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Abstract

The Pebble Bed Reactor Design, S&P 1963, of Sanderson and Porter is reviewed. Design problems, economic considerations, and the required research and development programs are discussed.

NOTICE

This document contains information of a preliminary nature and was prepared primarily for internal use at the Oak Ridge National Laboratory. It is subject to revision or correction and therefore does not represent a final report.

ERRATA

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Review of S&P 1963, The Pebble Bed Reactor

Page 11: Line 16 should read:

"...In general there is no reason ...."

Page 16: Line 20 should read:

"Million joint years"

Page ~~23~~ 33: Line 19 should read:

"...shield, however, this is definitely not the case ...."

Page 37: Line 20 should read:

"... leakage of contaminants from and if an oil seal is used ...."

REVIEW OF S&P 1963,

THE PEBBLE BED REACTOR

by

the Staff of ORNL

1. Introduction and Summary

Sanderson and Porter, under contract to the AEC, has prepared a design feasibility study for a Pebble Bed Reactor steam power plant of 125 Mw electrical capacity. The Oak Ridge National Laboratory has been requested by the AEC to review the design as reported in the Sanderson and Porter Report No. 1963 with "particular emphasis to the consideration of the potential of this type reactor for economical power recovery and the estimated scope of research and development program".

In accordance with this request ORNL has undertaken to consider the following aspects of the report:

1. The specific details of the design prepared by Sanderson and Porter in order to evaluate the validity of the basic assumptions of the design.
2. In the light of this study, the potential of the Pebble Bed Reactor concept irrespective of the specific proposal of Sanderson and Porter.

With regard to the proposal by Sanderson and Porter it is concluded that the design shows promise of being economically competitive if the following questions can be favorably resolved:

1. Is U-233 available for use in this reactor as a burner rather than as a breeder? It should be noted that the design conversion ratio is only 0.863 so that U-233 would necessarily be required as makeup fuel for the reactor lifetime.
2. Can the contamination problems within the reactor be successfully handled in a manner which would permit direct maintenance on the equipment external to the core? Unless this provision can be experimentally substantiated it will be necessary to plan for remote maintenance of the major equipment included in the design. The capital cost penalty for remote maintenance cannot be evaluated at this time.

3. When fuel fabrication costs have been weighed in the light of radiation damage experience, will the anticipated lifetime of the fuel be sufficiently long to justify the cost assumptions of the Sanderson and Porter design?

The answer to these questions can be developed only by further design and experimental effort and at this date can be only a matter of conjecture.

With regard to the Pebble Bed Reactor concept in general it would appear that if the answers to only the second and third questions regarding S&P 1963 can be resolved affirmatively this reactor type has a potential place in the overall gas-cooled reactor program.

Further work on core design is indicated to determine if the core would be suitable as a U-235 burner. In Section 2.1 of this review it is noted that some of the assumptions concerning nuclear cross sections are pessimistic while others are optimistic. It was not clear from the information in S&P 1963 whether the reactor's characteristics as a U-233 breeder would be improved if these assumptions were modified and the influence of these considerations should be examined to determine if the design could actually be a breeder under modified conditions. If the latter does not prove to be the case new core arrangements should be considered in order to determine whether a breeder potential can be realized from this type design.

A proposed research and development program has been outlined in Section 8 of this review. It would seem that of the items listed those most urgently required are:

1. Fluid flow and heat transfer data on pebble beds.
2. In-pile tests on the containment aspects of contamination from the reactor core.
3. In-pile radiation data on fuel balls manufactured by the several techniques proposed in S&P 1963.

A detailed discussion of the various aspects of the pebble bed reactor design are contained in the following review. It is worthwhile to note at this time that particular emphasis has been placed in the study on the problems associated with the containment of the contaminated coolant. The purpose of this emphasis has been to bring out the issues involved in this extremely nebulous area rather than to single it out as a consideration on

which the success of the reactor design must either stand or fall. It should be appreciated that if the fuel balls could effectively contain the activity then many of the problems would disappear and other design considerations mentioned are therefore equally important to the success of this reactor concept.

2. Reactor Core Design

2.1 Reactor Physics

Comment on the physics section of S&P 1963 centers around several points where further investigation is necessary before a detailed design of a specific reactor plant can be considered as having a firm basis. The following thus does not constitute an adverse criticism of the pebble bed concept itself.

The question of whether it is feasible to base the design of a system with a breeding ratio less than unity on the assumption that U-233 will be available is perhaps the most difficult to answer. It is almost certain that reactors of this type would have to be operated with U-235 as the fuel for a long enough period to establish a high inventory of bred material. This startup period is likely to have an important effect on the power cost and should be examined on that basis to determine its importance. During this period of operation with U-235 as the fissionable material the loading of the core must be substantially increased to compensate for the larger value of the capture-to-fission ratio for this material as compared with U-233. In addition, the reactor, if fueled with U-235, would have a lower conversion ratio. It is important to recognize that a reactor with a conversion ratio less than unity, as is the case with the S&P design, cannot be started with U-235 as the fuel and gradually changed over to a pure U-233 cycle using only its own bred material. If the reactor is to operate eventually with U-233 alone then enough of this material must be available from other sources to make up for the burnup of fissionable material. A less-than-unity conversion ratio will thus always return one to the problem of the supply of U-233 regardless of how the reactor is started up. It would then appear desirable to consider such systems on the basis of a mixed cycle where the consumed material is replaced by U-235 rather than U-233.

If the feasibility of a U-233 system with less-than-unity conversion ratio is accepted then the system must be analyzed strictly on the basis of using conversion to produce a low fuel cycle cost. This is what has been done by Sanderson and Porter. In order to realize a low fuel cycle cost the reactor system must have both a low fuel inventory charge and a

low fuel burnup charge. The first of these is achieved by producing a high specific power (kw/kg) in the fissionable material and the second by minimizing the losses in the breeding cycle. In addition, the core must have a long reactivity lifetime if the fuel fabrication charges are to be low.

The first two of the above items are competing processes in the S&P design since minimizing the losses in the breeding cycle requires a low rate of absorption in Pa-233 and hence a low neutron flux in the fertile material. With a value of $15/gram for U-233 and an interest rate of 4%/year the inventory charge in a plant operating with an efficiency of 35% and a load factor of 0.80 can be shown to be:

$$C = \frac{0.2}{X} \text{ mills/kwh}$$

where X is the specific power in the fissionable material in units of 1000 kw/kg. Since the specific power in the Pebble Bed Reactor (PBR) is about 4000 kw/kg a very low inventory charge is realized for the core material. The Pa-233 losses with the resulting flux of about 1.8×10^{14} n/cm^2-sec amount to 3% if the Pa-233 absorption cross section is assumed to be 50 barns (the value used at 1200°F in S&P 1963). This represents a 6% loss of neutrons and breeding ratio, however, since the neutron absorbed in Th-232 is also lost to the breeding cycle.

It must be emphasized, however, that the value of 50 barns is probably optimistic for the Pa-233 cross section in the PBR. The experimental data on the cross section of this element are quite incomplete, but measurements of the effective pile cross section[1] and resonance integral above 0.4 ev have been made.[2] These indicate an effective cross section in the neighborhood of 150 barns for <u>thermal</u> reactors and a resonance integral of about 700 barns. The large thermal cross section and resonance integral give strong indications that a prominent resonance structure occurs in the thermal or low epithermal regions. The assumption made by S&P that the cross section

(1) J. Halperin, R. W. Sloughton, C. V. Ellison, and D. E. Ferguson, Nucl. Sci. and Eng., 1, 1 (1956).

(2) R. R. Smith, IDO-16226 (1955), unpublished.

is 1/v is therefore probably not valid. It is not out of the question that the cross section in the somewhat intermediate neutron spectrum of the PBR is larger by a factor of 2 or 3 than the value assumed by S&P. Two effects of such an increase of this cross section would be a decrease of about 0.15 in the breeding ratio and a reactivity decrease of about 0.02. This decrease in reactivity would decrease the core life from 100 days to about 70 days.

This possible underestimate of the losses due to Pa-233 is perhaps compensated by the large estimate of the capture-to-fission ratio for U-233. Here, as with Pa-233, the present experimental data do not fully resolve the problem, but it would appear that a value near 0.2 for the ratio at intermediate energies is more accurate than the value of 0.5 used by S&P. The effect of this lower value for the capture-to-fission ratio is discussed in S&P 1963.

In summary the following items point to the regions where further study is indicated and more complete data are desired:

1. Re-examine the advisability of basing low conversion ratio systems on a pure Th-U-233 cycle.
2. Provide better cross section data for the important isotopes. In particular better information on the energy dependence of the Pa-233 absorption cross section and the energy dependence of α for U-233 in the epithermal and high-energy regions should be available.
3. Consideration should be given to systems where the fertile and fissionable materials are completely segregated in order to produce a higher conversion ratio.

2.2 Fluid Flow and Heat Transfer

The problem of steady-state heat transfer in beds of spherical particles will require development of a new technology. There are many chemical engineering processes which make use of mass transfer properties of packed beds, and there are instances where packed beds are used in non-steady state heat transfer, but prior to the pebble bed reactor concept there has been no application of steady state heat transfer.

