the circulator through a variable slip hydraulic coupling. Speed control is accomplished by remotely controlling the slip in the hydraulic coupling. An auxiliary drive is provided in the form of a 440 volt pony motor concentrically mounted on the motor shaft between the circulator and the hydraulic coupling. Operation of the circulators with the pony motors supplies about 10% flow which is more than adequate to remove decay heat while the primary coolant system is pressurized. In the event the main drives and the pony motors are unavailable, natural circulation in a single pressurized loop is adequate to cool the core sufficiently to prevent fuel damage.

The two main helium circulators located in the non-concentric portions of the cold leg piping are horizontal, centrifugal, single-stage machines, with an overhung impellor which requires only one shaft seal. The compressors, including the seals, have been extensively tested under various conditions, including sustained operation in surge, with no apparent damage. These tests indicate that the circulators should perform adequately and reliably under all conditions.

As a result of our review of circulator characteristics, we believe that centrifugal machines of the type used seldom suffer damage from being in a surge condition. Nevertheless, the applicant intends to further verify their ability to withstand surge by testing them in the plant under surge conditions.

The steam generators are the vertical U-tube type with water or steam in the tubes and the primary coolant helium on the shell side. The tube bundles are divided into economizer, evaporator, and superheater sections. Steam separation is accomplished in a spherical steam drum with the saturated steam from this drum being returned to the steam generator for superheating. The hot helium entering the steam generator first encounters an impingement baffle provided to reduce the possibility of erosion of tubes and then is directed across the superheater, evaporator and economizer sections which reduce the helium temperature to below 700°F before it first comes into contact with the shell.

The steam generators and the steam drums are designed in accordance with applicable sections of the ASME Boiler and Pressure Vessel Code. Detailed stress analyses were performed on all major internal and external portions of the vessel. These analyses indicate that the units are designed with a suitable margin of safety for all normal operating conditions and that failure is unlikely even under severe transient conditions. The steam generators are suspended from hanger rods which give freedom of movement horizontally in response to thermal expansion of piping. Hydraulic snubbers limit the rate of this movement in response to any externally induced forces.

Since the water and steam side of the steam generator operates at pressures as much as 1200 psi higher than the primary coolant helium side, water leakage into the helium is possible. Since leakage is most likely to develop at the tube to tubesheet joints, a baffle has been placed adjacent to the main tubesheet on the gas side to provide a purge space between the baffle and the tubesheet. A small helium flow is provided in this space to sweep any steam leakage away to the Coolant Purification System.

Moisture detectors which continuously sample the steam generator outlet helium are provided to detect any significant steam leakage and automatically isolate the effected steam generator from the reactor. An isolation signal shuts down the reactor, closes the three isolation valves in the affected loop, closes the feedwater valve to the affected steam generator, and dumps water in the affected steam generator until the steam generator pressure is reduced to approximately 50 psi above the helium system pressure. The applicant has stated that helium loop isolation can be completed in about 11 seconds from the initiation of a significant leak. From our review of the isolation system, we feel this is a reasonable estimate; however, it will be verified during preoperational testing.

Primary coolant system overpressure protection is provided principally by a dump valve set to operate at 390 psig which relieves to the helium dump tanks. If the dump valve should fail to operate properly,

additional protection is provided by three sets of safety valves set to relieve at 437 psig and 450 psig. Rupture disks, which relieve at a lower pressure, are upstream of each safety valve in order to minimize system leakage. Each pair of safety valves has an upstream block valve which permits shutting off one but not both of the discharge lines to permit maintaining or testing a valve or changeover in the event a valve fails to reseat.

We have reviewed the design of the primary cooling system for its ability to provide cooling to the core as well as for its mechanical integrity. Because of the rugged design of the main helium circulators and because of their auxiliary drives, we believe that one of these machines will be available to circulate coolant under all normal or abnormal conditions. We also believe that the careful attention paid to the design of the primary cooling system components and the liberal margins provided between design and operating conditions makes the probability of a major system rupture very small. In the event of a small leak in one of the cooling loops, the combination of radiation detectors in the loop cavities and the quick closing valves makes it probable that the damaged loop can be isolated without release of radioactivity beyond that contained in the coolant which escapes. In the less probable situation of a major rupture, pressure detectors in the loop and reactor vessel cavities should sense the increase in pressure associated with the release of coolant to the relatively confined cavity. These pressure detectors will also initiate closure of the loop isolation valves and provide an anticipatory signal to isolate the containment building before any radioactivity has escaped to the environment. Because of the generally conservative structural design of the primary cooling system and because of the equipment installed to limit the consequences of any accident which might occur, we have concluded that the system is adequate from a safety viewpoint.

B. Emergency Cooling System. The reactor vessel emergency cooling system is provided as a last-ditch system to be used following a depressurization accident if no other means of removing heat from the reactor vessel is available. Cooling coils for the system are welded to the outside of the cavity liner and heat from the reactor vessel is removed by once through water flow which is manually initiated on need. Valving, headers, and instrumentation for the system are all located outside the containment, directly under the main control room where they are accessible for operation and periodic testing.

The system contains two independent sets of full capacity cooling coils. Cooling water can be supplied by a number of means.

Normally, the emergency cooling system is dry and pressurized with gas. Because of the temperature difference between the dry cooling coils and the incoming cooling water, the procedure for placing the system in service is necessarily time consuming in order to avoid thermally shocking the tubes upon introduction of water. However, under conditions where the system might be needed, many hours are available for

initiation of its use and the operators are protected by ample shielding. We have reviewed the emergency cooling system in considerable detail and are convinced that the system has been designed to guarantee its adequacy if it should ever be needed.

VII. FISSION PRODUCT TRAPPING SYSTEM

As previously discussed, a significant fraction of the volatile fission products produced during operation is expected to diffuse from the graphite fuel compacts and enter the fission product purge stream within the elements. After passing through the internal traps in each element, the purge flow is taken to an external fission product trapping system outside the reactor vessel. Some degree of coolant contamination is expected by diffusion of volatile fission products through the graphite fuel element sleeves. However, as previously indicated, the level of coolant contamination is expected to be considerably less than the level of gaseous and suspended activity used in the design, i.e. 4225 curies. The Fission Product Trapping System operates on the principle of selective gaseous adsorption of fission products on activated charcoal. Since the volatility of the various fission products, and inversely their affinity for adsorption by the activated charcoal, is a function of temperature, the trapping system is divided into a series of separate beds, cooled to successively lower temperatures. The final bed is cooled to the temperature of liquid nitrogen in order to trap and retain the highly volatile isotope krypton-85. The preceding beds in the system are cooled to intermediate temperatures and delay or retain the other biologically significant isotopes.

The internal traps are located directly beneath the lower graphite reflector of each fuel element. Each of these traps consists of a graphite cylinder containing longitudinal slots filled with activated charcoal through which the helium purge stream flows. These traps are expected to remove or significantly delay the easily condensible fission products such as iodine, cesium, tellurium, barium, strontium, antimony, rubidium, and bromine from the helium purge stream that flows through the fuel element. For example, on the basis of test data it is expected that bromine and iodine will be delayed at least 50 days in the internal traps.

