DOCKET NO. 50-171

Journal File 4

PHILADELPHIA ELECTRIC COMPANY

1000 CHESTNUT STREET

PHILADELPHIA 5, PA.

VINCENT P. McDEVITT
VICE-PRESIDENT
AND GENERAL COUNSEL

August 7, 1961

Mr. Robert Lowenstein, Acting Director
Division of Licensing and Regulation
United States Atomic Energy Commission
Washington 25, D.C.

Re: Application of Philadelphia Electric Company For Construction Permit and Class 104 License
Docket No. 50-171

Dear Mr. Lowenstein:

There is transmitted herewith for filing with the Commission Amendment No. 2 to the Application of Philadelphia Electric Company in the above proceeding. See Report file

This Amendment includes, in Volume I of the Preliminary Hazards Summary Report, information on the matters raised in your letter of April 10, 1961 to L. R. Gaty of this Company and in the letter of the Advisory Committee on Reactor Safeguards to the Commission dated December 10, 1960. For your convenience there is attached to my letter a summary setting forth the matters raised, followed by a brief description of the information included in the Preliminary Hazards Summary Report in response, together with a reference to those parts of the Report where the information may be found.

Yours very truly,

*(3 cys. rec'd)

Vincent P. McDevitt

VINCENT P. McDEVITT

VPMcD:mbl
Enclosure

RECEIVED AUG 7 1961 U.S. ATOMIC ENERGY COMMISSION Director of Licensing & Regulation

DOCKETED USAEC Division of Licensing & Regulation AUG 7 1961 Docket Officer

A-172

Attachment to letter dated 8/7/61
to Robert Lowenstein.

DOCKETED USAEC Division of Licensing & Regulation AUG 7 1961 Docket Officer

50-171
Formal File

SUMMARY

There is set forth below a summary of the information included (by Amendment No. 2 to the Application) in Volume I of the Preliminary Hazards Summary Report regarding the matters raised in the letter dated April 10, 1961 from the Acting Director, Division of Licensing and Regulation, Atomic Energy Commission to L. R. Gaty, Philadelphia Electric Company, and in the letter dated December 10, 1960 from the Advisory Committee on Reactor Safeguards to the Commission. The quotations below are from the above referred to letters. References to the Report are to the pertinent sections of Volume I of the PHSR where the information may be found.

I. Letter from Acting Director, Division of Licensing and Regulation to Philadelphia Electric Co., April 10, 1961

"1. Fuel Elements

"The proposed fuel elements will require graphite of low permeability and hence of different mechanical properties from usual forms of graphite. This graphite will be used in very long fuel elements of complex structural design. The operating temperature, temperature gradients, and cumulative radiation exposures will be substantially different from similar parameters which have characterized earlier applications of graphite. Information is lacking as to the composite effects of these various factors and this leads to uncertainty regarding 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 the complex structure, and the mobility of fuel and fission products within the material."

RECEIVED AUG 7 1961 U.S. ATOMIC ENERGY COMMISSION Division of Licensing & Regulation

1. Fuel Elements

a) Effects of irradiation on the graphite fuel element. Extensive irradiation experiments have been carried out on the individual critical components of the fuel elements. These have included the fuel compacts and the low permeability graphite tubes. This information is given in Section II.J. In general, the results of these experiments show that the fuel elements will perform satisfactorily during the proposed fuel life and that there is no cause for concern as to over-all integrity. The particular properties of the fuel element which have been investigated are as follows:

The core of the reactor has been carefully designed so that changes in the over-all length or diameter of any of the fuel elements in the reactor, by as much as 1%, can be accommodated. (See Section II.A.1) This has been accomplished by providing a flexible positive lateral support for the core at the top by means of coolant pressure acting on the side reflectors rather than by use of a fixed top grid plate. (See Section II.B.9)

Radiation experiments carried out by General Atomic and by the Hanford laboratories show that graphite contracts under fast neutron irradiation at HTGR operating temperatures. The degree of contraction varies between types of graphite. The detailed data are summarized in Section II.J.1. These data show that the over-all dimensional changes will be within limits which are acceptable for the core configuration. In addition, the stresses resulting from the radiation-induced contraction under the fast neutron flux gradient across a single fuel element have been analyzed. (See Section II.A.1) The Report concludes that the graphite fuel element sleeves will not fail as a result of the most severe flux gradient in the reactor. Also, stresses resulting from the differential contraction of the fuel element sleeves and fuel compacts have been analyzed. The Report concludes that by proper matching of the sleeve and fuel compact materials undesirable interference of compact and sleeve can be avoided.

