Project 245

DEC 6 1960

Dr. Leslie Silverman
Chairman, Advisory Committee
on Reactor Safeguards
U. S. Atomic Energy Commission
Washington 25, D. C.

Dear Dr. Silverman:

Transmitted herewith for the use of the Committee are eighteen copies of a report dated December 5, 1960, prepared by the Division of Licensing and Regulation concerning the Peach Bottom Atomic Power Station, Philadelphia Electric Company.

Sincerely yours,

Harold L. Price, Director
Division of Licensing and
Regulation

Enclosure:
Hazards Analysis Phil. Elect. (18)

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December 5, 1960

ATOMIC ENERGY COMMISSION

DIVISION OF LICENSING AND REGULATION

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ON

PEACH BOTTOM ATOMIC POWER STATION
PHILADELPHIA ELECTRIC COMPANY

ACRS REPORT
12/5/60

Note by Director, Division of Licensing and Regulation

The attached report has been prepared by the Staff of the Division of Licensing and Regulation for consideration by the Advisory Committee on Reactor Safeguards at its December 1960 meeting.

Peach Bottom Atomic Power Station

Philadelphia Electric Company

The Philadelphia Electric Company has submitted an application requesting a construction permit for an advanced type gas cooled power reactor to be built on the shore of the Susquehanna River in York County, Pennsylvania. Information submitted in support of the application included a description of the site, environmental data, conceptual design information, and the outline of a research and development program currently being performed.

The proposed reactor system is of the homogeneous solid fuel type, graphite moderated and reflected, and helium cooled. The fuel (U-235) and fertile material (Th-232) are in the form of carbides homogeneously dispersed in a portion of the moderator and enveloped by a suitable cladding material. Steam produced from the heat content in the helium coolent is processed in a cycle typical of conventional power plants.

The Staff of the Division of Licensing and Regulation has reviewed the information submitted and the following observations are noted:

1. From the early submittals the applicant contemplated that the first fuel core would be a metal-clad hermetic sealed type, containing graphite-fuel compacts. Below the fuel compacts in the reflector region within the clad housing, an internal chemical trap for fission products was provided. Convective paths would allow the products to flow from the fuel to the cooler reflector and into contact with the trap material.

The applicant now believes that the results of recent development efforts on low permeability graphite warrant its selection as a cladding material for the first core, rather than for subsequent cores, as originally contemplated.

In this case, to prevent contamination of the primary coolant by fission products, a purge stream of helium would be passed through the fuel element and its internal trap to a series of fission product traps external to the reactor vessel.

2. The applicant is fully aware of the components and systems which require development and has initiated a program addressed to these problems. The following are comments upon important items in the reactor system believed to require substantial attention before an adequate design can evolve:

a. Fuel Elements: The applicant proposes to utilize a fuel element consisting of an impervious graphite sleeve surrounding an impervious internal fission product trap in the lower end of the element. A porous graphite reflector piece is connected to the top of the sleeve.

A great deal is known about the behavior of graphite in various reactor applications. In the present instance it is proposed that graphite of low permeability, and hence of different internal structural composition from usual forms of graphite, will be used. It is also proposed that this graphite will be used in complex structural arrangements within very long fuel elements and that temperature, temperature gradients - both radial and longitudinal, and cumulative radiation exposures will be substantially different from similar factors which have characterized earlier uses of graphite. Information is lacking as to the composite effects of these different parameters and this leads to uncertainty as to the effects of radiation in the long fuel cycle proposed, the structural integrity and dimensional stability of the fuel assemblies, the effective thermal conductivity of complex structure, and the mobility of fuel and fission products within the material.