Most of the external trapping system is located in the basement area of the containment building. The system includes piping, valves, helium compressors, heat exchangers, and the charcoal delay beds. Also included are moisture separators, a chemical cleanup system, graphite dust filters, and oil removal filters for the purpose of cleaning nonradioactive contaminents from the reactor helium coolant. The purge flow passes successively through two delay beds with intermediate water cooling, then through five low temperature delay beds with intermediate freon refrigerant cooling and then back to the primary cooling system. About 10% of the flow from the low temperature delay beds is shunted through the liquid nitrogen-cooled trap where Kr-85, is collected and stored for eventual disposal. The applicant calculates an effective delay of 35,300 hours for xenon passing through the system and 189 hours for krypton. Based on test results which the applicant has presented, and experience gained with a similar system at the chemical processing plant at the National Reactor Testing Station in Idaho, these calculated delay times appear reasonable.

Pilot operating experience with a fission product trapping system of the type proposed was gained in the prototype fuel element irradiation tests previously described. Additional verification of the basic principle of operation has been obtained from work done in connection with the AERE Dragon Project in Great Britain and from experience on similar installations in the Homogeneous Test Reactor at the Oak Ridge National Laboratory and at the Chemical Processing Plant at the National Reactor Testing Station. Review of operating experience available from these installations leads us to believe that the Peach Bottom design is fundamentally sound and workable.

Because of temperature conditions that range from 660°F at the head end of the system to -320°F in the Kr-85 trap, a number of different materials have been employed for construction of vessels and other components. From our review we are satisfied that materials are compatible with service requirements. The system is designed for 450 psig and will normally operate between 290 and 440 psig. In order to decrease the possibility of a depressurization accident and release of the relatively high concentration of activity in the inlet end of the system, the portion of the system from the reactor vessel past the second water-cooled delay bed is doubly contained, and is provided with a high pressure purified helium atmosphere in the intermediate space. This helium is monitored by pressure and high flow detectors to indicate rupture of either containment barrier. We believe the applicant has made a reasonable effort to reduce the likelihood of a mechanical failure of this system. If a leak or rupture should occur, the system is designed with sufficient valving and instrumentation to allow isolation and bypassing portions of the system. The consequences of ruptures in the external fission product trapping system are discussed later in the Accident Analysis Section.

A sudden loss of all helium purge flow through the trapping system is considered unlikely, but would be detected and called to the operators attention so that the reactor would be shut down immediately. Calculation of the temperature rises that would occur in the delay beds which are normally cooled by the purge gas itself indicate that the temperatures attained would be well below the design temperatures of the delay bed vessels. This is true even if the delay bed emergency cooling water system provided is not initiated by the operator as he should before vessel temperatures reach 200°F. Each delay bed has an emergency cooling jacket supplied from the Critical Cooling Water System. This system will provide partial, if not total, cooling in the event of a failure of the normal cooling water system. We have reviewed the design of the Critical Cooling Water System and believe it to be a highly reliable cooling water supply.

The probability of losing helium purge flow because of the coincident failure of both purified helium compressors which normally provide helium purge flow through the external fission product trapping system is very low. If both compressors did fail, purge flow could be continued for about 1-1/2 hours to the helium dump tanks. If one of the compressors could not be repaired during this time it would be necessary to resort to emergency cooling of the system. The probability of loss of refrigeration cooling in the low temperature delay beds is minimized by providing a complete spare

refrigeration system. However, if both systems should fail, emergency water cooling would be initiated and the limited amount of fission products that would leave the beds could be processed indefinitely by the liquid-nitrogen-cooled Kr-85 trapping section.

In the event of a sudden depressurization of the primary cooling system, a purge backflow protection system will automatically close the stop-check valves at the inlet to the trapping system and will isolate all the low temperature delay beds as discussed later in the Accident Analysis. In the event of such an isolation, the emergency cooling system discussed above would be used to cool the beds.

It is expected that periodically the Kr-85 trap will become saturated and need regenerating. This will be done by energizing built-in heaters and off-gassing the adsorbed krypton to a storage vessel provided for this purpose. The krypton may then be vented up the stack at a controlled rate or it may be pumped to a shipping container for off-site disposal.

Based on our review of the design of the fission product trapping system we have concluded that it can be used as proposed without undue risk to the health and safety of the public.

VIII. INSTRUMENTATION AND CONTROL

A. Plant Control. The Peach Bottom plant is designed to supply superheated steam to a turbine-generator in accordance with modern steam plant practice. Thus, the plant control systems allow the reactor to supply power as demanded by the turbine throttle setting. The automatic reactor control system adjusts the regulating control rod in response to a demand signal from steam temperature. The helium circulator speed controller in turn adjusts helium coolant flow in order to maintain steam header pressure. The operation and stability of the automatic control systems have been studied using an analog computer and it appears that the actual systems should perform satisfactorily.

In the event of a malfunction of the automatic control equipment, either of two undesirable events could occur: coolant flow could be changed to an unacceptable value, or the regulating rod(s) could be withdrawn. Either of these events would occur relatively slowly and ample time should be available for the operator to assume manual control. If the operator does not assume manual control, the reactor safety system will automatically place the reactor in a safe condition.

B. Reactor Safety System. The reactor safety system is designed to provide three distinct safety actions: (1) scram, all 36 normal control rods are driven into the core at high speed; (2) setback, three preselected control rods are driven into the core at their normal regulating speed; and (3) helium loop isolation, shuts off the helium compressor and closes the stop valves in the affected loop so as to isolate it from the reactor (restricted to only one loop at a time by interlocks). Insertion of the 19 emergency shutdown rods is initiated manually by the operator at the reactor control console only in the unlikely event that the normal control rods do not shut down the reactor.

Table VIII shows the conditions and tentative setpoints for causing safety system actions. We have reviewed these setpoints as well as the criteria for their being bypassed or inserted and believe them to be reasonable. The absence of any fuel burnout ratio or short-term fuel melting considerations allows greater than usual latitude in the absolute values of these setpoints.

The safety system uses 2 of 3 logic to trip each of the two series scram valves for each control rod. Both scram valves to a control rod must trip in order for that control rod to scram. We have conducted a detailed review of the safety system to determine whether logic circuits to each set of scram valves are sufficiently reliable. We have found that the system is well designed and that it should be highly reliable.

Most of the circuitry and components used in the safety system are similar to those used in other nuclear power applications. They are well proven and have been tested to military specifications by the equipment vendor. In addition, all of the installed equipment is being performance tested by the applicant, including testing at the expected extremes of supply voltage, supply frequency, and ambient temperature. One of the major features contributing to system reliability is the provision of sufficient redundancy to permit testing of the entire system, including the scram valves, on a regularly scheduled basis with the reactor operating.

Because of the proven design of the reactor safety system and the degree of testing to which it has and will be subjected, we have concluded that the system is adequate for this reactor.