A number of out-of-pile mechanical tests have been performed on the fuel element. These show that the fuel element has a safety factor of approximately 20 with respect to external loads which can be applied during core lifetime. (See Sections II.A.1 and II.J.1)

b) The effective thermal conductivity of the fuel element. The effect of irradiation on the thermal conductivity of low permeability graphite has been measured. (See Section II.J.1) The effective thermal conductivity of the gap between the compact and the sleeve has been calculated and the method of calculation has been verified by thermal mockup of irradiation capsules. In addition, the simplification of fuel element design has lowered the centerline temperature by approximately 300°C from the previous design value. The thermal performance of the fuel element is discussed in detail in Section II.A.3. The analysis in the Report shows that the thermal performance of the fuel element is satisfactory.

c) Mobility of the fuel within the fuel element. Fuel compacts of the type to be used in the fuel elements have been irradiated to the equivalent of two-thirds of the Peach Bottom fuel life with a periodic sampling and analysis of volatile fission product release during irradiation. The Report shows that there has been no significant change in release from these fuel

compacts observed during the irradiation. This result demonstrates that mobility of fuel is not a problem in the fuel element since harmful migration of uranium within the fuel element would be immediately evident in the volatile fission product release. (See Section II.J.2)

d) Mobility of fission products within the fuel element. The Report indicates that there will be a slow release of volatile fission products from the fuel elements to the fuel element purge stream. This release has been precisely measured in a large number of out-of-pile and in-pile experiments, and the data so obtained are being used in the engineering design of the fission product trapping system to provide that fission product activity in the coolant is maintained at acceptable levels. (See Section II.J.2)

e) General safeguards considerations of the fuel element. A fundamental feature of the HTGR fuel element is the continuous removal of that fraction of the volatile fission products which are released from the fuel compacts out of the reactor core to either the cold end of the fuel element or to an external trapping system. (See in General Section II.C) This means that the only effect of a failure of the outer graphite structural member of a fuel element is to produce a slightly increased level of fission product activity in the circulating coolant of the primary loop. (See Appendix B.1) The activity of the primary loop is continuously monitored. This subject is considered in detail in Section II.G.3 of the Report. A cracked fuel element and the consequent increase in primary system activity, therefore, does not cause the immediate release of a large quantity of stored fission product activity to the primary system nor, of course, to the secondary containment. (See Section II.A.4)

f) Fuel element design. Due to the successful development of improved fuel compacts which release less of the short-lived volatile fission products, it has been possible to simplify the design of the fuel elements. (See Figure 1) This simplified design reduces the centerline temperature approximately 300°C from that of the previous design with an attendant gain in over-all safety. (See Section II.A.3)

g) Loop Tests. As reported in Section II.J.2 a full diameter fuel element of approximately one-third full length will be irradiated in a pilot plant type of loop test as an overall demonstration of the fuel element performance. The fuel element used in this demonstration will contain all of the significant features of the Peach Bottom fuel element including the graphite matrix fuel compacts, low permeability graphite sleeve, and an internal fission product trap. In addition, the external components of the loop will contain an external fission product trap.

"2. Fission Product Trapping.

"Several problems must be resolved concerning the proposed fission product trapping system:

"(a) The proportional distribution of fission products between the fuel, internal traps, and external traps.

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

"(c) The engineering arrangements to accommodate safe handling and storage of fission products in the various traps.

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

2. Fission Product Trapping

a) As set forth in Section II.J.2 detailed measurements have been made on the release of fission products from the fuel compacts. Experiments have been done on the retention of fission products in the various portions of the trapping system. Based on these experiments, the expected distribution of fission product radioactivity can be stated to be as follows: 98.3% retained in the fuel compacts, 1.4% held in the internal traps, and 0.3% held in the external traps. The detailed inventories are given in Figures 54 and 55 and in Appendix B. For the purpose of designing the external trapping system, a safety factor of three has been applied to the release rates from uncoated fuel particles, and this system is being sized for an inventory of 2.6% of the fission product radioactivity.

b) Methods and materials to be used in the various traps.