Fluid flow data are available from previous work relative to mass transfer in packed beds, and these are applicable to the pebble bed reactor if used with

caution. A limited amount of heat transfer data are available, but details of the test are uncertain. The performance of the pebble bed reactor as outlined in S&P 1963 falls within the range of available data if such data are taken at their face value. No mention is made in available references as to the Mach numbers at which data were taken, but it is almost invariably the case when this omission is made that test conditions are well within the limits of subsonic flow. When compressible flow encounters "body"-type obstructions, as distinguished from surface friction, the loss coefficient can be strongly affected by Mach number. This can take place at Mach numbers which are normally regarded as incompressible. In the case of flow through a screen having a solidity (i.e., a significant ratio of blocked area to total area) of 0.65, the loss coefficient is a constant up to a Mach number of 0.1; between $M = 0.1$ and $M = 0.2$ the coefficient increases by 60%, and at approximately $M = 0.22$ the screen will choke. If heat is added to the gas the system will become more sensitive to Mach number. A factor of two error in pressure drop and pumping power is easily possible if loss coefficients obtained under incompressible conditions were used in regions where compressibility effects were being felt.

Enough data are not available to say whether or not the results of the Sanderson and Porter calculations are correct but this very lack of data indicates the need of a very extensive test program before design of a pebble bed reactor power plant can be fully evaluated.

2.3 Structural Problems

The PBR pressure vessel is of a conventional design for reactors and fabricated of a material (SA212B) commonly used by industry; hence, it should present no unusual problem in design. The design will be somewhat more complicated than that for a similar type pressurized-water reactor due to the lack of hydrogenous material to attenuate the fast neutron flux. This condition, however, would be true for any gas-cooled reactor and not peculiar to the pebble bed reactor. In the Sanderson and Porter study water is used outside of and adjacent to the pressure vessel to thermalize the fast flux before reaching the concrete shield. In so doing the water has a secondary effect of producing an unusual thermal gradient in the pressure vessel wall. This effect may be beneficial in reducing the overall stress in the pressure vessel; however, due to the unusual stress

pattern a careful and detailed stress analysis will be required. The report does not indicate that any rigorous stress analysis has been performed to optimize the pressure vessel or the thermal shield.

The pebble bed design lends itself to minimizing the number of penetrations into the pressure vessel. For example, the PBR has a total of 29 penetrations, three of which are major in size. Comparing this with Kaiser's 55-Mw gas-cooled reactor proposal[3] which has 177 penetrations, or the ORNL proposal[4] which has 77 penetrations, eight of which are major in size, it can be seen that a substantial reduction in penetrations can be achieved. Although the smaller penetrations simplify the stress problems, they are still a problem as potential sources for weld and seal leaks.

The report points out that the balls do not readily rise to reach their own level and takes advantage of this fact in the discharge mechanism. This same principle, however, along with any subsequent galling of the balls will result in a radial expansion (which amounts to about 1/4 in. across the core) as the reactor core heats up. It would seem that such an expansion could crack the graphite cylinders in the core. As these cylinders serve as the guides for the control rods, a hazardous condition could result. It is not apparent from the report that there exists any method of replacing the graphite pieces after reactor operation is initiated.

The use of concentric piping for the primary coolant has an advantage in reducing the number of penetrations of the pressure vessel, reduces the amount of high-temperature insulation required on the high-temperature primary piping system, and apparently eliminates the need for expansion joints. The use of high-temperature concentric piping is not commonly used in industry today, and although there is no apparent difficulty in its use, careful design is required particularly in the lower loop section in order to allow for differential expansion of the two pipes.

There is no apparent reason why the pressure vessel and piping design for a pebble bed reactor should be more difficult than for other types of gas-cooled reactors. In fact, the concentric piping concept and the mobility with which the small fuel balls can be inserted and discharged from the core should somewhat simplify the vessel and piping design.

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(3) Kaiser Engineers-ACF Industries Report IDO-2021.

(4) ORNL-2500, The ORNL Gas-Cooled Reactor.

3. Reactor Fuel

3.1 Fuel Fabrication

The S and P 1963 presentation notes the importance of structural integrity in graphite balls before and after irradiation. Three methods of fabrication are indicated as being available including: (1) impregnation of graphite spheres, (2) lumping of the fuel in a small sphere jacketed with graphite, and (3) admixture of fuel oxide, graphite, and binder. Of the three methods, preference appears to be for the impregnation technique although the decision is left open for further investigation. Since the final selection must be based on the overall fuel economics it is premature to judge the acceptability of these methods without further investigation. The Ceramics Department of the Oak Ridge National Laboratory has made some very preliminary investigations of the virtues of the admixture as opposed to the impregnation method of fuel production and believes there is reason to expect the radiation stability of the impregnated balls to be somewhat lower than that of the baked molded mixtures of method three above, but since overall physical properties both before and after operation must be considered, it is necessary to establish which method gives the best combination of physical properties. In any case, it will be important to obtain adequate data under radiation conditions comparable to the S and P 1963 value of 100 full power days before establishing the acceptability of a particular fuel material.

With regard to the impregnation technique the principal problem involves the amount of loading or saturation of the uranium salts in the graphite balls which can be tolerated. The manufacture of the balls using admixtures will be greatly dependent upon the baking temperature which is selected. At temperatures of approximately 1200°C the binder will tend to coke whereas the temperatures above 2500°C the binder would tend to graphitize. The higher temperatures would also lead to conversion of oxides of uranium to carbides which may be a desirable property. It is complicated, however, by the fact that at the higher temperatures there are likely to be losses of uranium by vaporization. The baking temperature to be selected will ultimately be determined by the physical properties desired and its effect on these properties needs to be investigated. The fuel lumping technique may have merit as a means of reducing the amount of fission gas

or fission product release through the fuel balls but is, of course, limited by the heat transfer properties of the fuel oxides when lumped in the center of the fuel ball. It is certainly premature to rule out this particular method of fabrication because of its inherent value as a containment medium.

Methods for establishing acceptability of a fuel ball for installation in the reactor also need to be investigated. Nondestructive testing techniques such as ultrasonics and careful visual examination may be necessary to establish acceptability. Uranium and thorium content will also need to be carefully controlled in order to maintain a uniform and predictable power production capacity within the balls. Means for testing the physical properties of the balls may also be required. (The simple bouncing technique suggested by S and P 1963 may be adequate, but it is also possible that hardness tests or other nondestructive testing means may be necessary.)

As indicated in S and P 1963, there is reason for optimism regarding the problem of sticking balls. In general there appears to be reason for expecting the uncoated graphite to stick. However, the fact still needs to be substantiated under radiation and under conditions where mass transfer of the graphite can occur. Further work in this direction is advisable. If the balls were coated there is more likelihood that sticking will occur, but again, this is dependent upon the properties of the coating and experimental investigation will be required to determine whether actual difficulties exist. It might also be noted that the coating proposed for the graphite cylindrical tubes surrounding the fuel balls could be a source of trouble.

3.2 Fuel Reprocessing

The S and P 1963 study indicates that the chemical processing method for the graphite fuel balls is likely to be cheaper than that of the zirconium and stainless steel clad fuel elements of other reactor types. Such a conclusion seems premature. Fuel element jackets have their merits as well as disadvantages, one of these being that the fuel is thoroughly contained until ready for reprocessing. In the case of the graphite fuel balls some type of container would be necessary to avoid contamination of the surroundings from the time the fuel element leaves the reactor until it is introduced into the reprocessing plant. The simple underwater storage technique used for the storage of clad fuel elements probably is not applicable in this case.

The entire process for recovering uranium from graphite fuel balls needs further investigation. It is desirable to study an integrated design for the recovery of fuel in this form before attempting to establish the reprocessing costs.

4. Instrumentation and Controls

Since the S&P 1963 design is only conceptual it does not detail the instrumentation and controls requirements of the reactor. Control rods and control rod drives were, however, described. The proposed cobalt control rod is of itself good from a nuclear standpoint as a control rod material, but has the objection that it reaches a high level of activity when subjected to nuclear radiation. The high activity would make servicing of the control rod difficult and might introduce problems in obtaining access to the control rod drive. It was not apparent from the study why materials such as boron steel, which would be subject to much lower levels of induced activity, were not considered. With regard to the drive mechanism the device appears to be conventional for this application except that the method of sealing the rod from the drive mechanism has not been clearly described. It would appear that some buffer provisions would be desirable to avoid outward leakage of contaminant from the control rods.

The scram provisions described in S&P 1963 are not clearly motivated. Scram operation might lead to serious thermal shock because of abrupt changes in the power level of the reactor and could possibly damage the reactor structure. From the hazards description in S&P 1963, it would seem that adequate provision has been made for avoiding serious nuclear excursions without requiring reactor scram so that the absence of this feature would impose no additional hazards and would eliminate an additional source of potential trouble.

The high-temperature differential between the helium inlet and steam exit of the steam generator naturally raises some doubt as to the sensitivity of the system to slight shifts in load. There seems little reason for concern about the effects of the changes in temperature on the steam side of the steam generator. The system there is sufficiently sluggish so that load swings could be handled without difficulty. On the gas side of the steam generator the problem is more difficult to evaluate without simulator studies of the reactor system, but the Sanderson and Porter 1963 design has apparently sidestepped the difficulty by provision for continuous dumping of all three steam generators in the event of a

change in system load. This provision should protect the design against any severe demand transients.