TABLE VIII REACTOR SAFETY SYSTEM

Condition	Scram	Setback	Helium Loop Isolation (Note 7&10)
High Flux, Rated Power (Power Range)	140% (Note 1)	130%	-
High Flux, Rated Power (Intermediate Range)	0.01% (Note 2)	-	-
High Flux, Rated Power (Source Range)	10^{-4}% (Note 5)	-	-
High Rate of Flux Change, at less than 10% Rated Power, DPM	5 (Note 4)		
High Coolant Temperature, °F	1500	1450	-
Low Feedwater Pressure, psig	800 (Note 7)	-	-
High Total Moisture, ppm	200 (Note 7)	-	-
Low Control Rod Hydraulic Header Pressure, psig	2500	-	-
High Differential Moisture, ppm	-	-	200 (Note 6)
Low Coolant Pressure, psig	210 (Note 7)	-	-
High Loop Cavity Pressure, in. H_2O	-	1 (Note 7)	1 (Note 9)
Low Compressor Speed, rpm	(Note 3)	75 (Note 7)	75
Low Steam Drum Level, in.	-	15 (Note 7)	15 (Note 8)
Low Recirculating Pump Δp, psi	-	3 (Note 7)	3 (Note 8)

NOTES:

(1) High power level scram will be reduced to 5% rated power during 1 Mw application.
(2) Used only during Low Pressure Operating Mode.
(3) Scram occurs on loss of both 2400 V buses (main circulator motors).
(4) Reduced to 2 DPM during fuel loading and zero power tests.
(5) Used only during fuel loading and zero power tests.
(6) Also actuates the Steam Generator Emergency Water Dump System.
(7) Bypassed when reactor is in Low Pressure Operating Mode.
(8) Bypassed on low feedwater pressure.
(9) Blocked on high reactor cavity pressure.
(10) It shall be possible to isolate only one loop at any one time.

IX. CONTAINMENT BUILDING

The containment building is a cylindrical steel pressure vessel with dished heads designed in accordance with the 1959 ASME boiler and pressure vessel code, Section VIII, and applicable nuclear code cases for a pressure of 8 psig at 150°F. Maximum specified leak rate at design pressure is 0.2% of the contained volume per day. The building is 100 feet in diameter, 162 feet in over-all height, and has a free volume of 720,000 cubic feet.

The containment building design pressure and temperature were established by calculating the peak pressure and temperature that would occur as a result of rupture of the primary coolant system and simultaneous failure of a steam generator tube. From our review of various potential accidents, we believe that selection of this accident as a design basis was adequate.

The containment vessel is fabricated of ASTM A-201 Grade B fire box quality steel made to ASTM A-300 specifications. Since the containment vessel was constructed in accordance with the 1959 Code, Charpy key-hole impact tests were performed on the material to demonstrate its impact strength. However, a number of V-notch specimens were also tested. Correlation of these V-notch tests with more recently established criteria of nil-ductility transition temperature (NDT) indicates that the NDT for steel is approximately -50°F. Taking no credit for heating of the steel by the atmosphere within the building, the minimum service temperature of the containment vessel will not be less than -15°F, which is the minimum instantaneous ambient temperature expected at the site. Under these temperature conditions the steel meets the 30°F NDT safety margin required by Section III of the 1963 ASME Code.

Containment building penetrations include 3 personnel air locks, one equipment access door, one fuel element transfer elevator penetration, one vacuum breaker, approximately 30 electrical penetrations, and approximately 74 mechanical or piping penetrations. Seven penetrations include stainless steel bellows to provide for thermal movements between the pipe and the containment building.

Closed loop lines not open to the containment atmosphere are not fitted with isolation valves, and the internal pipe loop is regarded as the containment boundary. All lines which are normally open to the interior of the containment have two valves in series. From review of each of these lines, we are satisfied that provisions for isolation are adequate and that non-isolable loops are adequately protected from being damaged during an accident.

Automatic isolation of the containment building will be initiated by the detection of high radiation levels in the exhaust gas or by high differential pressure between the general containment and the reactor cavity or either primary loop cavity. Containment isolation can also be initiated manually by the reactor operator. Four radiation monitors are provided in the ventilation exhaust system; one in each of the air and inert gas exhaust plenums and two in the stack. Each of these monitors will alarm on detecting an above normal activity level. Two-of-three logic is used to effect containment isolation. A simultaneous high

radiation signal from either plenum monitor and one of the stack monitors will cause isolation of the corresponding portion (air room or general containment) of the containment building. A simultaneous high radiation signal from both stack monitors will, after a short time delay, cause isolation of the entire containment building. Calibration of the ventilation exhaust monitors will be checked at least monthly. Satisfactory initiation of containment isolation by the radiation monitoring system will be tested at least quarterly. From our review of the containment isolation system and the periodic testing proposed we are satisfied that it will function reliably to effect containment isolation under all required circumstances.

The following conditions must be established before the containment building may be opened: (1) the reactor must be shut down, depressurized and cooled below 750°F so that a graphite fire cannot occur; (2) equipment must be operative for removal of decay heat; (3) removal of spent fuel must not be in progress; and (4) all vessels other than the Kr^{85} hold-up tank must be depressurized if they contain significant amounts of radioactivity. While the containment is open, the Kr^{85} trap will not be regenerated and the Kr^{85} hold-up tank, which is designed for 550 psig, will be at a pressure of less than 275 psig. Under these conditions the possibility of a Kr^{85} release is very remote; however, even if the maximum Kr^{85} inventory were released with the containment building open, potential off-site exposures would be well within the guide line values set forth in 10 CFR 100. Since the conditions outlined above preclude all conditions which could lead to a large release of radioactivity, we believe these provisions are satisfactory.

The containment building is divided into two volumes--the general containment volume and a much smaller volume called the air room space. Each has its own ventilation system. During normal power operation, the general containment volume is filled with a reduced oxygen atmosphere and the air room space is provided with normal fresh air. Most of the critical and high maintenance equipment has been located in the air room space so that operators have routine access via two of the three personnel air locks. The general containment will not be routinely accessible during operation because of the reduced oxygen atmosphere. However, access will be possible by the use of portable breathing equipment under emergency conditions.

When inerted, the general containment atmosphere will contain no more than 4% oxygen so that a concentration of less than 5% would result if the atmosphere were mixed with the air in the air room space. Limiting oxygen to 5% precludes the formation of an explosive mixture with combustible gases that might be generated by a steam-graphite reaction during the course of an accident. It also inhibits graphite oxidation under like circumstances.

The oxygen concentration in the containment atmosphere can be determined on samples taken from 28 locations. The concentration at all sample points will be monitored immediately after inerting the atmosphere and periodically thereafter as required. We believe that this will provide adequate assurance that the inert atmosphere is maintained when needed. Appropriate conditions under which the general containment atmosphere need not be inerted have been established and will be included in the Technical Specifications.

The containment ventilation system also includes an internal recirculation system consisting of two 2000 SCFM fans and a bank of high efficiency fire-proof filters. In the event of an accident, this system will be placed in service to reduce the amount of airborne particulate activity present in the building. We have reviewed the design and installation of this system in detail and have concluded that it will function reliably under accident conditions.

The containment building has been subjected to an initial integrity test at 115% of design pressure, as required by the ASME Code. An integrated leakage rate test at the 8 psig design pressure will be completed prior to initial operations to verify that the 0.2% per day maximum design leakage rate is met.