(i) Internal fission product traps. As reported in Section II.C.2 each internal trap will contain a trapping reagent consisting of activated charcoal and metallic silver. This trapping reagent will be retained in the lower reflector portion of the fuel element. Some fission products will be retained in this trap by adsorption on the activated charcoal. Other fission products will be held by chemical reaction to form stable compounds. Experimental data on the operation of the internal trap are given in Section II.J.2.

(ii) External fission product traps. The active reagent to be used in the external fission product trap is activated charcoal. The total quantity of approximately 16,000 lbs. will be contained in a series of steel vessels through which helium to be purified of

fission products will pass. Fission products will be held in the traps by adsorption on the charcoal. The external fission product traps are described in Section II.C.6.

c) The engineering arrangements to accommodate safe handling and storage of fission products in the various traps are described in Section II.C.6 of the Report. The flow sheet for the process appears in Fig.56 in this section. The trapping system consists basically of a series of activated charcoal adsorption beds which delay the passage of fission products for a sufficient period of time that substantially all of the noble gas fission products decay within the trapping system. The Report concludes that this system is simple in concept and is conservatively designed with respect to heat removal capability. Provisions have been made for adequately cooling all necessary components during emergency as well as normal operation. Experimental data on the performance of the external traps are given in Section II.J.2.

d) Facilities and procedures for handling and storage of the trapped fission products are described in Section II.C.6 of the Report.

(i) The internal traps are an integral part of the fuel elements and will be shipped with the fuel elements from the Peach Bottom site. (See Section II.A.1) Each fuel element will be individually canned in a gas-tight metal container and each container will be individually leak checked. (See Section II.H.)

(ii) The design basis for the external trapping system requires that every trap in this system have a life as great as that of the reactor plant. However, it is recognized that it may be desirable to service or to remove individual traps from the system at some time during the life of the plant. For this reason, all of the individual traps are being designed so that they can be removed from the system. The procedure to be followed is described in Section II.C.6.

e) General considerations for the external fission product trapping system. The total inventory in the external fission product trapping system is sufficiently small so that if the entire inventory were instantaneously released to the containment, at the design leakage of the containment (0.2% of the contained volume per day), the release to the environment is low. Specifically, using an inversion condition in the vicinity of the site, and an exposure period of two hours, the integrated whole body dose at the site boundary would be approximately 0.1 rem. (See Section VII.C.)

"3. <u>Control System</u>

"Inasmuch as the control rods will be constructed of graphite sections, the uncertainties with respect to stability under irradiation which were previously mentioned under Fuel Elements above are applicable

here. The concern is more acute here, however, since the capability of reactor shutdown is involved. A major concern of the proposed control rod design involves the possible effects of separation of the rod elements from their respective drives. There yet remains to be resolved the possibility that a secondary shutdown may be necessary."

3. Control Rod System

a) Stability of control rods under irradiation. The Report describes the preliminary tests which have been made to determine the irradiation stability of neutron absorbing materials to be used for the control rods. These irradiated specimens have shown no signs of deterioration or changes in properties which would lead them to malfunction in the reactor. These results are detailed in Section II.J.1 and II.J.4. Further specimens will be irradiated before loading of the reactor in order to obtain more information on this point. (See Section II.J.1)

b) Separation of the rod elements from their respective drives. The control rod design incorporates in each rod an electrical system which will give a positive indication of a separation of the absorber section of the control rod drive unit. See Section II.B.7 for a description of this feature.

c) Secondary shutdown system. The reactor design incorporates an additional nuclear shutdown system as a back-up to the normal complement of control rods and drives. This system is described in detail in Section II.B.8 of the Report. The system incorporates an independent set of 19 neutron-absorbing rods which can be driven slowly into the reactor core and which will have sufficient worth, in terms of reactivity, to shut the reactor down from any operating condition. Each of the 19 secondary shutdown rods has its own source of power. In addition a third system will be provided whereby poison will fall by gravity into the reactor core, released by a set of thermally actuated devices in the event of core overtemperature. See Section II.B.8 of the Report for a description of this system.