The method for monitoring steam leaks should be adequate to warn the operator of difficulty in the event small leaks develop. However, if a true rupture were to occur, it would be desirable to valve off the supply of water to the steam generator as quickly as possible. Some type of valving system to accomplish this purpose is desirable.

The whole problem of system flow stability is suspect in this design. The ability of compressors to operate in parallel, particularly during startup, must be demonstrated. It is conceivable that some type of bypass flow or variable speed drives may prove desirable to avoid flow instability problems.

It should also be noted that because of the lack of means for varying gas flow through the reactor the exit temperature of the reactor will change with load conditions. However, by the use of pre-whirl vanes as a continuous control device for gas flow the problem may be circumvented. The suitability of this system of control must be examined in conjunction with the compressor specifically selected for this application.

5. Materials Considerations

The compatibility of the materials in this reactor system is of concern both for the situation existing during normal operation and situations which may exist during various postulated accidents and failures. Of principal concern during normal operation is the mass transfer of the graphite which will occur at a significant rate if the CO_2 impurity of the gas stream is allowed to attain excessive values. On the other hand, different failures can introduce O_2 and/or steam into the helium system so that the potential reactions of these gases with the components of the reactor system are of concern. Although the nature of all of these potential reactions is known, the rate at which these reactions can occur in a system like the PBR is not. However, it may be concluded[5] that the reactions will not preclude the operation of such a system if the CO_2 level of the gas is maintained below 0.001 mole percent and that the hazard due to the system of O_2[5] reaction with graphite should not be excessive.

5.1 CO_2-Graphite Reaction

The report makes no mention of the CO_2-graphite reaction which at the operating temperatures of the PBR could be of concern. For example, if it assumed that there is 0.01 mole percent of CO_2 in the helium and if the reaction $CO_2 + C \longleftrightarrow 2CO$ attained thermal equilibrium in both the hot and cold portions of the gas system several thousand lb/hr of graphite could be transferred. This is obviously intolerable, but it is also obviously impossible for the reaction to attain thermal equilibrium with the helium circulation times in question. Actually how much graphite would be transferred is not known, but similar studies[5][6] based upon experience at Calder Hall had suggested a maximum mole concentration of CO_2 in the GCR-2 of 0.073%. Such analogies would suggest that the concentration of CO_2 in the PBR should be maintained $<$ 0.001%.

A source of significant quantities of O_2 in the PBR gas system must be presumed to exist since 1) the metal wall will contain occluded gases which will be replenished everytime the system is opened up, 2) there

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(5) The ORNL Gas-Cooled Reactor, ORNL-2505, p. 13.97.

(6) The ORNL Gas-Cooled Reactor, Part 3, ORNL-2500, p. 6.51.

will be residual air in the system everytime it is opened up, 3) the adsorptive properties of the Si-plated graphite chamber walls are not proven, 4) each fresh batch of fuel and/or blanket balls will introduce occluded oxygen and moisture, and 5) the reaction $UO_2 + 3C \rightarrow UC + 2CO$ will release CO to the gas.

In view of the above oxygen sources and the probable tolerance on oxygen in the helium during operation adequate provisions should be made for the control of the oxygen level. This can presumably be effected by a scrubber system[6] although high fractional flow rates may be required. Such a scrubber has the added advantage of removing tritium activity from the gas. The tritium activity from ${}_2He^3 + {}_0H^1 \rightarrow {}_1H^3 + {}_1H^1$ could otherwise amount to $\sim 10^4$ curies of β activity in the helium in both the gas system and that which had leaked to the containment vessel.

5.2 Steam Leak Into System

The most probable type of failure which is likely to occur in the steam generator is a tube leak although experience has shown that this type of leak is rare. Calder Hall has experienced none in over a year's operation and coal-fired generator data show about 15 leaks in 1-1/2 million years of operation. Since the pebble bed reactor will operate at a higher temperature and pressure it may be possible that these leaks may be more frequent, although it is felt that this probability is low in view of the otherwise favorable environment.

In any case, steam will enter the reactor on tube rupture and most likely will cause a reaction in the fuel balls;

$$C + 2H_2O \rightleftharpoons CO_2 + 2H_2 - 71{,}600 \text{ Btu}$$

$$\text{and/or} \quad C + H_2O \rightleftharpoons CO + H_2 - 70{,}900 \text{ Btu}$$

$$\text{and/or} \quad CO + H_2O = CO_2 + H_2 - 700 \text{ Btu}$$

$$\text{and} \quad CO_2 + C = 2CO - 70{,}200 \text{ Btu}$$

The reaction rates at temperatures below 1200°F are believed to be fairly small, (< 10%/1000 days), increasing rapidly with higher temperatures. However, preliminary calculations show, that given adequate steam, only about 3 to 4% of the fuel ball graphite would be consumed before temperatures

are reduced to a level of negligible reaction rates if the reactor is shut down immediately.

The behavior of the silicon-carbide coating of the graphite structural elements and moderator in contact with high-temperature steam is not known and should be tested. It is felt however that no significant reaction will occur due to the apparent stability of this material at these temperatures. The possibility also exists that cracks may occur in the SiC coating due to impacts and thermal cycling of the fuel balls. This does not appear to be very probable since preliminary tests with this material show that it has fairly good mechanical strength, and that it does not crack easily as a coating. This material also possesses fine resistance to thermal shock and good thermal conductivity. Even if cracks do occur as a result of the thermal expansion of the fuel balls it does not necessarily follow that the graphite will suffer significant structural damage in the time exposed to a steam leak.

Predication of the design on the use of this material appears justified, but further test data should be obtained.

6. Cleanup and Maintenance

The activity level throughout the system is one of the most important considerations in the proposed plant as the effect of the contaminated coolant on maintenance costs may outweigh the nuclear gains associated with the elimination of a cladding on the fuel. Sanderson and Porter have proposed a method for estimating the activity released to the coolant and attempted to limit the consequent system contamination by the use of a filter and bypass cleanup system. However, the PBR report presents no values for the maximum activity levels expected in the various portions of the system. Since it is only by inference that one can deduce the radiation fields which are expected to exist during maintenance of the system equipment there is no assurance that maintenance can be effected in the prescribed manner. It is possible that our estimate of the activity released may be high but it is also true that the effectiveness of the filter and bypass cleanup system in limiting the activity levels is doubtful.

The maintenance philosophy described in the report is to isolate the loop with the defective component by four valves and then to perform the required servicing from behind shielding and/or after decontamination. This approach is logical since, clearly, the unclad fuel must be separated from the defective component during direct maintenance. One must, however, consider the consequences if these valves failed (including poor seating) and how they in turn would be serviced. In view of the high activity level that would be expected on the valve surfaces, valve maintenance could only be undertaken remotely (since no means of decontamination is provided), but even this can be done only when the valve is isolated from the reactor. Although a way may be found to surmount this impasse, the other maintenance philosophies based on either remote maintenance or on removal of the fuel from the reactor should also be examined. The former is not discussed further in this review and it may be prohibitively expensive. The latter proposal is probably more attractive and would eliminate the valves entirely (although one valve with much greater tolerable leakage may be required in each loop, but only to prevent backflow through one "down" blower when the others are operative). In order to effect maintenance in such a system the fuel would be dumped into a cooled container and the system then evacuated

of contaminated gases before being subjected to the required maintenance. It must be appreciated that unless a means is provided to reload partially consumed fuel the cost of dumping this fuel must be weighed against the cost and servicing difficulties of the valves. However, since the fuel lifetime is $\sim$ 100 days (and the blanket $\sim$ 1000 days) such an approach may be feasible in a system where shutdowns for maintenance would be expected to occur less than once a year. If it is not economic to dump the fuel then the S&P 1963 approach may be the only out, but it should still be remembered that valves could only be serviced after removing the fuel from the reactor.

The fission products generated in a 350-Mw reactor operating for a period of six months represent an activity level of about 10^9 curies. Assuming that the primary system leakage is a negligible amount, these 10^9 curies will be located in the following places:

a. Fuel balls.

b. Fission product trap in the steam generator.

c. Filter and charcoal beds.

d. Surfaces of primary loop piping and equipment.

The actual amount of activity and relative importance of the activity at any of these locations will vary with each individual nuclide and will be determined by:

a. The half-life of the daughter nuclide and in some cases the half-life of the precursor nuclides.

b. The γ energy of the daughter nuclide.

c. The yield of the nuclide in the fission process.

d. The chemical and/or physical properties of the nuclide and/or its precursors (e.g., boiling point as it is assumed to control the release of activity from the fuel, affinity for metal surfaces, etc.).

The amount of activity and its location in the primary loop is of importance for three basic considerations; 1) the shielding requirements due to fission products during operation, 2) the contamination or hazard associated with fission products leaking from the primary system with the

helium (and tritium), and 3) the level of activity in the primary system due to fission product buildup and its influence on the methods of maintenance and/or decontamination.

The nuclides of importance for maintenance and for shielding calculations include all the nuclides emitting γ photons of high energy. However, for the case of maintenance only those nuclides with half-lives greater than 30 minutes are important since the other nuclides can be outwaited. An investigation of the hazards associated with leakage of helium along with the tritium and fission products would include an investigation of both the β- and γ-emitting nuclides of all energies.