In reviewing the containment building, we have noted that the design specifications were based on the most severe accident postulated and that the building is being constructed and tested in accordance with accepted industry standards. An acceptable design leakage rate has been established, and initial and periodic testing will be performed to confirm that this specification is met and maintained. The feature of a depleted oxygen atmosphere eliminates the possibility of excessive oxidation of core graphite or combustion of reaction gases, thereby assuring that the building cannot be over-pressurized in the event of an accident. Based on these considerations, we believe that the containment building is adequate from a safety standpoint.

X. FUEL HANDLING

Refueling of the Peach Bottom core is expected to be done on a batch basis on the order of once every three years. In order to implement the surveillance program, additional handling of a limited number of core components will be accomplished during the initial core life. Refueling is conducted with the containment closed but with air in the building and with the reactor shut down, depressurized, and cooled so that the exit helium temperature is less than 450°F.

Refueling is accomplished by individually removing fuel elements from the core, using a fuel transfer machine operating through the central fuel handling port, and placing the element in a parking hole in the core reflector region. The element is removed from the parking hole through another fuel handling port by a shielded charge machine which transfers it to a fuel element canning machine located in the reactor biological shield. After canning, the element is transferred through a fuel transfer chute to the spent-fuel storage pool located outside of the containment. New fuel is installed in the reverse manner except that the charge machine obtains a new fuel element from the new fuel pickup cell. Viewing equipment is provided to allow the in-core portion of the operation to be observed; however, this viewing is not required for refueling since the fuel transfer machine can be operated and positioned remotely.

The fuel transfer machine is essentially a mechanical arm which can be inserted in the reactor vessel, rotated, and indexed over each fuel and reflector element. A grapple engages the lifting knob of each core element and releases only when the element is set down. The fuel transfer machine can be adapted to handle other core components. A failed fuel element detector can be attached to the fuel transfer machine to assist in locating a failed element.

The charge machine is a shielded cask mounted on a carriage which moves on rails. A hoist and grapple inside of the machine picks up an element and raises it into the machine. The charge machine can also be used to make gamma scans of fuel elements and control rods in order to verify their integrity as part of the planned surveillance programs. The canning machine, located at operating floor level within the biological shield, is capable of welding a cap on a fuel element canister and test it for leaks. The machine is also capable of repairing a faulty weld.

Manually operated isolation valves and special flexible seals are provided to permit making and breaking connections between machines. Shielding and purging systems are provided to reduce the release of radioactive gas and the direct radiation during fuel handling operations. Interlocks and weight sensors which are included should prevent or indicate faulty operation.

The spent-fuel storage pool, located outside the containment building, is a water-filled, undrainable double-walled concrete pit with the air space between walls monitored for leakage. The canned fuel elements are stored underwater in safe-geometry racks until they are removed for shipment. If a leaking can is detected, it may be returned to the canning machine for rewelding or it may be placed inside a slightly larger can specifically designed for that purpose.

We have reviewed the design of the fuel handling equipment and can identify no serious hazards associated with its use. Fuel elements are handled throughout all transfer operations in a helium atmosphere. Even if this atmosphere were replaced by air, combustion would not occur since decay heat generation rate is too low to heat an element significantly above the refueling temperature. Similarly in the event of an equipment

breakdown, an element would not rise to temperatures high enough to cause undue activity release. Finally, the dropping or breaking of a fuel element is unlikely to lead to a significant release because the graphite matrix and the individually coated fuel particles in the fuel compacts would retard release from broken fuel compacts. It is possible, of course, that a local hazard could arise during the handling of fuel elements should equipment seals or valves fail. However, any released activity would be in the containment atmosphere. Procedures for minimizing personnel hazards in such events appear adequate. The fuel transfer chute is designed to maintain containment building integrity while elements are being transferred through it to the pool. Radioactive debris on the outer surface of a fuel element can could be transported into the pool. However, the likelihood that any such contamination could become airborne is small since the chute discharges beneath the surface of the pool. Cleanup equipment for the pool water is provided.

We believe the fuel handling operations proposed will not result in a significant fission product release to the environment and that adequate provisions have been made to minimize personnel hazards.

XI. CONVENTIONAL PLANT AND SERVICES

A. Steam Plant

The Peach Bottom Plant is capable of generating steam at modern conditions of 1000°F and 1450 psig. The superheated steam drives a single, outdoor, 45,900 Kw turbine-generator unit. A turbine bypass with desuperheating equipment can pass steam equivalent to 30% of rated reactor power to the main condenser when the turbine-generator is not operating. Condenser cooling water is circulated from Conowingo Pond by a single circulating pump. Condensate from the condenser is returned to the steam generators by duplicate condensate and feedwater pumps. An additional reduced capacity pump is provided to supply feedwater during startup, shutdown, or loss of feedwater emergencies.

The steam generator is a forced recirculating unit. If the steam generator recirculating pump were to fail, system elevations are such that each steam generator can continue to operate at 50% of its rated capacity. As a convenience following plant shutdown, a separate sub-boiling shutdown heat exchanger is provided to remove heat directly from the evaporator section of either steam generator and, thus, allow complete shutdown of the remainder of the steam plant.

We have reviewed the functional design of the steam plant and, because of the multiplicity of cooling methods provided, have concluded that some means should always be available for removing heat from the reactor by at least one of the two loops in the primary cooling system.

B. Electrical Power

The normal supply of power to operate plant auxiliaries is furnished by either the station turbine-generator or by a 220 Kv transmission line tied to the Philadelphia Electric Co. interconnected system. A reserve power source with sufficient capacity to supply all shutdown loads is provided by a separate 33 Kv line and transformer for use in the event that power is not available from the normal sources. Additional emergency power is available from an automatically started 1000 Kw diesel-driven generator and from an emergency station battery. The diesel-driven generator is capable of supplying the full emergency load of about 900 Kw for an indefinite period. In order to maintain continuity of power to emergency loads, the station battery can supply these loads until the diesel starts by serving as the power source to two d-c to a-c motor generator sets, each of which supplies power to separate essential buses. Normally these d-c to a-c machines are operated in the opposite direction to maintain a constant charge on the battery.

The major emergency electrical loads are comprised of key instrumentation, the critical service water pump, and the helium circulator pony motors. In the event that the normal power supplies are lost, continuity of instrumentation power is maintained by the station battery supplied critical power system. Continuity of main helium circulator power may be maintained by keeping the main turbine-generator connected to the auxiliary electrical system until its rotating inertia is no longer sufficient to allow it to supply useful power. A separate emergency storage battery is used to provide excitation for the main generator during this coastdown period. Power is then supplied from the 33 Kv reserve source or from the diesel-driven generator and the main helium circulators are switched to their pony motors.

In summary, adequate electrical power can be supplied to the Peach Bottom Plant to allow an orderly shutdown at any time from: (1) the 220 Kv transmission line, (2) the separate 33 Kv transmission line, or (3) the emergency diesel generator. Temporary power supplies are available in the form of the rotating inertia of the station turbine-generator and the plant's emergency battery which may be used until the load is assumed by one of the other power supplies. It should be noted that the station turbine-generator normally supplies station power so that loss of external station power sources would not necessarily result in a plant shutdown since the plant can continue to operate supplying only its own auxiliaries.

