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Comparative Evaluation of Pebble-Bed and Prismatic Fueled High-Temperature Gas-Cooled Reactors

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DEPARTMENT OF ENERGY

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ORNL-5694
Distribution Category UC-77
Gas-Cooled Reactor Technology

Contract No. W-7405-eng-26

COMPARATIVE EVALUATION OF PEBBLE-BED AND PRISMATIC FUELED HIGH-TEMPERATURE GAS-COOLED REACTORS

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Date Published: January 1981

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FOREWORD AND ACKNOWLEDGMENTS

This report describes a preliminary comparative evaluation of pebble bed and prismatic fuel configurations for high-temperature gas-cooled reactor cores and was prepared for the Gas Cooled Reactor Programs Division (G. A. Newby, Acting Director), U.S. Department of Energy. Reactor outlet coolant temperatures of 750, 850, and 950°C and reactor sizes of 3000 and 1000 MW(t) were considered. The basic studies were carried out over a period of about six months and involved specific relative evaluations, as well as a review of the general information that was available. Because of the time limitations, the evaluations were largely carried out utilizing reference designs in comparing the two reactor concepts. Although not evident on the basis of the understanding developed during this study, design reoptimization relative to some of the specific parameters considered to be important here might influence the comparative results; such reoptimizations were not carried out here.

The evaluation results given here are those determined by Oak Ridge National Laboratory; however, we wish to acknowledge the significant contributions made by other participants in providing information useful to this study. In particular, we wish to acknowledge the assistance of (1) General Atomic Company (E. O. Winkler, coordinator) in the areas of design, thermal hydraulics, fission-product behavior, safety studies, fuel cycle performance, maintenance requirements, and fuel reprocessing technology and costs; (2) General Electric Company (G. R. Pflasterer, coordinator) in the areas of reactor availability, control and design; (3) Gas-Cooled Reactor Associates (D. P. Harmon, coordinator) in providing utility perspectives on reactor maintenance and operations; (4) Management Analysis Company (L. W. Perry, principal) in providing the methodology for estimating the overall cost uncertainties of the two reactor concepts; and (5) the U.S. Department of Energy in providing guidance and comments on study emphasis, report organization, and presentation of information.

Finally, we wish to express our gratitude to Katie Lawhorn of ORNL who, working with the editors, typed numerous iterative drafts of the report, including this final one.

P. R. Kasten

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ABSTRACT

A comparative evaluation has been performed of the High Temperature Gas Cooled Reactor (HTGR) and the Federal Republic of Germany's Pebble Bed Reactor (PBR) for potential commercial applications in the U.S. The evaluation considered two reactor sizes [1000 and 3000 MW(t)] and three process applications (steam cycle, direct cycle, and process heat, with outlet coolant temperatures of 750, 850, and 950°C, respectively). The primary criterion for the comparison was the levelized (15-year) cost of producing electricity or process heat. Emphasis was placed on the cost impact of differences between the prismatic-type HTGR core, which requires periodic refuelings during reactor shutdowns, and the pebble bed PBR core, which is refueled continuously during reactor operations. Detailed studies of key technical issues using reference HTGR and PBR designs revealed that two cost components contributing to the levelized power costs are higher for the PBR: capital costs and operation and maintenance costs. A third cost component, associated with non-availability penalties, tended to be higher for the PBR except for the process heat application, for which there is a large uncertainty in the HTGR nonavailability penalty at the 950°C outlet coolant temperature. A fourth cost component, fuel cycle costs, is lower for the PBR, but not sufficiently lower to offset the capital cost component. Thus the HTGR appears to be slightly superior to the PBR in economic performance. Because of the advanced development of the HTGR concept, large HTGRs could also be commercialized in the U.S. with lower R&D costs and shorter lead times than could large PBRs. On the basis of these results, it is recommended that the U.S. gas-cooled thermal reactor program continue giving primary support to the HTGR. At the same time, the U.S. should maintain a cooperative PBR program with FRG, emphasizing work in the key areas of reactor control and instrumentation.

INTRODUCTION

In the United States the gas-cooled thermal reactor program has centered on the High Temperature Gas Cooled Reactor (HTGR) concept developed by the General Atomic Company and typified by the 330-MW(e) prototype Fort St. Vrain Reactor (FSVR), which is operated by the Public Service Company of Colorado near Platteville, Colorado. The HTGR design utilizes a prismatic-type core consisting of hexagon-shaped graphite moderator blocks loaded with fuel rods and arranged in an approximately cylindrical geometry. The fuel rods are packed with fissile UC_2 particles and fertile ThO_2 particles and are positioned in vertical holes in the graphite blocks, with parallel holes providing passageways for helium coolant. The active height of the core is determined by the number of blocks stacked in a column, the columns in turn being grouped into fuel regions consisting of a central column surrounded by six columns. Unfueled graphite blocks placed around the core comprise the side reflector.

The FSVR went on line in 1976 as only the second helium-cooled commercial reactor in the U.S., the first being Peach Bottom Unit 1, a small [40-MW(e)] prototype plant operated by the Philadelphia Electric Company between 1967 and 1974. Each FSVR fuel block is 79 cm high with 36-cm flats and 20.8-cm faces, six blocks comprising a fuel column. The overall height of the active core is 4.75 m and its effective diameter is 6 m. The uranium in the fissile particles is 93% ^{235}U-enriched, and the helium coolant passing downward through the core enters steam generator modules at a temperature of 770°C.

As designed, three graphite dowels served to align the individual fuel blocks and to ensure that the coolant holes in each stack were also aligned. However, as the FSVR initially approached power levels of about 60 to 70% of the design level, it experienced temperature/flow oscillations that have since been attributed to fuel block movements. Core restraint devices (Luci locks) have since been installed to interlock adjacent fuel regions, and the reactor is now operating within a 70% power limit specified by the Nuclear Regulatory Commission. This limit could be removed, however, if present testing shows that the core restraint devices have satisfactorily eliminated the temperature/flow oscillations at the higher power levels. In subsequent HTGR core designs, temperature/flow oscillations should be controllable by decreasing the gap width between fuel blocks, although it will be important that all factor influencing temperature/flow oscillations be carefully considered in specifying fuel block geometries and the associated gaps between blocks.

In spite of the initial shakedown problems of the FSVR, the many advantages of gas-cooled reactors have sustained U.S. interest, and the Department of Energy has continued a program to develop design concepts for large commercial HTGR systems for power production. More recently, DOE has also considered the development of HTGRs for process-heat production with outlet coolant temperatures up to 950°C.

Concurrent with the U.S. effort on the HTGR, the Federal Republic of Germany (FRG) has been developing a helium-cooled thermal reactor identified as the Pebble Bed Reactor (PBR). In the PBR concept the fuel elements are fabricated in the form of 6-cm diameter graphite-encased fuel balls, and the balls pass continuously through the core (into the top and out the bottom) during reactor operation. Cooling is effected by the interstitial flow of helium. The PBR core has a shape that resembles a somewhat flattened cylinder, and, it, like the HTGR core, is surrounded by a graphite reflector.

The on-line refueling feature of the PBR would appear to offer a significant advantage over the HTGR since shutdowns would not be required for refueling. However, operating experience with the reactor has been limited to a 15-MW(e) reactor [the Arbeitsgemeinschaft Versuchsreacktor (AVR)] that went on line in 1967 near Jülich, FRG, and some features of a large PBR remain unproven. Considerably more experience will be gained with a 300-MW(e) reactor now under construction [the Thorium Hochtemperatur Reaktor (THTR)], which is estimated to begin operation about 1983/84. A significant difference between the AVR and the THTR is that the control rods for the AVR operate in the side reflector, whereas in the larger THTR they will also operate above the core and some will penetrate the core. Thus the control system requirements will be much more stringent for the THTR than for the AVR.

Throughout the development of the HTGR and the PBR, the two reactor concepts have been considered as mutual backup systems, and during 1977 U.S. and FRG representatives signed a government-to-government umbrella agreement that covers exchange of gas-cooled reactor technology between the two countries. Late in 1979, the U.S. initiated a technical study to gain more insight as to how the two reactor concepts would compare for potential commercial applications in the U.S. It is that technical study, for which ORNL had the lead responsibility, that is described in this report. Other organizations contributing to the study were General Atomic Company, General Electric Company, Management Analysis Company, and Gas-Cooled Reactor Associates. On the basis of these combined efforts, specific conclusions and recommendations have been obtained. However, it should be acknowledged that this was a "best efforts" type study which was performed primarily over a period of only about six months duration.

DEFINITION AND SCOPE OF THE EVALUATION

The comparative evaluation of the HTGR and PBR was performed for a reference HTGR design developed by General Atomic and a reference PBR design provided to General Electric by FRG. Approximate representations are shown in Figs. 1 and 2. The reactors were compared at 3000 MW(t) and 1000 MW(t) for application to steam cycle systems, gas turbine systems, and process heat systems, the respective outlet coolant temperatures being 750, 850, and 950°C. Both once-through fuel cycles and recycle systems were considered, utilizing both medium enriched uranium (MEU)* and highly enriched uranium (HEU) fuel. Throughout the comparisons, primary emphasis was placed on those areas that were impacted by the choice of the specific core configuration (i.e., the prismatic core versus the pebble bed core).

The reactor pairs were examined from two perspectives: (1) the overall economic performance of the commercialized reactors in producing electricity or process heat and (2) the research and development (R&D) effort and the associated financial investment that would be required to bring them to commercialization. In arriving at the R&D costs, and also the capital costs required for the economic performance evaluation, the HTGR costs

*MEU is defined as uranium of about 20% enrichment in ^{235}U.

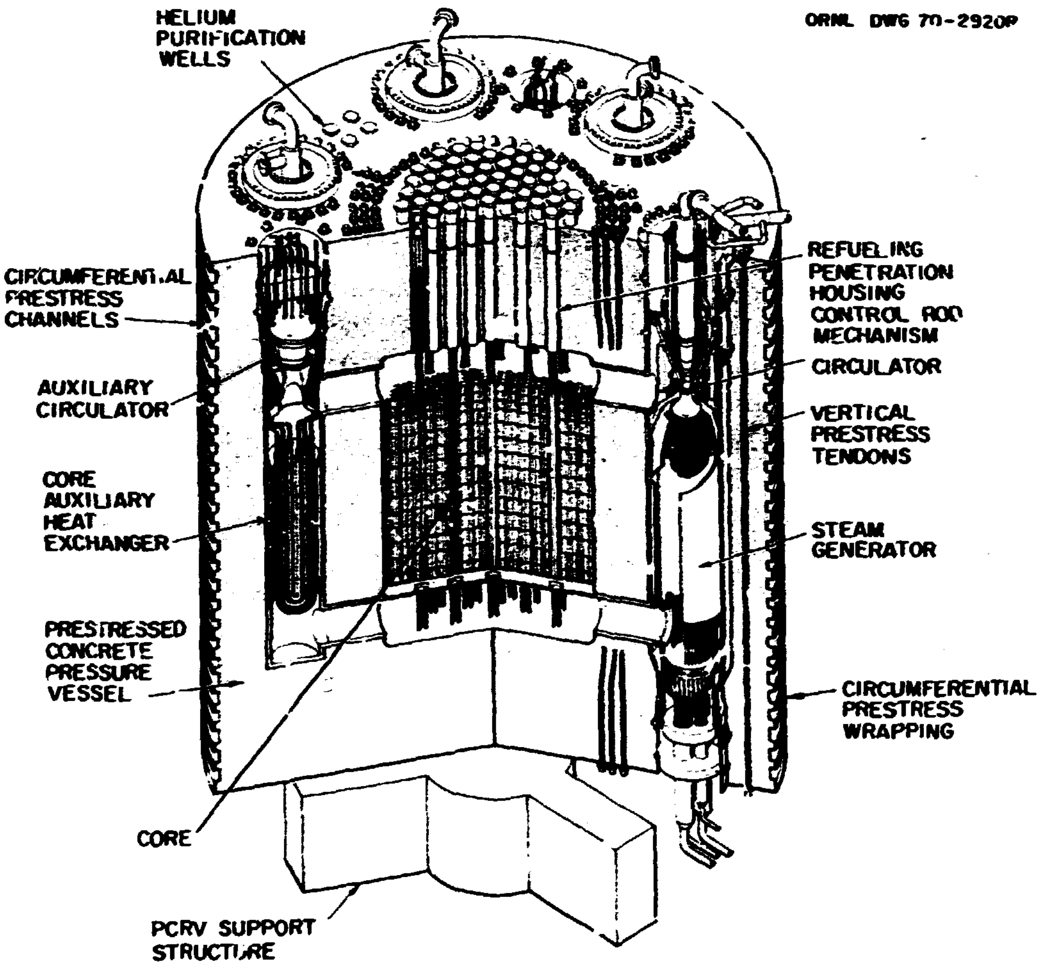

Fig. 1. HTGR with Prismatic Core. (Sketch includes steam cycle system.)

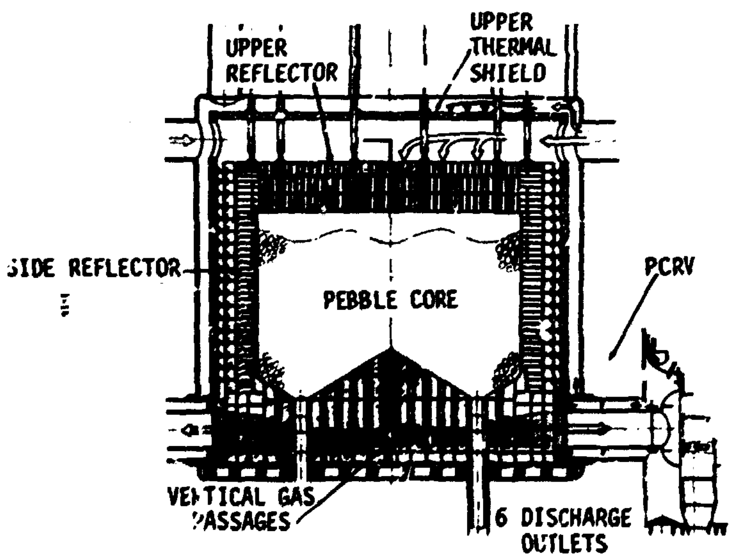

Fig. 2. PBR with Pebble Bed Core.

were first estimated ar ~ differences in expenditures required for the PPR were then projected. in all cases t reference costs were based on available information, since the work scope for this evaluation did not itself provide for detailed cost analyses of the systems.

ANALYSIS METHODS

Capital cost investment estimates for 3000-MW(t) HTGRs applied to all three types of systems were available in 1975 dollars and estimates for a somewhat smaller steam cycle system were available in 1979 dollars. A comparison of the costs for the two steam cycle systems was used as a basis for projecting all the costs for 3000-MW(t) systems to 1979 dollars. These costs were then scaled down for the 1000-MW(t) HTGR systems. In order to arrive at corresponding PBR capital costs, the two types of reactors in the various applications were studied to identify design differences and estimated incremental costs corresponding to those differences were added to the HTGR estimates.

In addition to the capital costs, the economic performance evaluation required estimates of the costs associated with the operation of the systems. Early on, however, it was judged that the uncertainties in the design of and operation of the reactors, as well as in their respective costs, made it inappropriate to compare the reactors on the basis of a single-value or deterministic criterion such as would be arrived at by adding total estimated costs. Hence a methodology was developed to produce probabilistic results that would account for the uncertainties. In addition, the methodology had the capability for treating the differences in design and operation so that uncertainties in the costs for the large components common to both systems would not dominate the results.

In order to determine the differences between the two types of reactor systems, including the differences in capital costs, the first step of the comparative evaluation was to identify and study key technical issues. The results of these studies, which are summarized later in this Executive Summary, indicated that the differences having the greatest impact were (1) the lower power density of the PBR (5.5 W/cm^3 compared with 7.1 W/cm^3 for the HTGR), (2) the more severe reactor control and instrumentation requirements for the PBR, (3) more costly consequences of invoking the secondary shutdown system in the PBR, (4) the requirement for a permanent radial reflector in the PBR but not in the HTGR, (5) the continuous refueling feature of the PBR versus periodic refueling for the HTGR, (6) the slightly better neutronic performance of the PBR, (7) the possibility of higher fission-product releases in the HTGR at high operating temperatures, and (8) differences in reactor availability.

Because of time limitations it was not possible to perform probabilistic analyses considering all these differences for all the combinations of reactor power, application, fuel enrichments, and fuel cycle options. Therefore, the analysis was performed only for 3000-MW(t) reference reactors applied in a gas turbine system and utilizing MEU fuel on a once-through cycle, with some perturbations introduced to consider process heat applications. Conclusions for other systems were deduced from the results and other analyses as discussed below.

CONCLUSIONS

The specific criterion for the economic performance evaluation was the overall energy production costs (in equivalent mills/kW-hr), which consisted of four cost components: capital costs, fuel costs, operation and maintenance costs, and nonavailability penalties. In calculating these costs, the impact of the design and operating differences between the two types of reactors on the individual cost components was determined. The studies of key technical issues showed the impact to be as follows:

(1) The lower power density (i.e., larger core) of the PBR requires a larger PCRV and containment building, which contributes to higher PBR capital costs.

(2) The more complex control system of the PBR, together with the more severe environment in which the control rods must operate, contributes both to higher PBR capital costs and to increased maintenance costs (to replace control rods).

(3) Replacing the PBR control rods requires more time than replacing the HTGR control rods plus refueling the HTGR; however, in most cases the times required for these activities are less than the times required for turbine-generator maintenance. Thus their relative times are important only for those systems for which the turbine maintenance is not the critical path.

(4) The requirement for a permanent PBR radial reflector to avoid a high nonavailability penalty for reflector replacement mandates the development of a superior grade of graphite and increases the PBR capital costs.

(5) The slightly better neutronic performance of the PBR (higher fuel conversion ratio) results in a smaller initial commitment of U_3O_8 and separative work for the PBR, an advantage that becomes more important with fuel recycle.

(6) Invoking the secondary shutdown system results in a significantly larger fuel cycle penalty for the PBR, especially for the once-through cycle.

(7) The HTGR may have a higher cost penalty associated with unscheduled maintenance shutdown than the PBR for process heat systems with high outlet coolant temperatures since for high operating temperatures the fission-product activity levels might be significantly higher in the HTGR. However, the effect of circuit activity level on the maintenance and nonavailability costs is very uncertain.

In the probabilistic analysis for the 3000-MW(t) gas turbine systems (i.e., the reference systems), the expected power cost of the HTGR was calculated to be 19.92 mills/kW-hr, with a probability distribution ranging from 15.69 to 24.30 mills/kW-hr. For the analogous PBR, the expected power cost was 0.66 mill/kW higher. The largest component of this incremental increase was +0.90 mill/kW-hr for increased PBR capital costs, which was partially offset by a -0.42 mill/kW-hr fuel cycle advantage of the PBR over the HTGR. The other components were +0.07 mill/kW-hr for higher PBR operation and maintenance costs and +0.11 mill/kW-hr for a higher nonavailability penalty due to control rod replacement requirements.

The increased PBR capital costs were due to the larger PCRV and containment building (+0.42 mill/kW-hr), the more complicated control system (+0.35 mill/kW-hr), and the assumption that a superior reflector graphite material would have to be developed (+0.17 mill/kW-hr). Substituting a standard grade of graphite for the PBR reflector reduced the capital costs but increased the PBR nonavailability penalty (due to the need to replace graphite) to the extent that the incremental increase in the PBR power cost was 0.97 mill/kW-hr rather than 0.66 mill/kW-hr. *Thus, in both cases the probabilistic analysis indicated that the economic performance of the HTGR may be slightly superior to that of the PBR for the gas turbine application, and the same conclusion would hold for the steam cycle application.*

For process heat systems with outlet coolant temperatures of 950°C, the costs associated with the possible higher fission-product activity in the HTGR coolant (item 7 above) must be balanced against the increased downtime for replacing PBR control rods (item 3 above). This was done cursorily by introducing both effects as perturbations in the probabilistic analysis for the reference systems. As a result, the expected value of the incremental increase in the PBR costs over the HTGR costs was reduced to 0.43 mill/kW-hr. However, the distribution about the expected value had a wide variance, ranging from a PBR advantage to an HTGR advantage. *The net result was no apparent preference between the two reactors for high-temperature process heat systems.*

With respect to the R&D effort required to commercialize the reactors in the U.S., it appears that either concept could be successfully commercialized. However, since the development of HTGR systems has been under way in the U.S. for some time, whereas the development of PBR systems has been carried out primarily in the Federal Republic of Germany, a larger R&D program would be required to bring the PBR on line. The estimated costs (in 1979 dollars) for the HTGR R&D were projected to be $300 million to $400 million for steam-cycle application, $450 million to $800 million for gas-turbine application, and $600 million to $1000 million for high-temperature process heat application. The increase in these costs for the PBR R&D program was estimated to be $100 million to $200 million, and the increased time was estimated to be up to four years (for the steam cycle application). For both the HTGR and the PBR, the R&D costs to develop fuel recycle capability were estimated to be $1400 million to $2100 million.

RECOMMENDATIONS

On the basis of these probabilistic results for the reference systems and the overall evaluation of all the systems considered in this study, *it is recommended that primary support of high-temperature gas-cooled reactors in the U.S. be given to the HTGR concept.* Key issues still to be resolved for the HTGR are (1) fuel performance (i.e., fission-product retention) as a function of temperature, temperature gradient and irradiation exposure, and (2) maintenance costs as a function of coolant circuit activity.

It is also recommended that the U.S. maintain a cooperative PBR program with FRG, emphasizing work in the key areas of reactor control and instrumentation requirements.

STUDIES OF KEY TECHNICAL ISSUES

As has been stated previously, the analyses summarized above were based on data obtained from studies of several key technical issues. These studies are described in detail in Chapter 3 of the report and are summarized below.

Reactor Control and Instrumentation (3.1)*

While the control and instrumentation systems for commercial HTGRs have been developed to the prototypical stage, comparable systems for the large PBRs are only in the conceptual design stage and definitive systems have not yet evolved. Thus the study of this key technical issue centered primarily on the development work still extant for the PBR. Of particular concern is the design and fabrication of the PBR control rods. While the HTGR rods will operate in channels in the fuel blocks and under normal operation will encounter no significant resistance force, considerable force must be applied to insert the PBR rods into the pebble bed. Moreover, the PBR backup shutdown mechanism must be powered, whereas the HTGR rods can be inserted by gravity.

Two types of control rods have been proposed for the PBR: a thrust-type rod designed to be pushed into the pebble bed; and an auger-type rod designed to be screwed into the reactor core. Since the primary drive mechanism for the PBR auger rods must be capable of both translation and rotation, each rod will require its own drive mechanism, whereas in the HTGR one drive will operate two rods. Also, because of the higher stress levels the PBR rods will experience, their material requirements will be more stringent than those of the HTGR rods and their replacement will be more frequent.

The consequences of invoking the secondary shutdown mechanisms for the two reactors in the two reactors will also differ significantly. In both cases small absorber spheres will be released into the core from the top, but in the PBR, removal of the spheres (called KLAK) will require that approximately 15% of the core fuel be removed via the fuel discharge machine, which would impose a high fuel cycle penalty (roughly 40% of the annual requirements on a once-through cycle). By contrast, the HTGR spheres can be removed after depressurization via a vacuum device.

Finally, the HTGR can more easily accommodate in-core instrumentation. The current assumption is that ex-core instrumentation will be adequate for the PBR, but this is still to be verified.

PBR Control Requirements. Primary considerations for normal operation of the PBR indicate that if short-time startup and peak xenon override capabilities are required, the reactivity needed in control absorption is 0.090. Without these capabilities, only 0.027

*Number in parentheses refers to section number in Chapter 3.

would be needed. Also, without short-time startup and override requirements, a once-through system could be controlled with rods operating only in the gas space above the core. But with the startup and override requirements, some of the rods would have to penetrate into the pebble bed. The probability that xenon override will be necessary and tha ntrol rods would be inserted exceeds 50%, with 37% of the primary rods involved [al : 1 rod per 65 MW(t)]. These would be subject to early replacement. It does appear, however, that xenon oscillation control will not be necessary for PBRs up to the 3000-MW(t) size considered here. (HTGR systems of this size would also be stable.)

Control Rod Damage, Replacement Requirements, and Costs. The structural integrity of proposed PBR control rod cladding materials (Alloy 800H, Hastelloy X, and Inconel 625) was evaluated for a range of mechanical loadings (insertion forces up to 147,100 N), temperatures, and neutron irradiations; the effects of the interactions of the rod cladding material with surrounding materials were also considered. The results indicated that under reasonable operating assumptions and a maximum temperature of 750°C, mechanical loading failures of the control rods by plastic yielding should be impossible and that failures by column buckling would be highly improbable; even at 950°C the rod integrity would be maintained. (The forces required for pebble bed insertion are greatly reduced by the injection of ammonia immediately prior to the rod insertion.) With the temperatures and irradiation levels expected (nominally 600°C and 10^{22} neutrons/cm^2), the hardness and ductility characteristics of the cladding material would be degraded, but no methods exist for determining what levels of toughness and ductility are necessary; therefore, conservative levels based on licensing experience would have to be set. Detrimental interactions of the rod cladding with impurities in the helium environment should be inconsequential, even at short-term temperatures of 950°C, but questions about the interactions of cladding material with the B_4C absorber and the fuel spheres are still to be addressed.

The lifetime for HTGR control rods (clad with Alloy 800H) has been estimated as four years for rods that remain in the core, and about twice that long for the average rod. This leads to an average of one-eighth of the HTGR rods being replaced annually. Studies of the more severe conditions for rods inserted in the PBR core indicate that a dual two-year/four-year cycle should be used for the PBR: one-third of the rods (those that penetrate the core) should be replaced every two years and the remaining two-thirds every four years. The most optimistic dual cycle considered is a four-year/ten-year cycle.

GA's estimate of the costs for a full complement of HTGR control rods is $2.3 million, and on a relative basis a full set of PBR rods should cost about $4 million (51 screw type for core penetration and 100 push type). The predicted per-year cost for control rod replacement is $590,000 for the HTGR and $1.3 million for the PBR.

Times Required for Control Rod Replacement. Annual replacement of the HTGR control rods and control drive assemblies would be performed concurrently with annual refueling, and an additional downtime penalty of only 5 hours per year is associated with the control rod replacement alone. Replacement of the PBR control rods would require reactor shutdown specifically for that purpose, access to the control rod drives, and time to replace the rods. Under the assumption of the most optimistic PBR dual replacement cycle (four-year/ten-year), the total PBR downtime incurred for control rod replacement would be 298 hours per year.

Reactor Instrumentation Costs. The costs for instrumenting the HTGR and PBR should be approximately equal if ex-core instrumentation is adequate for the PBR. If in-core instrumentation is required for the PBR, a cost penalty of up to $5 million is estimated.

Licensability of PBR Control Rod Systems. In general, the detailed technical data needed to judge the suitability of the PBR control system are lacking; however, a review of available information has shown that (1) stored energy in the amount of a few tenths of a kW-hr per rod would be sufficient for control rod insertion, (2) the control rods for routine operation and for safety operations could be identical but the "scram" action would have to be separated from the control action (presumably by using differential drives), (3) the design should ensure that no constraints exist to inhibit scram action, (4) the control rod force requirements could and should be met without a dependency on ammonia injection, and (5) potentially costly backup problems associated with testing and/or inadvertently actuating the KLAK backup safety system demand careful examination.

R&D Costs for Control and Instrumentation Systems. Research and development tasks to qualify control rod cladding materials are estimated to be $2.3 million to $10.8 million for the PBR and $0.3 million to $2.0 million for the HTGR. The costs include studies of PBR coolant-cladding interactions (necessary only if a new cladding material is selected), PBR fuel-cladding interactions, and absorber-cladding interactions for both reactor types. They also include in-reactor irradiation experiments for all candidate claddings and thermal history studies of all candidate claddings except Alloy 800H. If a completely new material must be sought for PBR control rod cladding, all costs are unknown.

The control rod drives for both reactor types can probably be based on proven design concepts. Prototype design, proof-testing, qualification, etc. of the drives per se should cost approximately the same for the two reactor systems (on the order of $2 million). Testing and qualification of the reactor control systems and obtaining statistically significant information for U.S. licensing could require extensive model tests, with associated R&D costs of tens of millions of dollars.

Fuel Cycle Analysis (3.2)

In the fuel cycle analysis for the HTGR/PBR evaluation, mass flow rates and fuel cycle costs were compared for 3000-MW(t) reference reactors operating on MEU/Th and HEU/Th throwaway cycles. The reference designs were a scale-down of the 3360-MW(t) conceptual HTGR and the German prototype direct-cycle HHT, both of which have specified fuel zoning patterns and fuel management schemes.

For the reference MEU throwaway fuel cycles considered in this analysis, the PBR has a higher heavy metal throughput and thus has higher costs for fabrication, waste disposal, etc. However, with its need for excess reactivity at beginning-of-cycle, the HTGR has higher ^{235}U requirements and therefore higher separative work and conversion costs. Based on the economic assumptions used here, the HTGR MEU fuel cycle costs, averaged over 30 years, are projected to be 5% higher than those for the PBR.

On the HEU/Th throwaway cycle, lower fuel cycle costs are realized for both reactors because of improved neutron efficiency due to a higher fraction of ^{233}U in the cycle. In this case the PBR heavy metal throughput rate is only slightly higher than that of the HTGR and the ^{235}U requirements only slightly lower. With the economic assumptions used, in which the HEU fabrication unit cost is higher for the HTGR, the HTGR fuel costs again are approximately 5% higher than the PBR costs.

Reducing the PBR power from 3000 MW(t) to 1000 MW(t) leads to a higher neutron leakage from the smaller core and requires approximately 6% more ^{235}U per GW(e) on both the HEU/Th and the MEU/Th cycle. Again the unit fuel cycle costs are projected to be 5% higher for the HTGR.

It had already been estimated in earlier studies that the ore requirements and fuel cycle costs would both be reduced for PBRs and HTGRs using recycled fuel, more so for the HEU/Th cycle than for the MEU/Th cycle. However, consistent data to compare the two reactors are not available. Based on initial estimates, the refueling scheme of the PBR gives the PBR a recycle cost advantage higher than that for the once-through cycles.

Reactor Pressure Vessel and Containment Building Capital Costs (3.3)

The larger core diameter of the PBR relative to the HTGR dictates that the PCRV (prestressed concrete reactor vessel) and containment building for the PBR will have larger diameters than those for a comparable HTGR system. Thus the costs for these structures will be higher for the PBR than for the HTGR. In the absence of detailed designs of these structures for the reactor systems of interest, PBR cost increase estimates were based on PCRV and containment building designs derived for 3000-MW(t) and 1000-MW(t) systems (850°C outlet coolant temperature) from the best available data (ERDA Report 109 and a General Atomic Company memorandum). Two PBR designs were considered: one based directly on data in the GA memorandum (PBR #1) and another in which the diameter of the PBR core cavity was reduced by 3 meters so that the PBR and HTGR would have equivalent core-cavity clearances (PBR #2). Cost estimates were then made for three unit costs of concrete: $185, $300, and $500 per cubic yard. The results indicate that the cost penalties for PBR #1 would range from $18.1 million to $26.5 million at 1000 MW(t) and from $25.1 million to $37 million at 3000 MW(t). The corresponding penalties for PBR #2 would range from $9.2 million to $14.2 million at 1000 MW(t) and from $15.5

million to $23.4 million at 3000 MW(t). These estimates were for gas turbine reactor designs, but the cost differentials for steam cycle and process heat applications should be similar. Also, the differential results should be approximately the same for all three coolant outlet temperatures considered in the PBR/HTGR evaluation (750°C, 850°C, and 950°C). It should be remembered, however, that these cost estimates do not represent cost extremes, since the installed cost of concrete for PCRVs would probably exceed the $500 per cubic yard maximum assumed.