6.1 Release of Activity

In the S&P 1963 analysis of the activity to be expected in the primary system one of the factors limiting the amount of activity in the gas stream was the assumption that only those nuclides with boiling points below 2500°F were able to escape from the fuel element. This assumption is not valid since any daughter of such a nuclide would also be expected to exist in the coolant regardless of the boiling point of the daughter. It would be more nearly correct to assume that nuclides with precursor elements having boiling points below 2500°F as well as nuclides with boiling points below 2500°F would be able to escape from the fuel elements. The correlation of boiling point with escape from the UO_2 is real as demonstrated by tests of the release of fission products from UO_2 in fuel capsules with defective cladding.[8] Such an assumption is, however, an oversimplification as the fraction of any nuclide escaping would increase continuously with increasing temperature and not with the discontinuity imposed by the assumption of the boiling temperature as a "cut-in" point. The resulting error would be most significant for nuclides whose boiling points were just above 2500°F. However, since most of the nuclides with high boiling points are daughters of nuclides which escape from the fuel the error is greatly minimized.

Two other factors make the use of the boiling point as the release criterion conservative. The fact that the fuel temperature is much less than 2500°F in a radial direction from the center of the sphere so that

(8) Irradiation Effects of UO_2, WAPD-183.

the greatest fraction of fission products is in this lower temperature volume, the amount of radioactive nuclides released is greatly reduced. The other factor tending to reduce the escape of fission products is the effect of the system pressure. With the system pressure of 1000 psi, the boiling points of all the nuclides will be effectively increased.

In order to obtain quantitative estimates of the activity in the gas loop we undertook an analysis of the nuclides present. The important fission product nuclides in the primary loop are listed in Table A. These nuclides have been selected as the dominant contributors to the shutdown dose external to the equipment based on the following:

a. Half-life greater than 30 minutes.

b. Fission yield greater than 0.1%.

c. Gamma energy greater than 0.5 Mev.

d. Volatilization or boiling point of the nuclides or any of their precursors less than 2500°F. (The half-life of the volatile nuclide must exceed one second.)

As may be seen from Table A some of these nuclides have boiling points greater than 2500°F. As previously noted it is possible to have these nuclides in the gas stream since the precursor nuclides in the chain have half-lives of sufficient length and low boiling points allowing them to escape from the fuel in the gaseous state.

6.2 Distribution of Activity

The actual amount of curies of the given nuclides in a particular place in the primary loop will depend upon several factors which are either not yet established in the basic design or are not known experimentally at this time. These include:

a. The rate at which the copper balls of the fission product trap are changed in the steam generator in order to collect decay products.

b. The flow rate and amount of charcoal in the bypass stream.

c. The actual method of impregnating the fuel balls, the degree to which fission products may be retained by applying a coating to the balls.

d. The deposition rates of particular nuclides on graphite, stainless steel or carbon steel surfaces. The effect of temperature, velocity

and radiation on deposition rates of atomic-sized particles in a gas stream.

e. The effectiveness of filters in retaining atomic-sized particles in a high-velocity, high-temperature gas stream.

Since it is important to estimate the activity in the loop after shutdown the activity for the various nuclides in Table A is shown for 1, 5, 10, and 100 hr after shutdown. In the last two columns are listed the activities remaining in the primary system after the helium has been removed to the storage tanks. In this case it is assumed that the gaseous fission products (Xe, Kr, and I) are removed with the helium. Since the number of curies is not as important as the number of photons of a given energy given off in the decay process, Table B lists the photons/sec of the various nuclides for the cases of 10 and 100 hr after shutdown assuming the gaseous products have been removed. From the values listed in Table B it is possible to list the number of photons/sec of similar energies (see Table C). From Table C, then, it may be seen that for shutdown times of 10-100 hr the level of activity is not decreasing at a rate fast enough to outwait the activity in the loop. It may also be seen that the doses adjacent to the equipment will be determined by the single nuclide La-140. This nuclide was not included in the S&P list because its boiling point is greater than 2500°F.

As was indicated previously the activity released from the reactor will be located in three places other than the fuel; 1) the fission product trap in the steam generator, 2) the filter and charcoal bed in the bypass stream, or 3) on the primary loop surfaces. The actual amount of activity in each of these places will have to be determined on the basis of experiments but it is important to examine the processes for removal of the active nuclides as now proposed.

6.2.1 Fission Product Trap

The copper-ball trap in the steam generator will be effective in removing the iodine (and bromime) from the gas. As these nuclides decay to the noble gases xenon and krypton, these gases will get into the coolant stream unless the copper is removed from the system before their iodine and bromine precursors decay. In this connection it should be noted that all of the iodine

nuclides except I^{131} have half-lives of less than one day so that the effectiveness of removing the iodine and therefore its daughters will depend upon the rate of change of the filter bed. On the other hand, since all of the nuclides of iodine decay to Xe and since both iodine and Xe may be removed by removing the helium by evacuating the system, the real value of removing iodine is not apparent.

6.2.2 Filter and Charcoal Beds

The charcoal beds will be effective in holding up Xe and Kr from the gas stream but will only be effective for reducing the amount of Xe and Kr of the longer lived nuclides of Xe and Kr. For the Kr chains 90-97 the bypass stream will be ineffective since these Kr nuclides have half-lives of only a few seconds. The same would be true for the Xe nuclides in the chains 139-144 and for some of the other Xe nuclides of longer half-life. This bypass would presumably remove all nongaseous nuclides but to be effective the flow rate to the cleanup system will have to be great enough for this removal process to be competitive with both of the other removal processes -- radioactive decay and deposition. Although the rate constants for deposition are not known it is readily conceivable that they may be so high that no bypass cleanup system could be effective.

6.2.3 Primary Loop Surfaces

If the deposition rate of nuclides which are not in the gaseous phase is greater than their removal rate by the cleanup system the nuclides will deposit on the surfaces or collect as dust in stagnant areas. Since the most important nuclide from the standpoint of doing maintenance is La^{140}, methods should be investigated for removing this nuclide. There is no experimental evidence to indicate that La^{140} or the other nuclides will not have a high deposition rate and it must therefore be assumed that these nuclides will be distributed uniformly in the primary loop surfaces. (The effect of dead legs, temperature, or radiation on the deposition of these nuclides is unknown; it is quite possible that most of the activity may be collected in one place which would create greater doses during maintenance.) With uniform deposition it is possible that the dose level at the surface of the steam generator might be of the order of 10^3 r/hr prior to any decontamination. It would be expected that if deposition or collection of radioactive

dust were the primary method for removing the nuclides from the gas stream, then the steam generator would have the greatest amount of activity since it has the greatest surface area. The amount of activity in the blowers would be of importance for maintenance procedures. The collection of activity in this equipment would be dependent upon the design of the blowers. Those parts of the blower through which gas did not sweep at loop velocities would tend to collect dust. Furthermore, since the blowers are in the cold part of the loop, if there is a temperature effect on deposition it would tend to deposit more activity on the cold parts of the loop, increasing the activity per unit area in this equipment.

6.3 Steam Generator Maintenance

Sanderson and Porter propose that a defective steam generator be isolated by means of butterfly valves, pumped down with the helium transfer equipment, and the steam blown off at a rate controlled by a pressure-regulating valve so that its pressure will always be above that of the helium in the steam generators. It is our belief that a large amount of development is necessary to insure butterfly valves capable of containing high-pressure helium in the reactor while the steam generator is evacuated. On the other hand, employing known technology, the whole system could be pumped down slowly, the afterheat being removed by the two remaining compressors, the third being blocked by a nonleak-tight valve to prevent a large amount of backflow through the inoperable loop. When the decay heat level becomes low enough fuel could then be dumped, the system pumped down to slightly below atmospheric pressure to prevent contaminated helium from flowing outwards, the steam header cut open, and the leak repaired.

The principal failure to be expected with the steam generator is a leaking tube. In both steam generator designs it is proposed to have the headers removed from the steam generator so that shielding may be placed in between. In addition it would be desirable to offset the headers so that there would be no direct radiation streaming from the steam generator. The leaking tube may then be located by opening the header and testing each tube and the defective tube may be isolated either by plugging within the header or by "pinching" outside the header.

Any other failure associated with the steam generator would be presumed to require the replacement of the generator since the dose level therefrom would be excessive and there are no provisions to reduce same. Decontamination does not appear practical because of the size of the vessel, the many closely spaced tubes, the hot gas baffle, internal insulation, and materials of construction. Although the probability of such other failures is remote, if it were not possible to remove the generator the failure could force the abandonment of at least a portion of the plant. This cost must then be weighed against the cost of removal provisions and the probability of such a contingency. If abandonment of that portion of the plant is economically unsound it would be necessary to have provisions for cutting the steam generator from the piping and removing it from the facility. This would require a crane of adequate capacity and headroom as well as some provision for egress through the containment wall. The dose at the surface of the steam generator would not permit approach to the steam generator (see section on activity).