Although a total loss of station power for up to several hours is tolerable, we believe that the multiplicity of available power sources is adequate to assure a continuous supply of power to vital auxiliaries under any credible circumstances.

XII. WASTE HANDLING

Design of the Peach Bottom Plant is such that there should be relatively little release of radioactive waste at the site incident to normal operation.

In addition to the detailed provisions described below for handling radioactive wastes, the applicant has and will continue a program of environmental monitoring. This program should provide a basis for detecting any unanticipated changes in radiation levels in air, water, milk, wildlife, or marine life as a result of plant operation.

A. Liquid Waste. Potential sources of radioactive liquid waste are water stripped from helium purge streams, water used in laundry and decontamination operations, the spent-fuel storage pool, and from heat exchangers cooling the reactor and fission product traps. All of those sources are expected to contain little if any activity.

All potentially radioactive liquids are collected in either of two 1000 gallon liquid waste receiver tanks. Waste is periodically pumped from the waste receiver tanks to one of two 1000 gallon liquid waste monitoring tanks, where it is sampled by health physics personnel for activity. If within prescribed safe limits, the waste may be discharged to Conowingo Pond by pumping at a controlled rate into the condenser cooling water discharge. An interlock prevents discharge unless the condenser cooling water pump is operating to provide dilution water. If activity levels in the waste monitoring tanks are too high, the effluent will be recirculated through a demineralizer and filter system until satisfactory levels are reached. Instantaneous concentration of effluents at the point of discharge to the Pond will not exceed the concentrations set forth in Appendix B, Table II of 10 CFR 20. A radiation monitor is provided in the discharge line from the waste monitoring tank and excessive concentrations will cause an alarm. A recording flow meter in this same line assures that a record of the total amount of released effluent is maintained. Although effluents discharged to the Pond will be within drinking water tolerances, further dilution will occur before reaching potable water intakes downstream. Based on lowest observed river flow, we estimate that a minimum dilution factor of 40 will be available, and based on the lowest recorded yearly river flow, an average annual dilution factor of 1000 would be available. Consequently, we feel that the downstream radioactivity monitor originally proposed by the applicant is not needed.

We have concluded that provisions for handling liquid wastes are adequate for this facility.

B. Gaseous Wastes. Due to the use of the fission product trapping system previously described, gaseous waste discharge is not expected to be large. Potential sources of radioactive gaseous wastes are leakage from the primary cooling system, purge gases used during refueling and maintenance, argon-41 and carbon-12 formed in the containment atmosphere, and possibly controlled release from the krypton-85 hold-up tank in the fission product trapping system.

The ventilation system is arranged so that sufficient dilution air is added to the potentially radioactive exhausts from the spent-fuel

storage pool area, and the containment building to achieve a total flow of 20,000 cfm. All gases from potentially radioactive areas are exhausted through high efficiency filters prior to mixing and discharge up the 150 foot stack. Pressure differentials in the various containment areas are controlled so that gas flow is always into areas of highest potential activity. All ventilation intake and exhaust lines to the containment building contain two quick-closing, butterfly-type valves in series. As indicated previously, these valves are closed automatically by the containment isolation system on indication of above normal radiation levels in exhaust gases.

Using conservative assumptions, the applicant has calculated an average gaseous release of less than 10% of the maximum permissible concentrations (MPC) given by 10 CFR 20. We are in agreement with this calculation. Since the Peach Bottom Plant is in a dairy area, the possible influence of iodine release on the cow-milk cycle is an important consideration. We have determined, however, that I^{131} release limits that will be in effect at the site are well below the recommended release limit where the cow-milk cycle is a consideration.

We have concluded that adequate provisions have been taken to assure safe discharge of gaseous wastes.

C. Solid Wastes. Solid radioactive wastes include spent fuel elements and other core components, radioactive lubricating and hydraulic oils, graphite dust from gas stream cleanup, filters, and other materials normally associated with nuclear plant operation. Proposed procedures for handling solid wastes appear to be satisfactory. Solid wastes are collected in appropriate containers and stored in a basement area of the containment building until sufficient quantity is collected to justify a shipment for off-site disposal. There will be no disposal of solid waste on the Peach Bottom site.

XIII. ORGANIZATION AND PROCEDURES

A. Plant Organization.

The Philadelphia Electric Company will be solely responsible for the operation and maintenance of the Peach Bottom Plant and the plant organization will be similar to that used in the conventional stations owned and operated by the utility. The line operating organization includes the Station Superintendent, Assistant Station Superintendent, Plant Engineer, Shift Reactor Engineers, Shift Superintendents, and Plant Operators. Also reporting to the Plant Engineer is the Supervisor of Chemistry and Health Physics.

During the one Mwt operations, the Shift Reactor Engineers will be in charge of shift operations. During this period, the Shift Superintendents will be in training, and in adddition to normal administrative duties, will assist the Shift Reactor Engineers in plant operations. It is

intended that the Shift Superintendents will ultimately assume full responsibility for shift operations and the Shift Reactor Engineer's position will be eliminated.

The Health Physicist, although a member of the plant operating staff, will not be under the supervision of the Shift Reactor Engineer and Shift Superintendent. He will report through the Supervisor of Chemistry and Health Physics to the Plant Engineer on matters pertaining to plant or personnel safety. During shift operations, the Health Physicist will report directly to the licensed senior operator in charge of shift operations, and his recommendations concerning radiological safety matters will be binding. During the period of the one Mwt license, the health physics staff will be supplemented by a consulting Health Physicist who will assist in establishing the plant health physics program and who will add to the depth of experience available during the initial operating period.

Each of the key positions in the initial plant organization with the exception of the Shift Superintendents is filled by a college graduate, each of whom has received extensive supplementary training. Training has been varied to include extensive formal classroom and laboratory work, assignments at the offices of the nuclear design contractor, and assignments covering six months to a year as shift personnel on operating power or test reactors. Additional training is being received by all operating personnel by utilizing the plant simulator.

The Station Superintendent has served as an operator and supervisory trainee on the Shippingport Reactor and for over two years as a Shift Supervisor at the Enrico Fermi Reactor Plant, and appears particularly well qualified. The Assistant Station Superintendent, Plant Engineer, and Shift Reactor Engineers have been engaged in planned work-training programs for years in preparation for assuming their duties at Peach Bottom. The Shift Superintendents have qualified for this position through years of experience operating conventional power stations and by a demonstrated aptitude for special nuclear training. Each of the key plant personnel has been associated with the Peach Bottom plant for a number of years and should be quite familiar with its design.

We have reviewed the plant organization and personnel qualifications in considerable detail and believe that the proposed staff is technically qualified to operate the plant.

B. Technical Support and Safety Review Committees. During initial criticality and testing, it is intended to have full-time coverage of operations by qualified personnel from the nuclear design contractor (General Atomics Division of General Dynamics Corporation). Under the general supervision of a full-time Startup Coordinator, these consultants will be made available as necessary in order to take fullest advantage of their specialized knowledge. Although they are not part of the operating staff, Philadelphia Electric intends to shut down the plant or take other "safe" actions

on the recommendations of these specialists. General Atomics personnel will not be responsible for operations, but they will be available to provide technical backup to Philadelphia Electric for at least the life of the first core.