Fuel Fabrication and Recycle Unit Costs (3.4)

In the absence of detailed fuel element and fuel cycle designs for the PBR and HTGR systems considered here, estimates of the unit costs for the fabrication of makeup (or fresh) elements and refabrication of recycle elements were based on data available from earlier studies of selected PBR and HTGR fuel cycles. For each reactor these consisted of a once-through case utilizing $^{235}U(<20\%)/Th$ fuel and a recycle case initiated with $^{235}U(93\%)/Th$ fuel. The fabrication requirements of each cycle were assessed and the cost of a commercial plant to support 20 reactors was estimated. The results were then reduced to process costs per kilogram of heavy metal, assuming that the heavy metal itself was customer supplied. Because less information was available for the PBR cycles than for the HTGR cycles, the uncertainty associated with each cost category was assessed and the estimates were adjusted to obtain a probabilistic range. Within those ranges, the most probable unit costs for makeup fuel fabrication in the once-through cycles were $1170/kg HM for the PBR and $[illegible]80/kg HM for the HTGR; in the recycle cases the makeup fuel costs were $640/kg HM for the PBR and $800/kg HM for the HTGR. The most probable unit costs for recycle fuel refabrication were $2190/kg HM for the PBR and $1740/kg HM for the HTGR, but the mid-life reload weighted averages were closer together, $1270 for the PBR and $1380 for the HTGR. In all cases, these unit costs would vary with the heavy metal loading of the elements, the coated particle design, the plant capacity, etc.

Estimates of the unit costs for fuel reprocessing were based on the same recycle data used to estimate the unit costs for fuel refabrication, again by first projecting the cost of a reprocessing plant to support a 20-GW(e) economy utilizing each reactor type. The resulting unit costs were $639/kg HM for reprocessing the HTGR fuels with TRISO-coated fissile particles and $484/kg HM for reprocessing the PBR fuels with BISO-coated particles. Using the TRISO-coated particles in the PBR fuel would incur about a 4% penalty, increasing the unit costs to $500/kg HM. (Note: These differences reflect the differences in the fuel loadings in the HTGR and PBR. For the same carbon and heavy metal throughput, reprocessing unit costs would be essentially independent of reactor fuel types.)

Transportation costs for spent fuels were estimated to be $870,000 per GW-yr for the HTGR and $1.3 million per GW-yr for the PBR, the higher PBR cost corresponding to a higher volume of fuel requiring transportation.

The technology for HTR fuel fabrication/refabrication and reprocessing is fairly well advanced and much is essentially common to both reactor types. Thus no significant differences exist in the outstanding RD&D required for commercialization.

Impact of Fission-Product Releases (3.5)

The normal and abnormal releases of fission products into a reactor's primary coolant system impact both the design of many system components and the plant operating and maintenance procedures. The relative effects of such releases on the PBR and HTGR were examined in the several studies summarized below.

Coolant Radioactivity During Normal Operation. Under normal operations, the release of fission products into the reactor coolant is directly proportional to the structural failures of the fissile and fertile particles, which, in turn, is a function of the operating temperature and burnup of the fuel. Utilizing fuel particle descriptions and fuel failure models from their 1979 Fuel Design Data Manual, GA calculated the temperature distributions and the percent of fuel failures that could be expected in PBR and HTGR pairs operating at coolant outlet temperatures of 700, 850, and 950°C (representing steam-cycle, gas-turbine, and process-heat applications, respectively). For each pair the calculated HTGR fuel temperatures were consistently higher than the PBR temperatures and peaked several hundred degrees above the outlet temperatures at fuel zone boundaries, where coolant/power mismatches occurred. By contrast, the PBR temperatures never exceeded the outlet temperatures by more than 100°C and exhibited no peaks; however, the absence of peak temperatures may have been due to the fact that an "idealized" PBR model was used in which control-rod actions were not considered.

The fuel failure mechanisms considered were (1) manufacturing defects, (2) gas pressure buildup within the particles, (3) migration of fuel inside the particles, and (4) corrosion of the silicon carbide layer on the particles due to attack by the fission product palladium and chemically similar materials. In all cases, the calculated percentage of fertile particle failures remained low, below 0.12% for the BISO-coated particles in the HTGR-SC and below 0.07% for the TRISO-coated fertile particles in all the other systems. For the 700°C and 850°C systems, the calculated fissile particle failures were also low (all just under 0.10%); however, the corresponding releases of fission products, principally ^{137}Cs and ^{88}Kr, were greater in the HTGRs because their higher core temperatures affect the fission-product retention properties of the graphite. At the 950°C outlet temperature, the HTGR fissile particles suffered significantly greater failures than the PBR particles, the dominant failure mechanism being palladium attack. When the calculations utilized an "Old Pd" attack rate and assumed a 4-yr fuel lifetime, the HTGR 40-yr ^{137}Cs plateout at 950°C was 484 Ci/MW and the circulating ^{88}Kr was 19 Ci/MW; however, with a recently proposed "Rev. Pd" attack rate these values were reduced to 40 and 2.3 Ci/MW respectively. Corresponding numbers for the PBR were 7 Ci/MW for ^{137}Cs plateout and 1.14 Ci/MW for circulating ^{88}Kr (based on "Old Pd" attack rate).

Operational and Public Exposures Due to Normal Coolant Radioactivities. The fission-product doses that could be delivered from the coolant to operations personnel and to the public were calculated for all three PBR/HTGR pairs under the conditions of both an open containment (containment building continuously purged up to 1 volume per hour) and a closed containment (semiannual purge). The results were compared with established ALARA limits* for occupational exposures (100 mrem/week) and public exposures (5 mrem/yr whole-body or 15 mrem/yr to the thyroid by inhalation).

In preliminary calculations using the Fulton HTGR design as a base case, it was determined that the whole-body offsite dose would always be reached before the thyroid dose and that the circulating noble gases comprised the major radionuclide class contributing to exposures of both the public and the operational personnel requiring access to the containment building. ^{88}Kr clearly dominates in all cases except for closed-containment occupational exposure, in which case ^{133}Xe and ^{88}Kr contribute approximately equally. The calculations for the Fulton HTGR also showed that in most cases the inventory of circulating noble gases could be substantially increased without exceeding the maximum permissible dose rates – under the condition of an open containment with a purge rate of 0.5 volume per hour, the noble gas inventory could be 74,000 Ci, of which 14,000 Ci would be ^{88}Kr, without exceeding either the occupational or the offsite dose rates.

The relative importance of ^{88}Kr having been established, the ^{88}Kr circulating inventories and the conservative "Old Pd" attack rate were used as the basis for calculating maximum (design) dose rates for the three PBR/HTGR pairs under normal operation with both open and closed containment conditions. In all cases the PBR dose rates were lower than the HTGR dose rates by factors of 2 to 4. Also, all the PBR offsite dose rates were well below ALARA limits and all the PBR operations personnel dose rates were below ALARA limits except for the PBR-GT and PBR-PH cases under closed-containment conditions. By contrast, the HTGR dose rates were below the offsite limits only for the HTGR-SC case and below the operations personnel limits only for the HTGR-SC case with open containment conditions. Thus for most HTGR cases and for at least two PBR cases, rotation of personnel requiring containment access would be necessary. On the other hand, it should be remembered that the design dose rates calculated are four times higher than expected dose rates and that they were calculated under assumptions that would lead to maximum values. Under expected dose rates, exposures of personnel do not appear to place practical limits on containment access times.

Impact of Coolant Radioactivities on Scheduled Maintenance Activities. In order to determine whether the amount of fission-product activity in the coolant would have a significant impact on regularly scheduled maintenance and inspection activities, 52 such activities for the reference 900-MW(e) HTGR-SC were analyzed for fission-product radioactivity levels of 0.1x, 10x and 100x, where x was the base case level. Of the 52 activities, six (12%) were to be performed inside the containment building during reactor operation, and for these the controlling source of radiation was gasborne activity plus direct radiation from the core. The remaining activities were to be performed at various locations during shutdown, and for these plateout was the dominant source. However, a

*ALARA = as low as reasonably achievable.

large fraction of the activities were to be performed by remote operations so that the level of coolant contamination was irrelevant.

As would be expected, decreasing the radioactivity to 0.1x would have no adverse effect on any of the activities. Increasing the level to 10x or 100x, however, would necessitate that temporary shielding be installed each time for some of the scheduled shutdown activities, adding 30 to 56 hours per year to the time required to perform those activities. For the six activities inside the containment during operation, a 10x level would require personnel rotation or an increased containment vent rate (assumed to be 0.5 volume per hour). At a 100x level, access to the containment would not be practical, and the activities, totalling about 123 hours per year, would have to be performed during shutdown. Thus at 100x, the additional downtime requirements for the regularly scheduled maintenance and inspection activities would total 153 to 179 hours per year, which would be equivalent to a 15% to 18% increase in scheduled downtime. However, many of the tasks could be performed concurrently so that the impact on plant availability due to varying circuit contamination could be small.

Impact of Coolant Radioactivity on Unscheduled Maintenance Costs. Although little information exists on which to predict the costs associated with unscheduled maintenance activities, reactor experience has shown that the associated downtime is proportional to personnel exposure. For this evaluation the ^{137}Cs plateout activities in the coolant were used as the basis for predicting personnel exposures, which were then converted to costs. The procedure was to first generate a family of curves showing PBR and HTGR ^{137}Cs coolant radioactivities versus outlet coolant temperatures for both the "Old Pd" and the "Rev. Pd" attack rates. Next, the curves were converted to exposures by normalizing the "Old Pd" HTGR activity at 850°C to an exposure of 400 person-rems per GW(e)-yr. It was then assumed that the actual exposures would lie between those predicted by the "Old Pd" and the "Rev. Pd" attack rates, leading to an adjustment of the exposure values at selected (high) temperatures where fission-product release by palladium attack becomes important. For the PBR, a further adjustment was made to account for the fact that an "idealized PBR" had been calculated (see above). Finally, it was assumed that 800 person-rems per GW(e)-yr could be accumulated by reactor plant operations personnel without penalty and that higher accumulations would effectively cost $25,000 per person-rem, including downtime costs.

The result of this procedure was that the effective cost of unscheduled maintenance for the PBR was predicted to be zero at all outlet temperatures and that the cost for the HTGR was predicted to be zero at outlet temperatures of 700°C and 850°C; at 950°C, however, unscheduled HTGR maintenance costs were estimated to be $5 million for a three-year fuel life cycle and $50 million for a four-year cycle. But it is to be emphasized that this method for predicting unscheduled maintenance costs is highly uncertain, and the penalty could be zero for the HTGR. For this evaluation, a value of $5 million per year was taken as the mean, which is comparable to the estimated maximum effect of high circuit activity on scheduled maintenance costs.

Relative Risks of Accidents Releasing Fission Products. Insufficient information was available to evaluate the relative safety performance of the PBR and HTGR in a completely quantitative or probabilistic manner; therefore, for this study the risk analysis was limited to an examination of the apparent relative safety of the two systems and identification of the analyses that must precede quantification. Under these limitations, the following conclusions were reached: (1) Enormous safety margins exist for DBDAs (design basis depressurization accidents), but should they occur, the consequences would be 2 to ~ times lower for the PBR because of a lower circulating noble gas inventory (based on the "Old Pd" corrosion relation). (2) Earthquake safety would be governed by the structural reliability of the HTGR fuel blocks and the PBR top cover reflector; relative reliabilities are considered comparable because of low failure probabilities of both concepts. (3) A postulated drop of HTGR or PBR spent fuel during unloading or transfer would not contribute significantly to the overall safety envelope. (4) The risk of water ingress in the systems and subsequent release of fission products to the atmosphere is similar for PBRs and HTGRs except at the 950°C outlet coolant temperature, where the PBR release would be five times smaller due to lower failed fuel fractions. (5) Core heat-up initiation events and the important subsequent consequences would be similar for PBRs and HTGRs. In all cases, no significant safety differences were identified.

Heavy Metal Loadings in PBR and HTGR Cores (3.6)

The economic performance of PBR or HTGR recycled fuel can be influenced by the conversion ratios that can be attained, which, in turn, depend directly on the amount of heavy metal that can be loaded into a fuel element. Thus, one of the studies in this comparative assessment was directed at predicting the maximum fuel element loadings attainable in the two types of reactors and determining the associated development required.

PBR Loading Capabilities. The PBR fuel design currently favored by FRG features a spherical element with a central core of overcoated HEU fuel particles surrounded by a graphite layer. Three particle variants are being considered: two that are one-particle designs utilizing $(Th,U)O_2$ kernels coated with HTI-BISO and LTI-TRISO, respectively (Variants 1 and 2); and one that is a TRISO-coated two-particle feed/breed design utilizing a UC_2 kernel in the fueled particle and a ThO_2 kernel in the fertile particle (Variant 3).

In current FRG fabrication techniques the overcoated particles and a matrix material are isostatically compressed in a rubber mold, a fuel-free graphite outer layer is added, and the sphere is cold-pressed to high density. The heavy metal loading is determined by the relative quantities of overcoated particles and matrix material in the central core of the element. The loading limits are influenced by the overcoating thickness, the allowable particle reject fraction (currently zero), and the required crushing strength of the fabricated fuel element (now $\geq$ 22 kN). FRG experience indicates that under current criteria and with current technology, the heavy metal loading limits are 20 g/sphere for Variants 1 and 2 and 15 g/sphere for Variant 3; however, they feel that with several years of development they could increase the loadings to 27, 23, and 18 g/sphere, respectively. At these loadings, the overcoated particles would comprise 65 vol.% of the fueled matrix, which FRG considers to be the maximum attainable with the cold-pressing fabrication technique.

The heavy metal loadings of PBR fuel elements could be increased by employing a U.S. HTGR feed/breed particle design using uranium oxide/uranium carbide (UCO) for the fuel particle and ThO_2 for the fertile particle (both TRISO-coated). ORNL calculations performed in this study for 65 vol.% particles in the fueled matrix indicate that loadings up to 25 to 28 g/sphere in the Th/U range of interest could be achieved. This is as high as the loadings predicted by FRG (and also by ORNL) for the FRG one-particle designs and considerably higher than those for the FRG two-particle design.

The long-range hopes of FRG are to develop a satisfactory hot-pressing fabrication method that would avoid the use of overcoatings and allow the 30- to 35-g/sphere heavy metal loadings that will be required for breeding and near-breeding systems. Under current technology, a PBR (with recycle) is limited to a conversion ratio of about 0.71 at a burnup of ~100 MW(t)-d/kg HM (assumed to be the most economic exposure). On the other hand, with the on-line refueling capability of the PBR, lower burnups might be economically feasible with concomitant increases in the conversion ratio.

<u>HTGR Loading Capabilities</u>. The HTGR core design currently being developed in the U.S. utilizes the feed/breed particle concept consisting of TRISO-coated UCO fuel particles and BISO- or TRISO-coated ThO_2 fertile particles packed into fuel rods that are loaded into 10-row or 8-row fuel blocks. The steam-cycle HTGR designs, which are the only systems for which much fuel cycle data are available, use the BISO-coated fertile particles. The TRISO-coated fertile particles are being developed for the higher fission-product retention thought to be required for gas turbine and process heat systems.

In this study calculations were performed by ORNL for three HTGR-SC cases to establish the volume percent of the 8-row and 10-row fuel blocks that would be needed for particles if the specified initial core and reload heavy metal loadings were to be met. The cases correspond to a reference plant, a lead plant, and a "high conversion ratio" plant for which some core design data were available.

HTGR loading calculations are complicated by the fact that the loadings are not constant throughout the core, flux flattening and other considerations mandating higher loadings at the top and edges of the core than at the center and bottom. Thus peak loading is the limiting factor. Also, because recycle fuel elements have higher concentrations of parasitic isotopes, they must have higher concentrations of fissile isotopes than fresh and makeup fuel. In addition, the relative concentrations of ^{233}U and ^{235}U in the recycle fuel elements affect the total concentration of fissile uranium required at different points in the cycle. Because the fuel cycle data available to ORNL for the three cases did not include information on the spatial variation in loading (the so-called "zoning factors"), nor on the isotopic concentration of the fissile species in recycle fuel elements, assumptions were made that were based on other cases and may have been overly optimistic. Also, it was assumed that 62% of the space in the fuel rods would be available for particles, which is definitely an upper limit.

The calculations showed that for the reference plant (CR = 0.66, power density = 8.4 W/cm^3), only 70% of the available space was needed for the initial core; recycle was not considered for this case because of inadequate data. For the lead plant (CR = 0.76, power density = 7.0 W/cm^3), not all the available space was needed for the initial core, but barely enough space existed for the recycle elements. For the high conversion ratio plant (CR = 0.82, power density = 6.0 W/cm^3), the reload requirements could not be met. The conclusion was that current technology limits the conversion ratio for HTGRs to 0.76 for a burnup of about 65 MW(t)-d/kg HM.

PBR vs. HTGR. The PBR has a greater potential for increased heavy metal loadings than the HTGR because improved fuel fabrication techniques could be developed for the PBR that would reduce the overcoating thickness on the fuel particles and thereby provide more volume for heavy metal in a fuel element. Corresponding improvements for the HTGR are not apparent.

Reactor Availability (?.7)

The percentage of time that a plant is available for producing electricity (or process heat) is an important cost factor that must be considered in any economic evaluation of a given system. On the surface it would appear that the PBR feature of continuous fueling during reactor operation would ensure that the PBR would have a higher availability than the HTGR since the HTGR must be shut down for refueling. However, if the HTGR refueling can be performed parallel to and within the same time frame as inspection and maintenance activities scheduled during shutdown, then the relative availabilities of the two systems would not be affected by the HTGR refueling requirements. In order to compare the scheduled downtimes of the two types of reactors, and also to determine whether HTGR refueling does indeed impact the comparison, work flow charts for scheduled shutdowns for a reference 900-MW(e) HTGR-SC were reviewed and differences with the requirements for a reference 3000-MW(t) PBR-SC were noted. In the comparison, the time intervals between refueling, inspection, and maintenance operations of the HTGR were considered to be 1, 2 or 3 years, with corresponding intervals assumed for the PBR.

The critical path for scheduled shutdown is determined either by the turbine-generator maintenance or the core-servicing activities, the latter including removal and replacement of fuel elements, control rods, control rod drives, and reflectors, plus maintenance of core-servicing equipment. The reviews of the work flow charts showed that for all three refueling intervals the turbine-generator maintenance comprises the critical path for the HTGR-SC, since even the estimated minimum time required for the turbine-generator maintenance exceeds the time required for core-servicing activities. The same turbine-generator maintenance times apply for the PBR, of course, but in general they do not comprise the critical path for PBR outages. Instead, the PBR critical path is determined by the core-servicing activities, primarily the removal and replacement of control rods and control rod drives. As a result, only in one case was the availability of the PBR as high as that of the HTGR, as is shown below:

Interval Between Shutdowns	Availability	
	HTGR	PBR
1-year, maximum T-G maintenance	92%	92%
1-year, minimum T-G maintenance	96%	94%
2-year	96%	95%
3-year	97%	96%

This analysis is, of course, preliminary and does not consider possible reductions in PBR downtime requirements by increasing the control rod lifetime and decreasing the number of control rods required. It also is to be pointed out that the equipment for the removal and replacement of the PBR reflectors is still in the conceptual design state, and if a decision is made to build a PBR in the U.S., then an R&D program for reflector-handling equipment should be undertaken. The probable cost of such a program would be about $10 million.

Graphite Reflector Damage (3.8)

The high temperatures and high neutron fluences in HTGRs and PBRs will produce stresses and dimensional instabilities in the graphite reflectors that will limit their lifetimes. This presents a problem that will be particularly acute for the PBR since the PBR side reflectors are also to provide lateral containment of the core and therefore their structural integrity must be maintained. One solution is to replace the reflectors periodically. But while this could be accomplished easily in the HTGR during refueling operations, replacing the reflector in the PBR could increase in reactor shutdown over that required for scheduled maintenance outages since it would require that fuel pebbles be emptied from the core at least to the depth to which the reflector must be replaced. Removing the fuel pebbles would also introduce a fuel cycle penalty that could be especially severe for once-through fueling schemes. The alternative is to develop a type of reflector graphite that will last for the expected 30-year lifetime of the reactor, and FRG has already developed several candidate graphites based on coal-derived fillers.

When irradiated, graphite initially undergoes contraction (to a maximum density) and then expands. By the time it returns to its original bulk density, its physical properties are degrading rapidly and microscopic damage increases rapidly as the graphite undergoes a net volumetric expansion. Thus the fluence at which the graphite returns to its original volume has been conservatively defined as its "lifetime." In ORNL studies of graphite lifetimes, six of the FRG-developed graphites, plus one developed by the United Kingdom, were irradiated in EDN fluences up to 2.0×10^{22} neutrons/cm^2 (EDN = fluence above 0.18 MeV ÷ 1.8) at 600 to 620°C. Two of the FRG graphites and the UK graphite had lifetimes as high as 1.7×10^{22} EDN, but this is less than one-half the lifetime projected to be required in a large PBR. Thus a superior grade of graphite must still be developed. This implies an increase in the capital cost of reflectors (in addition to increased development costs), which could be a factor of 10 greater than the costs of current graphites.

Seismic Effects (3.9)

The impact of seismic excitations on the safety of the HTGR and the PBR must be established from two different perspectives: the ability of the reactor to operate safely during low-level excitations; and the ability of the reactor to shut down safely during high-level excitations. Of major concern are the possible consequences of (1) a core disarray, (2) a core support failure, (3) a core lateral restraint failure (side reflectors), and (4) a failure of the top reflector and core to respond to an excitation as a unit.

All four of these safety issues have been investigated for the HTGR by GA, who observed in scale model tests that the core behaves as a single unit, and, on this basis, developed a number of computer programs for seismic response analysis. GA has also performed limited studies of seismic effects on the PBR; however, the safety issues were not addressed per se. In addition, General Electric's Advanced Reactor Systems Division has done some analytical seismic studies for the PBR but the results do not permit an adequate engineering assessment. Finally, HRB in Germany has performed experimental studies on a scaled-down model of the PBR core. Their results also have not been published, but HRB has stated that a seismic-caused PBR core disarray would not result in blockage of coolant passages or interfere with the insertion of control rods.

From the investigations performed to date it appears that the HTGR and PBR can both be developed into seismically safe systems; however, all these investigations should be reviewed and summarized to establish the precise state of the knowledge on seismic effects. Only then can the need for future work and the associated costs be determined. At this time, it is not evident that seismic effects are significantly different between HTGRs and PBRs.

Temperature/Flow Oscillations in HTGRs (3.10)

It is recognized that commercialization of the HTGR implies resolution of the temperature/flow oscillation problem that has been encountered in the Fort St. Vrain Reactor. The problem became evident when the FSVR was being raised to power levels that were about 60 to 70% of the design level. The coolant leaving specific regions of the core was observed to have significant temperature oscillations, with the period of an oscillation being about 10 minutes. Reactor neutron flux measurements, together with out-of-reactor flow tests by GA and FRG, indicated that the oscillations were caused by fuel block movements that opened up alternate coolant flow paths. GA developed a "Jaws theory" which postulates that periodic tilting of fuel elements near the top of the core opens up alternate coolant flow paths that could change the fluid flow in a region. However, modeling studies at ORNL showed that the flow variations required to provide the observed outlet temperatures (and validate the Jaws theory) were unreasonably large. On the other hand, bypass flow leakage into the thermocouple assembly sleeves probably gave erroneous in-core temperature readings. Thus, it is plausible that the oscillations were indeed due to block motion. This being the case, the motion should be controllable by proper core design.

Plant Capital Costs (3.11)

The capital costs of the HTGR and PBR were compared by first estimating the costs for the HTGR systems and then estimating the change in costs for the PBR systems. The reference capital cost estimates for all the HTGR systems are given as follows:

		HTGR Estimated Costs ($\$10^6$) (1979 Dollars)					
				3000-MW(t) Process Heat		1000-MW(t) Process Heat	
	Cost Category	3000-MW(t) Steam Cycle	3000-MW(t) Gas Turbine	W/O IHL	With IHL	W/O IHL	With IHL
20	Land and Land Rights	2	2	2	2	2	2
21	Structures and Improvements	135	147	135	144	78	83
22	Reactor Plant Equipment	282	335	396	514	205	266
23	Turbine Plant Equipment	105	78	0	0	0	0
24	Electric Plant Equipment	48	37	42	43	27	27
25	Miscellaneous Plant Equipment	12	10	14	14	10	10
26	Heat Reject System	56	37	0	0	0	0
2	Total Direct Costs	640	646	589	717	322	388
91	Construction Services	83	69	83	92	52	57
92	Home Office Engineering	109	90	109	120	87	95
93	Field Office Engineering	34	28	34	37	22	24
94	Owners Costs	47	39	47	52	30	34
9	Total Indirect Costs	273	226	273	301	191	210
Contingency (10%)		91	87	86	102	51	60
	Total	1,004	959	948	1,120	564	658

These costs were obtained by taking initial cost breakdowns given in 1975 dollars by GA and UE&C (United Engineers and Constructors) for 3000-MW(t) plants of each type (HTGR-SC, HTGR-GT, and HTGR-PH) and then updating the costs to 1979 dollar estimates on the basis of later GA-UE&E information for a somewhat smaller HTGR-SC system. In addition, the HTGR-PH estimates, given both with and without an intermediate heat transfer loop (IHL), were scaled down to a 1000-MW(t) plant. The nominal range of uncertainty in the total direct costs (Category 2) was taken to be -5% to +10%.

The resulting relative total direct costs estimated for HTGRs and PBRs, based on HTGR costs and estimated PBR incremental costs, are given below:

	HTGR Total Direct Costs ($\$10^6$)		PBR Total Direct Costs ($\$10^6$)	
	Reference Cost	Cost Range	Reference Cost	Cost Range
3000-MW(t), SC	640	608 - 704	714	657 - 816
3000-MW(t), GT	646	614 - 711	720	663 - 823
3000-MW(t), PH (with IHL)	767[a]	729 - 844	841	778 - 956
1000-MW(t), SC	346	329 - 381	374	344 - 427
1000-MW(t), GT	350	333 - 385	378	348 - 431
1000-MW(t), PH (with IHL)	420[b]	400 - 462	448	415 - 508

[a]Includes $50 million for simultaneous generation of electricity.
[b]Includes $32 million for simultaneous generation of electricity.

Reactor Research and Development Costs (3.12)

Research and development (R&D) costs for HTGRs and PBRs were estimated on a relative basis. In obtaining values, the procedure was to first estimate the HTGR costs and then to project the additional expenditures required to develop the PBR in the U.S.

The costs were divided into reactor costs and fuel recycle costs, with the reactor R&D costs estimated to be as follows:

System	HTGR Costs ($\$10^6$)	PBR Incremental Costs ($\$10^6$)
Steam Cycle	300 - 400	145 (range of 100 - 200)
Gas Turbine	450 - 800	145 (range of 100 - 200)
Process Heat	600 -1000	145 (range of 100 - 200)

These estimates cover both base R&D (development of fuels, structural materials, graphite, and containment vessels and development of an information base on fission-product behavior) and equipment R&D (design, development, fabrication and testing of equipment and associated systems).

The fuel recycle R&D costs were estimated to be $1400 million to $2100 million for all the systems. The costs cover base R&D for head-end reprocessing and fuel refabrication and the design and operation (for 5 to 8 years) of a hot pilot plant to demonstrate fuel recycle equipment and systems.

These costs, based on an assumed schedule for the introduction of the lead plants, are those considered to be above vendor/utility commercial investments. They do not, however, include first-of-a-kind costs for construction of the early plants.

1.0. INTRODUCTION

The definition and scope of this comparative evaluation of the High Temperature Gas Cooled Reactor (HTGR) and the Pebble Bed Reactor (PBR) have been described in the preceding Introduction and Executive Summary. Restated briefly, the comparative evaluation consists of a comparison of the economic performance of the two types of reactors as they were applied to the production of electricity or process heat. For electricity production the systems considered were a steam cycle system with an outlet coolant temperature of 750°C and a gas turbine system with an outlet coolant temperature of 850°C. For process heat production, the outlet coolant temperature was taken as 950°C. The reactors were rated at 1000 and 3000 MW(t), and both once-through fuel cycles and recycle cases were considered.

A first step of the comparative evaluation was the study of key technical issues to determine which design or operational differences between the systems would affect their relative economic performances. These studies are described in detail in Chapter 3, and brief descriptions of those differences that were judged to be sufficiently important to be considered in the one-to-one comparisons of the two types of reactors are described in Section 1.1 of this chapter.

As was expected, the studies of technical issues revealed that the design and cost information available on the HTGR was considerably more detailed than that available on the PBR; therefore it was determined that the comparative analysis should employ a probabilistic analysis technique that would incorporate the different levels of uncertainty regarding the two reactor concepts. It was also determined that the analysis should focus on the differences between the reactors so that large uncertainties associated with components common to both systems would not dominate the results. Because of time limitations, the probabilistic analysis was limited to reference cases, and a more cursory overall evaluation was performed for the other systems. The probabilistic analysis is described here in Section 1.2 and the overall evaluation in Section 1.3. The conclusions and recommendations resulting from the evaluations are then presented in Section 1.4.

It will be noted throughout this report that certain overall assumptions regarding each system were necessary owing to the insufficiency of information. For example, it was assumed that the specified instrumentation and controls systems for the PBR were both adequate and practical, whereas technical studies described in Chapter 3 indicate that the practicality of these systems remains an open question. Also, it was assumed that the flow-temperature oscillations observed in the Fort St. Vrain Reactor (FSVR) were not a generic HTGR problem and that future HTGRs will not be subject to this problem. In addition, reference core designs for the HTGR and PBR were used with no attempt to optimize them through redesign. Finally, with certain exceptions noted, it was generally assumed that the two reactors can be licensed in the U.S. and will operate at the designed power level. The results of these studies are, of course, contingent upon these assumptions; however, it should be pointed out that non-attainment of a specific assumption may only reinforce the conclusions, rather than change them.

1.1. IDENTIFIED DIFFERENCES BETWEEN HTGR AND PBR

1.1.1. Introduction

As has been stated, this comparative evaluation of the HTGR and PBR focused on the design and operational differences between the two reactor types, the differences in turn being limited to those expected to have a significant impact on the relative economic performances of the systems in which the reactors were applied. This section summarizes the differences judged to be the most important, the specific impact of each being discussed in more detail in Chapter 3.

1.1.2. Primary and Secondary Containment Structure

The core volume required by the PBR for a given thermal output is significantly larger than that required for the HTGR, primarily because the PBR has a larger coolant fraction (39% versus ~20% for the HTGR) and it operates at a lower power density (5.5 W/cm^3 versus 7.1 W/cm^3 for the HTGR). Moreover, due to a limited depth to which control rods can be inserted in the pebble core and a higher pressure drop across a core of a given height, the shape of the PBR core is somewhat like a "pancake," whereas the shape of the HTGR core is more like a right circular cylinder, which is the preferred geometry from a neutron leakage standpoint. With the larger volume and the pancake shape, the diameter of the PBR is larger than the diameter of the corresponding HTGR, and therefore the diameters of the PCRV and the surrounding containment building are larger. This, of course, translates into higher costs for the PCRV and containment building, as well as for the liner and insulation. (Note: Although PBR designs often consider use of a warm liner, a cold liner was assumed in this study in order to place the two systems on a common basis relative to performance requirements.)

1.1.3. Reactor Control and Instrumentation

The control and instrumentation requirements for the HTGR are well known and control systems prototypical of those envisioned for commercial-size HTGRs have been developed and are in use in the U.S. By contrast, control rod designs for large PBRs are not in commercial use, and while such designs appear feasible, the licensing requirements for the PBR control system have not been established in the U.S.

The control systems for the two types of reactors differ significantly in mechanical operation. In the prism core of the HTGR, the control rods are inserted into channels in the fuel blocks and under normal operation no significant resistance force is involved; hence, the rod insertion can take place due to gravity. This provides a backup shutdown mechanism should electrical power to the reactor be interrupted. By contrast, insertion of control rods into the pebble core of the PBR will require significant force and both the control rods themselves and the control rod drives must have a greatly increased mechanical capability relative to those in the HTGR. Similarly, a powered backup shutdown mechanism would be necessary for at least part of the control rods.

Two types of control rods have been proposed for the PBR: a thrust-type rod designed to be pushed into the pebble bed; and an auger-type rod designed to be screwed into the reactor core. The auger type would require that the primary drive mechanism be capable of both translation and rotation, whereas the thrust-type would require translation capability only. While the number of rods for the HTGR and the PBR is approximately the same, the mechanical requirements for the PBR control rods necessitates that each be linked to its own drive mechanism, whereas in the HTGR two rods are associated with each drive.