The applicant's research and development program on fuel elements was begun in January 1959. It is stated in the hazards report of September 1960, "There exists enough information at present to assure that a fuel element of the type described can be satisfactorily developed for use at the conditions specified; the exact design characteristics will be forthcoming from the research and development program." The applicant's program on the fuel element is scheduled to continue through the in-pile loop testing phase and be completed by mid year 1962. The scope of this program includes: establishment of permeability requirements, methods of producing low permeability graphite of the requisite dimensions (tubes 12' long, 3½" diameter), compatibility of graphite with possible contaminants in the coolant, thermal conductivities, thermal shock effects, structural strength requirements, properties of the uranium and thorium carbides, methods for inclusion of burnable poison in the core, irradiation effects on materials, mobility of fission products, development of internal traps and fittings and seals, flow model studies, and fuel assembly techniques.

According to information now available, the following experiments and tests are to be performed on the fuel assembly and its materials:

(1) Sealed capsules containing samples of fuel compacts will be irradiated to establish burnup, dimensional characteristics, fission product effluent, and effects on physical properties.

(2) Purge capsules containing fuel compacts will be irradiated to provide the information on the release of fission product from the fuel as a function of burnup at design temperatures.

(3) An in-pile loop irradiation program to test a full diameter, approximately ¾ length fuel assembly under reactor conditions will be carried out to provide information on physical stability, fission product release and evaluation of fission product deposition in the loop. This program will provide radiation exposure equivalent to only a small fraction of that which would be experienced by the element in a full fuel cycle in the reactor.

(4) Mechanical tests on full scale elements will be conducted to establish vibration characteristics, the effects of vibrations, and suitability of the fuel element design for core loading operations.

Our evaluation of the fuel element problem is the following:

(1) It is our opinion that the subjects and scope of investigation proposed to be carried out in the R & D program on fuel element assembly and materials are sufficient to furnish clarification of the uncertainties now existing.

(2) It must be recognized that the test program will not furnish answers to all the problems which may arise in the behavior of the fuel element in the composite environment and arrangements which will exist in the core, and particularly under the cumulative effects of radiation in the long fuel cycles proposed. We cannot, of course, predict now that the results of fuel development will turn out to be acceptable, but we see no reason from what is now known of graphite properties, for believing that satisfactory fuel elements cannot be developed. However, if the results obtained from the experimental

and test program outlined indicate favorable characteristics for the fuel element materials and assemblies, we believe this would furnish an acceptable basis for their use in limited and controlled operation of the reactor.

b. Fission Product Trapping: In the preliminary hazards report the applicant assumes that about 50 per cent of the fission products are passed from the core region into the thermal collection traps external to the reactor vessel. According to more recent oral discussion, preliminary test information indicates a possibility that a substantially larger portion of the gross products may be retained within the core fuel assemblies. The fission yields of elements and basic methods for trapping these nuclides are well understood. However, there are several major problems to be resolved regarding fission products and their removal and trapping. These problems are:

(1) The proportional distribution of fission products between (a) the fuel; (b) the internal trap, and (c) the external traps.

(2) The methods and materials to be used for collection of the fission products in the various traps.

(3) The engineering arrangements to accommodate safe handling and storage of fission products in the collection traps.

(4) Development of facilities and procedures for handling, storage, shipment and disposal of trapped fission products.

The research and development program on fission product trapping began in 1959 and is scheduled to continue through late 1961. The program is addressed to the four principal problems listed above. In addition to experiments on mobility of fission products mentioned earlier in the

discussion of fuel elements, the program will include experiments on the adsorption characteristics of various possible trapping materials, investigation of various arrangements for thermal trapping of fission products, the effectiveness of various materials and arrangements for removing fission products from flowing gas streams, experiments on trapping systems in the external circuit of the in-pile fuel element loop mentioned earlier, and mechanical equipment and physical arrangement for effective handling, storage and shipment of large quantities of fission products.

Considerable technology exists concerning the proposed fission product trapping system. This technology, supported by the planned R & D.program, should provide a sound basis for the design of this system, and the solution of the practical problems mentioned. We would expect that the R & D program would furnish a suitable basis for design of a system that could be used in initial operation of the reactor. The feasibility and safety of this system for extended operation could be evaluated as operations proceeded.

c. Control System: The proposed control rods will be composed of jointed lengths of boron carbide dispersed in graphite and operating within a graphite guide sleeve. The uncertainties with respect to structural stability under irradiation which were previously mentioned on Fuel Elements are applicable here. The concern is more acute here, however, because the capability of reactor shutdown is involved.