Review of reactor operations and safety will be conducted by two separate committees: (1) the Plant Operations Review Committee, and (2) the Operations and Safety Review Committee. The Plant Operations Review Committee is chaired by the Station Superintendent and is comprised of senior technical members of the station staff. This committee will meet at least biweekly to review operating procedures and operating plans.

The Operations and Safety Review Committee is chaired by the Manager-Nuclear Power Division of Philadelphia Electric Company. This committee is comprised primarily of qualified technical personnel from various departments of the Philadelphia Electric Company plus one qualified independent consultant. There are no members of the plant operating staff on this committee. This committee will meet at least monthly during initial operations and will perform an independent review of plant operations with particular emphasis on safety. It will also review proposed changes in tests, procedures, and other matters referred to it by the Plant Operations Review Committee. We have reviewed the technical support and safety review provisions proposed and believe them to be adequate for this plant.

C. Operating Procedures. Operating procedures have been prepared covering both normal and emergency conditions, and all operations will be performed in accordance with these procedures. During the initial startup operations, changes to procedures or tests must be approved by both the Philadelphia Electric and the General Atomic startup organizations. Following initial operations, procedural changes will be reviewed and approved by the Plant Operations Review Committee. We have reviewed a complete outline of plant operating and emergency procedures as well as sample detailed procedures. Based on this review, we believe that adequate procedures have been developed to assure safe operation of the plant. We also believe that procedure modifications and proposed new procedures will be adequately reviewed prior to use.

XIV. PROPOSED 1 MWT OPERATIONS

A preoperational test program will be conducted prior to initial operation of the plant. This program is intended to demonstrate that the plant has been built in accordance with drawings and specifications and is ready for initial fuel loading and startup. Systems will be functionally tested, with special emphasis placed on those systems important to the safe operation of the plant. Successful completion of these tests will be required before a license is issued by the AEC.

During most of the 1 Mwt license period, the applicant intends to operate the plant essentially at zero power in order to perform physics tests. Major tests will include initial core loading, flux distribution measurements, control rod worth measurements, reactivity worth measurements of core materials, and isothermal temperature coefficient measurements.

Initial loading of the reactor will be done remotely by replacing solid graphite (dummy) elements currently in the core with fuel-bearing elements using the fuel transfer machine. Special in-core nuclear instrumentation will be installed along with special startup neutron sources. Small groups of fuel-bearing elements will be added in a predetermined systematic pattern and inverse multiplication will be used after each group is loaded to predict the critical loading. Before each group of fuel elements is added, a number of control rods will be withdrawn, which, in addition to the emergency shutdown rods, will be available for scram. All instrumentation which normally initiates automatic scram in addition to manual scram will be active during initial loading of the core. In addition, the alarm and trip levels for certain neutron count and count-rate-increase measurements will be reduced so as to be appropriate for initial loading.

A flow verification test of the fission product trapping system will be conducted after the core is loaded since the dummy core elements used prior to initial fuel loading do not have the required internal structure. During initial operations a number of systems will be operated only to accumulate operating history and experience; others, such as the steam turbine equipment, will not be operated at all.

It is intended to conduct most initial operations with the containment closed, but with an air atmosphere in the entire containment building so that personnel access will be possible. We believe this is satisfactory because graphite temperatures will be limited to 750°F and neither graphite oxidation nor production of combustible gases by a steam-graphite reaction is a significant consideration at temperatures below this limit.

From our review of the proposed 1 Mwt program we are satisfied that initial operations will be conducted in a conservative and safe manner. We also believe that the reactor will be adequately protected by the safety systems that are provided. Since containment is provided and the fission product inventory in the fuel and in the trapping system will be very low, we believe the off-site consequences of any possible accident during 1 Mwt operations would be negligible.

XV. ACCIDENT ANALYSIS

Both the applicant and we have analyzed the consequences of a wide spectrum of accidents associated with full power operation of the Peach Bottom Plant. These accidents include those due to reactivity additions, primary coolant system depressurization, loss of forced circulation, leakage from the steam system into the primary coolant system, failures in the fission product trapping system, and fuel handling accidents. In each case, we have concluded that the accident would not result in undue hazard to the public because of the inherent safety characteristics of the reactor and its systems, the provisions that had been made in the design to accommodate the accident, or corrective operator action that could reasonably be expected to be performed. A discussion of the accidents of significance is presented below.

A. Reactivity Accidents. A series of reactivity accidents have been analyzed. These include reactor startup accidents, control rod fallout accidents, cold coolant accidents, and accidents initiated by rod withdrawal at full power. Even in the event of no automatic or manual scram action, the transients are not severe. Although high powers may occur very briefly, the total energy released would be small, since the excursion would be arrested by the large prompt temperature coefficient of reactivity. For the most severe accident postulated (rod ejection at full power), the final temperatures reached in the fuel compacts with no scram action whatsoever may be as much as 2150°C, or about 800°C higher than normal. We consider the possibility of such an excursion with no scram protection to be exceedingly remote. However, should such an unprotected excursion occur, we would expect some fuel damage, but no significant release of radioactivity from the primary system.

B. Fuel Handling Accidents. As discussed in the section on Fuel Handling, we can identify no significant accident associated with fuel handling. The reactor is shut down, depressurized and cooled before fuel handling is initiated. Only one fuel element is handled at a time, and it is sealed in a metal canister before it is taken outside the containment building. Calculations indicate that a spent fuel element could be stuck in any part of the fuel handling system indefinitely without over-heating or other serious consequences. We have concluded that the maximum quantity of radioactivity that could be released as a result of a fuel handling accident would be very limited and would represent no serious hazard to the public.

Fuel handling accidents during initial core loading were considered. Such an accident is considered improbable because of the conservative procedures that will be followed during initial loading. Initial loading is to be performed by replacing dummy elements in the core on a distributed pattern, with poisoned dummy elements located so that criticality is not predicted until approximately 704 fuel elements are loaded. For this condition the reactivity worth of one fuel element is so little that the excursion resulting from dropping an element into a critical core would be negligible.

C. Accidents Involving the External Fission Product Trapping System. We have examined the probability and the consequences of accidents that could occur involving the external fission product trapping system and believe that the most significant ones involve a rupture of the system. In order to minimize

the possibility of a rupture in the external fission product trapping system, the inlet end of the system is doubly contained, with the intermediate volume monitored for leakage. Numerous remotely controlled valves and check valves are provided throughout the system to allow components and sections of the system to be isolated in the event of a rupture. In the event of a rupture, the amount of radioactivity released would depend on the location and nature of the failure. The radioactivity contained in the helium and some of the radioactivity which had previously been adsorbed on the traps adjacent to the break would be released. Analyses indicate that a maximum release of 20% of the total activity contained in the entire system would be expected as a consequence of the worst rupture anywhere in the system. It is significant to note that the consequences determined for all accidents involving the external fission product trapping system were based on the design values for fission product activity inventories and decay heat loads which are more than a factor of seven greater than the values actually expected. In order to bracket all accidents involving the release of activity from the external fission product trapping system, the consequences of the release of the entire design inventory of the system to the containment building were evaluated. The doses at the site boundary during the first two hours following the accident would be less than 0.1 rem whole-body gamma and 25 rem to the thyroid. These calculated doses for this hypothetical accident are significantly less than the guideline values in 10 CFR 100 and the corresponding values for a more realistic release of activity would be even less.