Control of xenon-induced instability in the PBR appears possible without control-rod insertion, but it is not assured. In any case, xenon override requirements will necessitate the insertion of at least some of the PBR control rods, and because the rods will be inserted directly in the pebble bed, they will reach their ductility limit more rapidly than the HTGR rods and their lifetime will be shorter. One result of the increased number of control rod drives and the shorter rod lifetime is that the time required to replace and/or maintain the PBR control system will be greater than the corresponding time for the HTGR.

The material requirements of the PBR control rods also differ significantly from those of the HTGR control rods. Although radiation damage to the control rod materials is important for both reactors (for the HTGR, damage to the shock absorber is assumed to be the limiting factor), the stress levels are inherently higher in the PBR.

The secondary shutdown mechanisms for the reactors are somewhat similar. In both cases, small absorber spheres are released at the top of the core to effect shutdown. In the HTGR, these absorber spheres (pellets) flow down into previously machined channels in the fuel blocks. In the PBR, the absorber spheres, which are considerably smaller than the fuel spheres, filter down into the interstitial locations of the pebble bed itself. While the secondary shutdown systems are operated similarly, it must be noted that the consequences of the operation are markedly different for the two systems. In the HTGR, the absorber pellets can be removed after depressurization via a vacuum device. In the PBR, removal of the small absorber spheres (called KLAK) will require that approximately 15% of the core fuel be removed via the fuel discharge machine. On a throwaway fuel cycle this can amount to a fuel cycle penalty of roughly 40% of the annual requirements.

Finally, the HTGR can more easily accommodate in-core instrumentation. In-core instrumentation for the PBR, either permanent or occasional (i.e. flux traverses), will require the use of replaceable instrument thimbles which will themselves be subjected to significant forces within the bed. The design basis for the PBR assumes that ex-core instrumentation will be adequate for operational and licensing purposes. Under such circumstances more instrumentation will undoubtedly be required.

1.1.4. Radial Reflector

The two reactor concepts differ in the design of the radial graphite reflector. In the HTGR, the reflector is composed of hexagonal graphite prisms similar to the fuel blocks. Periodic replacement of the graphite will be accomplished with the fuel-handling machine during regular refueling operations; hence, no significant additional downtown penalty is foreseen.

In the PBR, the radial reflector needs to be permanent; however, the graphites presently available for reflector fabrication are limited as to their operational lifetime, and hence will have to be replaced at least once during the operating lifetime of the reactor. Replacing the radial reflector will require a partial unloading of the fuel, an operation that will impose a fuel cycle penalty, especially for once-through fuel cycles, and also introduce a significant nonavailability penalty. In addition, replacing the PBR reflector will require a separate reflector-handling machine, but since such a device would be needed at a given reactor only infrequently, it can be assumed that the cost of such a machine would be shared among a number of PBRs.

An alternative to replacing the PBR radial reflector would be fabricating it from improved graphites with lifetimes equal to the projected reactor lifetime. However, such improved graphites are not yet available and their development and fabrication would probably cost 10 to 20 times that of the conventional reflector graphite. Thus, a tradeoff would exist between the initial capital cost of the reflector and the availability cost incurred during reflector replacement.

Finally, in order to achieve a 30-year life for the PBR reflector, it will be necessary to maintain the temperature of the graphite below 550°C, which will probably necessitate a separate reflector cooling system.

1.1.5. Reactor Refueling Systems

Owing to the disparate refueling operations in the two systems, the requisite refueling mechanisms represent a significant difference between the two concepts. For the HTGR, which is shut down periodically to replace a fraction of the core fuel blocks with fresh fuel, a fuel-handling machine, fuel transfer casks, an auxilary transfer machine, and fuel storage wells will be required. Maintenance of this equipment can be accomplished during reactor operation.

For the PBR, on-line refueling is used, with a specified number of fuel spheres withdrawn from and added to the reactor daily without reactor shutdown. This system requires that fuel feeding and removal tubes be structured with helium locks to maintain isolation of the primary coolant circuit. Also, mechanical distributors must be included to disperse the fresh fuel being added. Thus, the on-line refueling mechanism of the PBR, and in particular the requirement for daily operation of the helium locks, represents a system for which no direct analog exists in the HTGR design. Such an arrangement will possibly require reactor shutdown for maintenance to be performed on the refueling valves. At the same time, however, the refueling mechanism in some HTGR designs is contained within the PCRV, and this will also require reactor shutdown for maintenance (not considered here).

Contrary to what one might expect, the continuous refueling feature of the PBR does not lead to improved availability. For systems dedicated to electricity production, the turbine-generator maintenance requirements control the availability of either system in most cases. For systems dedicated to process heat production, the control rod replacement requirements for the PBR (see above) could be the controlling factor.

1.1.6. Fuel Performance

In general, the neutronic performance of the PBR is slightly better than that of the HTGR. The on-line refueling feature reduces the parasitic neutron losses attributable to fission products, and a slightly higher conversion ratio of the PBR results in a decrease in the amount of control poisons required. On the other hand, the "pancake" shape of the PBR core (and, in particular, a 1-meter gap between the top of the core and the bottom of the top reflector) results in a slightly higher neutron leakage from the core; however, this only partially offsets the gains in neutronic performance.

Also, with its on-line refueling, the PBR can approach its equilibrium core configuration (i.e. fissile inventory) from an initial value that is less than the equilibrium value, whereas the HTGR must have excess reactivity initially and hence approaches equilibrium from above. This results in a smaller initial commitment of U_3O_8 and separative work for the PBR, which is a significant advantage.

Several mechanisms for fission-product release from the fuel particles into the external coolant loops of the reactors have been recognized, including uranium contamination of the particle matrix material, manufacturing defects in the particle coatings, and failure of the coatings due to reactions within the fuel kernel. If the fuel kernels for both reactors were manufactured to the same quality standards, circuit activity due to the matrix contamination or coating manufacturing defects for a given temperature would result in the same fission-product activity in both reactor systems. The degree of fission-product attack of coatings, however, is influenced by the fuel temperature (the retention ability of the fuel kernel degrading with increasing fuel temperatures, higher temperature gradients, and longer irradiation times). PBRs have lower fuel temperatures and smaller gradients for a given outlet coolant temperature; therefore, the radioactivity of the coolant loop (due primarily to SiC corrosion by the fission-product palladium at high temperatures) tends to be lower for PBRs than for HTGRs. The difference in the degree of palladium attack does not appear to be significant for coolant outlet temperatures below 850°C (a region where the fission-product activity is dominated by other factors). However, the increase in palladium-induced fuel failures with increasing fuel temperature is high enough that the difference in the coolant loop activity at the 950°C coolant outlet temperature could be significant. The level of coolant loop activity can influence the plant costs through requirements for additional equipment and personnel and through increased downtime (decreased reactor availability) for maintenance of the loop components.

1.1.7. Reactor Safety

In comparing the relative safety of the HTGR and PBR, consideration was given to earthquakes, depressurization events, spent fuel handling accidents, water ingress accidents, and core heatup events.

The seismic responses of the two systems are different due to differences in the physical characteristics of the core. In particular, the core barrel in the PBR is subjected to significant radial forces, requiring a different seismic restraint design. However, it was estimated that such a design modification would not impact the capital cost difference between the two systems. A second seismic question concerns the top reflector of the PBR, which is located 1 meter above the core. Failure of the reflector support or the reflector integrity would result in an increase in the reactivity of the core and possibly could interfere with control rod insertion. The same problem does not exist in the HTGR, since the top reflector is located directly on the core fuel blocks. However, a seismic event could produce motion in the HTGR fuel blocks, resulting in a misalignment between the control rods and the control channels. In both reactor systems, the secondary shutdown system (i.e. the small absorber spheres) would provide sufficient shutdown margin, and thus seismic events are not considered to be safety problems. But as explained in Section 1.1.3, when the secondary shutdown system is invoked, the PBR suffers the greater economic penalty. Therefore, while the consequences of seismic events will not differ for the two systems, the costs for the PBR could be greater. (It should be noted that even under normal operating conditions, the probability of control rod failure upon insertion is higher for the PBR due to the greater force required.)

The consequences of a depressurization accident would be lower for the PBR than for the HTGR, but the PBR is estimated to have a higher probability for such occurrences (due primarily to the fuel-handling system). Thus, the two systems are considered to be comparable with respect to depressurization events.

The consequences and probability of spent fuel handling accidents were also judged to be comparable.

With regard to air or water ingress and subsequent graphite oxidation, the probabilities for occurrence are approximately the same for both systems. While the graphite oxidation rate will be higher for the PBR (due to the larger surface area), the difference is not considered to be significant. Fission-product release to the environment as a result of water ingress is similar in both reactors for outlet coolant temperatures below 850°C. Above 850°C, the PBR will have a lower fission-product release due to a lower fraction of failed fuel particles. However, no significant safety difference was identified since both release values appear to be low compared to NRC standards.

Finally, for core heatup accidents, the time constants for both systems are long. To reach a graphite temperature of 3500°C, the HTGR is estimated to require 20 hours, while the PBR requires 25 to 30 hours. This difference is not significant.

On the basis of the above considerations, the PBR and HTGR were assumed to be comparable with respect to overall safety; however, the PBR would suffer the greater economic penalty if the secondary shutdown mechanism were to be actuated.

1.1.8. Reactor Availability

Although the differences in the projected reactor availability between the reactor systems have been discussed in the preceding paragraphs, it is useful to summarize those factors contributing to the availability of each system. As noted above, the time for refueling and control rod replacement in the HTGR and the time required for control rod replacement in the PBR both usually fall within the turbine-generator maintenance "envelope." Therefore, no significant difference in availability due to scheduled maintenance is anticipated except in those systems for which turbine maintenance is not the critical path, in which case the longer times required for PBR control rod replacement may result in an additional nonavailability penalty. Differences in availability due to unscheduled maintenance may be significantly higher for the HTGR at the 950°C outlet coolant temperature owing to a higher coolant circuit activity.

Another difference between the two systems is the potential requirement for replacing the radial reflector in the PBR, which would introduce an extended downtime. If, on the other hand, an improved grade of graphite is developed for the PBR reflector, the probability that reflector replacement would be necessary would be markedly reduced.

1.1.9. Miscellaneous Items

Among the other differences identified for the two reactor systems is a smaller heat exchanger surface area for the PBR core auxiliary cooling system required to limit the temperature rise of the fuel following a loss-of-coolant incident. Also, owing to its larger fuel volume, somewhat greater fuel storage capability is required for the PBR system (for both fresh fuel and spent fuel) to maintain an equivalent fuel supply.

Differences between the ex-reactor fuel cycles for the two systems have also been identified. It is estimated that the fabrication of nonrecycled fuel would be 15 to 20% less for the PBR than for the HTGR; however, the refabrication of recycle fuel would be 5 to 10% more expensive for the PBR. The reprocessing costs for the fuels of both reactors are proportional to the amount of graphite burned and are estimated to be about the same for the same equivalent fuel processing rates.

Two other differences should be noted. First, because of the more advanced state of HTGR design in the U.S., it is estimated that the operational date of the PBR will lag up to four years behind that of an equivalent HTGR system. For the same reason, it is anticipated that the incremental PBR R&D costs (over the HTGR R&D costs) will be approximately $145 million.

1.2. PROBABILISTIC ANALYSIS OF REFERENCE HTGR AND PBR SYSTEMS

T. J. Burns

1.2.1. Introduction

It has been pointed out earlier that because of larger uncertainties associated with the design and cost data for the PBR as opposed to the HTGR, it would be inappropriate to compare the two reactor systems on the basis of a single-value or deterministic criterion. Hence, an early requirement of the study was to develop a methodology structured to produce probabilistic results for each system -- that is, estimated costs that have specified confidence levels reflecting the uncertainties in the input data.

The methodology was also to have the capability for treating the differences between the two systems. It has already been shown in Section 1.1 that the primary differences between the PBR and the HTGR are associated with the differences in the fuel form and core geometry; thus much of the reactor plant (and also much of the process plant) is common to both systems. However, considerable uncertainty exists as to the performance and cost of these common components, and if they were to be included as cost elements in the probabilistic analysis, they would tend to dominate the results because of their relative magnitudes. Therefore, it was concluded that the elements common to both systems should not be considered in the probabilistic method. Using these criteria, Management Analysis Company developed the basic methodology,[1] which was subsequently modifed and extended by ORNL.

As noted earlier, the scope of this comparative evaluation of the HTGR and PBR was such that for each reactor type it included two reactor sizes [1000 MW(t) and 3000 MW(t)], two fuel enrichments (MEU and HEU), two fuel cycle options (throwaway and recycle), and three process applications (steam cycle, direct-cycle, and process heat, corresponding to coolant outlet temperatures of 750°C, 850°C, and 950°C, respectively). However, with the limited time available for the study, it was not possible to consider all combinations of these parameters in the probabilistic analysis. Therefore, the probabilistic analysis was limited to the following specific parameters chosen to define reference HTGR and PBR systems:

Size:	3000 MW(t) [1200 MW(e)]
Fuel:	Medium enriched uranium (MEU)
Fuel Cycle:	Throwaway
Process Application:	Direct cycle (referred to throughout the report as "GT" for gas turbine) for electricity production; coolant outlet temperature = 850°C

In addition, two perturbations were introduced in the analysis to roughly estimate the effect on the power costs of using systems dedicated to process heat production (coolant outlet temperature = 950°C).

In the discussion below, the probabilistic methodology is first described and is followed by a presentation of the results obtained by applying the methodology to the reference systems. The input data for the analysis are given in Chapter 2; and the studies of key technical issues from which most of the input data were derived are described in Chapter 3.

1.2.2. Methodology

Decision Measures and Logic Structure

The specific evaluation criterion selected for this study was the levelized (15-year) power cost, given in mills/kW-hr, that is commonly used by utility decision makers. It is recognized, of course, that in a practical situation the levelized power cost of a commercial reactor is only one of several decision measures utilized by a utility. Here, however, it was assumed that a commercial market for the product (electricity) will exist in the future and that the demand will be large enough to support the required ex-reactor infrastructure.*

The logic structure of the computer model used in the analysis was mandated to a large extent by the selection of the power cost as the comparison criterion (see Fig. 1.2.1). In it the overall energy costs are collected from four major cost categories: (1) plant costs, (2) fuel costs, (3) operation and maintenance costs, and (4) replacement supply costs. The replacement supply component is included to allow for possible differences in the operational availability of the two systems.

Although these four costs categories are sufficient to determine the overall energy production costs from an owner perspective, an additional cost category, research and development costs, is needed for the broader governmental perspective of overall project costs. This is particularly true for the high-temperature gas-cooled systems since different R&D levels for each system can be anticipated. Thus, the R&D component is included as a fifth category.

A description of how these various cost categories enter into the analysis is given below:

Plant Costs. The overall plant cost component was subdivided into a direct cost category, an indirect cost category, and an owner's cost category, with a ten-year

*The method provides for the levelized power costs in the process heat cases to be given in \$/MBTU(t) to facilitate comparison with other process heat sources. In this analysis, however, the process heat cases were treated as perturbations to the gas turbine cases, and thus the results are retained in units of equivalent mills/kW-hr.

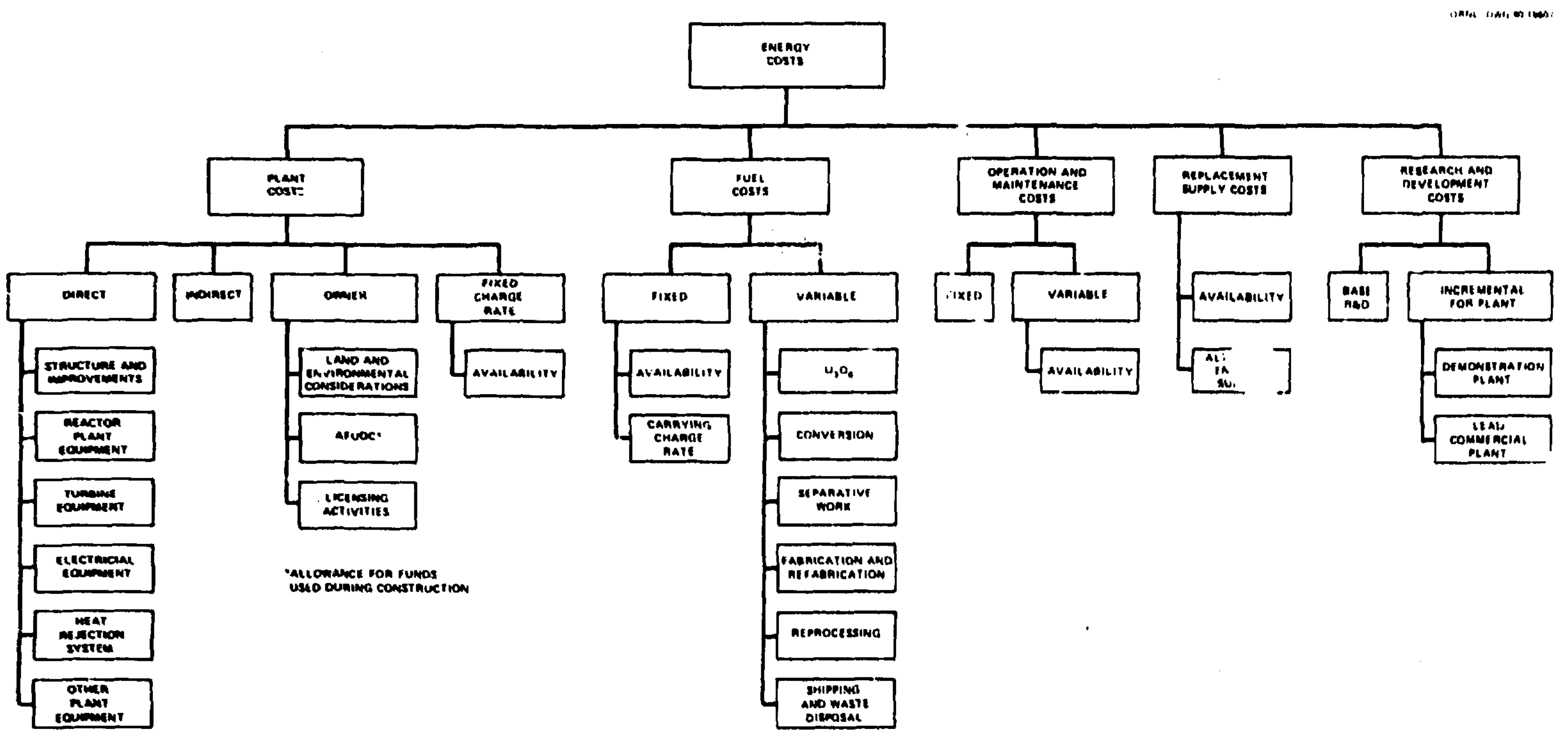

Fig. 1.2.1. Overall Energy Cost Model Structure.

construction schedule postulated for both reactor types. The direct cost category was constructed to match the plant cost estimating format employed by the CONCEPT computer code.[2] In order to adequately differentiate between the reactor concepts, the estimates were carried to the three-digit cost level. The indirect costs were calculated on the basis of 32% of direct costs. The owner's cost includes the cost of the plant site, licensing activities, and an allowance for funds used during construction. The plant costs are then levelized using a discount rate of 7% and the estimated capacity factor of the reactor for each year.

Fuel Costs. The fuel costs utilized in this study were based principally on the mass flows calculated for each reactor system. The breakdown used in the model allows specific charges for various fuel cycle activities such as U_3O_8 purchases, conversion to UF_6, enrichment, fabrication, reprocessing, and waste disposal. In accordance with utility practice, the carrying charges attributable to the fuel cycle expenditures were also included as a part of the fuel cycle costs. All fuel cycle component costs, with the exception of the price of U_3O_8, were assumed to remain constant in terms of 1980 dollars. The price of U_3O_8 was escalated at a rate of 2.5%/year to account for the depletion of the resource base and the attendant real price increase.

Operation and Maintenance Costs. The operation and maintenance costs, entered as fixed costs and variable costs (the latter dependent on the reactor capacity factor), were treated in a lesser degree of detail than the plant and fuel costs; however, the elements employed are deemed adequate to differentiate between the two reactor types.

Replacement Energy Costs. The costs for replacement power were calculated by assuming that the alternate energy will be provided by a typical base-loaded, coal-fired power plant. The amount of replacement power required is calculated from the difference between the overall HTGR and PBR plant availability factors. The charge for the replacement power is the difference between the coal-fired power plant costs and the nuclear power plant costs, i.e. the incremental costs incurred in switching to the alternative energy supply. Thus, a lower availability for one system compared to the other is reflected in the cost for replacement power, as well in the previous three cost categories through the cost levelization procedure.

Research and Development. The research, development, and demonstration costs required by the government to assess the overall plant costs are included as (1) base R&D program costs, and (2) incremental costs for the demonstration plant and the lead commercial plant. The incremental costs are included to incorporate the additional costs ("first of a kind" costs) inherent in the development of a new reactor system and are intended to represent those costs that are in excess of an equivalent proven reactor system.

Probabilistic Assessment

As noteu previously, an estimate of the energy costs of either the HTGR or the PBR entails considerable uncertainty, including uncertainty associated with the costs of components common to both systems. However, in acquiring the requisite uncertainties for this study, only those components in the two systems that differ were considered. The assessment of the uncertainties was based on work performed by GA, GE, and ORNL and described in Chapter 3.

Representation of the uncertainty inherent in each identified cost component was accomplished via a cumulative probability distribution such as that depicted in Fig. 1.2.2. This distribution is based on the nominal uncertainty characteristic of a developed industry. Figure 1.2.2 indicates the probability that the capital cost of the HTGR will be less than the percentage (of the base value) indicated on the abscissa. For example, there will be a probability of 0.5 that the capital cost of the reactor will be less than 100% of the base estimate, and a probability of 0.75 that the capital cost will be less than 105% of the base estimate. (The above distribution does not influence the evaluation performed but gives perspective on overall cost uncertainties.)

It should be noted that the 0.5 probability level corresponds to the *median* value, and that, in general, the *expected value* differs from the median for nonsymmetric distributions. The median value for the above capital cost is 100%, yet the expected value for the distribution is 101.25%. The shape or structure contained in the input probability distributions is related to the level of information available. The greater the degree of structure present in the distribution, the higher the presumed level of information relative to that particular component. The lo... level of structure possible is a straight line, reflecting only information concerning the expected minimum and maximum values. Such a distribution assumes that all intermediate values are equally probable.

With estimates of this type used for all the relevant variables, an overall probabilistic result for the anticipated energy cost difference for the two reactors can be constructed. To illustrate the process, consider the data for the four most significant factors in determining the overall energy costs for the HTGR: the capital costs shown in Fig. 1.2.2, the U_3O_8 price escalation rates presented in Fig. 1.2.3, the estimates of downtime for unscheduled maintenance shown in Fig. 1.2.4, and an estimated scheduled annual outage of four weeks. Combining these data yields the overall energy cost distribution depicted in Fig. 1.2.5. The expected value of this distribution is 19.92 mills/kW-hr, and the distribution ranges from 15.69 to 24.30 mills/kW-hr based upon the economic ground rules used in this study.

Probabilistic Sensitivity Analysis

As in other types of studies, the effects of specific uncertainties can be assessed through "sensitivity" analyses. Two different types of sensitivity analysis are useful

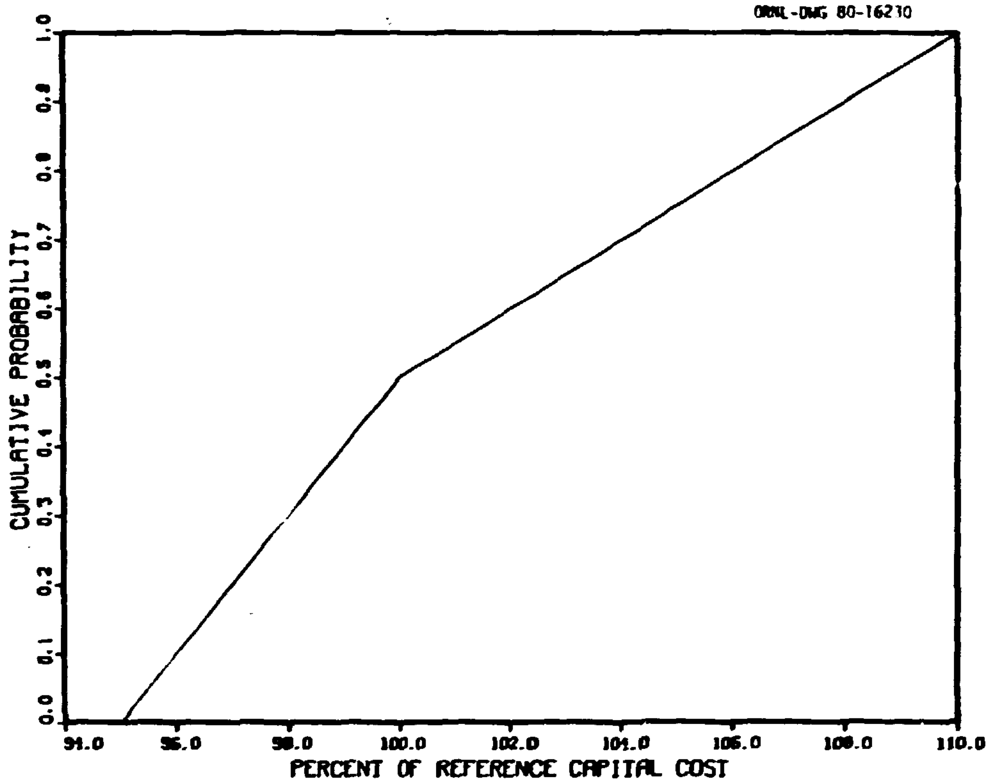

Fig. 1.2.2. Cumulative Probability of HTGR Capital Costs.

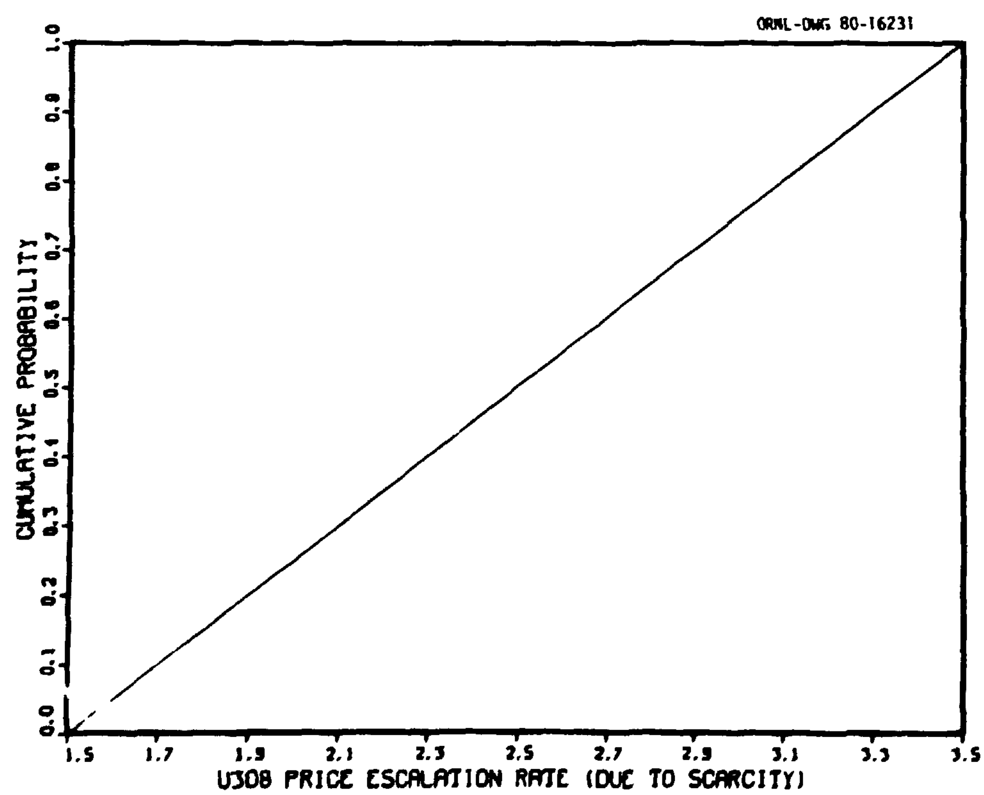

Fig. 1.2.3. Cumulative Probability of U_3O_8 Price Escalation Rate.

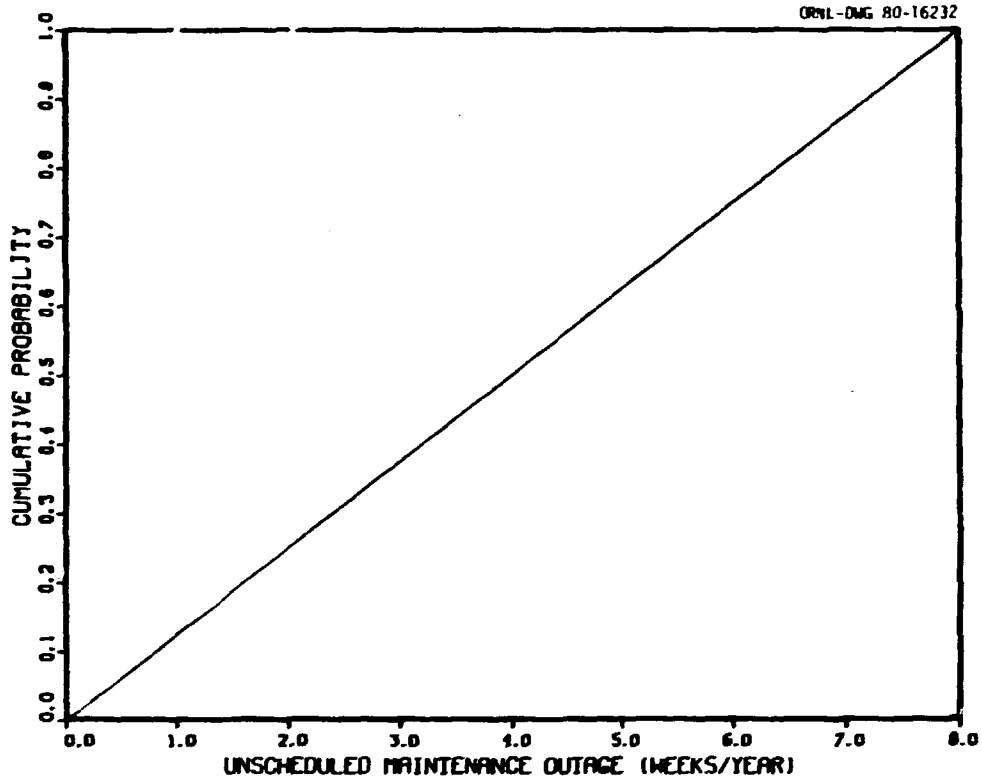

Fig. 1.2.4. Cumulative Probability of HTGR Downtime for Nominal Unscheduled Maintenance Activities.