The proposed rod drive system is a bottom entry type, mechanically driven by hydraulic motors for both control and scram operations. One of the major concerns in this design is the effect of rod failure. The bottom

drive removes poison by lowering the rod from the core. In the event of a failure which results in separation of the push rod or poison section, it is possible that the poison sections could stick in the core when the drive is lowered. If the stuck rod suddenly dropped out of the core, a hazardous transient could occur.

The applicant has presented details of an R & D program in support of the proposed control rod system. No R & D effort is directed toward development of a back-up shutdown system. The schedule indicates that work began in late 1959 and will continue until early 1962. The scope of this program includes: dynamic cold air tests on the rod, dynamic hot helium tests on the rod, irradiation of sample specimens of control rod materials, wear tests on material combinations, dynamic behavior tests of components, and finally full scale, simulated environment tests of a single control rod system but excluding any irradiation effects.

We believe the scope of development and testing proposed in the R & D program is sufficient to furnish clarification of the uncertainties relating to materials and arrangements of components in the proposed control rod system, and should indicate whether or not the characteristics of materials and dependability of mechanisms would be sufficient for use in limited and controlled operations of the reactor.

It is our opinion, however, that the possibility will exist even if the R & D program has been successfully completed that full dependability cannot be placed in the integrity of the graphite materials and that failures in the rod assemblies could permit rod fall out from the core and consequent hazardous reactor transients. Therefore, on the basis of information

now before us, we believe that plans should be developed for inclusion of an independent reactor shutdown system. Such a system is not now proposed by the applicant for investigation in the R & D program or for inclusion in the design of the reactor.

d. Core Design: The core arrangement proposed has an egg-crate type lower grid for support of the fuel assemblies and rod guide tubes, while fixed and movable graphite reflector rings are carried on a separate support ring attached to the vessel. Fuel assemblies, guide tubes and the inner reflector are radially fixed at the lower end while the upper extremities are contained in non-rigid support by lateral forces on adjacent parts created by differential coolant pressure across the movable inner reflector ring. There is not sufficient design information presently available for an adequate evaluation of the proposed core arrangement with respect to such questions as: flow disturbance, lateral stability, interferences with control rod movement, and oscillatory movement.

As a part of the R & D program, a one-half scale model of the reactor vessel and core will be constructed complete with blowers and ducting. These tests include: measurements of flow distribution, pressure drop measurements, determination of lateral forces, vibration characteristics, and general operational information. These tests will be conducted at ambient conditions.

A variety of other analyses and tests will be made to establish the suitability and dependability of arrangements of components within the core, e.g., tests on equipment and procedures for removal and replacement of fuel elements, sealing devices on the ends of fuel elements, etc.

These tests, in addition to those previously described for fuel elements and control rods, should provide a suitable basis for determining the mechanical stability and structural adequacy of the core arrangement. If satisfactory results are obtained in these tests on components and arrangements, a basis should be provided for a suitable design.

e. Facility Design: It is proposed that there be two external loops in the primary cooling circuit of the reactor system. Each loop will be equipped with a helium circulator, heat exchanger in which steam will be generated in the secondary side, stop valves in each leg of the primary lines to the steam generator, and the usual auxiliary lines and connections. Supplies of power to the circulators will be available from three independent sources, which assures high dependability of service..

Three features of the reactor system are of sufficient importance to the safety that further study should be given to the adequacy of safeguards provided and the possibility that additional protection may be needed:

(1) Emergency cooling: In one sense, each of the two loops may be considered to be an emergency cooling system, since either one alone could remove heat from the core during forced shutdown or other emergency conditions. However, if both the cooling loops should be inoperative, the arrangement of the core is such that convective cooling alone would not be sufficient to prevent core overheating. Hence, further study should be given by the applicant to the possibility that an additional loop for emergency cooling might be required.