D. Steam Generator Failure. We have investigated the possibility and the consequences of failures in the steam generators. Failure of a single tube in a steam generator is considered quite possible, and provisions have been made in the design so that the consequences would not be significant. The moisture monitoring system in the primary system would detect the failure and would initiate isolation of the loop and blowdown of the failed steam generator. Even if neither of these systems worked, any one of the primary system's two dump valves or three safety valves are capable of relieving the maximum flow resulting from a single tube failure without damage to the primary system.

If a steam generator should fail so as to release its entire inventory of steam and water to the containment building, a containment isolation would be initiated by the primary loop cavity pressure detectors. The containment building would not be pressurized above its design pressure. The reactor would be scrammed and would be cooled by the other primary loop.

As an upper limit consideration, we investigated the possible consequences of an hypothetical massive failure in a steam generator, assuming that a large quantity of water was suddenly released into the primary coolant system. The primary system dump and safety valves would immediately operate to relieve the over-pressure and return the system pressure to the 390 psig setting of the dump valves. The capacity of these valves is large enough to prevent primary system pressure from significantly exceeding the 450 psig set point of the safety valves for more than several seconds for

any realistic steam flow rate from a failed steam generator.

Conditions do not appear to be such that a significant amount of steam would react with core graphite to form non-condensible gases during the initial pressure surge. However the steam remaining in the primary system would react with core graphite after the initial transient and the gases generated would be adequately relieved by the dump valves. It is anticipated that primary system pressure would remain at about 390 psig. If, however, one of the safety valves failed to seat properly, the pressure in the primary system would slowly equilibrate with that in the containment building. Under these circumstances the pressure in the containment building would not exceed the design value of 8 psig. The core would be adequately cooled by the undamaged steam generator and the circulator in the other primary loop. The release of radioactivity to the containment would be small, consisting mainly of the gaseous activity inventory normally permitted in the primary coolant system. The doses at the site boundary from such an accident would therefore be negligible.

E. Sudden Depressurization of Primary Coolant System. In analyzing the possible consequences of a sudden depressurization of the primary coolant system the following items are significant:

1. Initial Release of Fission Products. If a large leak or rupture occurs anywhere in the primary coolant system, essentially all of the helium would escape to the containment atmosphere. Adequate core cooling capability would be expected to be preserved and the maximum initial release of radioactivity would be the 4225 curie inventory of gaseous activity permitted in the primary coolant system plus a potential additional release of 0.54 megacuries of activity from the fuel elements and internal traps. The estimated 2-hour doses at the site boundary from this release would be very small.

2. Backflow from External Fission Product Trapping System. In the event of a primary coolant system depressurization accident, backflow from the external fission product trapping system is prevented by stop-check valves. These stop-check valves are self-acting like a conventional check valve but, in addition, they can also be automatically closed by a powered operator. We believe that failure of these valves would be highly improbable. However, even if backflow should occur, the additional activity released and the doses that could occur would be significantly less than the guideline values given in Part 100 of the Commission's Regulations, as discussed in the section entitled Accidents Involving the External Fission Product Trapping System.

3. Core Levitation. The transient pressure and flow conditions that could occur in the core following a major rupture in the primary coolant system were evaluated to determine if the core could be lifted away from the bottom entry control rods. The only conceivable event which could lead to core levitation would be complete shear and removal of one of the large fuel handling nozzles in the top of

the reactor vessel. This is not considered credible because of a close fitting missile shield over these nozzles which would prevent a severed nozzle from lifting more than 2 inches. Nevertheless, the possible effects of this occurrence were analyzed and it was found that the core might be lifted a maximum of 0.8 inches. The associated increase in reactivity would be less than half of that for a rod ejection accident which would not have serious consequences as discussed under Reactivity Accidents above. Thus core levitation does not appear to be a significant consideration in analyzing depressurization accidents.

4. Effect on Steam Generators and Main Helium Circulators. Possible effects of a sudden depressurization accident on each of the major loop components were investigated. The applicant performed detailed analyses of the stresses occurring in the steam generator during a depressurization accident. These indicate that the total stresses and strains that would occur during a depressurization are within allowable values specified in the ASME code. On the basis of these results, we believe that a sudden depressurization would be unlikely to cause failures in a steam generator.

A large rupture in the cold leg of one of the primary loops could upset or damage the rotating shaft seal system of the main helium circulator in the failed loop. This could cause or require shutdown of this circulator. Both circulators could be subjected to stall or surge conditions but, as discussed previously, this should have no adverse effects on the integrity of these machines. Analyses indicate that the effects on the circulator in the unfailed loop would be very mild, consisting of a temporary small speed change and a possible momentary spurt of seal oil into the circulator casing due to the inability of the buffer gas control valve to follow the rapid decay of pressure. There should be no structural or operational damage to this unit. Thus it appears to us that it is a remote possibility that both gas circulators would fail as a result of any depressurization accident.

In the event of a large primary system rupture, the primary coolant pressure would equilibrate with the containment atmosphere at about 4 psig. At this pressure, one main helium circulator is capable of supplying more than 3% of full power flow through the core, so that the maximum temperatures that would occur with one circulator available after the rupture would not exceed the design limits on any equipment. There should, therefore, be no significant consequences of a primary coolant system depressurization accident beyond the initial release of radioactivity to the containment building.

F. Maximum Accident. The most severe accident postulated by the applicant is a rupture and sudden depressurization of the primary system in which it is assumed that both main circulators become inoperative, a steam generator tube fails, and the loop isolation valves fail to close. Under these conditions it is calculated that the containment building pressure would rise to 8 psig due to the combination of hot helium released from the primary system and the non-condensible gases formed by reaction of the released steam with core graphite. In our opinion this sequence defines an upper limit accident suitable for use in evaluating the maximum potential hazards to the public. As indicated in our previous discussion, we consider the occurrence of the failure of steam generator tubes simultaneously with a sudden depressurization to be exceedingly remote. In addition, the presence of automatically actuated loop isolation valves should limit the amount of water available to react with the core. The loss of both main circulators as a result of sudden depressurization is also considered to be exceedingly remote. However, if this combination of failures is assumed to occur, a rise in core temperature would result which increases the release of fission products from the core to the containment building.

The possible consequences of this accident were analyzed on the following basis:

1. Emergency cooling system operation is initiated within 4 to 6 hours after the rupture.

2. Fission products which are volatile at the core temperatures attained during the accident are released at a rate corresponding to experimental data obtained using uncoated particle fuel compacts. This assumption results in the conservative estimate that about 75% of all biologically significant fission products are released from the fuel compacts before the end of the accident. Low vapor pressure fission products are assumed to partially recondense on the cooler surfaces of the core top reflector.

3. The initial fission product inventory in the fuel is 100%, i.e. there has been no removal of fission products to the external trapping system.