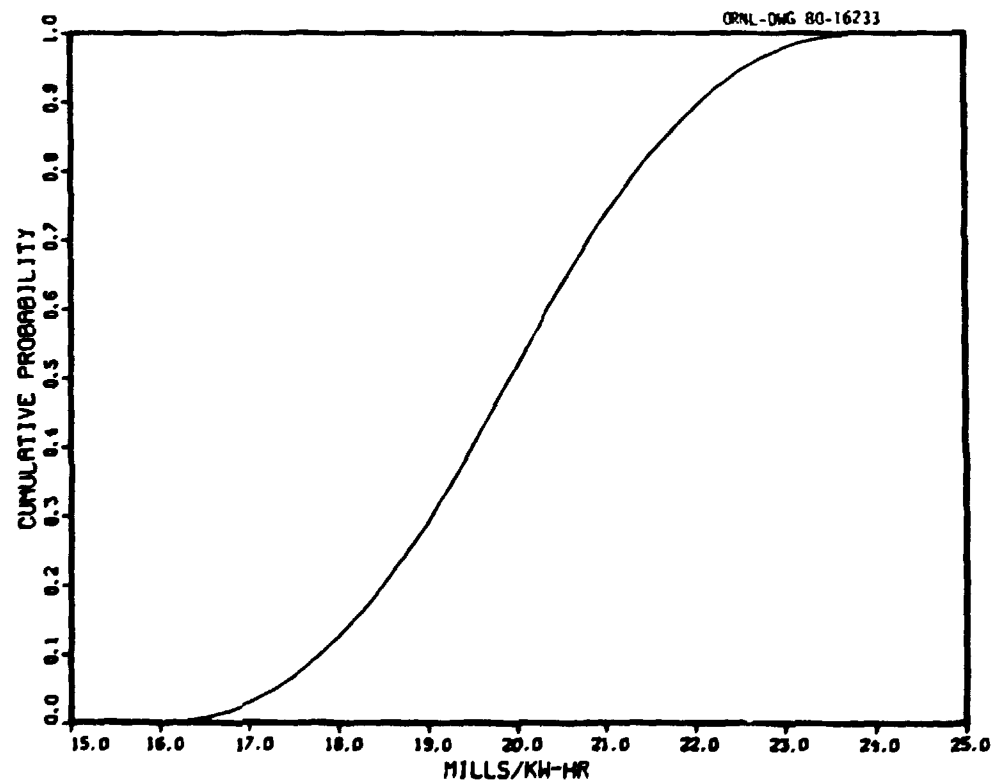

Fig. 1.2.5. Cumulative Probability of HTGR Total Energy Costs.

for probabilistic problems. In the first, specific values from one of the component distributions are selected as input to the model, thereby eliminating from the results the uncertainty related to that component. The calculated energy cost distribution curve then reflects a reduced uncertainty, as well as a possible shift due to the exact parameter value selected. Hence, the contribution of both the choice of the values (e.g., minimum, maximum, median, etc.) and the component uncertainty can be addressed. As an example of this type of analysis, the probabilistic sensitivity of the overall HTGR energy cost relative to the scarcity rate is given in Fig. 1.2.6. Here, the low and high designations correspond to the minimum and maximum values shown in Fig. 1.2.3.

The second type of sensitivity analysis concerns the shape of the various cost component distributions. As noted previously, the amount of structure inherent in the distribution can be viewed as the degree to which the uncertainty of the distribution can be quantified, i.e., the adequacy of the current level of information. By changing the shape of the input data distributions to reflect anticipated increases in the level of information, such as through a successful R&D program, the value of the information in terms of the impact on the decision measure (i.e., expected value and uncertainty) can be studied. Since this study was concerned primarily with differences between two reactor systems involving many similar components, this latter type of sensitivity analysis was not pursued.

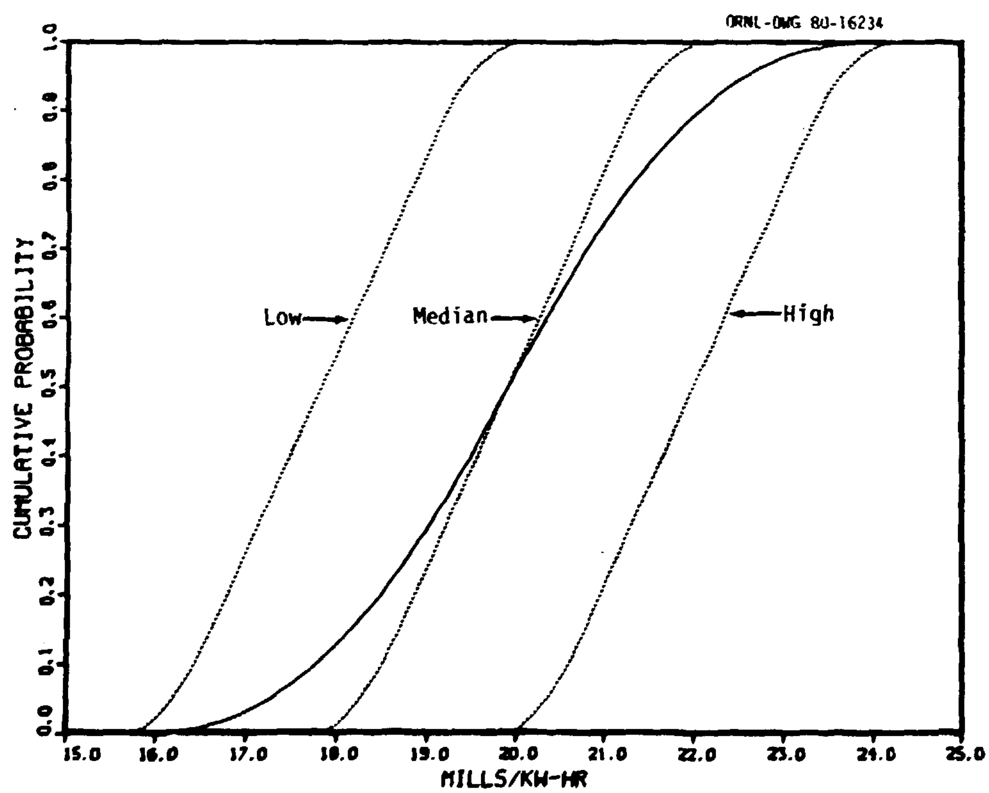

Fig. 1.2.6. Sensitivity of HTGR Energy Costs to Scarcity of U_3O_8.

1.2.3. Probabilistic Results

As noted in Section 1.2.2, the overall decision measure selected for this study consists of the levelized power cost of each system but within the framework of recognized differences and technology requirements. The basic economic parameters utilized in the study are shown in Table 1.2.1. The year by year costs (10 years construction and 15 years operation) are levelized and referenced to 1980. In order to eliminate the rather large effects of inflation (common to both systems), all costs are given in terms of constant 1980 dollars.

Although the methodology is structured to produce four basic cost components (plant, O&M, fuel, and availability), it is instructive to examine the impact of various design and operational differences via an alternate, and more generic, categorization. This categorization also has four basic components which derive from the fundamental differences between the HTGR and the PBR: (1) those differences resulting from a different design choice for power density (e.g., differences in the size of the containment vessel and the PCRV), (2) those differences relating to the instrumentation and control requirements, (3) those differences in the reflector replacement requirements, and (4) those modifications that are traceable to the physical form of the fuel (fuel handling, fabrication, performance, etc.). Table 1.2.2 shows how many of the cost area(s) are impacted by these four classes of design differences.

3000-MW(t) Gas Turbine Systems

The basic data describing the 3000-MW(t) gas turbine systems in terms of the plant (capital) costs, O&M costs, and fuel cycle parameters are given in Tables 1.2.3, 1.2.4, and 1.2.5, respectively. Although Chapter 3 provides the rationale and assumptions for these data, it is instructive to consider a specific item to illustrate the interactions between the various cost component categories. Consider, for example, the PBR control system. The issue of the control mechanism for the PBR is subject to a large uncertainty due to its lower level of specification for the 3000-MW(t) PBR-GT relative to the HTGR-GT. In particular, both the number and type of control rods and control rod drives required in the PBR are uncertain. Two types of rods are being considered: a thrust-type rod which would be pushed into the pebble bed and an auger- or helical-type rod which would be screwed into the bed, thereby displacing the pebbles upward. The operational characteristics of the PBR control rods are also not available at present; i.e., it is not yet known how frequently the rods, which will primarily operate in the void above the bed, will routinely penetrate the bed or at what depth they will penetrate. Until the operational mode is defined, the schedule for control rod replacements cannot be determined since bed penetration will require more frequent replacement.

For the analysis presented here, it was assumed that the PBR control system would consist of 46 helical rods and 105 thrust rods, each with a separate control rod drive.

Table 1.2.1. Economic Parameters Utilized in the Comparative Assessment of HTGR and PBR [Once-Through Fuel Cycles]

Inflation Rate		0%
Utility Interest Rate		3.5%
Fixed Charge Rate (Capital)		7.0%
Carrying Charge Rate (Fuel)		7.9%
Government Interest Rate		5.0%
Discount Rate		7.0%
Construction Period		10 years
U_3O_8 Price		\$100/kg
U_3O_8 Escalation		2.5%
Conversion Price		\$4.30/kg (UF_6)
Conversion Loss		0.5%
Separative Work		\$95/SWU
Tails Assay		0.002
Spent Fuel Shipping Cost		\$250/kg HM
Spent Fuel Disposed Cost		\$450/kg HM
Lead/Lag Times:	U_3O_8	12 months
	Conversion	9 months
	Enrichment	7.5 months
	Fabrication	6 months
	Offsite Shipment	9 months

Table 1.2.2. Cost Components Impacted by Design Differences

Reactor Design Difference	Impacted Cost Component			
	Capital	O&M	Fuel	Availability
Power Density[a]	X		b	
Control	X	X		X
Reflector	X	X	X	X
Fuel Type	X		X	X

[a]Includes impact of differences in core geometry.

[b]Fuel-cycle cost component affected by difference in power density is included under fuel type.

Table 1.2.3. Reference Plant Median Costs for 3000-MW(t) Gas Turbine Systems (January 1979 Dollars)

		Cost Category	Costs ($10^6) HTGR	PBR	(PBR - HTGR) Difference
21	Structures and Improvements				
	211	Yardwork	8.9	8.9	-
	212	Reactor Containment	63.2	71.5	8.3
	213	Turbine Generator Building	-	-	-
	214	Security Building	0.4	0.4	-
	215	Reactor Service Building	10.8	10.8	-
	216	Main Circ. Control Bldg.	0.6	0.6	-
	217	Fuel Storage Building	10.8	13.9	3.1
	218	Other Structures	48.5	48.5	-
		Subtotal	143.2	154.6	11.4
22	Reactor Equipment				
	221A	PCRV Structure	69.3	91.3	22.0
	221B	PCRV Liners & Penetration	44.3	48.4	4.1
	221C	Reactor Control Mechanism	10.8	38.8	28.0
	221D	PCRV Internals & Insulation	43.4	56.5	1.6
	221E	Reflector Graphite Upgrade	0	10.5	10.5
	222	Main Heat Transfer & Transport System	67.4	71.4	4.0
	223	Safeguards Cooling System*	23.0	19.0	-4.0
	224	Rad. Waste System	4.2	4.2	-
	225	Nuclear Fuel Handling	42.9	37.9	-3.0
	226	Other Reactor Plant Equip.	26.6	26.6	-
	227	Instrumentation & Control	8.7	9.7	1.0
		Subtotal	340.6	403.8	64.2
23	Turbine Plant Equipment		78.0	78.0	-
24	Electric Plant Equipment		37.0	37.0	-
25	Miscellaneous Plant Equipment		10.0	10.0	-
26	Heat Reject System		37.0	37.0	-
	Total Direct		645.8	720.4	74.6
90	Indirect Costs		206.7	230.8	23.[illegible]
	TOTAL		852.5	951.2	9[illegible].[illegible]

*Auxiliary Cooling System is cheaper.

Table 1.2.4. Reference O&M Median Costs for 3000-MW(t) Gas Turbine Systems (1979 Dollars)

Plant Type	Costs ($10³) HTGR	PBR	Difference (PBR - HTGR)
Staff (215 persons at $23412)	5034	5034	-
Maintenance Material	1850	1850	-
Supplies and Expenses			
Fixed	3700	4400	
Variable*	466	466	700
Subtotal	4166	4866	700
Insurance and Fees	408	408	-
Admin. and General	1587	1587	-
Total Fixed Costs	12,579	13,279	700
Total Variable Costs	466	466	0
Total Annual O&M Costs	13,045	13,745	700

*At a plant factor of 1.00.

Table 1.2.5. Reference Fuel Cycle Parameters for 3000-MW(t) Gas Turbine Systems

	HTGR	PBR
Plant Design Parameters		
Thermal Power, MW(t)	3000	3000
Electric Power, MW(e)	1200	1200
Thermal Efficiency, %	40	40
Core Power Density, W/cm³	7.1	5.5
Fuel Cycle Parameters		
Initial core		
Number of Fuel Elements	4760	2,941,955
Mass of ^{235}U	1701	1388
Mass of ^{238}U	6804	5552
Mass of ^{232}Th	28216	18875
Equilibrium cycle		
Fuel Residence Time, Full Power Years	3.2	2.2
Carbon/HM Ratio	478	450
Mass Flow, kg/yr at Full Power		
^{235}U	946	893
^{238}U	3784	3571
^{232}Th	3635	6487
Total HM	8365	10951

The following parameters were treated probabilistically:

(1) Direct capital cost of control rods.

(2) Direct capital cost of control rod drives.

(3) Lifetime of control rods,

  (a) Helical-type, operation in bed.

  (b) Helical-type, operation above bed.

  (c) Thrust-type.

(4) Operation mode - fraction of helical rods required to operate in the bed.

(5) Replacement time/control rod.

The last parameter was included so that an assessment of the impact of control rod replacement on the assumed annual maintenance outage for the reactor could be made.

Figure 1.2.7 illustrates the effect of the PBR control rod replacement on the length of the annual outage. Whereas for the HTGR the refueling and control rod replacement time is predicted to lie entirely within the assumed 28-day turbine maintenance envelope (and hence for the HTGR the turbine maintenance is the critical path item), there is a small probability that for the PBR the control rod replacement will be the critical path item, resulting in a slight decrease in availability for the PBR.

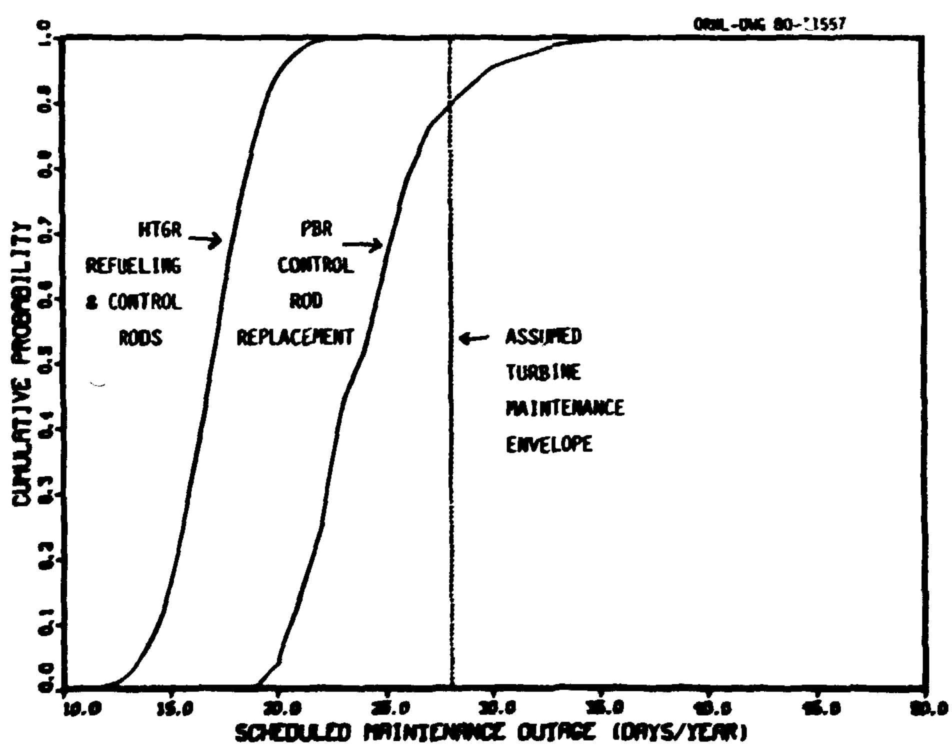

Fig. 1.2.7. Effect of Control Rod Replacement on Annual Downtimes of HTGR-GT and PBR-GT.

It should be noted that the correlation between the cost components must be accounted for in the analysis since in certain cases different cost categories contain common uncertainties and hence are not independent of each other. For the control rods, for example, the initial capital costs of the control rods and the replacement costs (O&M) are related: high initial capital costs imply high replacement costs. Also. the replacement time of the control rods (availability cost) and the replacement costs are correlated through the number of rods replaced. Although the expected value of the decision distribution is unaffected by the correlation, the existence of a positive correlation serves to reduce the dispersion of the probability distribution of the power cost.

Combining the probability distribution depicted by Fig. 1.2.7 with other probability distributions for control rod/drive direct costs, control rod replacement costs, rod lifetime, and operational mode (fraction of helical rods which must be operated in the bed of the reactor) allows the incremental power cost associated with the difference in instrumentation and control characteristics between the PBR and the HTGR to be calculated. The result is depicted in Fig. 1.2.8. As indicated, the expected total power cost of the PBR is anticipated to be 0.43 mill/kW-hr higher than that of the HTGR. Moreover, the distribution is skewed towards the higher values, essentially reflecting the unfavorable possibility of a high fraction of rods operating in the bed coupled with high capital and replacement costs.

The results of a similar analysis for the reflector differences is given by Fig. 1.2.9. The primary component of the distribution (the expected value) is the initial capital cost increment attributable to the PBR improved graphite. However, the distribution is highly nonsymmetric due to the inclusion of a small (10%) probability that the improved graphite may have to be replaced once during the operational lifetime. Such a possibility entails a significant availability penalty (up to 6 months) and thus represents a high-cost, low-probability event for the PBR - leading to the skewed shape of the curve.

In contrast to the skewed shapes of the curves showing the I&C effects and reflector differences, the probability curve representing the differences in the fuel form and fuel cycle has a relatively symmetrical shape (see Fig. 1.2.10). The negative expected value of -0.43 mill/kW-hr is attributable to a generic PBR fuel cycle advantage implicit in the mass flows given in Table 1.2.5. Moreover, this advantage is a net effect since it also presumes a 2-yr delay for the PBR (and subsequently higher U_3O_8 costs). The major contributor to the uncertainty in the fuel effect distribution is the underlying uncertainty of the escalation rate in the price of U_3O_8 due to its scarcity. Although not shown, the remaining incremental cost component, the increment attributable to power density differences, is also symmetric about an expected value of +0.42 mill/kW-hr.

The combination of the four probability distributions, each of which represents a specific category of differences between the two reactor systems, allows the overall PBR

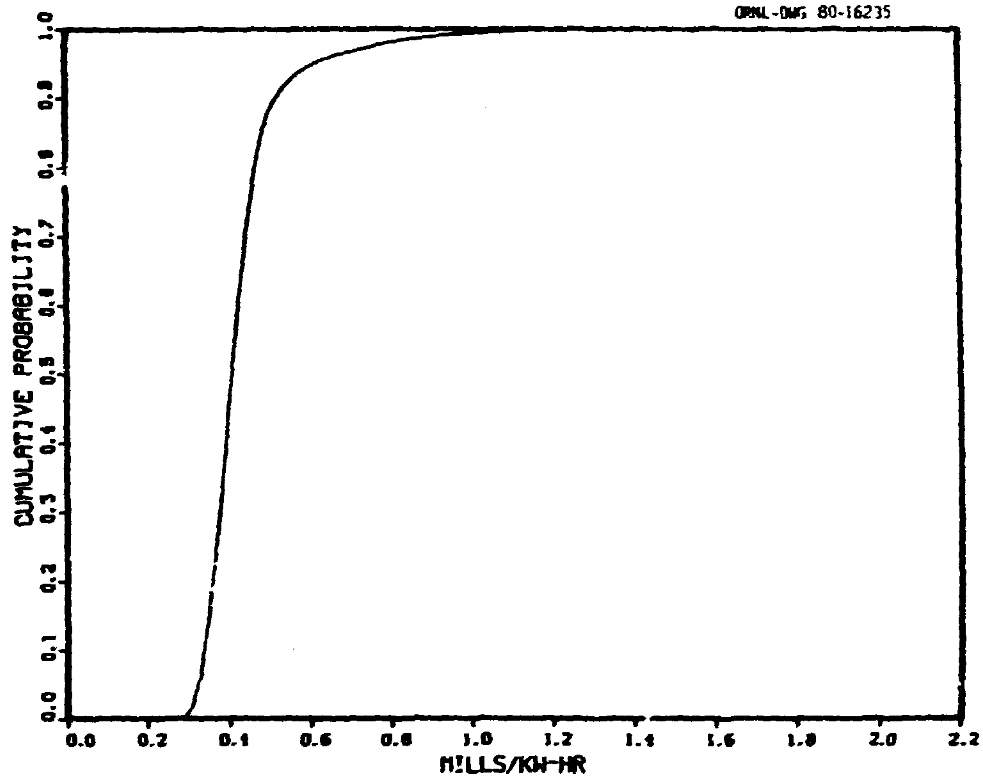

Fig. 1.2.8. Incremental Power Cost Increase for the PBR-GT Relative to the HTGR-GT Due to Differences in Instrumentation and Control Characteristics.

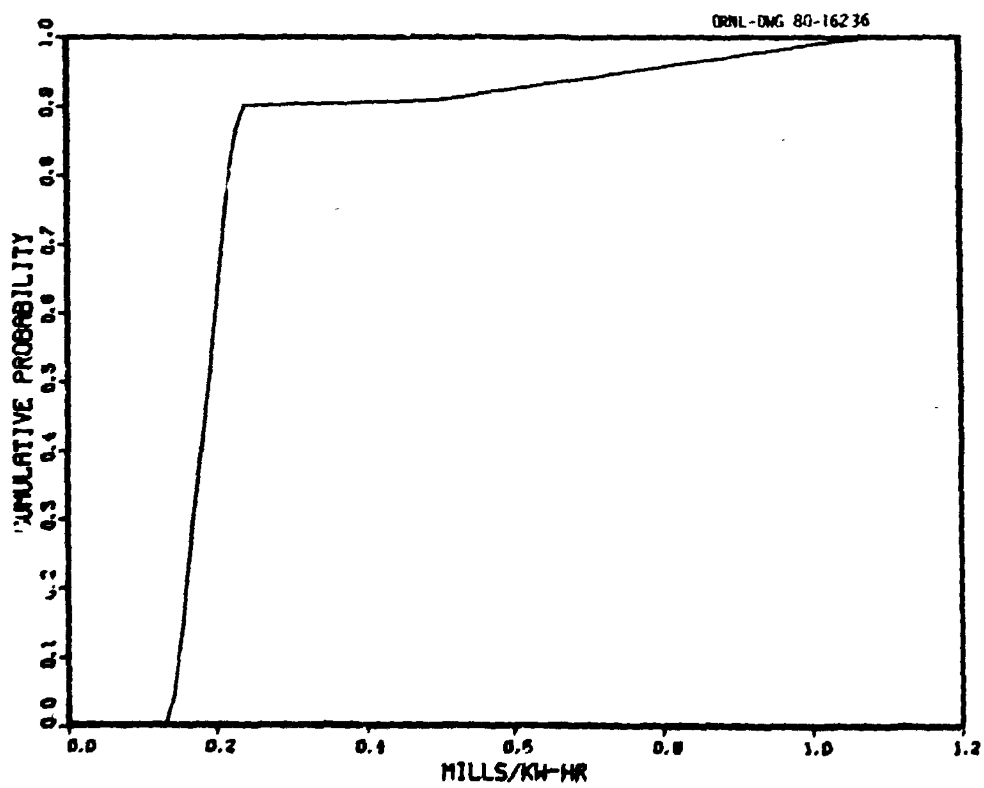

Fig. 1.2.9. Incremental Power Cost Increase for the PBR-GT Relative to the HTGR-GT Due to Reflector Fabrication and Replacement Differences.

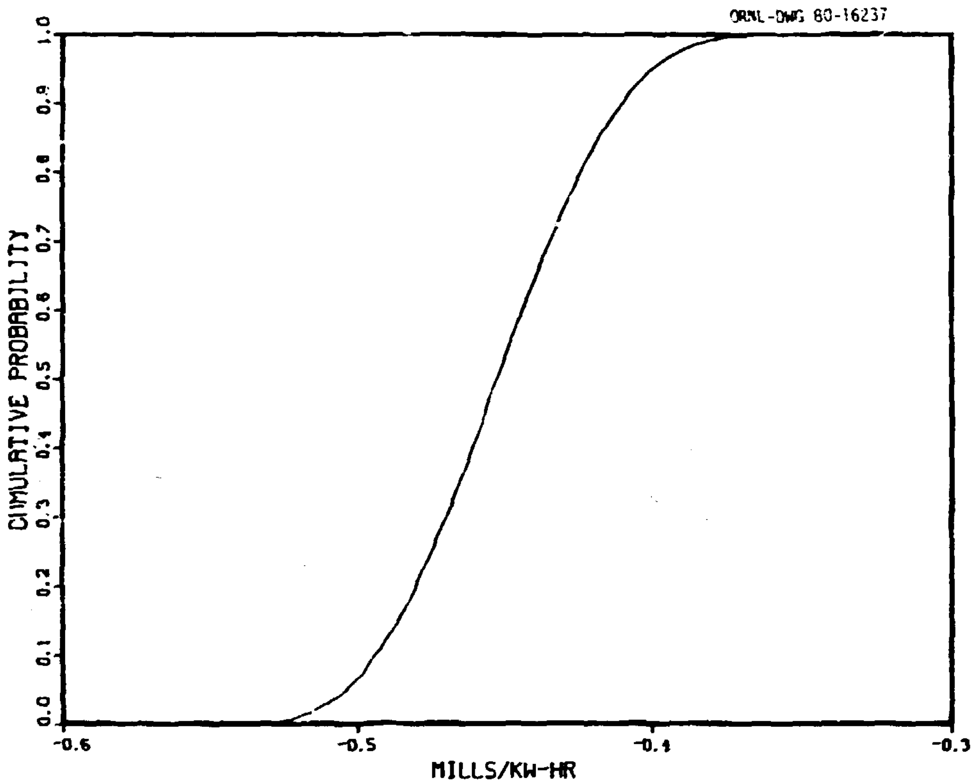

Fig. 1.2.10. Incremental Power Cost Increase for the PBR-GT Relative to the HTGR-GT Due to Differences in the Fuel Form and Fuel Cycle.

incremental power cost distribution to be calculated. The result is displayed in Fig. 1.2.11. The incremental distribution has an expected value of 0.66 mill/kW-hr and ranges from 0.08 mill/kW-hr to well over 3 mills/kW-hr. Thus the PBR power cost is anticipated to be slightly higher than that of the HTGR. Moreover, due to the use of incremental differences, the analysis also points out that, irrespective of the absolute cost of the power from the HTGR, which is shown in Fig. 1.2.5 to be about 20 mills/kW-hr (with a distribution range of 15.69 to 24.30 mills/kW-hr), the PBR is expected to be 0.66 mill/kW-hr more expensive. However, it should also be noted that there appears to be a 95% probability that the PBR incremental cost will be less than 1 mill/kW-hr.

Table 1.2.6 summarizes the contributions of the various effects, both by cost components and by design and operational differences. As indicated, the major cost category is the capital cost and of this approximately one-half is due to the lower power density of the PBR, which increases the costs for the PCRV and containment building. The more complex control system of the PBR (and the more severe operating environment of the control system) also leads to higher capital costs, as well as to higher operating and

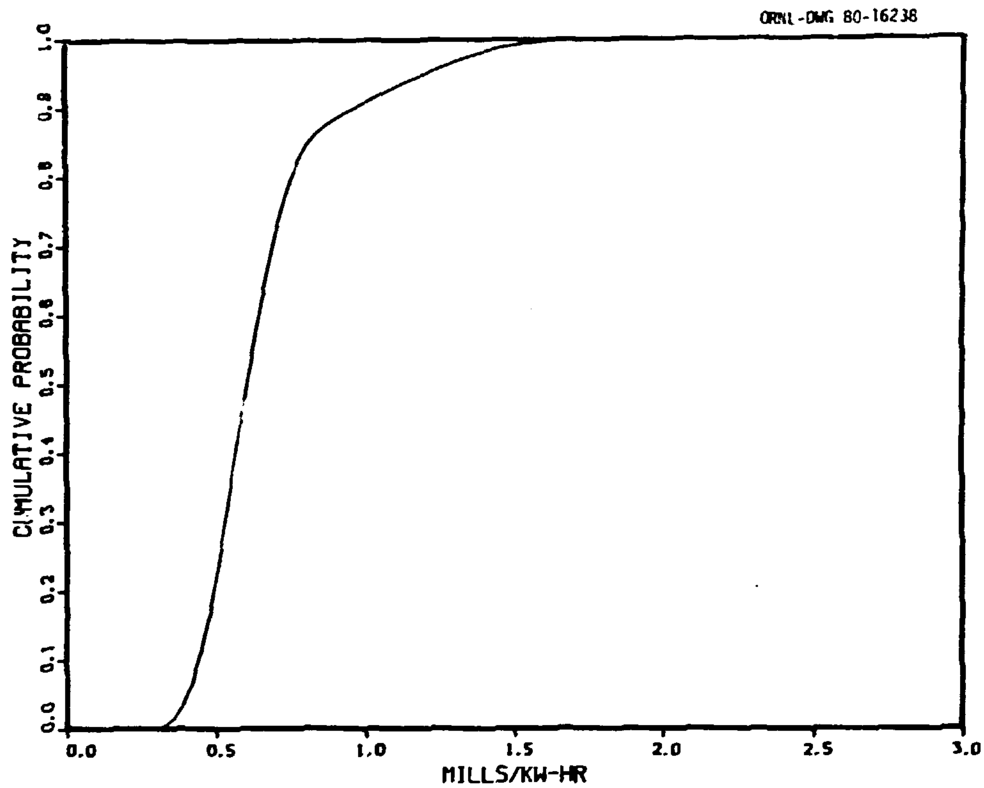

Fig. 1.2.11. Total Incremental Power Cost Increase for the PBR-GT Relative to the HTGR-GT.

Table 1.2.6. Breakdown of Incremental PBR Cost Differences (Expected Values) for 3000-MW(t)-GT Comparison

Design/Operational Difference	Cost Component (mills/kW-hr)				
	Capital	O&M	Fuel	Availability	Total
Power Density	+0.42	-	-	-	+0.42
I&C	+0.35	+0.06	-	+0.02	+0.43
Reflector	+0.17	+0.01	+0.01	+0.05	+0.24
Fuel Type	-0.04	-	-0.43	+0.04	-0.43
Total	+0.90	+0.07	-0.42	+0.11	+0.66

maintenance costs. Similarly, the permanent (or semipermanent) nature of the PBR radial reflector significantly impacts the capital cost component, and also contributes to a lesser extent to the other three cost components. Overall, the PBR fuel cost component is negative, but this advantage is not large enough to offset the three unfavorable cost components.

In terms of the design and operational differences, the power density, I&C, and fuel effects are equal in magnitude. The magnitude of the reflector effect is a factor of two smaller than the other three effects, primarily because the use of an improved graphite for the PBR was specified. This results in a significantly smaller effect on the power cost than does the use of standard graphite with periodic replacement.

The assumption that the PBR reflector is fabricated from a superior grade of graphite implies that it is advantageous to incur a higher capital cost in order to reduce the probability of incurring a nonavailability penalty for reflector replacement. In order to verify this assumption, the incremental power cost of the PBR was calculated neglecting the $10.5 million extra capital cost attributable to the advanced graphite but with the probability of replacement increased from a 0.1 to 1.0. The result is depicted in Fig. 1.2.12. As indicated, the expected power cost increment is increased (from 0.66 to 0.97 mill/kW-hr). Additionally, the perturbation results in a decrease in the dispersion of the possible power cost increment reflecting the change from an uncertain reflector replacement to a certain replacement. However, clearly from an expected value viewpoint, the use of upgraded graphite for the PBR radial reflector is the more advantageous choice.

3000-MW(t) Process Heat Systems

When the HTGR and PBR are applied in systems dedicated to process heat production (outlet coolant temperature of 950°C), two significant differences from the 850°C gas turbine base cases can be anticipated. First, if the process heat application is assumed to generate no electricity, or if the system is designed such that turbine maintenance can be performed during reactor operation (i.e., steam bypass of the turbine), the annual turbine maintenance envelope can no longer be considered the critical path that determines (in a majority of cases) the length of the annual scheduled outage. As indicated by Fig. 1.2.7, elimination of the turbine maintenance requirements would introduce a significant difference in the expected length of the scheduled maintenance outage for the two systems due to the PBR control rod replacement requirements, which translate into a lower availability for the PBR. In order to estimate the effect that this would have on the relative power costs of HTGR-PH and PBR-PH systems, the turbine maintenance requirements were eliminated from the base case (gas turbine) calculations. Figure 1.2.13 shows the resulting change in the probability cost difference distribution.