6.4 Compressor Maintenance

Although it is not clear from the S&P report what activity level is expected in the blower, it is quite within the realm of probability that this activity level will be so high as to require substantial shielding. It is not known how tightly the activity therein would be attached to the system surfaces. Experimental information must be obtained to clarify each of these points but it is not unlikely that there could be sufficient activity attached to the surface in such a fashion as to require the highest decontamination factor available in order to permit direct maintenance of the blower. Repeated flushing with various alkaline and acid reagents would be required and this in turn would indicate that the decontaminated surfaces should be of stainless steel although it is possible that the mild steel might suffice for this service. In any event the blower and associated piping would have to be designed so as 1) to prevent trapping of pools of the reagent solutions, 2) to permit spraying of all surfaces (built-in spray nozzles), and 3) to be provided with plumbing to the various decontaminating solutions which after use must be held in "hot storage areas"

for ultimate disposition. In addition the removal of the decontaminating solution is of concern as well as the problem of preventing the decontaminating reagents from entering the seal oil system.

The suggested prewhirl vanes may be particularly troublesome since they will require linkages or gear mechanism within the gas system. Some type of external actuator will be required which would be sealed by a bellows. The entire mechanism would be virtually impossible to decontaminate and would also be an undesirable trap for decontamination fluid used in other regions of the compressor. Maintenance of this feature alone is therefore a trouble spot and the situation is aggravated by the fact that it is one of the more likely parts to require servicing.

6.5 Gas Valve Maintenance

The advisability of including isolation valves in the gas loop is questionable since even if the valves were to perform initially as indicated in the design they would constitute another source of potential trouble in the loop and provisions for their maintenance would be needed. This may involve decontamination provisions similar to that provided for the blower (see above) but in this instance there would be no way to isolate the rest of the system from the decontaminated area. Accordingly it appears more realistic to simplify the contaminated portion of the system by the elimination of the valves and to shut down the entire plant whenever maintenance on any portion of the contaminated system is required. Without the valves in the lines the blower would have to be located so that the decontaminating solutions would be restricted to the portion of the system being decontaminated.

6.6 Fuel-Loading and -Unloading System Maintenance

Few details were given on the fuel-handling system but it is apparent that it must always be operable and there must be provision for maintenance service. The loading system should present little difficulty as it should be clean and outside the principal reactor shielding. However, the charge tubes which permit the fuel and blanket balls to be charged into the reactor need to be adequately shielded from the core in order to minimize the activation of the metal or residual activity streaming from the reactor core. Since the gas in the reactor would otherwise contaminate the area above the shutoff

valve in the charge tubes, a buffer of clean gas should be used in this operation. This clean gas would also minimize contamination of the reactor system with atmospheric air. The consequences of buffer gas failure would have to be considered. The development of valves for these tubes would be fairly simple if tight shutoff were not required. However, the problem of how to replace one of these valves should it fail must be considered. Since the valves or piping would probably be highly contaminated this would have to be done remotely.

An area of hazard in all fluid or drainable systems is associated with the means for removing or draining the fuel. The pebble bed design is dependent upon the reliability of the single drain valve. If this valve were to stick there would be no way to remove the fuel and one would be faced with the most unattractive prospect of having to service a valve which is holding back $\sim 10^9$ curies of activity. An obvious solution is to provide another means of removing the fuel. In most reactor systems where drainage is required at least two means are provided. Then, if the valve fails, the system can still be drained and repairs made. Without this provision, if a valve fails, it cannot be removed for repair for to do this might spill the fuel charge. In any case, to remove a highly activated drain valve which must be under a reactor for gravity flow is an extremely difficult design problem. The high activity level may require that this be done remotely although it is conceivable that given an alternative means of removing the fuel, the valve may be provided with means of decontamination so that it may be serviced in place.

6.7 Maintenance of Fission Product Traps

In various locations in the containment vessel will be carbon traps, chemical traps, and filters which are used to remove various fission products from the helium stream. Although the copper-ball filter at the inlet of each steam generator is of questionable value (see section on activity), if such a filter is really effective it would be desirable to divorce this filter from the steam generator. With the filter located in a separately shielded compartment its failure would not require that the steam generator be discarded, as would otherwise be the case since there is no means of gaining access to the filter. The other filters would have to be located

so that they are adequately shielded and at the same time capable of being replaced and/or serviced. It is possible that these units may be so hot as to require remote removal and perhaps even remotely placing the new unit due to activity on pipes, etc. If casks are used to contain these units during removal, they will be heavy and cumbersome; thus, good access and heavy handling equipment are required. If fission products can be released from the cooled carbon traps the units must be plugged during handling. One difficult development and design problem where cold traps are used is the danger resulting from loss of coolant. This could lead to the release of a large amount of activity. Another problem associated with charcoal traps is the danger of plugging them should moisture get into the helium stream. When these traps are removed, a storage or burial area will be required for these and other items of equipment.

6.8 Control Rod Drive Maintenance

In order to conduct maintenance operations on the control rod drive a valve should be provided between the main system and the atmosphere with some provision for maintaining a gas buffer. This will be some protection in the event the valve leaks. If the valve should leak, it may also be necessary to provide a bleed from the reactor system to insure gas flow in the correct direction.

6.9 Maintenance of Gas Piping, Expansion Joints, etc.

The maintenance of all other portions of the primary system not mentioned in the preceding sections (exclusive of the reactor) that may require repair should be considered.

6.10 Reactor Maintenance

No provision is made to service the reactor -- nor does it seem feasible. In the event of the failure of the reactor vessel provision should be made to remove it or abandon the reactor.

TABLE A

Activities in the Primary System during Operation and for Various Times after Shutdown

Element	Mass No.	Fission Yield	Activity in Curies after 6 Months Operation at 350 Mw x 10^{-6}	Max Gamma Energy	Half-Life t 1/2	Fraction of Activity at 1 hr	Activity 1 hr after Shutdown Curies x 10^{-6}	Fraction of Activity at 5 hr
Se	83	0.21	0.62	0.950	25m	0.19	0.1178	2.5×10^{-4}
Br	84	0.872	2.55	1.89	30m	0.26	0.663	1×10^{-3}
	87	2.7	7.92	5.4	55.6s	10^{-5}		
Kr	87	2.7	7.92	2.3	78m	0.60	4.752	0.07
Rb	88	3.78	11.1	2.8	17.8m	0.87	9.657	0.32
Sr	91	6.24	18.3	1.413	9.7h	0.96	17.57	0.70
Y	90	6.1	0.254	1.4	61h	1.0	0.254	1.0
Y	91m	2.5	7.32	0.551	51m	0.98	7.17	0.76
	91	3.74	9.02	1.4	61d	1.0	9.02	1.0
	92	6.45	18.9	0.6	3.6h	0.98	18.52	0.60
	93	6.87	20.2	0.7	10.0h	0.92	18.58	0.68
	94	6.85	20.1	1.4	16.5m	9×10^{-2}	1.81	10^{-5}
Te	129	1.0	2.93	0.8	72m	0.56	1.64	0.38
	131	2.97	8.7	0.7	24.8m	0.30	2.61	0.05
	133	5.6	16.4	1.0	2m	0.43	7.05	0.03
I	131	2.97	8.7	0.722	8.14d	1.0	8.7	0.98
	132	4.45	13.0	2.0	2.4h	1.0	13.0	0.97
	133	6.62	19.4	1.4	20.8h	1.0	19.4	0.90
	135	5.6	16.4	1.8	6.68h	0.95	15.58	0.58
	136	3.1	9.09	2.9	86s	10^{-5}		
Xe	135m	1.7	4.98	0.52	15.6m	0.95	4.73	0.6
Cs	138	6.12	14.3	1.44	32m	0.38	5.434	4×10^{-3}
Ba	137m	5.69	0.17	0.661	2.60m	1.0	0.17	1.0
	139	6.3	18.5	1.05	85m	0.68	12.58	0.1
	140	6.07	17.8	0.54	12.8d	1.0	17.8	1.0
La	140	6.07	17.8	3.0	40.0h	1.0	17.8	1.0
	141	4.6	13.5	1.5	3.7h	0.9	12.15	0.42
I	134	7.81	22.8	1.78	52.5m	0.74	16.87	5×10^{-2}
Totals			3.29×10^{8}				2.43×10^{8}	

TABLE A (continued)

Activities in the Primary System during Operation and for Various Times after Shutdown

Activity 5 hr after Shutdown Curies x 10^{-6}	Fraction of Activity at 10 hr	Activity 10 hr after Shutdown Curies x 10^{-6}	Fraction of Activity 100 hr after Shutdown	Fraction of Activity 100 hr after Shutdown Curies x 10^{-6}	Activity 10 hr after Shutdown with Gas Removed Curies x 10^{-6}	Activity 100 hr after Shutdown with System Gas Removed Curies x 10^{-6}
	10^{-5}					
	10^{-5}					
0.554	$5x10^{-3}$		10^{-5}			
3.55	$9.6x10^{-2}$		10^{-5}			
12.81	0.5	9.15	$8.5x10^{-4}$		9.15	
0.254	1.0	0.254	1.0	0.254	0.254	0.254
5.56	0.55	4.03	$9x10^{-4}$		4.03	
9.02	1.0	9.02	0.96	8.66	9.02	8.66
11.34	0.29	5.48	10^{-5}		5.48	
13.74	0.51	10.30	10^{-3}		10.3	
1.11	0.34	0.996	0.31	0.908	0.996	0.908
0.435	0.02	0.174			0.174	
0.492	0.061	0.016			0.016	
8.526	0.96	8.352	0.70	6.09		
12.61	0.93	12.09	0.42	5.46		
17.46	0.75	14.55	0.04	0.776		
9.512	0.36	5.904	$3x10^{-5}$			
2.988	0.36	1.743	$3.3x10^{-5}$			
	$1.5x10^{-5}$		10^{-5}			
0.17	1.0	0.17	1.0	0.17	0.17	0.17
1.85	$8.7x10^{-3}$		10^{-5}			
17.8	0.98	17.44	0.82	14.60	17.44	14.60
17.8	1.0	17.8	0.88	15.66	17.8	15.66
5.67	0.16	2.16	10^{-5}		2.16	
1.14	$1x10^{-3}$		10^{-5}			
$1.55x10^{8}$		$1.20x10^{8}$		$5.26x10^{7}$	$7.93x10^{7}$	$4x10^{7}$

