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ATOMIC ENERGY COMMISSION

DIVISION OF LICENSING AND REGULATION

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ON

PEACH BOTTOM ATOMIC POWER STATION

PHILADELPHIA ELECTRIC COMPANY

Note by Director, Division of Licensing and Regulation

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

A-190

PEACH BOTTOM ATOMIC POWER STATION
PHILADELPHIA ELECTRIC COMPANY

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

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

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

Fuel Elements

Several basic design changes have been made to the fuel element since the previous review. The annular fuel compacts are now contained within a single impervious graphite sleeve, whereas the previous design provided a double impervious sleeve arrangement. In the prior design the purge flow through the element entered near the grid plate stand-off pin, passed upward between the outer and inner sleeve, back down around the fuel compacts, through the internal trap and out through the stand-off. With the new single sleeve design purge flow is introduced through the porous upper reflector end piece, down around the compact and internal trap to the stand-off.

Other basic changes in fuel element design include the use of pyrolitic coated fuel particles in the graphite-fuel compacts, addition of Rhodium-103 to the fuel compacts, and placement of Boron Carbide as a burnable poison in the graphite support spine. Each fuel particle (100 to 400 microns in diameter) is pyrolitically coated with a 50 to 60 micron thickness of dense carbon to protect the fuel from oxidation during compact fabrication and to increase the retention time of fission products during operation. Particles of Rhodium-103 will likewise be pyrolitically coated prior to their addition in the compact materials. The purpose of the Rh^{103} is to ensure a more extensive negative temperature coefficient throughout the core life; this will be discussed further in a later section of this report.

Following our previous review, the applicant was informed that additional information was needed regarding the effects of radiation during the long fuel cycles on structural integrity and dimensional stability, effective thermal conductivity, and the mobility of fuel and fission products in the compacts.

The latest amendment includes information on the research and development program to establish long term integrity of graphite components, including low permeability graphite tubes, fuel compacts, control materials, and reflector materials. The information contains the results of the programs conducted thus far to support the design, involving investigation of the effects of irradiation, temperature, and chemical impurities on the mechanical, physical, and dimensional stability of all graphite materials in the reactor. The R & D program includes both in-pile and out-of-pile testing.

Based on our review of the results to date of the R & D program and the further tests proposed we have concluded that there is now reasonable assurance that the impervious graphite diffusion specification of 1×10^{-6} cm^2/sec can be met, that the structural integrity against mechanical loads is adequate, that stress loads resulting from dimensional changes can be controlled by proper dimensional control, that the minimum specification for thermal conductivity of 15 Btu/hr/ft^{o}F can be met, and that the use of pyrolitic coatings to the fuel particles adequately prevents the mobility and improves the retention of fission products.

Fission Product Trapping

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

The design presented in the amended report provides several mechanisms to limit the fission products in the primary system. First, release of volatile fission products from the fuel compacts is delayed until the short-lived products have decayed. Second, most of the condensible products which do escape the fuel compacts are removed in an internal trap located in the cold end of the fuel element. Third, the fission gases and other fission products which are not held in the fuel compact or internal trap are removed by a helium purge stream to a system of traps external to the reactor vessel. Fourth, the outer sleeve of the fuel assembly, which encloses the fuel compacts and internal trap, is fabricated from very low permeability graphite to limit release of products directly to the core coolant.

The amended report presents results from experimental work carried out to give quantitative information relating to the above four mechanisms designed to limit fission product release. These experiments include: measured releases of fission products from fuel compacts at elevated temperatures as a function of burnup for both coated and uncoated fuel particles, quantitative determination of the trapping efficiency for the reagents to be used in the internal traps, experimental data for the low permeability graphite, and the capacity of the external trapping system.

Because the long term burnup experiments duplicating end-of-life conditions have not in all cases been completed, the applicant has chosen to base the design requirements of the traps by applying a factor of 3 to the activity and heat load values corresponding to uncoated fuel particles. Data collected to date on coated fuel particles at a burnup equivalent to 50% of the end-of-life value indicate that the coated particles will decrease the amount of fission products released from the fuel compacts by an additional factor of 2.5.

An in-pile loop test, which incorporates a quarter length prototype fuel element and prototype traps is currently in progress. Preliminary results of this test indicate that the design is conservative.

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

Control System

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

The amended information recently submitted indicate that extensive modifications of the control system have been accomplished and that several new design additions have been incorporated.

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

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

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

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

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

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

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

Based on the amended information showing the extent of development of the control rod materials and results of the testing program, it is our opinion that the design of the control rod system is feasible. It also appears that suitable methods have been provided for indication of rod separation, and that the backup shutdown system would provide assurance that adequate shutdown capability is available in the event of abnormal conditions, or loss of the normal control rods.

Core Design

On the basis of previous information submitted it was concluded that sufficient information was not available for an adequate evaluation of the proposed core arrangement with respect to flow disturbances, lateral stability, interferences with control rod motion, and possible osillatory movements.

The information now submitted provides data from test programs recently conducted intended to resolve these questions. A detailed ½ scale hydraulic model of the entire pressure vessel and core was constructed and operated to determine the feasibility of the hydraulic design. This hydraulic model has now been in operation for about 10 months. In addition to the model a test chamber was built in which a cluster of 19 full scale graphite fuel elements were subjected to flow conditions equivalent to core conditions. Vibrations caused by flow conditions were monitored and externally excited vibrations were intentionally introduced to determine damping characteristics. Under no flow conditions did the fuel elements vibrate nor were the forced vibrations affected by the flow.

The applicant has recently described results of tests on a prototype, control rod drive wherein the guide tube was deflected 1¼" at the top with no apparent effects on rod operation. In addition the rod drive unit was deflected 9 inches from centerline below the vessel and only a slight change in rod speed was noted.

It is our opinion that information now available provides assurance that the problems previously identified with respect to core design will be resolved in a satisfactory manner.

Facility Design

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

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

In addition to the information contained in the amended report on the reactor pressure vessel, the applicant has been requested to provide more information regarding the design specifications and fabrication and operational limits of the pressure vessel. The applicant has agreed to submit this information prior to the October ACRS meeting.

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

If an abnormal amount of moisture is detected the reactor is scrammed and the following action takes place:

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

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

It now appears that the applicant has provided sufficient assurance that the problems of emergency cooling, water leaks and air entry have been resolved. We have not reviewed the outstanding information regarding the design and control of manufacture for the reactor pressure vessel.

Safety Analysis

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

Contained in the safety analysis are studies on the following:

1. Incidents involving the reactor

    a. Reactivity accidents
    b. Loss of fission product barriers
    c. Loss of both main loops following rupture.

2. Incidents involving the Fission Product Trapping System

    a. Loss of full cooling capacity
    b. Loss of system integrity
    c. Change in purge environment.

3. Safety of fuel handling

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

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

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

5. Environmental consequences of accidents

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

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

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

Associated with the safety analysis of the reactor the applicant has completed more extensive work to verify the Doppler coefficient and over-all temperature coefficient. The Doppler coefficient was established by both refined calculational methods and verification by experimental work using the linear accelerator as a source of pulsed neutrons. This work was also compared with measurements on the Zenith critical assembly in England and measurements with the HTGR critical. As a result of adjustments to the loading of thorium in the fuel, the lower fuel compact temperatures, and better calculations the over-all temperature coefficient is shown to be negative throughout core life. To provide added assurance of negative temperature coefficient, 5 kg of rhodium-103 has been added to the core.

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

Conclusions:

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

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

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