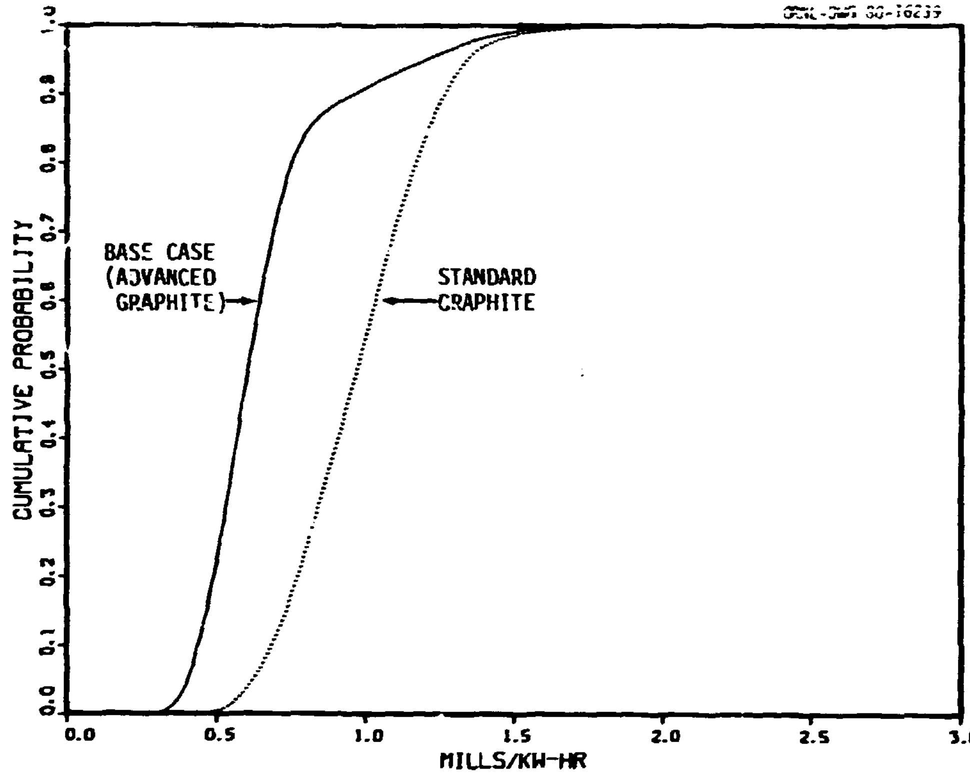

Fig. 1.2.12. Incremental Power Cost Increase for the PBR-GT Due to Substitution of Standard Graphite for Advanced Graphite in Reflector. (Base case curve is from Fig. 1.2.11.)

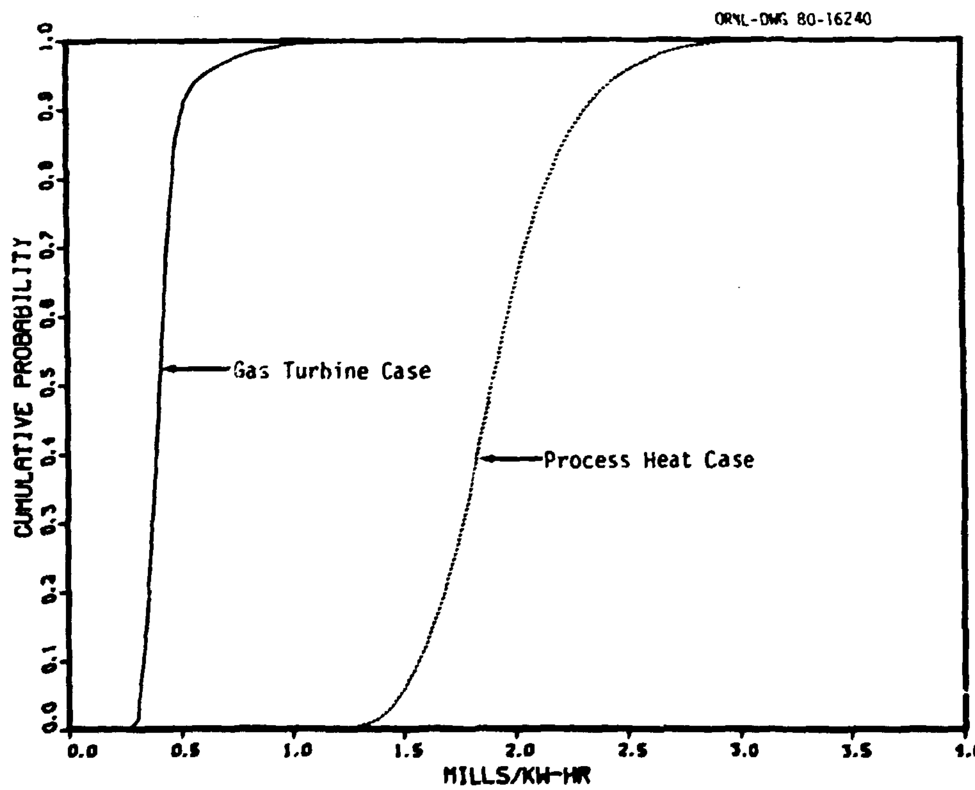

Fig. 1.2.13. Incremental Power Cost Increase for the PBR-PH Relative to the HTGR-PH Due to Differences in Instrumentation and Control Requirements as They Influence Reactor Availability. [Curve at left shows incremental increase of PBR-GT relative to HTGR-GT (see Fig. 1.2.8); curve at right shows effect of removing turbine maintenance requirements from base case.]

The second major difference is in the fission-product release rates of the HTGR and PBR cores at the 950°C process heat outlet helium temperature. As indicated in Section 1.1, at the 950°C temperature, the fission-product activity in the HTGR coolant circuit is expected to be significantly higher than that in the PBR circuit. It was assumed that for scheduled maintenance the increased circuit activity would not result in a large cost increment for the HTGR since extra shielding, revised maintenance procedures, etc. could be incorporated at the design stage of the reactor. However, if unscheduled (and hence unanticipated) maintenance on the components within the circuit were required, it is clear that the increased activity would impact both cost and availability. Due to the difficulty inherent in quantifying the impact of the increased activity based on the design detail available, this effect was modeled in a gross sense, again by a perturbation to the base case. Figure 1.2.14 represents the incremental cost (above that required for 850°C operation) attributable to the increase in circuit level activity. The distribution is intended to represent all costs incurred due to the activity level difference, that is, costs associated with increased personnel requirements, additional equipment, and increased downtime due to the necessity for remote and unplanned operations.

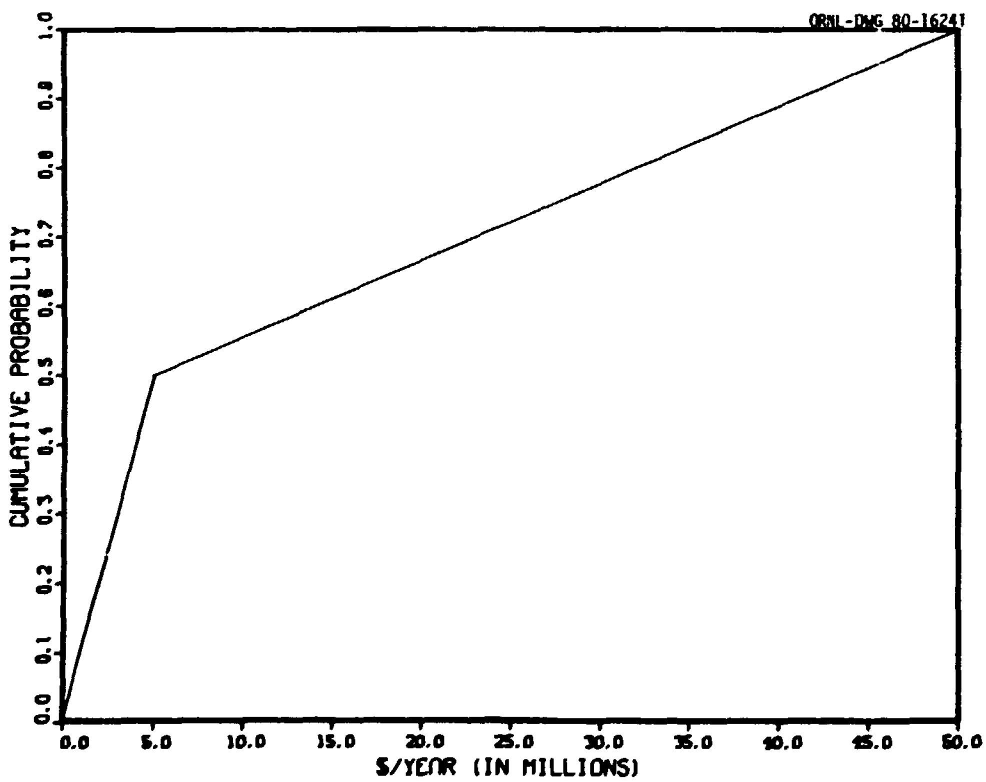

Fig. 1.2.14. Incremental Power Cost Increase for the HTGR-PH Relative to the PBR-PH Due to Increased Temperature-Induced Fission-Product Radioactivity in Coolant Circuit.

Figure 1.2.15 illustrates the combined results of these two perturbations on the incremental power cost for the PBR. The expected value is 0.43 mill/kW-hr, which represents a decrease from the base case value of 0.66 mill/kW-hr. However, it should be noted that the revised distribution incorporates a marked increase in uncertainty relative to the base case. The remarkable shape of the distribution can be traced to the dominance of the circuit activity cost curve (Fig. 1.2.14) for the negative values (HTGR disadvantage) as opposed to the dominance of the control rod effects (elimination of the turbine as the critical path item for the scheduled outage) for the positive values (PBR disadvantage). If a cogeneration plant should be specified so that the turbine-generator maintenance could not be bypassed, then only the fission-product effect would be significant, resulting in a negative expected value (PBR advantage) but with a large uncertainty in the distribution.

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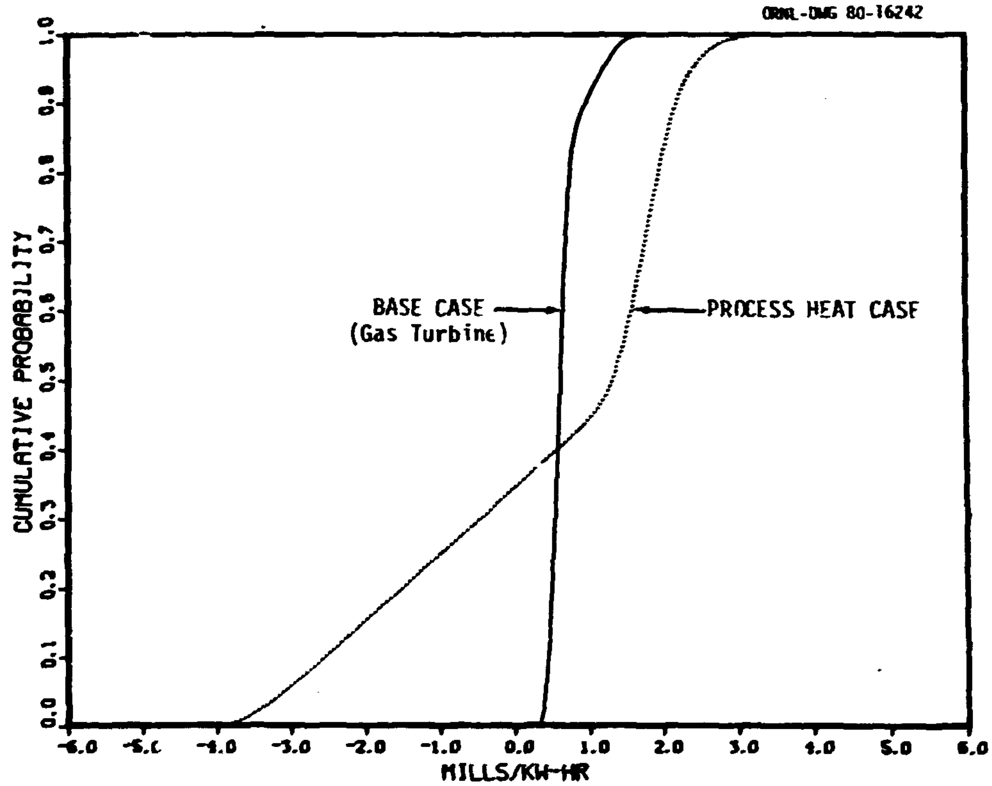

Fig. 1.2.15. Comparison of Incremental Power Cost Increases for the PBR Relative to the HTGR for Gas Turbine and Process Heat Applications.

References

1. "Probabilistic Comparative Study - Prismatic Fuel and Pebble Bed Gas Cooled Reactor Power Plants," Management Analysis Company, San Diego, California, May 31, 1980.

2. C. R. Hudson, II, "Concept-5 User's Manual," ORNL-5470 (January 1979).

1.3. OVERALL EVALUATION OF HTGR AND PBR SYSTEMS

P. R. Kasten

Section 1.2 has presented probabilistic cost evaluations for 3000-MW(t) HTGR and PBR systems operating on a MEU throwaway fuel cycle with a coolant outlet temperature of 850°C, plus estimates of the effect of increasing the outlet temperature to 950°C. In addition to these, more cursory evaluations were performed for all the systems considered in the study, and the results of these evaluations are included here. Specifically, the direct construction costs, the operating and maintenance costs, the fuel cycle costs, the unscheduled maintenance costs, and the research and development costs that would be incurred in deploying each of the systems are compared.

1.3.1. Capital Costs

Table 1.3.1 presents the estimated capital cost differences between the HTGR and the PBR for the 3000-MW(t) and 1000-MW(t) power levels: these cost differences are estimated to be largely independent of the coolant outlet temperature and thus apply to all three process applications (steam cycle, gas turbine, or process heat). The capital costs are given as the difference between the PBR and the HTGR, both in terms of a nominal range and a mean value.* For example, the range of the capital cost difference of the PCRV, liners, insulation, and containment building for the 3000-MW(t) systems is given as $25 million to $50 million (more expensive for PBR), while the mean value of the incremental construction cost increase of the PBR relative to the HTGR is $36 million. The range of the overall capital cost increase for the 3000-MW(t) PBR is $49 million to $112 million, with the mean incremental cost being $74 million. The corresponding range for the 1000-MW(t) systems is estimated to be $15 million to $46 million, the mean value being $28 million (PBR more expensive).

While the capital costs given in Table 1.3.1 are independent of the reactor application, there will, of course, be other capital costs that will differ with the application. However, the differential costs between PBRs and HTGRs as given in Table 1.3.1 would still apply. This is shown in Table 1.3.2, which gives estimated direct construction costs for steam cycle, gas turbine, and process heat applications of 3000- and 1000-MW(t) HTGRs, along with a nominal range for a developed industry. The incremental increases in direct construction costs for the PBRs are the same as in Table 1.3.1.

1.3.2. Operating and Maintenance Costs

The estimated differences in annual operating and maintenance costs between the two reactor concepts are given in Table 1.3.3. The net difference in O&M costs is a $530,000

*These range and mean values are not always consistent with the data used in the probabilistic analysis since the input data varied with time. However, the differences do not influence the the overall results obtained, and so it was not necessary to repeat the analyses.

Table 1.3.1. Estimated Direct Construction Cost Differences Between HTGR and PBR Systems* (Incremental Increases for PBR in 1979 Dollars)

	Range of Cost Increase ($\$10^6$)		Mean Value of Increase ($\$10^6$)	
	3000 MW(t)	1000 MW(t)	3000 MW(t)	1000 MW(t)
PCRV, liner, insulation, containment building	25-50	13-25	36	18
Costly graphite (POCO) to obtain full reactor life from inner reflector	8-15	3-5	11.5	4
Control rods and drives	23-39	8-13	28	10
Nuclear instrumentation	0-5	0-2	0	0
Fuel storage	3	1	3	1
Refueling/graphite replacement	-10-0	-10-0	-5	-5
Core auxiliary cooling systems	0	0	0	0
Total	49-112	15-46	74	28

*Direct cost differences are estimated to be largely independent of outlet coolant temperature.

Table 1.3.2. Comparison of Direct Construction Costs for Various Applications of HTGRs and PBRs* (1979 Dollars)

Outlet Temperature (°C)	Type System	HTGR Cost ($\$10^6$)				Cost Increase for PBR ($\$10^6$)			
		3000 MW(t)		1000 MW(t)		3000 MW(t)		1000 MW(t)	
		Ref.	Range	Ref.	Range	Ref.	Range	Ref.	Range
750	SC	640	608-704	346	329-381	74	49-112	28	15-46
850	GT	646	614-711	350	333-385	74	49-112	28	15-46
950	PH	767	729-844	420	400-462	74	49-112	28	15-46

*Excludes inflation, scheduling delays, and regulatory impacts; range covers -5% to +10% for HTGR.

Table 1.3.3. Estimated Operating and Maintenance Cost Differences Between HTGR and PBR* (Incremental Increases for PBR in 1979 Dollars)

	Range of Cost Increase ($\$10^3$/yr)		Mean Value of Increase ($\$10^3$/yr)	
	3000 MW(t)	1000 MW(t)	3000 MW(t)	1000 MW(t)
Control rod replacement costs	300-1400	100-500	530	170
Fueling valve replacement costs for PBR = Fueling machine maintenance costs for HTGR				

*Annual cost differences are estimated to be largely independent of outlet coolant temperature.

per year higher cost for the 3000-MW(t) PBR and a $170,000 per year higher cost for the 1000-MW(t) PBR. Again, the cost differences are estimated to be largely independent of the outlet coolant temperature, but the total costs will vary with the reactor application.

1.3.3. Fuel Cycle Costs

With regard to fuel utilization, the HTGRs and PBRs are not significantly different for comparable conditions, but the PBR tends to have a slightly higher fuel conversion ratio because of its on-line fueling feature. This higher conversion ratio holds for both once-through and recycle conditions, but more so for recycle. Based on present technology and zoning limitations, the fuel loading in an HTGR can be slightly higher than in the PBR, but this leads to about the same conversion ratio in the two reactors at the same fuel exposure under fuel recycle conditions. In the long term, however, the HTGR appears to be more limited in its ability to go to higher fuel loadings; the PBR probably can have 15 to 20% higher fuel loadings (in terms of kilograms of heavy metal per unit volume). This is important only under fuel recycle conditions, however. Thus, the fuel cycle advantages of the PBR are most significant under fuel recycle conditions and at high U_3O_8 prices.

The fabrication of PBR fuel is estimated to be 15 to 20% less expensive than the fabrication of HTGR fuel under the reference conditions. However, the refabrication of PBR fuel appears to be about 5 to 10% more expensive than HTGR fuel. Reprocessing costs of PBR and HTGR fuel are essentially the same for equivalent conditions. For the once-through fuel cycle and reference conditions, the fuel cycle costs of the PBR are about 0.4 mill/kW-hr less than for the HTGR. Under fuel recycle conditions, the PBR cost advantage would increase probably to about 0.6 to 0.8 mill/kW-hr. Also, if the HTGR were to go to a two-year interval between refuelings instead of a one-year interval, or if the HTGR were to use a three-year fuel cycle time instead of a four-year cycle, the fuel cycle cost of the HTGR would rise relative to the PBR.

The relative fuel cycle costs of HTGRs and PBRs are essentially the same for 1000-MW(t) plants as for 3000-MW(t) plants. While the HTGR gains slightly at the lower power level, the gain is insignificant. Overall, the higher capital costs of the PBR for the reference conditions more than offset the higher HTGR fuel cycle costs.

1.3.4. Unscheduled Maintenance Costs

With regard to the effect of unscheduled maintenance downtimes on relative performance, only at the 950°C outlet coolant temperature is there a significant difference (favoring the PBR), with a large uncertainty in that difference (ranging from insignificant to very significant). Thus, for the 950°C outlet coolant temperature, it is not clear whether the HTGR or the PBR has an advantage in power costs.

1.3.5. Research and Development Costs

The above discussion has not taken into consideration the differences in the R&D costs associated with developing the two concepts. Table 1.3.4 summarizes estimates of the R&D expenditures (base technology plus equipment development) required for HTGRs and PBRs as a function of outlet coolant temperature. The differences are based on estimates of the incremental PBR R&D costs; the incremental R&D difference is estimated to have a mean value of $145 million (higher for the PBR), with a range of $100 million to $200 million. The higher PBR costs are on the basis that U.S. vendors will furnish the PBRs. The R&D required would be largely independent of whether 3000- or 1000-MW(t) commercial plants were being considered. As shown in Table 1.3.4, the plant R&D increases with increasing coolant temperature. This is primarily due to increased costs of developing materials and of associated equipment; for the 850°C case, the gas turbine development requirements were also included. In all cases, the incremental R&D costs for the PBR over the HTGR were the same; however, the different cases give perspective on overall R&D costs.

The performance of either the HTGR or the PBR is enhanced by reprocessing and recycle of spent fuel. The estimated R&D expenditures for basic recycle technology and for a fuel recycle pilot plant are summarized in Table 1.3.5; no significant differences are identified between HTGR and PBR fuel recycle development.

1.3.6. Alternative Reactor Designs

Although this study did not specifically evaluate the potential of alternative reactor designs, the results obtained indicated certain effects. For example, by going to higher core power densities and deeper bed depths in PBRs, PCRV and containment costs could be brought back to HTGR levels. However, HTGRs could also increase their power density at the expense of higher fuel temperatures. Further, it appears that deeper bed depths in PBRs would make control-rod insertion more difficult, as well as increase the incremental control system costs relative to those of HTGRs.

Use of annular cores in PBRs appears to show some potential in alleviating control and instrumentation problems but tends to lead to lower nuclear performance and higher fuel cycle costs. The annular core permits improved PCRV head support in the middle of the reactor, but this type design is not unique to PBRs.

A decrease in HTGR fuel temperatures appears possible through use of reactor cores with higher core pressure drops. A proposed use of thorium blankets to protect the graphite reflector from high fluences would lead to a significant reduction in outlet coolant temperature for a given maximum fuel temperature, and thus does not appear to have an overall advantage. On the other hand, use of twice-through fueling rather than once-through-then-out fueling in the PBR would tend to decrease graphite reflector fluences and to reduce the irradiation damage to control rods above the core. However, twice as much fuel handling would need to be carried out.

Table 1.3.4. Estimated Research and Development Expenditures for HTGRs and PBRs* (1979 Dollars)

Reactor System/Cost Category	R&D Costs ($\$10^6$) HTGR	PBR
Steam cycle, $\bar{T}$ = 750°C		
Base technology	200 - 250	300 - 450
Plant equipment	100 - 150	100 - 150
Total	300 - 400	400 - 600
Gas turbine, T = 850°C		
Base technology	250 - 400	350 - 600
Plant equipment	200 - 400	200 - 400
Total	450 - 800	550 -1000
Process heat, T = 950°C		
Base technology	400 - 600	500 - 800
Plant equipment	200 - 400	200 - 400
Total	600 -1000	700 -1200

*Cost estimates refer to costs required above vendor/utility commercial investments; R&D costs are same for 1000- and 3000-MW(t) plants.

Table 1.3.5. Estimated Research and Development Expenditures for HTGR or PBR Fuel Recycle Systems (1979 Dollars)

Cost Category	R&D Costs ($\$10^6$)
Base technology*	500 - 800
Pilot plant	900 - 1300
Total	1400 - 2100

*Includes 5-8 years operating cost/interactions with pilot plant.

1.4. CONCLUSIONS AND RECOMMENDATIONS

On the basis of the information available, it appears that high-temperature gas-cooled reactors based on either the HTGR design concept or the PBR design concept can be successfully commercialized in the U.S. However, since the development of HTGR systems has been under way in the U.S. for some time, whereas the development of PBR systems has been carried out primarily in the Federal Republic of Germany, a larger R&D program would be required to bring the PBR on line in the U.S. The increased cost of the PBR R&D program over that of an HTGR R&D program is estimated to be $100 million to $200 million, and the increased time is estimated to be up to four years.

Under the assumption that either reactor could be developed, the HTGR and PBR were evaluated and compared in this study on the basis of their relative economic performance when they were applied in comparable systems for the production of electricity (steam cycle or gas turbine systems) or the production of process heat. The criterion for the economic performance was the overall energy production costs, which consisted of four cost components: capital costs, fuel costs, operation and maintenance costs, and nonavailability penalties. In calculating these costs, the impact of design and operating differences between the two types of reactors on the individual cost components was determined.

The comparative evaluation began with an examination of key technical issues to identify the most important design and operating differences. The results of these studies can be summarized as follows:

(1) The lower power density (i.e., larger core) of the PBR requires a larger PCRV and containment building, which contributes to higher PBR capital costs.

(2) The more complex control system of the PBR, together with the more severe environment in which the control rods must operate, contributes both to higher PBR capital costs and to increased maintenance costs (to replace control rods).

(3) Replacing the PBR control rods requires more time than replacing the HTGR control rods plus refueling; however, in most cases the times required for these activities are less than the times required for turbine-generator maintenance. Thus their relative times are important only for those systems for which the turbine maintenance is not the critical path.

(4) The requirement for a permanent PBR radial reflector to avoid a high nonavailability penalty for reflector replacement mandates the development of a superior grade of graphite and increases the PBR capital costs.

(5) The on-line refueling feature of the PBR results in a slightly better neutronic performance (higher fuel conversion ratio) and a smaller initial commitment of U_3O_8 and separative work for the PBR, an advantage that becomes more important with fuel recycle.

(6) Invoking the secondary shutdown system results in a significant fuel cycle penalty for the PBR, especially for throwaway cycles.

(7) For process heat systems with high outlet coolant temperatures, the HTGR may have a higher cost penalty associated with unscheduled maintenance shutdown than the PBR since for high operating temperatures the fission-product activity levels will probably be higher in the HTGR. However, the effect of circuit activity level on the maintenance and nonavailability costs is very uncertain.

The above differences were considered in a probabilistic analysis performed for reference HTGR and PBR systems [3000 MW(t), direct gas turbine, 850°C outlet temperature, and MEU fuel on a once-through cycle]. With 15-year levelized power costs (in mills/kW-hr) used as the criterion, the analysis yielded an expected power cost of 19.92 mills/kW-hr for the HTGR, with a probability distribution ranging from 15.69 to 24.30 mills/kW-hr. For the PBR, the expected power cost was 0.66 mill/kW-hr higher. The largest component of this incremental increase was +0.90 mill/kW-hr for increased PBR capital costs, which was partially offset by a -0.42 mill/kW-hr fuel cycle advantage of the PBR over the HTGR. The other components were +0.07 mill/kW-hr for higher operation and maintenance costs and +0.11 mill/kW-hr for a higher nonavailability penalty.

The increased PBR capital costs were due to the larger PCRV and containment building (+0.42 mill/kW-hr), the more complicated control system (+0.35 mill/kW-hr), and the assumption that a superior reflector graphite material would have to be developed (+0.17 mill/kW-hr). Substituting a standard grade of graphite for the PBR reflector reduced the capital costs but increased the PBR nonavailability penalty to the extent that the incremental increase in the PBR power cost was 0.97 mill/kW-hr rather than 0.66 mill/kW-hr. *Thus, in both cases the probabilistic analysis indicated that economically the HTGR may be slightly superior to the PBR for the gas turbine application, and the same conclusion would hold for the steam cycle application.*

For process heat systems with an outlet coolant temperature of 950°C, the costs associated with the possible higher fission-product activity in the HTGR coolant must be balanced against the increased downtime for replacing PBR control rods. This was done cursorily by introducing both effects as perturbations in the probabilistic analysis for the reference systems. As a result, the expected value of the incremental increase in the PBR costs over the HTGR costs was reduced to 0.43 mill/kW-hr. However, the distribution about the expected value had a wide variance, ranging from a PBR advantage to an HTGR advantage. *The net result was no apparent preference between the two reactors for process heat systems.*

On the basis of these probabilistic results for the reference systems and the overall evaluation of all the systems considered in this study, *it is recommended that primary support of high-temperature gas-cooled reactors in the U.S. be given to the HTGR concept.*

Key issues still to be resolved for the HTGR are (1) fuel performance (i.e., fission-product retention) as a function of temperature, temperature gradient and irradiation exposure, and (2) maintenance costs as a function of coolant circuit activity.

It is also recommended that the U.S. maintain a cooperative FBR program with FRG, emphasizing work in the key areas of reactor control and instrumentation requirements.

CHAPTER 2

INPUT DATA FOR PROBABILISTIC ANALYSIS

B. A. Worley and T. J. Burns

Outline

2.0. INTRODUCTION

Since the total power cost was used as the figure of merit in this analysis, all data incorporated into the analysis were either costs or data from which costs could be calculated. In this chapter an effort is made to document all the cost data actually used in calculating the probabilistic distributions for 3000 MW(t) PBR and HTGR total plant costs. In most cases the data were arrived at through studies of key technical issues, which are described in Chapter 3. However, in some cases time limits or lack of a sufficient technical base precluded detailed studies, and the corresponding input data were selected on the basis of judgment. In the analysis, the uncertainty in the data was considered through the use of probability distributions. For those cost categories that were interdependent, the total costs of the categories were summed and a probabilistic distribution was estimated for the total. An example is the capital costs of the PCRV and the containment building, whose sizes are interdependent and both of which depend on the cost of concrete.

It should be noted that a comparison of this chapter with Chapter 3 will reveal that the data used in the decision-making analysis do not always correspond exactly to the data given in Chapter 3. This is because the tight time schedule for the analysis sometimes necessitated the selection of input data before the detailed studies were completed and it is these data that are included here. Chapter 3, on the other hand, presents the results of the detailed studies updated to the time of publication of this report. Where differences exist, use of the updated data would not change the essential results of the comparative analysis.

2.1. REACTOR CONTROL SYSTEM COSTS

2.1.1. Direct Capital Costs of Control Systems

The large differences between the PBR and HTGR control requirements result in significant differences in the costs of both the control rods and the control rod drives for the two types of reactors (see Sections 3.1.3 and 3.1.3). For the HTGR, a large technical base exists and relatively firm estimates can be given. The reactor will utilize 168 control rods and 84 control rod drives (two rods per drive), for which the capital costs are summarized in Table 2.1.1.

Table 2.1.1. Capital Costs for HTGR Control Rods and Control Rod Drives

	Unit Costs ($\$10^3$)	Total Costs ($\$10^6$)
Control rods (168)	14	2.4
Control rod drives (84)	100	8.4
Total		10.8

The estimated costs for the PBR control rods are higher: \$25,000 per thrust-type rod and \$30,000 per auger-type rod. For a total of 151 rods, it is assumed that 105 rods will be the thrust type (at a total cost of \$2,625,000) and 46 will be the auger type (at a total cost of \$1,380,000). The estimated unit costs for the PBR control rod drives (one per rod) are also higher since the PBR backup capability must include a driving force for the thrust or rotating action of the rods, whereas the HTGR backup system is based on rod drop by gravity. The estimated costs for the PBR control rod drives are \$200,000 for for the translation or rotating action of the rods, whereas the HTGR backup system is based on rod drop by gravity. The estimated costs for the PBR control rod drives are \$200,000 for the thrust-type and \$300,000 for the auger-type. Thus, the control rod drive probabilistic distribution of the costs was used in the analysis (see Table 2.1.2).

Table 2.1.2. Probabilistic Estimates of Costs for PBR Control Rods and Control Rod Drives

	Auger Rods		Thrust Rods	
	Cost ($\$10^3$)	Probability	Cost ($\$10^3$)	Probability
Cost for Control Rod	25	0.1	20	0.1
	27.5	0.2	22.5	0.2
	30	0.4	25	0.4
	37.5	0.2	27.5	0.2
	45.0	0.1	30	0.1
Cost for Control Rod Drive	250	0.1	150	0.1
	275	0.2	175	0.2
	300	0.4	200	0.4
	325	0.2	225	0.2
	350	0.1	250	0.1

2.1.2. Cost of Control Rod Replacement

Given the unit cost for a control rod, the direct cost for additional rods was determined by the total number to be replaced during the reactor lifetime, which, in turn, was based upon the projected practical lifetime of a control rod.