(2) Water in-leakage to the core: At operating temperatures any water admitted to the core would react rapidly with the graphite causing damage to the structures and releasing combustible products which could cause further damage. If a leak should occur in the He-water heat exchangers, water would leak into the primary system, - its pressure is only 350 psi while the steam pressure is 1450 psi. Presumably, in case of major water in-leakage, the stop valves could be operated to prevent most of it from reaching the reactor. However, details for this have not been worked out, and we believe that further study should be given by the applicant to the question of whether this alone is sufficient protection against water in-leakage accidents.

(3) Air in-leakage to the Primary System: At operating temperature, any air admitted into the core will react immediately with the graphite, causing its oxidation. Major in-leakage of air would cause extensive burning of the core, with hazardous consequences. If a major break should occur in the primary system, the outflow of helium under high pressure would prevent any immediate in-leakage of air. In time, however, when the pressure has been relieved, air could gain entrance, and an accelerating sequence of damages could occur. We believe that further study should be given by the applicant to the possibility that additional safeguards against such potential accidents should be provided.

It is our opinion that appropriate studies can reveal whether or not additional safeguards in the three areas mentioned above are necessary, and if they are, that ways can be found within the conceptual plan of the project to incorporate them into the design.

f. Containment: The applicant proposed in the hazards summary report that the reactor and the entire primary cooling system, including the steam generators, will be contained in a steel shell designed for 5 psig internal pressure and a leak rate of 0.2% per day at this pressure. The applicant estimates that 5 psig could result if the He from the primary system and the steam from 1 generator were released simultaneously into the building. In subsequent discussions it has been agreed that a firm opinion cannot now be reached on the magnitude of credible accidents that may occur, and hence, it must be determined later, when firmer opinions on credible accident potentialities can be reached, whether or not the containment specifications proposed will be adequate.

If accidents more serious than those now visualized by the applicant should be considered credible, the leakage rate specification on the vessel may have to be reduced. This can be done if necessary. For example, by present technology and established construction techniques, buildings with 0.1% per day leak rate at much higher pressures have been constructed.

3. The maximum credible accident postulated in the hazards report is a leak in the fission product purge line connecting the fuel assembly to the external traps, with a release of the fission product inventory then in the purge line into the containment building. This accident would provide no appreciable pressure build up in the containment vessel. Such an accident would not result in any serious hazard to the public, because, since no pressure build up is assumed, very little leakage to the atmosphere would occur.

It, of course, cannot be determined at this time, what fraction of the current core inventory and what additional stored inventory of fission products could be in the external traps at the time of an accident and what accidents might possibly release this material into the containment building. Furthermore, we cannot agree at this point that serious accidents involving major releases from the core itself could not occur. These would involve such inventory as might be in the core at the time and would involve the possibilities of pressure in the building, e.g. from helium escape, which would act as a driving force in the escape of radioactivity to the atmosphere.

Hence, we are not convinced at present that the accident described represents the upper limit of possible hazard. Further study of this will be necessary.

## Summary

In summary, there appear to be two general categories of issues involving questions of safety evaluation which cannot be answered fully at this time:

(1) Those matters, such as materials and design of fuel elements, control rod assemblies, core design, and fission product trapping systems, which depend for final choices on R & D programs. Safety evaluation on these issues cannot be made until R & D programs have been completed, but it is our opinion that R & D programs of appropriate scope are now being devoted to them.

(2) Those matters, such as an emergency shutdown system, addition of emergency cooling system, and additional protection against possible air and water in-leakage, on which further study by the applicant is necessary to determine whether or not added features of design are needed. If studies reveal that added features are necessary there will then be two additional questions to be resolved: (a) whether or not the applicant would be willing to add these or could add them without major alterations in the general plan of the facility; and (b) whether or not satisfactory designs for the added features could be agreed on.