TABLE B

Nuclide	Activity in Loop with Gas Removed (curies x 10^{-6}) 10 hr	100 hr	Gamma Energy (Mev)	%	(Photons/sec) x 10^{-16} 10 hours	100 hours
Sr^{91}	9.15		1.413	(7%)	2.37	
			1.025	(33%)	11.17	
			0.747	(7%)	2.37	
			0.66	(22%)	7.45	
			0.64	(33%)	11.17	
Y^{91m}	4.03		0.55	(100%)	14.9	
Y^{91}	9.02	8.66	1.22	(0.3%)	0.1	0.096
Y^{92}	5.48		0.6	(100*)	20.28	
Y^{93}	10.3		0.7	(100%)	38.11	
Te^{129}	0.996	0.908	0.8	(100%)	3.69	3.36
Te^{131}	0.174		0.7	(45%)	0.29	
Ba^{137m}	0.17	0.17	0.661	(100%)	0.63	
Ba^{140}	17.44	14.60	0.54	(30%)	19.36	16.21
La^{140}	17.8	15.66	3.0	(1%)	0.66	0.58
			2.5	(5.4%)	3.56	3.13
			1.596	(94%)	62.04	54.52
			0.815	(29%)	19.14	16.82
La^{141}	2.16		1.5	(5%)	0.40	
Y^{90}	0.254	0.254	1.4	(0.4%)	0.004	0.004

* Assumed at 100%.

TABLE C

Energy Range (Mev)	Group	Energy (Mev)	Photons/sec (10 hours)	100 hours
0.5 - 0.815	I	0.8	1.37×10^{18}	3.64×10^{17}
1.0 - 1.6	II	1.6	7.61×10^{17}	5.46×10^{17}
2.5	III	2.5	3.56×10^{16}	3.13×10^{16}
3.0	IV	3.0	6.6×10^{15}	5.8×10^{15}

7. Hazards Considerations

The various potential nuclear excursions were not considered in the report and the failure and accident analysis was somewhat incomplete. However, there is no apparent reason why the control system cannot be designed to compensate adequately for the various potential excursions. With regard to the potential accidents and failures the principal concern is for the reliability of the decay heat removal provisions. In view of the large quantities of activity released to the gas coolant all portions of the plant associated with the coolant are necessarily enclosed in a containment vessel. Again, there do not appear to be any fundamental issues which would preclude the realization of either reliable decay heat removal provisions or an adequate containment vessel. In the following review a few details are discussed. It is worth noting that items such as insuring adequate missile protection for the containment vessel will probably increase costs above those estimated in S and P 1963.

7.1 Containment Vessel

The design of the containment vessel to contain the simultaneous release of the helium system and the water from one steam generator is valid. Consideration should be given to the use of a cylindrical vessel with hemispherical heads as well as that indicated.

In the unlikely event that either the reactor vessel or a steam generator were to rupture the escaping gas would expand into the concrete chamber containing the component in question. There exists sufficient energy in the high-pressure coolant system to blast the concrete shielding plugs covering the reactor and steam generator through the containment vessel. The height of rise of the plugs is, of course, a function of the size of the postulated leak. Although the velocity and height of rise of the plug may be decreased by assuming longer time for the gas release it is not obvious from the information in S and P 1963 that the chambers can reasonably be designed to prevent penetration of the containment vessel if the gas leak is large. The mode of failure of pressure vessels of this type will have to be examined in order to ascertain the design of concrete shielding chambers, plugs, and vents that will prevent containment vessel rupture via missiles. Other items of equipment such as steam drums which might fail and violate the containment vessel should also be carefully studied.

Where a containment vessel is designed for personnel access for maintenance, fuel charging, etc., the access hatch which is provided should be of the double-door type such as is used on the Dresden station to insure that one door is always closed and the integrity of the vessel is never violated. The ambient within the containment vessel needs to be controlled for activity level (see subsequent discussion of helium leak-tightness), temperature, humidity, and suitability for human habitation. This is a problem which will affect the quality of the maintenance work done and the hazards associated with it. These design features are complicated by the requirement that the integrity of the containment vessel not be compromised.

7.2 Shield Design

The primary shields consist of 2 ft of water surrounded by 4 ft of concrete. The secondary shield consists of 2-1/2 ft of concrete located outside the containment vessel. Furthermore, it is stated that the shutdown dose (presumably within the containment vessel, but outside the primary shield) is initially 20 mr/hr decaying to 10 mr/hr in 8 hr. While these figures have not been checked they seem within reason in view of the primary shield thickness if it is assumed that all the activity in question is within the primary shield. This is definitely not the case since even if the activity did not settle on the gas surfaces (as seems probable to us), but remained in the copper and/or graphite filters that are provided in the design, these filters could then contain a large fraction of the total fission product activity and would require substantial shielding (as well as provisions for decay heat removal and maintenance and/or replacement). If, as seems probable, the activity were to be distributed in some manner over the exposed gas surfaces the cell would probably not be accessible unless, as would be necessary, the entire gas system were shielded.

It is further noted that the presence of significant amounts of activity in the helium (or upon the surfaces of the helium system) would cast doubt on the adequacy of the secondary shield during operation. In either event the report does not indicate the source strengths that are expected external to the reactor -- presumably because this information is not known.

7.3 Helium Leak-Tightness

The incentive for a leak-tight helium system is even greater here than in the ORNL gas-cooled reactor design[4] even though an extra containment vessel was not provided in the latter design. This is a direct consequence of the large amounts of activity which are in contact with, if not released

to, the circulating coolant stream. The coolant leakage even though within the containment vessel is of concern not only because of the loss of helium but also because of the fission product activity which is transported by it. The activity could plate out within the containment vessel and prevent access therein even if all other sources of activity were acceptable.

The S and P report states that there is no activation of the helium. Apparently tritium formation due to a (n, p) reaction with the isotope He^3 has been ignored. The tritium is a beta emitter and does not present a shielding problem outside the containment vessel, but would be a distinct inhalation hazard upon entering the containment vessel. In order to reduce this hazard and also to prevent the surface contamination by fission products in helium leaking into the containment vessel, this atmosphere should be continuously recirculated and filtered.

7.4 Afterheat Removal

Fission product decay heat is to be removed from the fuel by the helium blowers which are each provided with a small emergency power motor in addition to the main drive motor. Since the geometry of the coolant loop makes it unlikely that natural circulation of the gas would be of much value in removing the afterheat it is essential that at least one of the two motors on one of the three helium compressors be operative to remove the afterheat. The three per cent flow which can be maintained by the small motor on a single compressor can remove the afterheat with a temperature rise of less than 180°F in the hottest part of the core.

Since by its very nature the reactor will contain but a fraction of its total fission products and hence decay heat, the rest of the activity will reside elsewhere. Circulation of the gas through the loop should remove the decay heat from the nuclides whether in the reactor or elsewhere in the loop, but should one loop be isolated to perform maintenance it may be necessary to provide afterheat removal for those portions of the system which tend to concentrate activity, i.e., traps, cold surfaces, etc.

It seems fairly safe to assume that if the steam drum is positioned at a higher level than the heat exchanger enough natural water circulation will occur to remove afterheat. If sufficient water is not available in the boiler and boiler drum to remove the afterheat until power is restored, provisions must be made to operate the feedwater pumps from emergency power.

The afterheat in the fuel will determine the fuel removal cask design and the hazards associated with not providing adequate cooling in the removal mechanism or handling casks. The consequences of a spill in this hazardous operation make it important that the removal area be readily decontaminated and not inaccessible to equipment which might be required during such an emergency.

7.5 General Plant Layout

The exclusion area for the plant would apply as equally over water as land. Thus, where it is presumed that the source of water is a navigable stream or lake the plant should be set back from the river or lake a distance equal to the exclusion radius. This is necessary to protect any fishermen, ship traffic, etc., from being too close in the event of a reactor catastrophe. A setback of this type from the river would increase the cost of installing the cooling water inlet and outlet piping and also increase cooling water pumping cost.

In laying out a plant of this type where there is always the possibility of a need for quick evacuation, the means of transportation or parking areas should be located such that they are in line with the general evacuation direction. This would appear preferable to a location which requires passing the containment vessel while evacuating.

8. Design and Development Problems of the PBR

8.1 Design Problems

It is recognized that S&P 1963 study is a conceptual one and therefore subject to considerable refinement, but it seems important to avoid minimizing design problems in major components. Many of these are common to other gas-cooled reactor concepts. The discussion here is intended to remind the reactor designer of some of these difficulties.