"4. Core Design

"The proposed core arrangement involves fuel assemblies, guide tubes, and reflector elements which are fixed at the lower end while the upper extremities are supported laterally by differential coolant pressure forces. There is not sufficient design information presently available for an adequate evaluation of the proposed core arrangement with respect to flow disturbance, lateral stability, interferences with control rod motion, and oscillatory movement."

4. Core Design

a) Flow disturbance, lateral stability, and interferences of the control rod motion. Section II.J.3 of the Report describes a detailed model of the reactor pressure vessel and core which has been constructed on a 1/2 linear scale at General Atomic. This model was constructed for the purpose of investigating the flow distribution through the pressure vessel and core, and to demonstrate any mechanical problems which might arise from the method of fuel element and reflector support proposed. This flow model has been operated for some months and has shown completely satisfactory performance of the core arrangement. No evidence of problems with fuel element motion, from the standpoint of interference with the control rods or any other cause, has been discovered. The Report concludes that the core design is satisfactory. (See Section II.B.9)

b) Possible oscillatory movement of the core. An experimental search has been made for vibrations which might be excited in the fuel elements by the passage of coolant through the core. A bundle of 19 full size fuel elements was assembled for such a vibration test. It has been demonstrated that no vibration is excited by the passage of coolant through the bundle and moreover that any vibration excited by external forces is actually damped by the coolant flow. The Report states that there is no evidence whatever of oscillatory movements of the entire core during operation of the half-scale core model. (See Sections II.J.3 and II.B.9)

"5. Facility Design

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

"(a) Provision of an emergency cooling system in event of disablement of both primary loops.

"(b) Precautions to prevent water in-leakage to the core, which would cause rapid deterioration in the graphite components.

"(c) Safeguards against accidents or other events which would allow air to enter the primary system which would result in oxidation of the graphite."

5. Facility Design

a) Emergency cooling system. As described in Section II.F.2, the design of the reactor plant now incorporates an emergency

cooling system to remove heat from the core in the event that neither primary loop is available for cooling. With this emergency cooling system, heat is removed from the reactor core by means of a water-cooled steel jacket surrounding the pressure vessel. During normal operation, water is not supplied to the jacket. In the event that emergency cooling is required, water is pumped to the cooling tubes (See Section III.F) of the steel jacket and by this means stored energy and afterheat are removed from the core. The temperatures in the center of the core attained during such emergency cooling are higher than the normal operating temperatures, but are sufficiently low to prevent a failure of the primary system. (See Appendix C)

b) Precautions to prevent water in-leakage to the core. In the event that a high moisture level is detected in one loop, that loop can be isolated from the reactor core by means of valves in the primary system piping. In addition, the steam and water inventory in that loop would be rapidly dumped, and thus the total quantity of moisture entering the primary loop can be limited. Under this highly unlikely coincident combination of a steam tube rupture, a primary system rupture and a failure of the primary loop isolation valves, gas from the primary loop will be vented to the secondary containment vessel rather than to the dump tanks provided. In this event, further chemical reaction of the products of the steam-graphite reaction is prevented by the reduced oxygen atmosphere of the containment vessel. A description of the moisture detecting instruments and dumping is given in Section II.G, and the dumping process is described in Appendix D.

c) Entry of air into the primary system and consequent oxidation of graphite. The plant is now designed so that the secondary containment is filled with reduced oxygen gas rather than air. This precludes oxidation of graphite in the event of a primary system rupture. For information on this system see Section III.L.

"6. Accident Analysis

"Inasmuch as the Peach Bottom reactor represents a new reactor concept, experience is lacking regarding the possible types of accidents which could result in hazardous conditions. Consequently, further study should be directed toward the identification of all failures which could lead to on-site or off-site hazards. In particular it appears that more attention should be given to accidents which could result in radioactivity releases from the core. We are not convinced at present that the accident represented as the maximum credible in the hazards report actually is indicative of the upper limit of potential hazard

for this reactor. Further study of this matter will be necessary."