4. The post-incident recirculation and filter system removes strontium and other particulates from the containment atmosphere with 50% efficiency at a flow rate obtained with only one of the two available fans.

5. For the 2-hour dose calculations, no credit is taken for condensation of fission products on the cooler surfaces within the containment. For the 30-day dose at the outer boundary of the low population zone calculation, 50% of the iodine released to the containment is assumed to have condensed on cooler surfaces.

6. The atmospheric dilution factors to the site boundary were determined by the application of conservative empirical meteorological parameters to Sutton's diffusion equation.

7. Direct radiation from the containment building is calculated using a 50% self-shielding factor and whole-body gamma doses are calculated on the basis of immersion in a semi-infinite cloud.

Since the core is assumed to be cooled only by the emergency cooling system, the fuel would gradually heat up and additional fission products would be released after the primary system depressurization. Combining this additional release with the maximum initial activity release, the estimated doses at the site boundary for the first two hours under the worst possible meteorological conditions are still essentially negligible.

The gradual heatup and release of fission products from the core would continue for many hours after the start of the accident. The applicant has performed extensive digital and analog computer calculations to evaluate the core and reactor vessel temperatures that will occur during this period. The gradual release of fission products to the containment atmosphere is based on these calculated temperature transients. The results of these calculations indicate that the maximum vessel temperature will not occur until about 64 hours after loss of normal cooling and will be about 1100°F, occurring at the center of the bottom head. This temperature is not of concern since the vessel would be depressurized and stresses low. The highest average core temperature occurs about 40 hours after the start of the accident and is calculated to be about 2900°F. The peak core temperature occurs in a small region after about 30 hours and is calculated to be about 4200°F. It is significant to note that the temperature at which the uranium-thorium carbide fuel particles would melt is about 4450°F. The properties of the graphite matrix would not be significantly affected at these temperatures.

We have reviewed these temperature calculations by confirming that the general approach, the geometric model used, the assumptions made, and the values of the physical constants used all appear to be reasonable. In addition, we performed a one dimensional, steady-state calculation to determine the peak temperatures that will occur in the reactor vessel and at the core boundary with emergency cooling system cooling only. The results obtained indicate that the peak temperatures reported by the applicant given above appear to be reasonable. We have also noted that the over-all results obtained would not be altered significantly even if the actual temperatures were considerably higher than those that were calculated. We therefore believe that the results of these calculations can be used as a basis for analyzing the possible consequences of this accident.

The applicant has calculated doses based on the conditions outlined above. For diffusion conditions representative of a prolonged inversion at the site, it is calculated that the maximum guideline dose value of 300 rem to the thyroid, as defined in the Commission's Regulation 10 CFR 100, could be received at a distance of 3 miles upstream or downstream along Conowingo Pond for a 30-day exposure. The calculated dose at a distance of 3 miles directly inland from the plant is only about 15 rem because of the difference in meteorology effects. We have verified that these calculations are essentially correct. In addition we have made comparative calculations based on the release assumptions set forth in TID-14844, using the design leak rate for the containment building and the site diffusion parameters. Results of these calculations are summarized in the following table.

Potential Dose Calculation Summary[1]

	Applicant's Release Model	TID Release Model
Two-Hour Dose at Site Boundary (0.6 mi)		
Whole body, rem	0.0007	-
Thyroid, rem	0.02	34
Bone, rem	0.0008	4
30-Day Dose at 3 Miles		
Thyroid, rem	300	250
Bone, rem	2	140 (Assumes no internal cleanup)

1/ Based on constant 0.2% per day leak rate and site meteorology.

From these results it is evident that use of the applicant's time-temperature dependent release model has a significant influence in reducing the calculated short-term (2-hour) doses. For the long-term (30-day) thyroid dose, the difference between the applicant's model and the TID model diminishes.

The applicant's release model predicts that a large fraction of the bone seeking fission products would be retained by condensation on the top reflector, thereby limiting the calculated bone doses to relatively low values. We have investigated the validity of this assumption in considerable detail and have concluded that theoretical and qualitative considerations indicate that it should be generally valid. However, since there is not sufficient experimental evidence to confirm that the reflector would retain fission products as

efficiently as predicted, we investigated the consequences of the release to the containment atmosphere of a considerably larger fraction of bone-seeking fission products than was predicted by the applicant. The results show that even if no credit is given to retention of fission products by the reflector, the bone dose would not be excessive if the post-accident recirculation and filter system is assumed to operate with an efficiency of about 50% of its design specification.

As a result of our review of the system design, we have concluded that the post-accident recirculation and filter system will perform reliably under the conditions of the assumed accident. It is significant to note that additional reduction in the fission product burden in the containment atmosphere may also be provided by other systems. The external fission product trapping system could be placed into operation to remove fission products from the containment atmosphere. In addition, the helium transfer pumps could be utilized to pump part of the post-accident atmosphere in the containment into the helium dump tanks. This would reduce the pressure in the containment building by about 3 to 4 psi, and thereby significantly reduce the rate of fission product leakage from the building. However, considering the post-accident recirculation and filter system to be the only effective engineered safeguard available at the time of the postulated accident, we have concluded that bone dose will not be a limiting consideration in determining the required low population zone; therefore the potential thyroid dose will be governing.

As indicated previously, the potential dose to the thyroid due to radioactive iodine from the most severe accident places the required low population zone boundary at 3 miles. The applicant has determined that about 1100 persons reside within this distance, and has outlined the provisions that have been made and the evacuation procedures that would be implemented to remove all persons from this area in the event of an accident. We believe that these provisions are adequate.

Since the river downstream of the site is utilized as a source of potable water, we have considered the contamination that might conceivably result from the maximum accident. As a limiting case, we have assumed that gaseous effluents from the containment building travel over Conowingo Pond under conditions of continuous heavy precipitation. Under these conditions, concentrations of I-131 in surface water near the immediate vicinity of the plume could conceivably exceed the allowable concentrations set forth in 10 CFR 20 for unrestricted areas. Other contaminants may also be present, but in concentrations so low that their presence can be neglected. Our calculations indicate that under all anticipated conditions, sufficient dilution would take place in the pond to reduce concentrations well below allowable values prior to transport as far as Conowingo Dam. Considering stratification of pond water more severe than anticipated and a period of heavy rain greater than 8 hours, it is possible to calculate as an upper limit that water in the vicinity of the dam would reach concentrations slightly higher than those specified in 10 CFR 20 for a short period of time. In no case

that we can reasonably anticipate for this assumed upper limit case would the average annual concentrations approach permissible values. We have concluded, therefore, that under conditions of the assumed accident, there is negligible likelihood that water downstream from the plant would be contaminated to the extent that it would be unusable for domestic water supply.

XVI. CONCLUSION

On the basis of the foregoing considerations and the recommendations of the Advisory Committee on Reactor Safeguards, we have concluded that the applicant is technically qualified to engage in the activities to be authorized by the proposed provisional operating license, and there is reasonable assurance that the Peach Bottom facility can be operated as proposed at power levels up to 1 Mwt without endangering the health and safety of the public.

Original Signed by

S. Levine

Saul Levine, Chief
Test & Power Reactor Safety Branch
Division of Reactor Licensing

Date: JAN 29 1965