For the HTGR, the estimated average lifetime of a control rod was assumed to be 7 years (see Sections 3.1.3 and 3.7.3). If one-seventh of the 168 control rods are replaced during each shutdown for annual refueling, the lifetime for each rod will be 7 years. Thus 24 control rods will have to be replaced on an annual basis.

The number of PBR control rods to be replaced on an annual basis is difficult to ascertain since the control rod lifetime is dependent upon the number of times (if any) a rod must penetrate into the fueled pebbles from the gas space above the core. The thrust-type rods are assumed to penetrate the bed of pebbles only during shutdown under normal operation. Therefore the lifetime of the thrust-type rods was estimated to be greater than that for the auger rods and comparable to that of an HTGR control rod (7 years). On the basis of the discussion in Section 3.1.3, it was assumed that the auger rods would have an expected lifetime of 2 years if they had to penetrate the core and an expected lifetime of 4 years if they did not have to penetrate the core. The probability distributions for the control rod lifetimes in a PBR are given in Table 2.1.3

Table 2.1.3. Probability Distribution of Control Rod Lifetimes in a Pebble Bed Reactor

	Lifetime (Years)	Probability
Thrust Rods (penetrate pebble core only during shutdown)	4	0.05
	6	0.45
	8	0.45
	10	0.05
Auger Rods (penetrate pebble bed)	1	0.1
	2	0.8
	3	0.1
Auger Rods (do not pentrate pebble bed)	3	0.1
	4	0.8
	5	0.1

The fraction of auger rods that penetrate the pebble bed core was assumed to have the probability distribution given in Table 2.1.4.

Table 2.1.4. Probability Distribution for Fraction of Auger Rods Penetrating PBR Core

Fraction of Penetrating Rods	Probability
0.23	0.03
0.29	0.17
0.37	0.60
0.45	0.17
0.51	0.03

In the analysis this probability distribution was combined with the one given in Table 2.1.3 to determine a probability distribution for the average number of control rods replaced per year in a PBR.

2.1.3. Times Required for Control Rod Replacement

Removal and replacement of the control rods and control rod drive assemblies in the HTGR would be accomplished in parallel with refueling such that only 5 hours would be added to the refueling time (see Section 3.1.4). But even this extended time would be less than the time required for the turbine-generator maintenance (see Fig. 3.7.1); thus refueling and control rod replacement probably would not affect the availability of the HTGR.

The situation is different for the PBR. The time required for control rod replacement in the PBR is estimated to be greater by approximately a factor of two than the time required for refueling and control rod replacement in the HTGR (Table 3.7.4). Moreover, the amount of time required for control rod replacement in the PBR can be such that the turbine-generator maintenance is not the critical path for the required shutdown time (see Fig. 3.7.2). Therefore, for the PBR, the replacement time per rod and the average number of rods replaced per year must be determined in order to predict reactor unavailability due to scheduled shutdowns. From the information given in Section 3.1.4, the average number of hours required for replacing one control rod in the PBR is 13 (298/23). Therefore, the probability distribution given in Table 2.1.5 was assumed for control rod replacement time in the PBR.

Table 2.1.5. Probability Distribution for Control Rod Replacement in a Pebble Bed Reactor

Replacement Time Per Rod (days)	Probability
0.50	0.10
0.55	0.50
0.60	0.30
0.65	0.10

2.2. FUEL CYCLE COSTS

The 30-year levelized fuel cycle cost was calculated with the cost model described in ref. 1 and the cost assumptions listed in Table 1.2.1. The cumulative probability distribution for the U_3O_8 price escalation rate used in calculating fuel cycle costs is shown in Fig. 1.2.3.

For calculation of the fuel cycle costs of the throwaway cycles considered in this study, one must know the initial core uranium and thorium loadings, the subsequent ^{235}U, ^{238}U, and ^{232}Th reloads (makeup fuel), and the total heavy metal discharges. These data are given in Table 2.2.1 for the 3000-MW(t) PBR and HTGR and were taken directly from data in Section 3.2.

Table 2.2.1. Mass Balance Data for Reference 3000-MW(t) HTGR and PBR on MEU Throwaway Fuel Cycles

	HTGR	PBR
Initial Core Inventory[a] (kg)		
^{235}U	1,701	1,388
^{238}U	6,804	5,552
^{232}Th	28,216	18,875
Equilibrium Loadings[b] (kg)		
^{235}U	946	893
^{238}U	3,784	3,571
^{232}Th	3,635	6,487
Equilibrium Discharge[b] (kg)		
Total Heavy Metal	7,249	9,652

[a]Initial core loadings shown are the fuel requirements for operation during first year of reactor life at a load factor of 0.8.

[b]Equilibrium mass flows are for one year of full power operation.

2.3. PCRV AND CONTAINMENT BUILDING COSTS

In the analysis the increases in the costs of the pressure vessel (PCRV) and the containment building for the PBR over those for the HTGR were added to the base HTGR capital costs given in Section 2.9. As shown in Table 2.9.1, the capital cost three-digit breakdown levels are such that the PCRV structure, its liners and penetrations, and the containment building are listed separately. Based upon the detailed study of the PCRV and containment building costs discussed in Section 3.3, the range of the increase in these cost categories for a 3000-MW(t) PBR is as shown in Table 2.3.1. The cost ranges of the three items are not independent since the quantity of material for one item depends upon the design of the other two and since they all depend upon the assumed unit cost of concrete. Also, the values shown here are based on the estimated mean value for installing concrete, with the range associated with uncertainties in the PCRV design of the PBR. In the comparative evaluation the three items were treated as a single cost category, for which the probabilistic estimates of cost are as shown in Table 2.3.2, and consider a range of installed concrete costs.

Table 2.3.1. Range of Capital Cost Increase for PBR PCRV and Containment Building*

Cost Category	Cost Increase ($$10^3$)
Containment building	5,912-10,026
PCRV structure	15,258-22,961
PCRV liners and penetrations	2,228-4,050
Total	23,398-37,037

*Based on cost of installed concrete of $500 per cubic yard.

Table 2.3.2. Probabilistic Estimates of Combined Cost Increases for PBR Containment Building, PCRV Structure and PCRV Materials*

Cost Increase ($$10^6$)	Cumulative Probability
25.0-36.0	0.0-0.5
36.0-50	0.5-1.0

*Costs listed increase linearly over ranges for cumulative probabilities.

2.4. FUEL FABRICATION UNIT COSTS

The fuel fabrication unit cost estimates for the MEU throwaway cycle are taken directly from the detailed study described in Section 3.4 and are shown in Table 2.4.1. A probabilistic distribution of unit cost is not assumed since the total fuel cycle cost is dominated by the U_3O_8 escalation rate for which a distribution is assumed (Fig. 1.2.3).

Table 2.4.1. PBR and HTGR MEU Fuel Fabrication Costs

Reactor	$/kg HM
PBR	1170
HTGR	1380

2.5. COSTS RELATED TO FISSION-PRODUCT RELEASES

Costs due to nominal fission-product releases to the coolant are included in operation and maintenance costs and are not considered to be different for HTGRs or PBRs. In all cases, the input to the comparative evaluation accounting for the release of fission products was limited to the *increase* in unscheduled maintenance as a result of increased coolant circuit radioactivity. The releases, in turn, were limited to routine releases; that is, accident-related fission-product releases were not considered to be significantly different between HTGRs and PBRs.

Unscheduled maintenance costs are difficult to estimate by their very nature. Given a probability of occurrence of unscheduled maintenance and a nominal cost range within which this maintenance cost is expected to vary, it is assumed that unscheduled maintenance costs will increase above the values within the nominal range as the circuit activity level increases. The additional expenditures for higher circuit activity levels are due to increased downtime and additional personnel and equipment costs.

In order to quantify costs relative to circuit activity, the analyses given in Section 3.5.5 were utilized. Based on those analyses, the penalty associated with increased circuit activity is very uncertain, and could range from zero to $50 million per year for the HTGR with an outlet coolant temperature of 950°C; the mean value was estimated to be $5 million per year. However, for outlet coolant temperatures of 750°C and 850°C, no unscheduled maintenance penalty was estimated for the HTGR. For an outlet coolant temperature of 950°C, the increase in unscheduled maintenance cost is estimated to have the probabilistic distribution given in Table 2.5.1.

For the PBR, no maintenance penalty due to increased circuit activity was found at any of the three outlet temperatures because of the lower fission-product release associated with that system.

Table 2.5.1. Relative Increase in HTGR and PBR Annual Unscheduled Maintenance Costs for Coolant Outlet Temperature of 950°C

Cost Increase* ($10^6)		
HTGR	PBR	Cumulative Probability
0-5.0	0	0.0-0.5
5.0-50.0	0	0.5-1.0

*Cost increases linearly over the range of cumulative probability.

2.6. OPERATION AND MAINTENANCE COSTS

The operation and maintenance costs projected for the 3000-MW(t) HTGR are taken directly from ref. 2 and are listed in Table 1.2.4. The corresponding costs for the PBR are expected to differ only for the capital cost of the control rods being replaced during reactor lifetime. As discussed in Section 2.1, the replacement rate and cost of control rods for the PBR are projected to be greater than for the HTGR. The actual difference of the control rod replacement cost on a yearly basis between the PBR and HTGR was calculated directly from the data in Section 2.1.2.

2.7. UNAVAILABILITY COSTS

The cost of the power produced by a plant is dependent upon the time the reactor is available to provide the power. When the reactor is not available, not only do routine O&M expenses continue, but also replacement energy supplies must be purchased. In the comparative analysis, these costs were factored in the appropriate cost categories and were based on the following analyses of availability for the two reactor types.

The information in Section 3.7 provides the basis for estimating the difference in expected availability between a plant with an HTGR core and one with a PBR core. The base case considered in this evaluation was one for which the turbine maintenance was assumed to require 26 days for a total shutdown period of 28 days including depressurization and pressurization. For the discussion in Section 3.7.3, it is noted that 26 days will be greater than the time required for refueling and control rod replacement in the HTGR, so that the unavailability per year of the HTGR is fixed at 28 days. However, the PBR unavailability may exceed 28 days per year, depending upon the time required for replacing the control rods (refer to Section 3.7.3). From the probabilistic distribution data in Section 2.1.1 concerning rod types used, rod lifetimes, and replacement times, the probability that the PBR unavailability will exceed 28 days per year can be calculated. For the reference case, the resulting probability distribution for PBR unavailability in days per year is illustrated in Fig. 1.2.7. Note that the turbine maintenance envelope leads to at least a 28-day downtime for the PBR with a 10% probability that the downtime will exceed 28 days.

For evaluation of process heat applications, it was considered that the plant turbine-generator could be bypassed in order to provide high plant availability to process heat applications. For this case the unavailability due to scheduled maintenance will be dominated by the time required for refueling and control rod replacement (see Fig. 3.7.1). The PBR control rod replacement time is discussed in the preceding paragraph. The probabilistic distribution for scheduled shutdown for the HTGR was calculated by assuming a normal distribution for the time required to replace one fuel assembly. The normal distribution used in the analysis has a most probable value of ~20 min/assembly and maximum and minimum values of 50% higher and lower, respectively. The probabilistic distribution for the total scheduled maintenance time was calculated by multiplying the replacement time per assembly by the number of assemblies replaced during each shutdown period and adding the times required for other service requirements shown in Fig. 3.7.1. The calculated scheduled maintenance outage probability distributions for the reference HTGR and PBR are shown together in Fig. 1.2.7.

The influence of high coolant circuit activity can also influence reactor availability; this is discussed in Section 3.5.5 and presented in Table 2.5.1 in terms of annual cost differences between HTGRs and PBRs.

2.8. GRAPHITE REFLECTOR COSTS

It was assumed in the analysis that the costs of HTGR and PBR reflectors using the same grade of graphite would be approximately equal. On the basis of information provided in Section 3.8.3, the capital cost penalty for using an improved graphite in the PBR was estimated to be $10.5 million. The probabilistic distribution used in the analysis in order to reflect the uncertainty in the estimate is given in Table 2.8.1.

Table 2.8.1. Probabilistic Estimates of Cost Increase for Improved Grades of Graphite in PBR Reflectors

Additional Reflector Cost ($10^6)	Cumulative Probability
7.0-10.5	0.0-0.5
10.5-14.0	0.5-1.0

2.9. CAPITAL COST DATA

Plant capital costs for the prismatic HTGR have been calculated in various prior studies. By contrast, the plant capital costs for the PBR are difficult to calculate given currently available design details.

The lack of information on the PBR was circumvented by determining the capital cost of the reference HTGR plants (see Section 3.11) and then estimating the change in cost for various capital cost categories for a plant with a pebble-bed core. Cost differences were estimated only for those cost categories in which the differences in design were expected to impact the cost differential significantly – i.e. PCRV costs, control equipment costs, containment building costs, etc.

The capital costs used in the evaluation were estimated to the three-digit level and are listed in Table 2.9.1. This level of breakdown in cost categories was essential for determining cost differentials between the concepts. The cost categories designated as "different" for the PBR were projected to have costs that deviate substantially from those for the HTGR.

As was explained in Section 2.3, the estimated differences between the PBR and HTGR for the reactor containment, PCRV structure, PCRV liners and penetration, and PCRV internals and insulation were interdependent and therefore treated as a summation of costs. The difference between the reactor control mechanism capital costs for the two systems was calculated from the data by a technique described in Section 2.1.

The fuel storage building cost was projected to be greater for the PBR than for the HTGR because the PBR core fuel volume is greater. The fractional volume increase is proportional to the ratio of the HTGR power density to the PBR power density (7.1/5.5 = 1.29). Assuming the cost is proportional to the volume, the fuel storage building cost for the PBR was estimated to be $13.9 million (1.29 × $10.8 million), for a difference of $3.1 million.

The reference PBR case was assumed to include the use of a special grade of graphite capable of lasting the entire reactor lifetime. With the higher grade of graphite, the cost increase for the PBR reflector was estimated to be $10.5 million (see Section 2.8).

Table 2.9.1. Reference Plant Capital Costs for 3000-MW(t) Gas Turbine Systems (January 1979 Dollars)

		Cost Category	Costs ($10⁶) HTGR	PBR	How Cost Difference Probability Distribution Was Determined
21	Structures and Improvements				
	211	Yardwork	8.9	Same	-
	212	Reactor Containment	63.2	Different	Calculated from estimated designs
	213	Turbine Generator Building	-	-	-
	214	Security Building	0.4	Same	-
	215	Reactor Service Building	10.8	Same	-
	216	Main Circ. Control Bldg.	0.6	Same	-
	217	Fuel Storage Building	10.8	Different	Estimated
	218	Other Structures	48.5	Same	-
22	Reactor Equipment				
	221A	PCRV Structure	69.3	Different	Calculated from estimated designs
	221B	PCRV Liners & Penetration	44.3	Different	Calculated from estimated designs
	221C	Reactor Control Mechanism	10.8	Different	Calculated from estimated designs
	221D	PCRV Internals & Insulation	43.4	Different	Calculated from estimated designs
	221E	Reflector Graphite Upgrade	0	Different	Estimated
	222	Main Heat Transfer & Transport System	67.4	Different	Assumed
	223	Safeguards Cooling System	23.0	Different	Assumed
	224	Rad. Waste System	4.2	Same	-
	225	Nuclear Fuel Handling	42.9	Different	Assumed
	226	Other Reactor Plant Equip.	26.6	Same	-
	227	Instrumentation & Control	8.7	Different	Estimated
23	Turbine Plant Equipment		78.0	Same	-
24	Electric Plant Equipment		37.0	Same	-
25	Miscellaneous Plant Equipment		10.0	Same	-
26	Heat Reject System		37.0	Same	-

The cost of the main heat transfer and transport system was projected to be higher for the PBR because of the additional reflector cooling system needed to cool the super-graphite liner. The cost increase over that of the HTGR was assumed to be $4 million. On the other hand, the auxiliary cooling system cost should be lower for the PBR since it has a lower power density compared to the HTGR and therefore has lower shutdown (normal and abnormal) heat removal requirements; the cost advantage to the PBR was assumed to be $4 million.

Nuclear fuel handling costs include capital costs for the equipment to transfer the fuel elements. Although the PBR has many more fuel elements than the HTGR, the handling mechanisms were projected to be less costly in the PBR by a projected 12%.

Instrumentation for the PBR requires sophisticated equipment for detecting core characteristics in the pebble bed. The cost for the PBR instrumentation was estimated to be higher than that for the HTGR by an assumed 11%.

A large uncertainty exists in the capital cost differentials between PBR and HTGR plant systems, particularly for the cost differentials of the fuel storage building, reflector graphite upgrade, heat transfer components, nuclear fuel handling and core instrumentation. Projected cost estimates for these cost categories were assumed to be independent; thus a cumulative probability distribution of the difference between a particular PBR cost category and the corresponding HTGR cost category was assumed and used in calculating a total plant cost probability distribution. The resulting probabilistic estimates of costs are shown in Table 2.9.2.

Table 2.9.2. Probabilistic Estimates of Differences in Capital Costs Between PBR and HTGR

Cost Category	Plant System	Capital Cost Difference ($\$10^6$) (PBR - HTGR) CP* = 0.0-0.5	CP = 0.5-1.0
212, 221A, 221B, 221D	PCRV Structure, Liners, etc.	See Table 2.3.2	
221C	Reactor Control Mechanism	See Section 2.1	
217	Fuel Storage Building	2.48-3.10	3.10-3.72
221E	Reflector Graphite Upgrade	7.0-10.5	10.5-14.0
222	Main Heat Transfer and Transport System	2.4-4.0	4.0-5.5
223	Safeguards Cooling System	-2.5- -4.0	-4.0- -5.5
225	Nuclear Fuel Handling	-1.0- -3.0	-3.0- -5.0
227	Instrumentation	0.0-1.0	1.0-5.0

*CP = cumulative probability.

References

1. "Probabilistic Comparative Study Prismatic Fuel and Pebble Bed Gas Cooled Reactor Power Plants," prepared for Oak Ridge National Laboratory by Management Analysis Company, San Diego, California, May 31, 1980.

2. M. L. Myers and L. C. Fuller, "A Procedure for Estimating Non-Fuel Operation and Maintenance Costs for Large Steam Electrical Power Plants," ORNL/TM-6467 (January 1979).

CHAPTER 3

STUDIES OF KEY TECHNICAL ISSUES

Outline

3.0. INTRODUCTION

The purpose of this chapter is to summarize the separate detailed studies of key technical issues performed for the comparative evaluation of the two types of high-temperature, gas-cooled reactors. As has been stated in earlier sections, the intent of the studies was twofold: (1) to perform a technical evaluation that would compare the merits, disadvantages, and feasibility of the HTGR and PBR and also specify and describe the development effort required to deploy the systems, and (2) to estimate the differences in cost which might arise from the selection of one reactor concept over the other for particular applications. This cost information was then used as the basis for a probabilistic cost comparison.

The studies summarized here focussed on the major differences between the HTGR and PBR and/or on major technical issues. They are described in the following sections:

Section 3.1. Reactor Control and Instrumentation
Section 3.2. Fuel Cycle Analysis
Section 3.3. Reactor Pressure Vessel and Containment Building Capital Costs
Section 3.4. Fuel Fabrication and Recycle Unit Costs
Section 3.5. Impact of Fission-Product Releases
Section 3.6 Heavy Metal Loadings in PBR and HTGR Cores
Section 3.7. Reactor Availability
Section 3.8. Graphite Reflector Damage
Section 3.9. Seismic Effects
Section 3.10. Temperature/Flow Oscillations in HTGRs
Section 3.11. Plant Capital Costs
Section 3.12. Reactor Research and Development Costs

Although the research performed to date for the HTGR (mainly a U.S. effort) and the PBR (mainly an FRG effort) encompasses the above work areas and more, the work presented herein attempts to provide a consistent, comparative evaluation, at least with respect to the key technical issues and estimated plant costs for particular HTGR and PBR plant designs. It is to be pointed out, however, that in the time allotted for the work, it was not possible to optimize the HTGR and PBR plant designs for particular measures of performance (i.e. plant cost).

The source of information summarized in this chapter is the work performed by the staff at Oak Ridge National Laboratory (ORNL), General Atomic Company (GA), and General Electric Company (GE). Individuals responsible for the material presented in this report (or for the material that has been summarized for this report) are identified; in addition, documentation of analysis performed at GA and GE, but not specifically referenced, is listed in the bibliography at the end of the report.

3.1. REACTOR CONTROL AND INSTRUMENTATION

3.1.1. Introduction

The objective of the work presented here is to assess, technically and economically, the major advantages and disadvantages of the control and instrumentation systems for the PBR core as compared to the systems for the HTGR core. This comparison is necessarily limited in that the control and instrumentation requirements for the HTGR are better known than for the PBR, practical control systems for the HTGR having already been licensed and developed in the U.S. Also, proper evaluation of the feasibility and cost of the control rod systems for the PBR is hampered by a lack of pertinent design information for the reactor sizes of interest. Nevertheless, the design and operation of the control systems of the PBR and HTGR are such that basic comparisons can be made.

First, the mechanics of control rod movement in the two reactors differ significantly. The HTGR is controlled by the insertion of control rods into open spaces in the fuel blocks and the insertion can take place under the force of gravity. By contrast, the PBR is controlled by rods that must be inserted into the core by mechanical means, with significant forces required on the rods. Some control can be achieved by manipulating control rods in the gas space above the PBR core, but a number of rods will probably have to actually penetrate into the core under both normal and abnormal situations. The number of penetrations will depend, of course, upon the actual control requirements.

The design of the control rod drives for the two reactor types also differ. The HTGR has two control rods per control rod drive, while the PBR has only one rod per drive and the drive train of the PBR is more complicated and expensive. Thus, the time and cost required to replace a fixed number of control rods will be less for the HTGR than for the PBR since there are fewer control rod drives that will have to be removed in the process.

The HTGR and PBR backup control systems required for safety considerations also differ considerably. The HTGR backup control system consists of small control pellets that are suspended above the control rod channels so that they can be dropped into the channels if required. The PBR system is somewhat similar in that control pebbles, smaller than the fuel pebbles and referred to as KLAK, are suspended above the core so that they, too, can be dropped. However, in the PBR the control pebbles filter down in between the larger fuel pebbles and to subsequently remove them would require that a substantial portion of the fuel pebbles also be removed. Thus, the PBR would suffer a higher fuel cycle cost penalty than the HTGR in the event that the backup system was required.

Radiation damage to control rod materials is important in both reactor types, but the stress level occurring in the PBR rods leads to higher material ductility requirements than for the HTGR rods. Therefore, the life for PBR control rods inserted into the bed during reactor operation is expected to be shorter than the life of HTGR control rods.

While control of xenon-driven instability appears possible without rod insertion into the PBR bed, such operation is not assured; moreover, a full xenon override requirement would necessitate operation with control rods inserted.

In determining the role the control system has with respect to effects upon total plant cost of a PBR as compared to an HTGR, the following items have been assessed: (1) capital costs of the control rods, (2) capital costs of the control rod drives, (3) replacement costs of the control rods, (4) the cost penalties associated with downtime to replace the rods, (5) the cost penalties associated with use of the backup control system, either for testing or following an accident, and (6) the costs of developing any advanced materials required. In addition, licensability of the PBR control system is addressed.

3.1.2. PBR Control Requirements

D. R. Vondy

Primary considerations for normal operation of the PBR core indicate that 0.027 reactivity is needed in control absorption without short-time startup capability, and that 0.090 reactivity is required for full, warm, peak xenon override startup capability. These requirements decrease somewhat as the C/HM ratio decreases, but the rod worth decreases even more owing to the decrease in the associated thermal-neutron flux level if no changes are made in the rod design.

For the PBR control requirements discussed here, only once-through continuous fueling is considered. (The data would not be applicable even for once-recycle of the pebbles; this would require reevaluation.) A relatively large worth of rods in the gas space above the core is considered, and insertion of the rods into the bed (if required) is assumed to be relatively shallow, certainly for a worth of less than half the total. Fairly symmetrical insertion patterns would be necessary for the full effectiveness of the use of less than the total number of rods. No credit is taken for the worth of rods in the top reflector, the assumption being that they would reside in the reflector, not above it. Radial reflector rods are considered for the small PBRs, but they have little merit for large ones (unless the design were changed to include a central graphite column). Additional control requirements beyond the primary requirements (independent cold long-term shutdown) are not considered.

A composite picture of rod requirements is presented in Table 3.1.1. Note that even for the 1,500-MW(t) size, the use of reflector rods may not lead to practical control. These estimated insertion requirements have considerable uncertainty because accurate confirmation calculations are incomplete, specifically for the worth of the removal of rods selectively inserted for normal operation. The more rods inserted into the bed, the shorter the required depth of insertion and the higher the replacement rate (rods/year) -- but further study is needed for optimization. Considerable neutron absorption and high fast-neutron flux exposure occurs with rod insertion into the gas space.

Table 3.1.1. PBR Control Requirements as a Function of Reactor Size

Reactor Size [MW(t)]	Primary Control Rods	Reflector Rods	Rods Inserted for Normal Operation: In Gas Space, With No Xe Override	In Gas Space, With Xe Override	In Bed, With Xe Override
500	15	15	0	12	0
1,500	60	0 (24)[a]	25 (20)[a]	25 (20)[a]	22
3,000	125	0	60	60	48

[a]The use of reflector rods would reduce the number of core rods that must be inserted.

The data in Table 3.1.1 indicate that control rod insertion into the bed is not needed for normal operation without peak xenon override. A reduction in reactor power from 100 percent to 40 percent is allowed; however, the window for warm restart is so narrow that restart after full shutdown would have to be rather quick and likely would often not be possible. If full restart capability is required, normal operation would have to be with some rods inserted into the bed for the large plant sizes. It is likely some applications will require full restart capability. In some cases such capability might be needed only part of the time, and this seems to be more readily accomplished with continuous fueling. Without full restart capability, the availability would be less and could either force the installation of more excess capacity or increase the demand on an energy source external to the system. The probability that override capability is required directly affects the probability that the rods must be inserted into the bed during normal operation.

The major uncertainties regarding whether control rods are to be inserted into the bed and how many are to be inserted center on:

(1) The override capability requirement,
(2) The worth of rods in gas space,
(3) The once-through fueling scheme,
(4) The worth of selective rod insertion in the bed,
(5) The reactor size.

Items 1, 2, and 4 have relatively high values of uncertainty; also the reference once-through-then-out (OTTO) fueling scheme (Item 3) has not been utilized in an operating reactor. With more than one pass of the pebbles relative to once-through, we predict lower heavy metal temperatures (lower fission-product release), slightly lower pressure drop through the bed, lower high-energy flux exposure to the reflectors, higher fuel conversion and higher fuel (ore) utilization. Thus if control and pebble-handling requirements can be satisfied (practically and economically), the reference design probably would specify at least once-recycle of the pebbles without reprocessing, and possibly more than two-pass. However, the greater fuel-handling requirements augur against implementation of multipass refueling.

Note that since the estimated fraction of the control rods that would have to be inserted into the pebble bed is roughly 2/5 of the primary rods for the large plants (with some uncertainty), it would not be realistic to treat the fraction of the primary rods (or rods per unit power) directly as the variable of uncertainty.

It is concluded that the probability that the control rods must be inserted into the bed during normal operation exceeds 50 percent (say 60 percent) and that 37 percent of the primary rods [about 1 rod per 65 MW(t)] would be inserted and subject to early replacement, with a standard deviation of say 15 percent uncertainty. Any reasonable probability distribution is acceptable.

Finally, the control required for stability against xenon oscillations depends upon the stability bound of the reactor. Generic results show the core sizes above which xenon oscillation is expected and for which special control would be required (within certain restrictions). The corresponding stability bounds are shown in Table 3.1.2.

Table 3.1.2. Stability Bounds of PBRs

	Stability Bound [MW(t)]		
Power Density Flattened	No	Yes	Yes
Temperature Feedback	No	No	Yes
Power density [$W(t)/cm^3$]			
2.81	1,760	1,660	2,300
5.62	2,630	2,500	3,160
11.24	4,480	4,240	5,150

While there is uncertainty regarding control of xenon-driven oscillation, based on the azimuthal harmonic for the reference low height-to-diameter ratio, the 3,000-MW(t) size seems to be near the stability bound, its stability decreasing with increasing size. It is highly probable that up to the 3,000-MW(t) size, xenon control will not be necessary, and that if it were necessary, adequate control could be exercised in the gas space above the bed with once-through fueling.

Coarse calculations for a reference HTGR core indicate that it would be stable up to 3,000 MW(t) even without temperature feedback.

3.1.3. PBR Control Rod Damage, Replacement Requirements, and Costs

P. L. Rittenhouse

Control Rod Materials Assessment

An assessment of the structural integrity of the materials used in PBR control rods requires a mechanical loading analysis, studies of the hardening and ductility properties as a function of temperature and neutron irradiation, and an examination of the interactions of the rod cladding material with other materials in the rod environment. The alloys primarily being considered as rod cladding materials are Alloy 800H, Hastelloy X, and Inconel 625.

Mechanical Loading. Control rods being considered for use in the PBR are of two types: push-type rods, which are forced directly through the bed of fuel spheres; and rotating-type rods, which are rotated as they penetrate the fuel bed.* The force, F, required for full insertion (4.5 m) of a push-type rod is about 127,500 N. However, this can be lowered to about 32,400 N if ammonia is injected immediately prior to the start of rod insertion. The forces required for insertion are reduced significantly when rotating-type rods are employed: even without ammonia, only 9800 to 19,600 N is needed.†

For the mechanical loading analysis the following assumptions were made for both types of rods:

(1) The insertion rate of the control rods is 2 cm/s to minimize temperature increases in the cladding of the control rods.
(2) Maximum temperature of the rods during insertion is 750°C.
(3) The column length, L, of the rods is 6.5 m.
(4) The outer cladding tube is 130 mm OD by 110 mm ID; the inner cladding tube is 90 mm OD × 80 mm ID; and the cladding cross-sectional area, A, is 5100 mm^2.
(5) There is no support from the B_4C absorber material.
(6) No irradiation damage has occurred.

Failure of the PBR control rods during their insertion into the core could occur by plastic yielding (i.e., when $F/A >$ yield strength of the cladding) or by column buckling (i.e., when $F > F_{cr} = \pi^2 EAr^2/4L^2$). F_{cr} is the Euler critical load (buckling load), E is the elastic modulus of the cladding material, and r is the least radius of gyration of the concentric cladding tubes. Safety factors (i.e., force to cause failure divided by the applied force) have been calculated for both failure modes and the three proposed cladding alloys over a range of forces encompassing those discussed earlier for push- and rotating-type rods. The results, presented in Table 3.1.3, show that in all cases the plastic yielding safety factor is $>$ 5 and therefore failure of the control rods by this

*These two types of rods are referred to in earlier sections as thrust-type and screw- or auger-type rods respectively.

†It must be noted here that values for F obtained from various sources were not entirely consistent. Those quoted above appear to be reasonable.

mode should be impossible even under the most severe conditions (i.e., push-type rods without ammonia, F = 127,500 N). The significant differences in the safety factors for the three alloys are directly due to their differences in yield strengths.

Column buckling safety factors are very much lower than those for plastic yielding but still seem sufficient for all cases except the push-type rods without ammonia. Values of E for the three alloys are essentially identical, and this results in the small variations between the safety factors for the various alloys.

Should, for some reason, the temperature of the cladding reach 950°C* during control rod insertion, safety factors would be reduced to those shown in Table 3.1.4. The plastic yielding safety factors are reduced by 50% or more, depending on the alloy, because yield strength falls off rapidly from 750 to 950°C; those for buckling are relatively unaffected because E is almost constant over this temperature range. However, even with these lower safety factor values, the conclusions on control rod integrity are unchanged relative to the reference 750°C case. Even more severe temperature conditions were considered in calculations of yielding safety factors for Hastelloy X and Inconel 625 at 1100°C and 98,100 N and in calculations of the buckling safety factor for Alloy 800H under the identical

*A temperature of 950°C would be achieved if the insertion rate were increased to 10 cm/s; under such conditions, F might also be higher.