8.1.1 Reactor Core

It is probable that the temperature structure within the reactor pressure vessel will be extremely complex and a thorough stress analysis will be required for the vessel. Thermal expansion of the graphite core and fuel balls could lead to cracking of the core and/or balls. Mounting arrangements for the graphite core structure must also be thoroughly examined. There is no apparent reason to expect any serious difficulty with the proposed arrangement, but until the design has been fully detailed final judgement must be reserved.

8.1.2 Piping System

The principal problems in the piping system have to do with provisions for thermal expansion, internal support of the concentric piping arrangement and the valves. Success of the concentric piping arrangement is contingent upon the designer's ingenuity in providing suitable provisions for thermal expansion and internal support, and particularly in a manner which is conducive to good fabrication practices.

8.1.3 Gas Loop Valves

As noted in the discussions on hazards, maintenance and reactor structures, the problem of obtaining valves which will be tight is a major one. To our knowledge there are no valves with metal seats of the size required in the PBR which would be expected to maintain suitable leak tightness as required in a contaminated system. Perhaps by buffering between the two block valves it will be possible to avoid diffusion of radioactive materials into areas where maintenance is being carried out. The acceptability of this arrangement, however, must be demonstrated before the maintenance premises of the design can be accepted. As an alternative, the economics of a system without valves in which components are serviced remotely or by unloading the fuel should be examined.

8.1.4 Compressors and Seals

The helium compressor deserves particular attention because of the importance of flow stability to this design. S&P 1963 suggests that a conventional two-stage centrifugal compressor could be adapted to this application. Such a premise is not inconceivable. It should be noted, however, that centrifugal compressors are particularly noted for their flat characteristics curves and are thus subject to cross coupling instability[9] when operated in parallel. In addition variable pitch inlet guide vanes for centrifugal compressors are of somewhat limited value since they can produce only small variations in the compressor flow characteristics. Bypass flow or variable speed drives would be necessary if larger flow variations are desired. Further attention to the compressor design problems is therefore very desirable. Perhaps the examination should extend to a study of axial flow compressors as an alternative because of their superior performance characteristics in this application.

The design of shaft seals for the compressors is also of major importance because of the high level contamination in the gas stream. Only very low leakage levels will be tolerable and probably some type of seal will be required to prevent leakage of contaminants from, and oil into, the gas stream. Difficulties with radiation instability of the oil from the seal will need to be investigated in order to establish that the seal itself has the necessary high degree of integrity.

8.1.5 Instrumentation, Controls, and Operation

A thorough analysis of the instruments and controls of this design is required before the system can be intelligently evaluated. It is certain that there will be development problems in connection with the control rods and drives. Need for temperature monitoring also requires study, and though no difficulties with operational instability under load shift are apparent further examination of this problem is desirable. No comprehensive discussion of operational procedures was presented in S&P 1963 and it would be surprising if detailed consideration of startup, routine load changes, shutdown and emergency procedures did not uncover

(9) See ORNL-CF-57-12-24, CF-58-4-80.

design problems.

8.1.6 Steam Generators

The steam gnerator is one of the most critical items within the reactor system. Unless the steam generator is insulated on the outside the heat loss to ambient will be very high. If it is insulated on the outside the problem of maintaining the vessel wall temperature below 650°F requires careful design. In some instances this leads to internal insulation as illustrated in the B&W design, AP-337. The problems associated with keeping insulation out of the gas stream and dry during decontamination are manifold.

An area which should be investigated for a reactor of this type is the question of venting the steam system to the containment vessel. If such venting occurs the consequences to instruments, motors, etc., resulting from high humidity and perhaps droplets should be investigated and may require more expensive equipment.

Aside from technical considerations the basic cost of the steam generator is of great significance and a detailed design and cost analysis for this component is essential to a thorough evaluation of the design.

8.1.7 Helium Storage

If it proves practical to remove contaminants from the helium in the system before pumping into storage no serious problems are likely to exist. However, if the contaminants cannot be removed from the system then consideration should be given to difficulties with the transfer compressors in pumping the contaminated helium to storage. In this latter eventuality consideration also should be given to a hot storage system. In addition, the type of compressors usually used for this application are oil-lubricated and may face problems due to the instability of the oil in the radiation field.

8.1.8 Maintenance Features

Rather extensive discussion of maintenance problems in Section 6 points out many of the problems in maintaining the reactor system. The design of all equipment which is decontaminated by reagent flushes should permit the convenient flushing of all contaminated surfaces and the removal of the flushing reagents. Consideration should also be given to techniques for

removing moisture and other residue, if any, from the decontaminated system. Particular attention should be drawn to the equipment associated with the reactor proper. Maintenance of the fuel unloading valves and the block valves deserve special attention.

An alternative means of removing fuel from the reactor in the event of the failure of the single fuel drain valve as well as means of serving these alternative fuel removal devices should be incorporated in the design.

Should further investigations show that the direct maintenance approach is not feasible then attention should also be directed toward the remote maintenance of important items (e.g., the compressors and compressor seals) since special remote maintenance tools would be required. Control rod drives should also be thoroughly examined to make certain that difficulties with the equipment do not incapacitate the reactor. Finally, the layout of the plant should be prepared in a manner which permits replacement of any piece of equipment which may fail including the steam generator.

The distribution of activity throughout the system and the effectiveness of the proposed activity traps should be re-evaluated. Inasmuch as La-Y-Sr appear to be the most predominant activities after shutdown special considerations should be given to means of removing these specific elements.

8.1.9 Fuel Storage and Handling

Because of the unclad nature of the fuel balls spread of contamination will be a serious problem. Methods of storing and shielding the fuel while removing afterheat should be carefully examined.

8.1.10 Tightness and Cleanliness Requirements

The contaminated nature of the coolant places great emphasis on the tightness of the system. Probably mass spectrometer leak-tightness specifications will be necessary for all parts of the system including the steam generator. The problem of testing the system for the desired 1,000-psi pressure is a serious one. Hydrostatic testing is normal practice for such high-pressure systems, but the final closures in this design cannot be tested using any liquid which would contaminate the reactor vessel. Probably high-pressure gas testing could be used, but fabricators always

have some reservations about applying high-pressure gas tests to equipment without prior hydrostatic testing operations.

Cleanliness in the system is also an extremely important consideration. As noted previously oxide scales are conducive to the formation of CO leading to possible erosion of the graphite fuel and of even more importance, the graphite core structure. Other contaminants might be deposited on the fuel balls and lead to sticking difficulties.

8.1.11 Fuel Reprocessing

An integrated fuel reprocessing plant using graphite balls as the basic fuel should be examined in order to establish a true basis for determining fuel reprocessing costs.

8.1.12 Accidents

An accident analysis should be performed to define the design criteria imposed by these abnormal conditions.

8.1.13 Containment

The system and containment vessel must be designed so as to insure the integrity of the containment vessel in the maximum creditable accident. It would appear that the current design will permit the generation of missiles which could replace the containment vessel. The integrity of the containment vessel must also be examined in view of the many penetrations, each of which must be sealed in the event of the "maximum creditable accident". Cost figures for the containment vessel should also reflect the testing which because of stringent leakage requirements may run more than normally allowed.

8.2 Experimental Program

The experimental program for the PBR if initiated can be divided into two categories; out-of-pile testing and in-pile testing. Suggested studies are discussed below:

8.2.1 Fluid Flow and Heat Transfer

The information available on fluid flow and heat transfer pebble bed reactors as previously noted is extremely sketchy. It is imperative that both in-pile and out-of-pile tests on pebble beds be carried out in order to determine heat transfer and fluid flow characteristics.

8.2.2 Fuel Fabrication

Much attention needs to be given to methods of manufacture of fuel balls. All of the several methods of fabrication suggested in S&P 1963 should be investigated further and correlated with the anticipated mechanical properties of the balls. In-pile tests of the fuel balls under conditions comparable to S&P 1963 specifications should be carried out to determine the radiation effects on the fuel.

8.2.3 Distribution of Contamination

The entire maintenance philosophy of this reactor is dependent upon control of radioactive contamination. Distribution of radiation products should be examined in in-pile loops under design conditions. The effectiveness of filters and absorbed beds for removing contamination should be studied on a laboratory scale and then confirmed by in-pile loop operation.

Specific objectives of the in-pile program would include the following:

1. The escape rate of the fission products as a function of temperature for the type of fuel element proposed.
2. The effect of velocity, temperature, and radiation on the deposition or settling out of the various nuclides of importance.
3. The effectiveness of various filters, precipators or reagents for removing atomic-sized particles with particular regard for the removal of Ba, La, Y and Sr.
4. The deposition rate of the nuclides on the graphite, stainless steel and carbon steel surfaces and the nature of the deposition; i.e., if the nuclides are plated out or collected as dust.

8.2.4 Decontamination

Methods of decontamination of the various surfaces must be developed. Closely related to this is the problem of removing the decontaminating reagent and residue from the system.

8.2.5 Materials Problems

The erosion problems of the graphite core structure should be evaluated with respect to presence of foreign oxides in the reactor (such things as vessel wall scale). The reaction of CO_2 with the graphite and silicon coating in the core should also be investigated.

8.2.6 Physics

There is a need for better cross section data on the important isotopes involved in the neutron economy. More reliable information on the energy dependence of the Pa-233 absorption cross section and the energy dependence of α for U-233 in the epithermal and high-energy regions should be obtained if possible.