6. Accident Analysis

In the course of plant design and analysis a number of postulated accidents have been evaluated to determine the adequacy of the plant. The accidents analyzed show that the many safety features incorporated into the plant design provide adequate protection to the health and safety of the public and are indicative of the upper limit of potential hazard to the public. (See Section VII, Plant Safety Analysis.)

The following summarizes briefly the different kinds of postulated incidents which might lead to release of appreciable fission product activity to the secondary containment vessel.

A. Accidents which release fission product activity directly from the primary loop.

1. Ruptures of the primary loop.

a) A crack or weld failure in the primary loop piping would release slowly to the secondary containment the fission product activity contained in the helium of the primary loop. (See Section VII.B.2) Under normal conditions this is expected to be about 100 curies. The plant design assumes it might be as high as 4000 curies. (See Appendix B.1)

b) The complete rupture of one of the large ducts of the primary loop would release quickly the helium and contained activity of the primary loop plus an appreciable contribution of fission product activity from the resultant sudden depressurization and backflow of the external fission product traps. The attendant release to the secondary containment vessel is approximately 2.0 megacuries if the purge line check valve fails to close and only about 0.5 megacuries if the check valve closes. (See Section VII.B.2)

2. Tube rupture in one steam generator

In the event of a tube rupture in one steam generator the primary system and dump tanks are not overpressured even if the hot valve fails to isolate the faulty loop. Thus, no activity is released to the containment vessel during this incident. (See Section VII.B.2)

3. Simultaneous loss of both main coolant loops

The Report shows that in the eventuality of simultaneous loss of both main coolant loops the plant would be put on the emergency cooling system. This could occur if a rupture of a coolant loop

makes one loop inoperative and if the circulator in the second loop subsequently fails during the period that the radioactivity released to the containment precludes access for an extended period of time. This results in a slow increase of the temperature within the core and the consequent release to the primary system of part of the volatile fission product inventory from the reactor core. The activity in the containment rises to a maximum of about 5.2×10^6 curies at 16 hours and then drops slowly as a result of the continuing release from the reactor, radioactive decay, and removal by the emergency filtration system. (See Section VII.B.3)

B. Release of fission products from the external fission product traps. (See Section VII.C.)

The actual inventory of the external fission product traps is expected to be 1.4 megacuries. The design inventory, consisting of 11 megacuries, is approximately 9 megacuries of noble gas fission products and approximately 2 megacuries of other fission product elements. Various accidents to the fission product trapping system are postulated in the Report. (See Section VII.C.) None of these accidents would release more than 20% of the total activity in the fission product trapping system at the time of the accident.

C. Release of radioactivity from spent fuel elements after removal from the reactor core. (See Section VII.D.)

Each spent fuel element is individually canned in a strong metal container immediately on removal from the reactor pressure vessel. Each such can is individually leak tested. However, the failure of one of these cans could release up to approximately 11,000 curies of gaseous activity to the specially designed spent fuel storage building, where its release through the ventilation stack would be controlled. This release leads to a whole body gamma dose of 0.1 rem and a negligible thyroid dose at the site boundary under adverse meteorological conditions.

In case of water contamination in the pit the double-wall construction of the spent fuel pit precludes any loss of water to the Conowingo Pond. (See Section II.H.2)

II. <u>Letter from ACRS to the Commission, dated December 10, 1960.</u>

a) "The long-term integrity of the graphite under the proposed design conditions, which lie far outside past experience with graphite, has yet to be established."

Information appears under I.1 above.

b) "The experimental data required to determine that the fission products emitted continuously from the homogeneous fuel components can be collected, stored and disposed of safely have yet to be developed."

Information appears under I.2 above.

c) "A novel design is proposed for the control rod system, but the development is not yet sufficiently advanced to permit adequate evaluation of its reliability."

Information appears under I.3 above.

d) "The applicant has not provided either a secondary back-up safety system or an emergency coolant system, but has not yet established to the satisfaction of the committee that these systems are unnecessary."

Information appears under I.3 and I.4 above.

e) "The relatively large assumed Doppler coefficient has yet to be confirmed."