Table 3.1.3. Calculated PBR Control Rod Safety Factors for Normal Insertion Rates and Temperatures

Control Rod Insertion Force (N)	Safety Factor					
	Failure by Plastic Yielding			Failure by Column Buckling		
	Alloy 800H	Hastelloy X	Inconel 625	Alloy 800H	Hastelloy X	Inconel 625
9,800	78	132	216	16	16	18
19,600	39	67	108	8.0	8.0	8.8
49,000	16	27	43	3.2	3.2	3.5
98,100	7.8	14	22	1.6	1.6	1.8
147,100	5.2	9.0	14	1.1	1.1	1.2

Table 3.1.4. Calculated PBR Control Rod Safety Factors at 950°C

Control Rod Insertion Force (N)	Safety Factor					
	Failure by Plastic Yielding			Failure by Column Buckling		
	Alloy 800H	Hastelloy X	Inconel 625	Alloy 800H	Hastelloy X	Inconel 625
9,800	39	52	91	14	14	15
19,600	19	26	45	7.0	7.0	7.5
49,000	7.8	10	18	2.8	2.8	3.0
98,100	3.9	5.2	9.1	1.4	1.4	1.5
147,100	2.6	3.5	6.1	0.94	0.94	1.0

temperature and stress. The yielding safety factors calculated were 2.6 and 4.0, respectively, for Hastelloy X and Inconel 625, a reduction of approximately 50% from the 950°C case; the buckling safety factor of Alloy 800H was 1.3 versus 1.4 for 950°C and 1.6 for 750°C.

Thermal Aging and Irradiation Damage. During the major portion of the service life of the control rods, the cladding will be at temperatures in the range 575-650°C and hardening of the cladding materials, with an accompanying reduction in ductility, will occur by thermal aging. Also, neutron doses on the order of 10^{22} neutrons/cm^2 (thermal) will be experienced and will further add to hardening and ductility reduction. Effects on such properties will begin to be seen at about 10^{16} neutrons/cm^2.

Evaluation of the performance of the control rods (as in the section above) as a function of service time will require tensile properties (yield strengths and elastic modulus values) as a function of neutron dose at a nominal temperature of 600°C. However, it is to be expected that both yield strength and elastic modulus will, at least initially, increase with exposure and result in calculated safety factors which will be enhanced - or at least not be reduced - by service exposure.

More realistically, one must worry about the ductility and toughness characteristics of the cladding material. Both will probably be degraded significantly by the service exposure and this will increase the possibility of failures through thermal shock, impact loads (e.g., the insertion forces generally occur as "spikes" rather than as a continuous loading), seismic events, etc. There is probably no method for determining with certainty what levels of toughness and ductility are necessary for assurance of the integrity of the control rods. (These properties are not involved in the safety factor calculations described earlier.) Therefore, it is not improbable that ductility minimums for the control rod cladding at end-of-life will be specified on an arbitrary but conservative basis. Input relative to the licensing experience of U.S. and other reactor vendors on control rods would be valuable in further analysis.

Cladding-Environment Interactions. The alloys being considered as control rod cladding are known to react with the impurities present in the helium coolant and, as a result, to undergo changes in mechanical properties, usually in a detrimental fashion. However, for temperatures up to about 750°C the degree of such reactions should be inconsequential; therefore, environmental degradation of the cladding alloys during their life at 650°C would not be expected. Under normal full-insertion conditions, the temperature is only slightly higher, and no damage should occur during the short periods in question. Even a few hours at temperatures as high as 900-950°C should cause no problems. Additional compatibility questions which need to be addressed are the possibilities of reaction between the B_4C absorber material and the cladding and between the fuel spheres and the cladding. Both could affect the properties of the cladding and its subsequent performance.

Control Rod Service Life, Replacement Requirements, and Costs

Service Life and Replacement Cycles. The control rods for both PBRs and HTGRs will have service lives considerably shorter than the design life of the system and, therefore, they will need to be replaced periodically. The allowable service lifetime for HTGR control rods clad with Alloy 800H was fixed by GA at four years for rods that remain in the core and about twice that for average rods. The rationale for their selection was that the shock absorber (also of Alloy 800H) at the bottom of the canned neutron absorber sections and the spine of the control rod would be the most critical component in terms of mechanical failure. To minimize the possibility of failure, it was decided to specify an end-of-life ductility value of no less than 2% for the shock absorber material. Further, it was known from experiments that this ductility was reached after an irradiation dose of about 6×10^{21} neutrons/cm^2 (E > 0.1 MeV). This, combined with an expected dose rate for the rods of 1.5×10^{21} neutrons/cm^2/yr, resulted in the specification of the four-year life for in-core control rods. In practice, an average of one-eighth of the control rods is expected to be replaced during each shutdown for refueling (nominally at one-year increments).

A dual cycle is being considered for control rod replacement in large PBRs (i.e., approximately one-third of the rods after x years and the remainder after nx years, where $n > 1$). The one-third of the rods replaced most frequently would be those expected to be inserted into the fuel bed one or more times during service; the remainder would probably operate only in the space between the top reflector and the core. The most optimistic cycle which has been suggested is one-third of the rods for four years and two-thirds for ten years, a "four-year/ten-year" cycle. This appears overly optimistic given today's state of knowledge.

Since the service required of the PBR control rods that penetrate the core is more severe than that of the HTGR control rods, it seems reasonable to set a minimum residual ductility requirement of 5% for the PBR control rod cladding as opposed to 2% for the HTGR control rod cladding. Assuming that Alloy 800H is used, the ductility will fall to 5% at about 3×10^{21} neutrons/cm^2 (E > 0.1 MeV). The expected maximum dose rate at the PBR top reflector is 1.6×10^{21} neutrons/cm^2/yr. If this also applies to the control rods, a level of 5% tensile ductility is reached in approximately two years of operation. Therefore, the optimistic service life of PBR control rods mentioned above (two-thirds for ten years and one-third for four years) does not appear achievable in practice. A more reasonable replacement schedule using Alloy 800H is one-third after two years and the remainder after four years, i.e., a two-year/four-year cycle. Although FRG is conducting research to identify alloys offering improved life for control rod cladding, it is improbable that near-term developments will result in much better than a three-year/six-year cycle.

Control Rod Costs. The large differences between PBR and HTGR control rods (i.e., a column of heavy-walled concentric tubes for the former versus short, thin-walled, spine-supported absorber sections for the latter) result in a significant difference in the cost of the two types of rods. Each of the PBR control rods would be constructed of about 580 lb of Alloy 800H cladding, and tubes of the required dimensions currently cost about $8/lb. Based on this, the cladding for a complete set of control rods would be approximately $700,000 (151 rods × 264 kg/rod × $17.6/kg). The cost of cladding for the relatively thin-walled tubes of the 84 pairs of control rods in a comparable HTGR system would be only about $80,000. However, it is apparent that the cladding material for the HTGR rods is only a minor factor in total cost since GA's estimated price for a full complement of control rods is approximately $2,300,000 ($14,000/rod). About 90% of the cost is involved in the machining of parts (e.g., guides, spacers, support fitting), assembly and welding of the absorber sections, and inspection and quality assurance. Since the PBR rods will require even more complex and demanding manufacturing and inspection operations, the cost of a full set of rods has been estimated as $4,000,000.*

Based on the costs given above, and assuming a four-year HTGR control rod cycle and a two-year/four-year PBR cycle, the per-year cost of replacement control rods is approximately $590,000 for the HTGR versus $1,390,000 for the PBR. Even the most optimistic service life being suggested for the PBR control rods (four-year/ten-year) results in a cost of replacement rods of about $630,000, still slightly in excess of that for the HTGR. This differential would be larger based on an average eight-year replacement cycle for HTGR rods.

3.1.4. Times Required for Control Rod Replacement

(Note: The following information has been extracted by the editors from a General Electric Company report[1] authored by C. R. Davis and W. B. Scott.)

The assessment of reactor availability for the HTGR and PBR leads to the conclusion that the availability advantages of the on-line refueling for the PBR may be more than offset by the control rod replacement time. The control rod replacement time is dependent on the control rod lifetime, the number of control rods, and the effort required to clear the head access of the fuel-handling equipment to permit access and movement of the control rods. There is considerable uncertainty in these parameters for the large PBR.

Outage Effects of Control Rod Replacement in HTGRs

Refueling of the HTGR is accomplished on an annual basis with one-fourth of the fuel assemblies replaced. During the same outage (reactor shut down and depressurized), one-tenth of the reflector elements and one-eighth of the control rod and drive assemblies are also replaced. Since removal and replacement of the control rods and drive assemblies are

*This assumes one-third of the rods (51) are screw-type for core penetration and two-thirds (100) are simple push-type. The costs of these rods are estimated as $30,000 and $25,000 each, respectively.

accomplished in parallel with refueling, only 5 hours will be added to the refueling time, 2 hours to remove the first control rod and drive assembly before start of refueling, and 3 hours to replace the last control rod and drive assembly after the end of refueling. This time was confirmed at a recent refueling outage of the Fort St. Vrain reactor.

Outage Effects of Control Rod Replacement in PBRs

Refueling of the PBR is accomplished during power operation and requires no plant outage time. However, removal and replacement of control rodand drive assemblies (on an annual basis or at some other predetermined interval) and replacement of top and radial side reflectors (upper one-third, once during the 40-year plant life) would require reactor shutdown and depressurization. Since the PBR control rod and drive assemblies are similar in design and installation to those used in LMFBRs, it is expected that handling equipment and removal and replacement times will be the same as for the Clinch River Breeder Reactor (CRBR) or the Conceptual Design Study (CDS) plant. The equipment has been designed, built and will be performance tested for the CRBR. The time is predicted to be 7.2 hours per control rod and drive assembly, which is the same (7 to 8 hours) as estimated for the THTR in West Germany.

Consider a reference PBR with 151 control rod and drive assemblies; under the most optimistic schedule (see Section 3.1.3) 51 control rods are estimated to have a four-year life and 100 are estimated to have indefinite ($\sim$10-year) life. Therefore, on an annual basis, one-fourth of the 51 (13) and one-tenth of the 100 (10) will be removed and replaced. Removal, inspection and replacement of the 100 control rod and drive assemblies at the rate of 10 per year will satisfy in-service inspection (ISI) requirements. (Section XI, Division 2, of the ASME Code requires that components traversing the primary coolant boundary be inspected 100% within a 10-year period.)

Removal and replacement of control rod and drive assemblies also requires the removal and replacement of some of the fuel feeding tubes (estimated as one-fourth of 43) with their valves, hoppers, distributors, and other components. It is estimated that on an annual basis removal of the fuel feeding tubes would require 66 hours, removal and replacement of the control rod and drive assemblies would require 166 hours, and replacement of the fuel feeding tubes would require 66 hours. The total time would be 298 hours.

3.1.5. Reactor Instrumentation Costs

S. J. Ditto and E. P. Epler

Relative to core instrumentation, the HTGR can be more readily instrumented than the PBR, with in-core instruments located in stationary fuel blocks. For the reference PBR, nuclear instrumentation external to the core may provide adequate information for reactor operation. On that basis, the reactor instrumentation requirements of HTGRs and PBRs should be about the same. But if internal core instrumentation is required in the PBR, use of replaceable thimbles may be necessary.

Given the state of knowledge, it is assumed that any ex-core instrumentation for the PBR and HTGR are comparable in complexity and net cost. If the PBR ex-core instrumentation is not acceptable to U.S. licensing and in-core instruments are required, the instrumentation for the PBR could presumably cost up to an additional $5 million compared to that of the HTGR.

3.1.6. Licensability of PBR Control Rod Systems

S. J. Ditto and E. P. Epler

Overview of the Problem

In order to determine the costs associated with attaining licensability and acceptability of a reactor control rod system, one needs to know the relationships of specific rod-drive features to the needs of a specific plant (as determined through detailed analysis of plant operation and accident scenarios). Such characteristics as reactivity rates, response times, and reliability vary widely among plants. In judging the suitability of a control system, it is important that both the response requirements and the consequences of failure of the rod system be known, and in general this information is lacking for the PBR. Moreover, of the several features that have been described for PBR rod drives, some are undesirable and possibly even unacceptable. But this cannot be determined unequivocally without more information on the specific requirements. The suggestion that safety rods must be forced into the core under more-or-less controlled conditions implies rather large forces (and therefore large energy sources), and this casts doubt on the reliability of such systems. Unless it can be demonstrated that fast negative reactivity insertion is never a requirement, the proposed rods would be unable to qualify for protection service. The suggestion that a two-stroke drive meets the "separation" criterion is not responsive to the real issue. Also pneumatics in such devices are usually avoided because of springiness and sensitivity of response characteristics to pressure variations.

There still appears to be a great deal of uncertainty regarding the necessity to use rods (and instruments) within the core of a large PBR to handle power distribution and potential xenon oscillations. Until these uncertainties are resolved, one must assume that in-core rods and instruments will be required, and these must obviously withstand the very high temperatures of the core or have separate cooling. It is not clear that suitable materials are available.

Bridging problems have also been experienced with safety stysems similar to the proposed KLAK system; however, bridging can apparently be avoided. Recovery from secondary-shutdown system actuation can also be a problem. (Boron-containing balls have been retained in HTGR arrays.) The potential problem of retrieval from a PBR core (involving unloading a substantial fraction of the core) would lead to extreme measures to prevent inadvertent release of the KLAK. This would undoubtedly impact upon the reliability to perform as needed.

Required Development

In this assessment an attempt has been made to estimate what developmental features might be required to make the basic rod drive and KLAK proposals for the PBR licensable. We recognize that licensability is not an easy thing to judge, yet our experience tells us that, assuming these systems must fulfill a safety function, the following requirements probably must be met satisfactorily and rather promptly:

(1) There must be stored energy available to assure the ability to insert the poison in a timely fashion under a number of conditions, including loss of "normal" power.

(2) The safety function may use the same poison and mechanical supports as those used for routine control of power level and power distribution, but if that arrangement is used, the "scram" or protective action must be separated from the routine control action. That is, malfunctioning of control mechanisms and circuits shall not defeat nor inhibit protective action.

(3) Force requirements and lubrication with ammonia are coupled in the current proposals. We are of the opinion that scram forces should be adequate to operate the rods even in the event of failure of the proposed ammonia injection.

(4) There should be no constraints that would cause the design to require inhibition of scram motion on the basis of things like rod temperature limits. When required, the scram should proceed unhampered.

(5) The proposed secondary system (KLAK) must not impose costs as a result of test or inadvertent actuation that would cause design efforts to respond to fear of actuation that could lead to low reliability.

Instead of attempting to identify candidate concepts for meeting the above requirements, this assessment has concentrated more on establishing scopes of feasible developments that might be pursued in order to help us estimate the effort. Specifics are not proposed. Rather, approaches that suggest feasibility are discussed.

With regard to stored energy, it seems that forces of 26,700 or 31,200 N through a distance of about 46 m would be required for full insertion of a rod. If this were to be accomplished in about 1 min, the net power required would be about 3 hp or about 2.25 kW, for a total energy about 0.04 kW-hr. Because of losses, the energy available must then be perhaps a few tenths of a kW-hr per rod – about the capability of an automobile battery. Thus the energy requirements are seen to be not too great.

The separation of control and safety functions can be achieved by using differential drives. In such applications two motors, one an electric motor and one an air motor, could be coupled to the same drive train. Or two electric motors could be used with safety grade decoupling of control by means of a conventional magnetic clutch to allow overdrive by safety signals. Directional constraints could also be imposed, as could different speed requirements.

The force requirements do not seem to pose any particular problem that could not be taken care of by adequate sizing of motors, gears, etc. The temperature limits on the poison rods themselves are more or less independent of the drive and must be treated in the design of the rods.

With regard to the KLAK system, the problems of testing and inadvertent actuation are real. Unless the spheres can be removed easily, the impact on the fuel cycle will be large. It is possible that the test could be handled by a partial test involving some kind of interceptor that could check one hopper at a time. This would require involved, and presumably costly, equipment and procedures and would do nothing for the inadvertent actuation. Manual actuation leaves much to be desired if the consequences of unneeded actuation are expensive, as they would be.

3.1.7. Research and Development Cost Estimates for the Control and Instrumentation Systems

R&D Costs for Control Rod Cladding Materials (P. L. Rittenhouse)

This analysis applies only to the "assessment/proof of performance" of the control rod cladding materials. It does not take into account possible proof tests which might be required of the PBR control rods themselves (e.g., penetration of a simulated PBR core using a prototypic control rod fabricated with artifically embrittled cladding). Moreover, it is assumed that an acceptable cladding material exists for use with PBR control rods and that "proof of performance" is simply the process of identifying this material and demonstrating its acceptability in terms of strength, ductility, etc. needed at control rod end-of-life. It is also assumed that these needs (i.e., cladding materials properties or characteristics), plus minimum control rod service life consistent with economics, would be identified early in the R&D study.

As noted in Section 3.1.3, demonstration of the acceptability of a cladding material will require study of:

(1) coolant-cladding interactions and their effects,
(2) fuel-cladding interactions and their effects,
(3) absorber-cladding interactions and their effects,
(4) effect of irradiation,
(5) effect of thermal history.

The first of these, coolant-cladding interactions, should be minimal for both the PBR and the HTGR because of the relatively low temperatures involved, especially as control rod cooling is planned. No R&D costs should be necessary unless a material entirely new to gas-cooled reactor experience is selected and this is unlikely.

Interaction of the cladding with the PBR fuel pebbles needs to be investigated. Such interactions could result in degradation of the strength and ductility characteristics of the cladding. This concern does not apply to HTGRs since the control rods will not be in intimate contact with the fuel blocks.

Much information on absorber-cladding interactions should become available from post-service inspection of control rods from Fort St. Vrain and the THTR. Additional, more basic, work will need to be done in the interim.

Data on the effects of irradiation on the properties of the candidate cladding materials is in most cases minimal. The four-year life quoted by GA for HTGR control rods (clad with Alloy 800H) is based on a specified minimum tensile ductility of 2% and the observation that this is reached at 6×10^{21} neutrons/cm^2 ($E > 0.1$ MeV), the expected total dose over four years of reactor operation. (This is based on a 1972 analysis which has not been updated.) Additional information should become available from post-exposure testing of Fort St. Vrain control rods. In-reactor experiments, perhaps one for HTGR control rod cladding and up to about five for PBR candidate claddings, should be used in evaluating cladding acceptability.

The thermal history of the cladding will influence its end-of-life properties both through its interaction with irradiation (degree of damage will depend upon irradiation temperature) and metallurgical changes which occur on thermal aging. The effects of thermal aging on the properties of Alloy 800H are well known. This is not true of all of the candidate materials.

Highly subjective estimates of the minimum and maximum R&D costs to "qualify" cladding materials for the PBR and HTGR control rods are given in Table 3.1.5. Maximum and minimum costs for the PBR R&D might well represent the end points of a normal distribution. Values given for the HTGR are probably "one or the other" (i.e., the lower value will apply if a four-year life is accepted, the higher value if improvement in life is to be sought through refinement of information relating ductility and irradiation dose).

Finally, if none of the current crop of candidate cladding alloys should prove to be acceptable for PBR use, all estimated costs are unknown. Selection of other classes of materials or development of new materials could increase R&D costs by at least an order of magnitude.

Table 3.1.5. Estimated R&D Costs to Qualify PBR and HTGR Cladding Materials

Investigation	Costs ($\$10^6$) PBR	HTGR
Coolant-cladding interactions	0-0.05	0
Fuel-cladding interactions	0.10-0.25	0
Absorber-cladding interactions	0.15-0.35	0.10-0.20
Cladding irradiation effects	2.00-10.0	0.20-1.80
Cladding thermal aging effects	0.05-0.15	0
Total	2.3-10.8	0.3-2.0

R&D Costs for Control Rod Drives and Instrumentation (S. J. Ditto)

No purely developmental efforts appear to be required for the control rod drives, since designs embodying proven concepts in new applications can probably suffice. However, it is certainly true that prototype designs and proof-testing, qualification, and other activities normally associated with development would be required. A 20-manyear effort could be required at a cost on the order of $2 million.

The instrumentation requirements per se do not seem to differ greatly between the PBR and the HTGR. The uncertainty in the costs probably exceed the differences between the concepts. Therefore no basis exists for assuming any difference at this time.

The largest costs would be associated with large scale testing of control systems in reactor type environments, to obtain statistically significant data on control rod response to shutdown requirements. Much of the work done in the FRG may have to be repeated in the U.S. and expanded in order to meet NRC licensing needs. The costs of such a program could run several tens of millions of dollars.

References

1. C. R. Davis and W. B. Scott, "Maintenance and Availability Due to Core Servicing of High Temperature Reactor Plants," General Electric Report, March 1980.

3.2. FUEL CYCLE ANALYSIS

B. A. Worley

3.2.1. Introduction

For a high-temperature gas-cooled reactor operating on a fuel throwaway/stowaway cycle, that is, on a cycle in which the spent fuel is assumed to be discarded or to be stowed away for possible future reprocessing and recycling in some other reactor, the cost of fuel accounts for 35 to 45 percent of the total plant cost over the expected 30-yr life of the plant. The actual fuel cycle cost will depend upon a combination of such factors as average fuel burnup, reactor design, fuel enrichment, heavy metal loadings, operational constraints, etc., and for a particular reactor application, the minimum fuel cycle cost must be determined by an economic analysis that considers a wide range of reactor designs and modes of operation. Even then, other considerations such as safety and overall plant cost may take precedent, leading to a reactor design and/or operation other than that which would result in the minimum fuel cycle cost. Thus the choice of fuel cycle and reactor design must consider not only the fuel cycle cost component of the total power cost but also the total balance-of-plant operation.

The comparison presented here of the fuel cycle costs between a plant with a prismatic core design (HTGR) and one with a pebble bed core design (PBR) is largely made on the assumption that the optimum choice of core design and fuel management for each concept has already been determined. Parametric studies have been made previously for the PBR[1-4] and the HTGR[5-8] for a broad range of applications. The objective here is to compare on a consistent basis the mass flow rates and fuel cycle costs for reference 3000-MW(t) HTGRs and PBRs operating on MEU/Th and HEU/Th throwaway fuel cycles. In addition, performance parameters of reactors having smaller sized cores are compared with those of the 3000-MW(t) reactors, and the relative benefits of reprocessing and recycling spent fuel from the two types of reactors are discussed.

The reference core design data used for the 3000-MW(t) PBR and HTGR are given in Tables 3.2.1 and 3.2.2. The PBR core approximates the German prototype direct-cycle HHT[9,10] and the HTGR core is scaled down from the GA 3360-MW(t) conceptual design.[11]

3.2.2. 3000-MW(t) PBR and HTGR on MEU Throwaway Cycle

MEU Throwaway Cycle Considerations

Given the reference designs described in Tables 3.2.1 and 3.2.2 and assuming that the reactors will operate on an MEU/Th throwaway cycle, a particular choice of moderation ratio, average discharge burnup, and fuel management scheme will result in a minimum fuel cycle cost for a particular concept (PBR or HTGR).

Table 3.2.1. 3000-MW(t) PBR Core Design Data

Reactor Core	
Power, MW(t)	3000
Core-average power density, MW/m^3	5.50
Average inlet coolant temperature, °C	450
Average outlet coolant temperature, °C	850
Effective core height, m	5.50
Effective core diameter, m	11.24
Reflectors	
Top reflector thickness, m	1.0
Gas space thickness, m	1.0
Bottom reflector thickness, m	1.0
Side reflector thickness, m	1.0
Fuel Elements	
Pebble diameter, cm	6.0
Graphite shell thickness, cm	0.5
Total number in core (0.61 vol. fraction)	2,941,955
Coated Particles, BISO	
Kernel material	$(U/Th)O_2$
Kernel diameter, μm	400
Kernel density, g/cm^3	9.50
Carbon coating thicknesses, μm	85/30 /80
Carbon coating densities, g/cm^3	1.0/1.6/1.85

Moderation Ratio. The moderation ratio (C/HM) for the HTGR is influenced by the need to maintain criticality over the cycle length fuel volume constraints, region peaking factor constraints, and fuel temperature constraints. Within these constraints and on the basis of minimizing fuel cycle costs, the optimal equilibrium C/HM ratio for the HTGR on the MEU throwaway cycle is 478 (ref. 12). For the PBR on the MEU cycle, the optimal C/HM ratio, chosen with consideration of tested pebble heavy metal loadings, control rod worth requirements, and fuel costs, is 450 (ref. 2).

Burnup. Handling co , include charges for fabrication, on-site waste storage, shipping, and waste disposal. These costs decrease monotonically with an increase in burnup. The cost of fuel, however, passes through a minimum and starts to rise again at a high burnup because of increasing indirect charges and increasing fissile requirements due to a high buildup of fission products and parasitic plutonium isotopes. The optimal burnup will be that at which the sum of the handling costs, indirect charges and fuel costs is a minimum, provided the burnup is not so high that fuel-particle burnup limits are exceeded. This is true for both the PBR and the HTGR. For the HTGR using MEU fuel, the minimum fuel cycle cost occurs at a burnup of 130 MW(t)-d/kg HM (ref. 12). Parametric

Table 3.2.2. 3000-MW(t) HTGR Core Design Data

Reactor Core	
Power, MW(t)	3000
Core-average power density, MW/m^3	7.09
Average inlet coolant temperature, °C	450
Average outlet coolant temperature, °C	850
Effective core height, m	6.34
Effective core diameter, m	9.21
Reflectors	
Top reflector thickness, m	1.19
Bottom reflector thickness, m	1.19
Side Reflector thickness, m	1.47
Fuel Elements	
Hexagonal block distance across flats, cm	36.0
Hexagonal block height, cm	79.3
Total number of fuel elements	4752
Standard 10-Row Fuel Elements (number = 4080)	
Fuel holes per element	216
Fuel hole O.D., cm	1.27
Effective fuel height, cm	71.25
Fuel hole pitch, cm	1.88
Coolant holes per element	108
Large coolant hole (number = 102) O.D., cm	1.5875
Small coolant hole (number = 6) O.D., cm	1.27
Control Elements (number = 672)	
Fuel holes per element	118
Fuel hole O.D., cm	1.27
Effective fuel height, cm	70.56
Coolant holes per element	60
Large coolant hole (number = 45) O.D., cm	1.5875
Small coolant hole (number = 15) O.D., cm	1.27
Reserve shutdown control rod hole (number = 1) O.D., cm	9.53
Control rod hole (number = 2) O.D., cm	10.16
Coated Particles	
BISO: Kernel material	ThO_2
Kernel diameter, µm	500
Kernel density, cm	9.9
Carbon coating thicknesses, µm	85/75
Carbon coating densities, g/cm^3	1.05/1.90
TRISO: Kernel material	UC_2
Kernel diameter, µm	210
Kernel density, g/cm^3	10.8
Carbon-carbon-SiC-carbon thicknesses, µm	105/ 30 / 30 / 40
Coating densities, g/cm^3	1.05/1.90/3.20/1.80

studies that have been performed for the PBR (ref. 2) identify an optimum burnup of 100 MW(t)-d/kg HM for MEU fuel cycles.

Fuel Management. Typical HTGR fuel zoning patterns, chosen to reduce local power peaking and maximize fuel usage, are described in ref. 11. The HTGR has a 4-yr cycle length with 1/4 of the core refueled annually.

The fuel zoning pattern used in the PBR is a two-zone scheme in which the outer portion of the reactor is loaded with an increased U/Th ratio in order to reduce the maximum pebble power density and to flatten the outlet-coolant temperature profile across the core.

Reactor Performance Results

Performance characteristics of the reference 3000-MW(t) PBR and HTGR designs are summarized in Table 3.2.3. Note the greater throughput of heavy metal in the PBR at a burnup of 100 MW(t)-d/kg HM compared to the HTGR at a burnup of 130 MW(t)-d/kg HM. However, the ^{235}U requirement is higher for the HTGR because a certain amount of excess reactivity is required at the beginning-of-cycle for a batch-fueled reactor that is not required for the continuously fueled PBR. The U_3O_8 requirement for the HTGR is calculated to be higher than that of the PBR by approximately 7%.

Comparison of Fuel Cycle Costs

Fuel cycle costs were calculated for the PBR and the HTGR using cash-flow discounting. The assumption was made that the annual fuel requirement for the PBR would be purchased at 1-yr intervals in the same manner as for the HTGR. In order to determine the effects of unit costs and economic assumptions upon the comparison of fuel cycle costs for the two concepts, two sets of economic assumptions were used in this analysis, one set similar to those used for earlier HTGR and PBR assessments,[2,12] and another set developed for the current evaluation. The cost assumptions and resulting fuel cycle costs are shown in Tables 3.2.4 and 3.2.5.

Note that with the first set of economic assumptions, the higher heavy metal throughput in the PBR (due to a shorter fuel residence time, i.e., lower burnup) leads to higher fabrication and waste disposal costs for the PBR relative to the HTGR. With the current set of assumptions, the PBR waste disposal costs remain higher, but the fabrication costs are approximately equal to those for the HTGR.* By contrast, with both sets of assumptions the lower fissile requirements of the PBR translate into lower costs for U_3O_8, separative work, and conversion processes. The result is that with either set the MEU fuel cycle costs are higher for the HTGR than for the PBR. The differential is 3% with the earlier set of economic assumptions and 5% with the current set.

*See Section 3.4 for fabrication costs in current set of economic assumptions.

Table 3.2.3. Comparison of Performances of 3000-MW(t) PBR and HTGR on an HEU Throwaway Cycle

	Reference HTGR[a]	Reference PBR[b]
Power density, W/cm^3	7.1	5.5
Fuel residence time in core, calendar days[c]	146[c]	1009
Burnup, MW(t)-d/kg HM	130	100
Moderation ratio, C/HM atom density	478[d]	450
Conversion ratio	0.55	0.58
Heavy metal loadings at equilibrium, kg/GW(e)-d		
^{235}U	2.16	2.04
^{238}U	8.64	8.15
^{232}Th	8.30	14.81
Heavy metal discharged at equilibrium, kg/GW(e)-d		
$^{233}U + ^{235}U$	0.44	0.54
$^{239}Pu + ^{241}Pu$	0.12	0.10
Total heavy metal	16.55	22.04
Heavy metal loadings for initial core, kg[d,e]		
^{235}U	1701	1388
^{238}U	6804	5552
^{232}Th	28,216	18,875
30-yr U_3O_8 requirement, metric tons[f]	5402	5050

[a]Results shown are from ref. 12.

[b]Results shown are from calculations by ORNL.

[c]Load factor = 0.8. Full power days = 0.8 × calendar days.

[d]Note that C/HM = 478 corresponds to C/Th = 850 for the HTGR at equilibrium; the initial core C/Th = 350 for the HTGR which explains the high Th loading for the initial core relative to the PBR.

[e]Initial core loading for the HTGR is that required for one year at 0.8 load factor. Data are estimated from information in ref. 5. The initial core loadings quoted for the PBR are the initial core loadings plus the heavy metal makeup requirements for one year at 0.8 load factor.

[f]Assumes 0.8 load factor and 0.2% tails.

3.2.3. 3000-MW(t) PBR and HTGR on HEU Throwaway Cycle

HEU Throwaway Cycle Considerations

Compared with the MEU/Th throwaway cycle, the HEU/Th throwaway cycle results in lower costs that are directly attributable to lower U_3O_8 requirements (refs. 2, 4, 12). The lower ore requirements are due to a better neutron efficiency of ^{233}U (relative to ^{235}U and the plutonium isotopes), which, in turn, impacts the moderation ratio and burnup.