9. Cost Analysis

9.1 Capital Costs

Any economic evaluation of the PBR must be based largely on conjecture and there is every reason to hope that the ultimate costs of this reactor-power station would approach the estimate of S&P 1963. We believe, however, that there is a need to establish some basis for comparing the S&P 1963 estimates with other reactor designs. For purposes of discussion, therefore, the following tabulation compares direct costs in S&P 1963 with direct costs for the GCR-2 reactor described in ORNL-2500.

Item	S&P 1963	ORNL-2500
Gross Electrical Output, MWe	139.1	252
Net Electrical Output, MWe	126.8	225

	Est. Cost $/10^3	Cost/kw gross-$	Est. Cost $x10^3	Cost/kw gross-$
Structures and Improvements	2,819	20.3	7,695	30.5
Reactor Plant	8,361	60.2	19,414	77.3
Turbine Plant	6,003	43.5	15,379	61.0
Accessory Electrical Plant	854	6.3	4,091	16.2
Miscellaneous Power Plant Equipment	257	2.1	875	3.5
	18,324	132.4	47,454	188.5

It will be noted that the ORNL-2500 design reports considerably higher costs for the turbine plant, accessory electrical equipment, and miscellaneous power plant equipment. Since both turbine designs have the same speed and approximately equivalent rating, it is not apparent that there should be a great difference in their costs. Similarly, comparing accessory electrical equipment shows the following:

	Electrical Auxiliaries*	Cost per Installed kw
ORNL-2500	27 Mw	$175.00
S&P 1963	12.3 Mw	69.50

(* Cost per installed kw of auxiliary equipment differs from cost per kw of plant capacity in that it is a measure of physical equipment to perform equivalent functions. In power plants they are not particularly sensitive to steam system design.)

There is no obvious reason for these differences. Furthermore, there appears to be no basis for the difference in miscellaneous steam plant equipment costs which are substantially higher in ORNL-2500.

In view of the incompatibility of the two cost estimates, it seems of interest to compare only those parts of the capital costs which should show cost differences. These could be "structures and improvements" and the "reactor plant". The difference in these two items between ORNL-2500 and the S&P 1963 design is $27.3/kw. For comparison purposes the difference is multiplied by the 1.75 factor[11] to add contingency, engineering, overhead, and expense in the manner established by AEC in its review of previous gas-cooled reactor concepts. On this basis, the maximum difference in capital cost between ORNL-2500 and S&P 1963 is $47.8/kw or 0.96 mills/kwh. If, indeed, the value is a real incremental difference, it is a significant one and would most certainly be worth claiming in advanced GCR designs.

Particular emphasis should be placed on the fact that the S&P 1963 is a 125-MWe station compared to the larger 225-MWe unit described in ORNL-2500. Although there is assuredly an adequate market for either capacity system if it can produce power on a competitive cost basis, the smaller

(11) At the recent hearing on gas-cooled reactors before the JCAE, the AEC used the following factors:

Indirect	15% of direct costs	Contingency	25% of direct and indirect and escalation
Escalation	15% of direct and indirect costs	Design	12% of direct and indirect and escalation

Total factor for top charges: approximately 1.75.

system would probably have a broader market than the larger one. To illustrate the point, a city of 50,000 population might be required to utilize 125 MWe of capacity. The 225-MWe station would be suitable only for a city of 100,000 population. In the densely populated areas of Northern Europe or the Eastern United States, the advantages would be trivial but in the more sparsely settled and less developed areas of South America, Africa, Australia, and even Canada and the Western United States, the size factor is a major consideration.

Some of the important components of the S&P 1963 are of interest with respect to their potential effect on the ultimate power cost. Perhaps the most important item to be considered is the possibility that contamination outside the reactor may reach a level which makes direct maintenance unsafe. If direct maintenance were unsafe, then access to the system would require elaborate tooling, particularly for the blowers and motors which are massive machines. A thorough study is necessary to evaluate the cost of this complication but it seems probable that the additional expense would cancel much of the cost advantage which is presently indicated for the PBR design.

A second important component of interest is the steam generator. The design proposed in S&P 1963 is a controlled-circulation boiler with spiral tubes assembled in banks which penetrate the vessel wall. Except for the tube arrangement these units are comparable to the British Calder Hall design, although of smaller size. The S&P 1963 study again points up the inherent advantages of the "once-through" monotube boiler but tentatively recommends the controlled-circulation type with external steam drum pending further design investigation. The importance of establishing the "once-through" boiler concept on a firm footing is thus re-emphasized.

The S&P 1963 design also shows the virtues of capitalizing on large temperature differentials between the coolant exit and steam temperatures as indicated by the comparison of the ORNL-2500 steam generator costs with that of S&P 1963.

	ORNL-2500	S&P 1963
Reactor Coolant Exit Temperature	1000°F	1200°F
Steam Throttle Temperature	950°F	1000°F
Steam Generator Cost ($/kw installed)	21.70	13.30

The higher temperature differentials reduce the steam generator costs in S&P 1963 almost 40% below those in ORNL-2500 when compared on a $/installed kw basis. The important question at issue here is the base or reference cost for the steam generators. (One of the major differences in the cost estimates between the Kaiser-ACF and ORNL-2500 studies of gas-cooled reactor systems was in the cost of steam generators. The Kaiser-ACF estimate[12] was almost double that of ORNL-2500 for equivalent capacity units, yet both estimates were based on responsible sources of information.) In view of the significance of steam generator costs in indirect gas-cooled reactor cycles, a need for a thorough evaluation of these costs is clearly indicated, and nothing less than a completely detailed design and cost analysis is useful in such an evaluation.

9.2 Fuel Cycle Costs

The fuel costs given in S&P 1963 appear to be very attractive. The report indicates a maximum value 1.99 mills/kwh. The cost is, however, contingent on three major factors as follow:

9.2.1 U-233 Availability

Perhaps the most serious question about the S&P 1963 design concerns the availability of U-233 as a fuel. Since the reactor has a breeding ratio less than 1.0 it is dependent on external sources not only for the initial charge but for subsequent makeup of fuel. The availability of breeder reactors is therefore an inherent part of the economy of the S&P 1963 design. Justification for building such a reactor to operate in conjunction with breeders will hinge on a very delicate economic balance in which power can be produced more economically by burning excess fuel from the breeder in the PBR than by using it to initiate new breeder reactors. A negative answer to this question does not invalidate the applicability of the PBR concept but it points up the desirability of evaluating the reactor with U-235 as the fuel. There is insufficient information available to determine the significance of such a change but conceivably the design could still be economically competitive on this basis.

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(12) Kaiser-ACF estimated their 253-MWe gross plant to have a steam generator cost of $11,128,000. Private communication between M. Bender, ORNL, and J. Jackman, Kaiser Engineers.

9.2.2 Fuel Lifetime

As noted in section 3 there is little information on radiation damage for the PBR fuel element. Before fuel costs can be established, the stability of the fuel under radiation for 100 full power days must be demonstrated. The lifetime is equivalent to 367 Mw days/kg of initial fissile material. It is of interest to note that the equivalent value for UO_2 in ORNL-2500 is 370 Mw days/kg of initial fissile material, but because of the difference in fuel bodies there is no basis for evaluating the relative conservatism of the two designs.

9.2.3 Fuel Fabrication Costs

While the basic cost factors for manufacturing fuel balls are acceptable at face value, it is not clear how inspection, rejection rate, and recovery of rejected fuel have been factored into the fuel costs. Manufacturing experience and acceptance standards will have to be established before these considerations can be evaluated.

With respect to evaluation of fuel cycle costs some omissions of lesser significance deserve attention. The first of these is the capital charge against the fabrication of the first core and blanket. It is assumed that amortization will be absorbed in the spent fuel fabrication and reprocessing costs but the initial capital must earn profit, interest, taxes, insurance, etc. Subtracting 2.5% amortization from the 14% rate for other capital leaves an annual rate of 11.5% for this item. If the maximum fuel fabrication cost of the core is $1,474,000, the annual cost will be $170,000 or 0.2 mills/kwh.

The second item deserving examination is the fuel reprocessing "turn-around" charge. The assumption that this charge would be lower for small batches than large ones does not seem to be realistic. Presumably if the "turn-around" charge is prohibitive for a small batch of fuel it would be held in storage until a larger one is collected. The effect which the latter approach has on the S&P 1963 design has been examined for its contribution to fuel cycle costs. At the fuel reprocessing rates presumed for S&P 1963, collection of 3-4 cores before reprocessing seems to be the most economical basis with an 8-day "turn-around" period. The net difference between the annual cost of processing each batch separately and

in groups of 3 or 4 is about $125,000 or 0.14 mills/kwh.

If the U-233 were available for this reactor design and fabrication costs and fuel lifetime can be demonstrated, its competitive position with respect to other power plants seems strong. Recent information on coal-fired plants in the USA indicates that a fuel cost of 2.75 mills/kwh[13] is representative of the more efficient power plants of equivalent size in the USA. The ORNL-2500 design reported fuel cycle costs of 2.92 mills/kwh. The fuel cycle costs for S&P 1963 could thus be substantially higher than estimated without changing its competitive position. The addition of the two corrections mentioned previously would increase the estimated cost to 2.33 mills/kwh leaving a difference of 0.42 mills/kwh between it and competitive coal-fired plants.

(13) Electrical World: 9th Steam Power Plant Survey, October 1957.

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