Further theoretical and experimental work has verified completely the existence of a negative prompt temperature coefficient for the reactor based on the Doppler broadening of the thorium resonance neutron absorption. In addition, it can now be stated, as the result of further nuclear design work, that both the slow and prompt temperature coefficients of reactivity of the reactor will be negative throughout reactor life and at all obtainable temperatures. (See Section II.A.2)

ORIGINAL

BEFORE THE

UNITED STATES ATOMIC ENERGY COMMISSION

In the Matter of	:	
	:	Docket No. 50-171 — Formal File
PHILADELPHIA ELECTRIC COMPANY	:	

DOCKETED USAEC Division of Licensing & Regulation AUG 7 1961 Docket Officer

AMENDMENT NO. 2

TO

APPLICATION OF

PHILADELPHIA ELECTRIC COMPANY

FOR

CONSTRUCTION PERMIT AND CLASS 104 LICENSE

Vincent P. McDevitt
Eugene J. Bradley
Robert P. Garbarino

1000 Chestnut Street
Philadelphia 5, Pa.

Attorneys for
Philadelphia Electric Company

BEFORE THE

UNITED STATES ATOMIC ENERGY COMMISSION

In the Matter of	:	
	:	Docket No. 50-171
PHILADELPHIA ELECTRIC COMPANY	:	

AMENDMENT NO. 2

TO

APPLICATION OF

PHILADELPHIA ELECTRIC COMPANY

FOR

CONSTRUCTION PERMIT AND CLASS 104 LICENSE

Philadelphia Electric Company hereby amends its Application in the above entitled proceeding as follows:

Part A--General Information

1. Delete pages 3, 7, and 8 and insert in lieu thereof pages 3, 7, and 8 dated August, 1961.

2. Delete Exhibit D, entitled "Annual Report For The Year 1959, Philadelphia Electric Company", and insert in lieu thereof Exhibit D, entitled "Annual Report For The Year 1960, Philadelphia Electric Company".

3. Delete Exhibit F, entitled "Estimated Schedule Of Transfers Of Special Nuclear Material", and insert in lieu

thereof Exhibit F, entitled "Estimated Schedule Of Transfers Of Special Nuclear Material" dated August, 1961.

Part B--Preliminary Hazards Summary Report
Volume I--Plant Description And Safeguards Analysis

4. Delete Volume I, Plant Description And Safeguards Analysis, and substitute in lieu thereof Volume I, Plant Description And Safeguards Analysis dated August, 1961.

Part B--Preliminary Hazards Summary Report
Volume II--Site And Environmental Information

5. Delete pages I-10 and I-11 and insert in lieu thereof pages I-10 and I-11 dated August, 1961.

6. Delete page II-3 and insert in lieu thereof page II-3 dated August, 1961.

7. Delete pages II-8 through II-12 and insert in lieu thereof pages II-8 through II-12b dated August, 1961.

8. Delete pages II-20 and II-23 and insert in lieu thereof pages II-20 and II-23 dated August, 1961.

9. Delete pages III-9 through III-16 and insert in lieu thereof pages III-9 through III-16b dated August, 1961.

10. Delete Figure I-9 entitled "Environmental Radiation Sampling Stations" in Appendix A--Figures, and insert in lieu thereof Figure I-9 entitled "Environmental Radiation Sampling Stations" dated August, 1961.

11. Delete Table I-12 in Appendix B--Tables, and insert in lieu thereof Table I-12 dated August, 1961.

12. Delete Tables II-6(A), II-6(B), II-6(C), and II-7 in Appendix B--Tables, and insert in lieu thereof Tables II-6(A), II-6(B), II-6(C) and II-7 dated August, 1961.

13. Insert in Appendix B--Tables, Tables II-13(A) and II-13(B) dated August, 1961.

Respectfully submitted,

PHILADELPHIA ELECTRIC COMPANY

By _____
President

Commonwealth of Pennsylvania)
) SS.
County of Philadelphia)

R. G. Rincliffe, being first duly sworn, deposes and says:

That he is President of Philadelphia Electric Company, the Applicant herein; that he has read the foregoing Amendment and knows the contents thereof, and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

R. G. Rincliffe

R. G. Rincliffe

Subscribed and Sworn to
before me this 4th day
of August, 1961

Jean T. Moshirook
Notary Public

Notary Public, Philadelphia, Philadelphia Co.
My Commission Expires January 17, 1963