Moderation Ratio. The better neutron efficiency of ^{233}U leads to the choice of a low moderation ratio (high heavy metal loading) for the HEU/Th cycle since a higher thorium loading increases the ^{233}U production rate. The optimum C/HM ratio from an

Table 3.2.4. Economic Assumptions

	Previous Cost Assumptions[a]	Current Cost Assumptions[b]
Reactor startup	1982	2007
U_3O_8 cost escalation, %/yr	5	2.5
1980 U_3O_8 cost, \$/lb U_3O_8	30	45
SWU cost, \$/kg	90	95
Tails assay, %	0.25	0.20
Load factor	0.80	0.80
Discount factor	0.04	0.07
Fab cost, \$/fuel element		
HEU PBR	3.89	10.60
HEU HTGR	2523 (2565)[c]	9536 (9695)[c]
MEU PBR	3.45	9.44
MEU HTGR	2452 (2695)[c]	7444 (8182)[c]
Waste cost, \$/kg HM		
HEU PBR	450	700
HEU HTGR	450	700
MEU PBR	550	700
MEU HTGR	550	700

[a]Cost assumptions similar to those used for cost calculations in refs. 2 and 12.

[b]Cost assumptions similar to those used in Chapter 1 of this report.

[c]Value shown in parentheses refers to initial core.

Table 3.2.5. Comparison of 30-Year Fuel Cycle Costs for 3000-MW(t) PBR and HTGR on an MEU Throwaway Cycle[a]

	30-Year Fuel Cycle Costs (mills/kW(e)-hr)			
	Previous Cost Assumptions		Current Cost Assumptions	
Cost Categories	HTGR	PBR	HTGR	PBR
U_3O_8	3.54	3.31	5.65	5.24
Separative Work	1.94	1.80	2.16	2.00
Fuel Conversion	0.08	0.06	0.07	0.06
Thorium	0.02	0.02	0.02	0.02
Fabrication	0.42	0.50	1.40	1.40
Waste Disposal	0.34	0.46	0.44	0.58
Total	6.34	6.15	9.74	9.30

[a]Load factor = 0.8, tails = 0.2%, thermal efficiency = 0.4.

economic standpoint occurs at the heavy metal loading for which the benefit of increasing the ^{233}U production is just offset by the increasing heavy metal charges. The choice of C/HM ratio is also influenced by fuel element loading constraints and control rod worth considerations. Based on prior parameter studies (refs. 2, 3, 4 and 12), the C/HM ratios chosen for this comparison are 300 for the HTGR and 325 for the PBR.

Burnup. For the HEU/Th fuel cycle the optimal fuel discharge burnup is approximately 110 MW(t)-d/kg HM for the HTGR (from ref. 12) and 100 MW(t)-d/kg HM for the PBR (refs. 2 and 4).

Fuel Management. Fuel shuffling schemes similar to those described for the 3000-MW(t) PBR and HTGR using MEU/Th fuel are also used for the HEU cases.

Reactor Performance Results

The performance characteristics of the PBR and HTGR using HEU/Th feed with throwaway of spent fuel are summarized in Table 3.2.6. Again the shorter fuel residence time in the PBR (lower burnup of discharged fuel) results in a higher heavy metal throughput rate, but the difference is not nearly so great as for the MEU/Th cycle. And again the HTGR fissile and U_3O_8 requirements are higher than those of the PBR, but only slightly so. Thus the performance parameters of the two types of reactors are much more similar on the HEU/Th cycle than they are on the MEU/Th cycle.

Comparison of Fuel Cycle Costs

The 30-yr fuel cycle costs for the 3000-MW(t) PBR and HTGR, calculated on the basis of the cost assumptions shown in Table 3.2.4, are listed in Table 3.2.7. Note that for these HEU/Th reference cycles, fabrication unit costs based on the earlier set of economic assumptions result in approximately the same fabrication costs for the PBR and HTGR since the heavy metal throughput rates are similar (recall that these rates were quite different for the reference MEU/Th cycles). And since the slightly lower fissile charges for the PBR balance the slightly lower waste disposal charges for the HTGR, the total fuel cycle costs are approximately the same. With the current set of economic assumptions, however, the PBR has a fabrication unit cost significantly lower than that of the HTGR, such that the total fuel cycle cost of the PBR is projected to be approximately 5% lower than that of the HTGR.

3.2.4. 1000-MW(t) PBR on HEU and MEU Throwaway Cycles

The effect of the power of a high-temperature gas-cooled reactor on its performance parameters was studied by decreasing the power of the 3000 MW(t) PBR to 1000 MW(t). The results are summarized in Table 3.2.8. The core-average power density was kept constant such that the reduction in power required a proportional reduction in core volume. For

Table 3.2.6. Comparison of 3000-MW(t) PBR and HTGR Performance on an HEU Throwaway Cycle

	Reference HTGR[a]	Reference PBR[b]
Power density, W/cm^3	7.1	5.5
Fuel residence time in core, calendar days[c]	14[illegible]0	1388
Burnup, MW-d/kg HM	110	101
Moderation ratio, C/HM atom ratio	300[d]	325
Conversion ratio	0.55	0.58
Heavy metal loadings at equilibrium, kg/GW(e)-d		
^{235}U	1.94	1.93
^{238}U	0.14	0.15
^{232}Th	20.30	22.69
Heavy metal discharged at equilibrium, kg/GW(e)-d		
$^{233}U + ^{235}U$	0.48	0.59
$^{239}Pu + ^{241}Pu$	0.004	0.002
Total heavy metal	20.20	22.08
Heavy metal loadings for initial [illegible]re, kg[d,e]		
^{235}U	1353	1312
^{238}U	100	99
^{232}Th	31,834	36,504
30-year U_3O_8 requirement, metric tons[f]	4842	4814

[a]Results shown are from refs. 12 and 13.

[b]Results shown are from calculations by ORNL.

[c]Load factor = 0.8. Full power days = 0.8 × calendar days.

[d]Note that C/HM = 300 corresponds to C/Th = 330 for the HTGR at equilibrium; the initial core C/Th = 300 for the HTGR.

[e]Initial core loading for the HTGR is that required for one year at 0.8 load factor. Data for initial core of HTGR is taken from ref. 13. The initial core loadings quoted for the PBR are the initial core loadings plus the heavy metal makeup requirements for one year at 0.8 load factor.

[f]Assumes 0.8 load factor and 0.2% tails.

Table 3.2.7. Comparison of 30-Year Fuel Cycle Costs for 3000-MW(t) PBR and HTGR on an HEU Throwaway Cycle[a]

	30-Year Fuel Cycle Costs (mills/kW(e)-hr)			
	Previous Cost Assumptions		Current Cost Assumptions	
Cost Categories	HTGR	PBR	HTGR	PBR
U_3O_8	3.18	3.16	5.03	5.00
Separative Work	1.91	1.90	2.11	2.09
Fuel Conversion	0.06	0.06	0.06	0.06
Thorium	0.03	0.03	0.03	0.03
Fabrication	0.42	0.43	1.67	1.22
Waste Disposal	0.34	0.38	0.53	0.54
Total	5.94	5.96	9.43	8.94

[a]Load factor = 0.8, tails = 0.2%, thermal efficiency = 0.4.

Table 3.2.8. Effect of PBR Size on Performance Parameters

	HEU/Th Throwaway Cycle		MEU/Th Throwaway Cycle	
	3000-MW(t)	1000-MW(t)	3000-MW(t)	1000-MW(t)
Core-average power density, W(t)/cm^3	5.5	5.5	5.5	5.5
Effective core height, m	5.50	5.50	5.50	5.50
Effective core diameter, m	11.24	6.49	11.24	6.49
C/HM, atom density ratio	325	325	450	450
Fuel residence time in core, full power days	1110	1110	807	807
Burnup, MW(t)-d/kg HM	101	101	100	100
Conversion ratio	0.58	0.55	0.57	0.55
Heavy metal loading, g/pebble	11.21	11.21	8.23	8.23
Fissile inventory at equilibrium, kg/GW(e)	959	1005	771	828
Equilibrium feed rates, kg/GW(e)-d				
^{235}U	1.929	2.050	2.038	2.169
^{238}U	0.145	0.154	8.154	8.678
^{232}Th	22.686	22.557	14.809	14.159
30-year U_3O_8 requirement, metric ton/GW(e) installed[a]	4012	4255	4208	4477
30-year fuel cycle cost, mills/kW(e)·hr[a,b]	8.94	9.40	9.30	9.76

[a] Load factor = 0.8; tails = 0.2%; thermal efficiency = 0.4.
[b] Assumes nominal cost assumptions listed in Table 3.2.4.

the same C/HM ratio and burnup, the smaller core volume of a 1000-MW(t) PBR leads to a greater fractional loss of neutrons due to leakage compared with the 3000-MW(t) PBR, as is evident from the decrease in conversion ratio. Note that on a per GW(e) basis, the smaller core requires approximately 6% more ^{235}U (and thus 6% more U_3O_8) for both HEU and MEU throwaway cycles. Also, the 30-yr fuel cycle cost for the 1000-MW(t) reactor is projected to be greater than that of the 3000-MW(t) reactor by approximately 5% for both HEU and MEU throwaway cycles.

Information needed to determine the effect of a decrease in reactor power for the HTGR was not available because of a lack of calculational results in which the assumptions made were consistent with those made for the 3000-MW(t) cases. However, the neutron loss fraction by leakage is slightly higher for the PBR compared to the HTGR so that the penalty for a reduction in core size is expected to be slightly less for the HTGR than the PBR; however, this effect is not large enough to be significant.

3.2.5. PBRs and HTGRs with Fuel Reprocessing and Recycle

The PBR and HTGR both exhibit improved performance if the discharged fuel is reprocessed and recycled. The ore requirements and fuel cycle costs can both be decreased compared to the requirements for throwaway cycles, and, as was the case for throwaway cycles, use of HEU/Th fuel leads to better performance than does the use of MEU/Th fuel. These conclusions are supported by KFA, GA, and ORNL and are discussed in more detail in refs. 3, 4, and 12.

For fuel reprocessing and recycle, the main design and operational considerations are the technological constraints on heavy metal loading in the fuel elements, the optimum fuel in-core residence time from an economic viewpoint, and the costs associated with fuel reprocessing and refabrication. A high heavy metal loading in the fuel elements will increase the conversion ratio and thus require less fissile ^{235}U feed to the system. But the amount of heavy metal fuel that can be loaded into the core is limited by constraints on the amount of fuel which can be packed into an HTGR fuel rod or into a PBR pebble. These limits are discussed in Section 3.6.

The conversion ratio also increases if the fuel in-core residence time (fuel burnup) decreases, thus requiring less fissile ^{235}U feed to the system in the long term. However, the greater heavy metal throughput rate increases the handling charges (fabrication, reprocessing, and waste disposal) such that the overall fuel cycle cost passes through a minimum and then increases as the fuel residence time is decreased. Finally, the cost of reprocessing and refabrication of the recycled fuel is high and recycling is economically practical only when the costs for uranium ore and/or uranium enrichment are high enough to warrant recycling. The cost of ore, in turn, is directly related to its availability, and for an uncertain ore supply, recycling becomes increasingly attractive since the U_3O_8 requirement is much less with recycle of spent fuel than with fuel throwaway/stowaway cycles.

In comparing reactors with recycle, however, the objective for recycling must be the same for the two reactors: that is, the objective will be either to minimize the ore requirement or to minimize the fuel-cycle cost. To date the FRG work on the PBR has concentrated on greatly reducing the ore requirement at the expense of a higher fuel cycle cost, while the U.S. work on the HTGR has concentrated on operating at the economic minimum with a much less substantial savings in the ore requirements. Therefore a consistent comparison of HTGRs and PBRs with recycle of spent fuel could not be made with existing information in the time available for this study.

Qualitatively, the PBR is expected to have a greater potential for improved performance with recycle because the fuel cycle cost will pass through a minimum at lower and lower fuel residence times as ore prices increase. The fuel residence time can be decreased in the PBR by simply passing the pebbles through the core at a faster rate without impacting reactor availability. However, the HTGR must be shut down for refueling and a short fuel residence time causes more frequent shutdowns for refueling and thereby decreases reactor availability.

Quantitatively, for reactors with fuel recycle, the fissile penalty for a fixed-fueled reactor compared to a continuously fueled reactor (due to excess reactivity requirement for fixed-fueled reactor) was determined by fixing all other parameters (power density, burnup, C/HM ratio, leakage, and feed enrichment) and performing zero-dimensional, 30-yr history calculations using the nominal cost assumptions shown in Table 3.2.4. The results are shown in Table 3.2.9.

Table 3.2.9. Comparison of U_3O_8 Requirements and 30-yr Costs for Fixed-Fueled and Continously Fueled Reactors with Fuel Recycle

	U_3O_8 Requirement [kg/MW(e)]	30-yr Fuel Cycle Cost [mills/kW(e)-hr]
Fixed 1/3 refueling	2835	10.1
Fixed 1/4 refueling	2697	9.6
Continuous refueling	2348	8.5

Note that no decrease in availability is expected in going from 1/3 to 1/4 fixed refueling, and so no effect is considered in Table 3.2.9. Given that the reference PBR has a slightly higher neutron loss fraction by leakage than does the HTGR, the results in the table overestimate the cost difference between the HTGR and the PBR. Nonetheless, the 30-yr fuel cycle cost advantage of the PBR over the HTGR, as shown in Tables 3.2.5 and 3.2.7 for once-through fuel cycles, is projected to become more favorable for fuel reycle conditions.

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3.3. REACTOR PRESSURE VESSEL AND CONTAINMENT BUILDING CAPITAL COSTS

D. J. Naus

3.3.1. Introduction

Because the PBR will have a lower power density than the HTGR (5.5 W/cm^3 vs. 7.1 W/cm^3), for the same power rating the PBR will have a larger core diameter and thus require a larger core cavity than the HTGR. Other differences will also lead to a larger PBR core diameter, all of which will translate into increased capital costs for the PBR PCRV (prestressed concrete reactor vessel) and containment building. The purpose of this phase of the comparative evaluation was to determine the significance of these cost increases for 1000-MW(t) and 3000-MW(t) power reference systems, considering a coolant outlet temperature of 850°C.

3.3.2. Estimation Method

Since investigations of the PBR core concept by the Federal Republic of Germany and by the General Electric Company in the United States and investigations of the HTGR concept by General Atomic Company have all proceeded somewhat independently, comparable designs were not available which could be used directly for establishing relative PCRV and containment building costs.

Background information for the PCRV for a 3000-MW(t) HTGR-GT was obtained from ERDA-109 (ref. 1) and from a letter memorandum from GA (ref. 2). Information for a comparable PCRV design for a PBR core (identified here as PBR #1) was also obtained from ref. 2. In addition, a second PBR case (identified as PBR #2) was considered so that the PBR and HTGR could be compared on the basis of equivalent core-cavity clearances. In this second case the PBR #1 design was modified by reducing the diameters of both the core cavity and the PCRV by 3 meters.

Containment buildings for the 3000-MW(t) HTGR and PBR systems were designed by modifying the containment design presented in ref. 1. Height and diameter clearances between the containments and the PCRVs were maintained at the same values as those presented for the direct-cycle gas-cooled reactor described in ref. 1.

Reference PCRV designs for 1000-MW(t) systems were not available for either the HTGR or the PBR. A design for the HTGR was obtained by determining the PCRV diameter from a graph of PCRV diameter vs. plant power output contained in ref. 3 and maintaining the PCRV height the same as for the 3000-MW(t) design. The PCRV diameter for PBR #1 was determined by scaling the net concrete section so that it was reduced in the same proportion as for the HTGR in going from 3000-MW(t) to 1000-MW(t). The PCRV geometry for PBR #2 was determined by reducing the diameters of the core cavity and PCRV by 3 meters for the reasons cited above. PCRV heights for the two PBR cases were maintained the same as for the 3000-MW(t) cases.

The containment buildings for the 1000-MW(t) cases were sized in the same manner as for the 3000-MW(t) cases.

The final design parameters for the various systems are presented in Table 3.3.1.

3.3.3. Summary of Results

The development of cost data for the PCRVs and containment buildings for the 3000-MW(t) and 1000-MW(t) plants involved several iterations which are documented in refs. 4-13. Estimated cost factors (materials, quantity take-offs, and unit costs) for the PCRVs are presented in Table 3.3.2 and for the containment buildings in Table 3.3.3.

Differences in total costs (PCRV + containment) between the two PBR cases and the reference HTGR are summarized in Table 3.3.4 for the two power levels considered. The results indicate that the cost penalty for the PBR with an 850°C outlet temperature ranges from $9.[illegible] million to $26.5 million for the 1000-MW(t) power level and from $15.5 million to $37 million for the 3000-MW(t) power level.

While these results are cited for gas turbine designs, the PCRVs and containment structures associated with the other reactor applications, namely, steam cycle and process heat cases, should have relative cost differentials about the same as those given above. Further, to a first approximation, the results should also be applicable for all three coolant outlet temperatures (750°C, 850°C, and 950°C). It should be remembered, however, that these cost estimates do not represent cost extremes, since the mean cost of installed concrete could exceed the $500 per cubic yard value used in the overall evaluation. The comparative evaluation of the HTGR and PBR considered a maximum total cost differential of $50 million (PBR disadvantage) for the PCRV and containment buildings of the 3000-MW(t) systems.

Table 3.3.1. HTGR and PBR Design Parameters Used to Develop PCRV and Containment Building Capital Costs

Parameter	1000-MW(t) Systems			3000-MW(t) Systems		
	HTGR	PBR #1	PBR #2	HTGR	PBR #1	PBR #2
Pressure (psi)	700	700	700	1120	1120	1120
Coolant outlet (°C)	850	850	850	850	850	850
Core cavity, diameter (ft)	24.0	37.7	27.9	37.0	52.8	43.0
Core cavity, height (ft)	47.3	50.8	50.8	47.3	50.8	50.8
PCRV, diameter (ft)	103.0	123.3	113.5	128.5	152.0	142.2
PCRV, height (ft)	115.0	130.3	130.3	115.0	130.3	130.3
Containment diameter (ft)	133.5	153.8	144.0	159.0	182.5	172.7
Containment height (ft)	205.0	220.3	220.3	205.0	220.3	220.3

Table 3.3.2. Estimated Pressure Vessel (PCRV) Capital Cost Factors for 1000-MW(t) and 3000-MW(t) HTGR and PBR Systems

Material	Unit Cost	HTGR Quantity	HTGR Cost ($10³)	PBR #1 Quantity	PBR #1 Cost ($10³)	PBR #2 Quantity	PBR #2 Cost ($10³)
1000-MW(t) Systems							
Concrete	$185/CY*	27,843	5,151	47,913	8,864	40,067	7,412
Liners, penetrations, cooling tubes	$13,500/ton	3,159	42,647	3,398	45,873	3,284	44,334
Horizontal tendons	$74/LF	36,288	2,685	43,109	3,190	39,816	2,946
Vertical tendons	$74/LF	51,360	3,801	74,956	5,547	62,103	4,596
Circumferential steel	$1,240/ton	1,944	2,411	2,999	3,719	2,043	2,533
Reinforcing steel	$1,560/ton	898	1,401	1,599	2,494	1,337	2,086
Insulation	$300/SF	72,023	21,607	75,844	22,753	73,270	21,981
Total		*@ $185/CY	79,703		92,440		85,888
		*@ $300/CY	82,905		97,950		90,496
		*@ $500/CY	88,474		107,533		98,510
3000-MW(t) Systems							
Concrete	$185/CY*	46,500	8,603	75,840	14,030	66,301	12,266
Liners, penetrations, cooling tubes	$13,500/ton	3,300	44,550	3,600	48,600	3,465	46,778
Horizontal tendons	$74/LF	44,856	3,319	52,752	3,904	49,459	3,660
Vertical tendons	$74/LF	87,840	6,500	124,747	9,231	117,035	8,661
Circumferential steel	$1,240/ton	3,736	4,633	5,194	6,441	4,620	5,729
Reinforcing steel	$1,560/ton	1,500	2,340	2,530	3,947	2,212	3,451
Insulation	$300/SF	75,200	22,560	80,400	24,120	77,361	23,208
Total		*@ $185/CY	92,505		110,273		103,753
		*@ $300/CY	97,852		118,995		111,377
		*@ $500/CY	107,152		134,163		124,638

Table 3.3.3. Estimated Containment Building Capital Cost Factors for 1000-MW(t) and 3000-MW(t) HTGR and PBR Systems

Material	Unit Cost	HTGR Quantity	HTGR Cost ($10^3)	PBR #1 Quantity	PBR #1 Cost ($10^3)	PBR #2 Quantity	PBR #2 Cost ($10^3)
1000-MW(t) Systems							
Concrete	$185/CY*	67,519	12,491	74,006	13,691	71,159	13,164
Liner steel	$13,500/ton	1,596	21,546	1,712	23,112	1,669	22,532
Reinforcing steel	$1,560/ton	5,785	9,025	6,341	9,892	6,097	9,511
Structural steel	$1,200/ton	487	584	561	673	525	630
Tendons	$5,126/ton	987	5,059	1,310	6,715	1,148	5,885
Total		*@ $185/CY	48,705		54,083		51,722
		*@ $300/CY	56,470		62,594		59,906
		*@ $500/CY	69,974		77,395		74,138
3000-MW(t) Systems							
Concrete	$185/CY*	74,700	13,820	83,104	15,374	79,871	14,776
Liner steel	$13,500/ton	1,700	22,950	1,872	25,272	1,795	24,233
Reinforcing steel	$1,560/ton	6,400	9,984	7,120	11,107	6,843	10,675
Structural steel	$1,200/ton	580	696	666	799	630	756
Tendons	$5,126/ton	1,400	7,176	1,844	9,452	1,652	8,468
Total		*@ $185/CY	54,626		62,004		58,908
		*@ $300/CY	63,216		71,561		68,093
		*@ $500/CY	78,156		88,182		84,068

Table 3.3.4. Differences in Total Costs (PCRV + Containment Building) for PBR Systems Relative to HTGR Systems

	Total Cost Difference ($\$10^6$)			
	1000-MW(t) Systems		3000-MW(t) Systems	
Concrete Cost	PBR #1	PBR #2	PBR #1	PBR #2
\$185/CY	+18.1	+ 9.2	+25.1	+15.5
\$300/CY	+21.2	+11.0	+29.5	+18.4
\$500/CY	+26.5	+14.2	+37.0	+23.4

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14. "PCRV Estimate Review, HTR Core Evaluation," Letter from E. O. Winkler, GAC, to P. R. Kasten, ORNL dtd. January 18, 1980.

15. "Comments on PCRV Costs," Letter from A. J. Lipps, GE, to P. R. Kasten, ORNL, dtd. January 11, 1980.

3.4. FUEL FABRICATION AND RECYCLE UNIT COSTS

3.4.1. Introduction

The primary objectives of the tasks described here were to develop and document consistent unit cost estimates for (1) the fabrication of makeup fuel elements and refabrication of recycle fuel elements for the PBR and HTGR and (2) the reprocessing of the same fuel elements, together with costs for spent fuel transportation. In addition, the tasks were to identify the required research, development, and demonstration (RD&D) needed to bring each process to a level suitable for commercial application, and to provide estimates of the cost and time required for the RD&D effort.

3.4.2. Fabrication and Refabrication Unit Costs

A. R. Olsen

While all fuel fabrication/refabrication cost estimates should be specific to a given fuel element design and fuel cycle, it was not possible to provide such estimates for the fuel cycles assumed in this comparative evaluation of the PBR and HTGR because detailed fuel element and fuel cycle descriptions were not available. Instead, the cost estimates were based on four selected fuel cycles which had been analyzed earlier and for which the necessary fuel element descriptions and reactor mass flow data needed to define fabrication process requirements and to derive commercial-scale plant production capacity requirements were available. This approach is reasonable, since for a given type of fuel element the fabrication processes and equipment requirements are similar, and consistent and realistic estimates can be obtained for other cycles by the appropriate combination of major process equipment items and their corresponding costs. The four base cases were:

(1) A PBR once-through cycle utilizing U/Th fuel enriched to <20% ^{235}U and identified as the M020 cycle;[1]

(2) An HTGR once-through cycle utilizing U/Th fuel enriched to <20% ^{235}U and identified as the MEU(5)-Th(OT) cycle;[2]

(3) A PBR recycle case utilizing 93% ^{235}U/Th in the initial fuel and identified as the Th/U cycle;[3]

(4) An HTGR recycle case utilizing 93% ^{235}U/Th in the initial fuel and identified as the HEU(5)/HEU(3)-Th cycle.[4]

The parameter details for these cycles are summarized in Tables 3.4.1 through 3.4.5. From these, fabrication requirements for the cycles were assessed and the cost estimates of commercial plants that would support 20 reactors of each type were obtained. These then provide the basic process concepts and equipment costs to be used in assessing any other fuel element designs or cycle definitions in the PBR vs. HTGR assessment.

Table 3.4.1. Comparison of PBR and HTGR Characteristics in Four Base Cases*

	Once-Through Cases		Recycle Cases	
	PBR, MO20	HTGR, MEU(5)-Th(OT)	PBR, Th/U	HTGR, HEU(5)/HEU(3)-Th
Design MW(t), Q_t	3000	3360	3000	3360
Design MW(e), Q_e	1200	1332	1200	1332
Plant efficiency Q_e/Q_t	0.40	0.396	0.4	0.396
Load factor	0.80	0.75	0.8	0.75
Electricity generation [MW(e)/yr]	960	999	960	999
Fuel elements per core	3,233,370	5288	3,233,370	5288
Fuel elements replaced per year	1,111,720	1322	1,497,450	1322
Core power density (W/cm^3)	5.0	7.1	5.0	7.1
Average burnup (MWd/kg HM)	100	133	36.5	59.5
Fuel residence time (days, avg.)	872	1460		
Conversion ratio (avg. during equilibrium)	0.575	0.54	0.824	0.75
N_C/N_{HM}	458	395	220	169

*Note: Cycle designations used here are consistent with those used in the corresponding referenced publications: MO20 in ref. 1; MEU(5)-Th(OT) in ref. 2; Th/U in ref. 3; and HEU(5)/HEU(3)-Th in ref. 4.

Cost Estimates

The fuel fabrication cost estimates presented here[5] have been derived from a variety of previous studies and publications and must be considered preliminary in that plant design requirements, equipment designs, and capacities are in many instances based on concepts and not on tested equipment. This is particularly true for the PBR fuel, for which less information was available than for the HTGR. It had been hoped that through visits to HOBEG in Germany and to General Atomic Company the process descriptions could be reviewed and that additional functional flowsheet process information could be obtained, together with as much information as possible on process status, equipment designs, and operating experience. This type of consultation would have been particularly valuable for a realistic evaluation of the PBR element fabrication processing and costs because of the limited amount of detailed information on the processes and the equipment requirements available in the literature. The HOBEG visit could not be arranged, however, and in developing the PBR estimates it was necessary to depend on telephoned comments of the HOBEG staff on draft material forwarded to them in early December 1979. By contrast, the HTGR estimates were based on a large amount of detail from several recent U.S. studies.

Table 3.4.2. Fuel Element Loadings and Carbon Ratios for Four Base Cases[a]

	Once-Through Cases			Recycle Cases					
							HTGR, HEU(5)/HEU(3)-Th		
								Refabrication	
	PBR, MO20		HTGR, MEU(5)-Th(OT), Makeup M	PBR, Th/U			Fabrication, Makeup M	^{233}U Recycle, 23R	^{235}U Recycle, 25R
	Makeup MI[b]	Makeup MO[b]		Fabrication F1	Refabrication R1	Fabrication T1			
Fuel elements per year	898,806	221,912	1322	171,000	609,000	716,000	500	733	89
Loading (g/element)									
Total U	2.76	4.60	2920	1.24	1.86	-	740	650	1240
^{235}U	0.55	0.92	0560	1.15	0.90	-	690	30	530
^{233}U	-	-	-	-	0.20	-	-	490	-
Th	5.31	3.47	2490	15.22	14.60	16.45	11,240	11,240	11,240
Tot. HM[c]	8.07	8.07	5410	16.46	16.46	16.45	11,[illegible]0	11,890	1[illegible],480
Carbon ratios									
C/Th			950				180	180	180
C/U			740				2730	3110	1630
C/HM	458	458	395	220	220	220	169	170	162
Shipping requirements[d]	U	U	U	U	S	U	U	S	S

[a]Note: Cycle designations used here are consistent with those used in the corresponding referenced publications: MO20 in ref. 1; MEU(5)-Th(OT) in ref. 2; Th/U in ref. 3; and HEU(5)/HEU(3)-Th in ref. 4.

[b]MI, MO = inner (outer) makeup elements; all element notation on this line corresponds to that used in refs. 1-4.

[c]HM = heavy metal.

[d]U = unshielded; S = shielded.

Table 3.4.3. Description of Fissile and Fertile Particles Employed in Four Base Cases[a]

	Fissile Particles							Fertile Particles		
	Once-Through		Recycle					Once-Through[b]	Recycle	
			PBR		HTGR					
	PBR,[b] MI,MO	HTGR, M	FI	RI	M	23R	25R	HTGR, M	PBR, TI[c]	HTGR, M,23R,25R
Composition	(U,Th)O_2	UCO	(U,Th)O_2	(U,Th)O_2	UCO	UCO	UCO	ThO_2	ThO_2	ThO_2
Enrichment	20% ^{235}U	20% ^{235}U	93% ^{235}U	5% Fissile	93% ^{235}U	80% Fissile	43% ^{235}U	-	-	-
Kernel diameter (μm)	400	350	400	400	195	300	300	500	500	500
Kernel density (g/cm^3)	9.5	10.8	9.5	9.5	10.8	10.8	10.8	9.9	9.9	9.9
Buffer thickness (μm)	85	105	85	85	110	105	105	85	95	95
IIL thickness[d] (μm)	30	35	30	30	35	35	35	-	-	-
SiC thickness (μm)	-	35			35	35	35	-	-	-
OIL thickness[d] (μm)	80	45	80	80	40	40	40	75	80	80
Avg. particle diameter (μm)	790	790	790	790	635	730	730	820	850	850

[a]See Table 3.4.2 for identification of notation used in column headings; see Fig. 3.4.1 for sketches of particles.

[b]On PBR once-through cycle, only one type of particle is employed.

[c]Numbers for PBR recycle case are assumed.

[d]IIL, OIL = inner (outer) isotropic layer. For PBR fissile particles, these are high-temperature isotropic (HTI) layers; for PBR fertile particles and for HTGR fissile and fertile particles, they are low-temperature isotropic (LTI) layers (see Fig. 3.4.1).

Table 3.4.4. Descriptions of HTGR Fuel Rods and Fuel Elements[a] in Base Cases

	Once-Through, M	Recycle, M	Recycle, 23R, 25R
Fuel Rods			
Diameter (cm)	1.17	1.58	1.58
U content (g)	1.96	0.40	0.75
Th Content (g)	1.67	6.79	6.79
Fuel Elements			
Number of fueled holes	138/76[b]	138/76[b]	138/76[b]
Diameter of fueled holes (cm)	1.20		
Fuel rods/element	1656/851[b]	1656/851[b]	1656/851[b]
Number of cooling holes	72/43[b]	72/43[b]	72/43[b]
Diameter of cooling holes (cm)	2.10	2.10	2.10
Poison content (kg)	0.01	0.01	0.01
Graphite (kg)	99	94	94

[a]See Table 3.4.2 for identification of notation used in column headings.
[b]Standard elements/control elements.

Table 3.4.5. Description of PBR Pebbles in Base Cases

	Once-Through and Recycle, (MI, MO) and (FI, RI, TI)*
Ball diameter (cm)	6
Graphite shell thickness (cm)	0.5
Graphite density (g/cm^3)	1.7

*See Table 3.4.2 for identification of notation.

The fuel fabrication costs were estimated by breaking the fabrication process down into functional areas or sequential steps and then estimating the capital and operating costs for each step, using a methodology described elsewhere.[6] In order to permit uniform determinations of product costs to the reactors, the cost estimates for each of the four base cases were accumulated into four categories: facility capital costs, equipment capital costs, annual materials costs, and annual operating costs. The economic assessment used to define the costs to the reactor for these preliminary assessments was that used in a recent comparative cost estimate study.[7] Although this economic assessment can include a variety of financing assumptions, only those associated with a typical industry will be reported here.

Because of the inherent uncertainties associated with such estimates, the uncertainty associated with each cost category was assessed and the estimates were adjusted accordingly to obtain a probabilistic range. Then the cost to the reactor was calculated for both a high cost (i.e., a 90% probability that this cost will not be exceeded) and a low cost (i.e., a 20% probability that the actual cost will be less than this cost) to provide a range of cost estimates for each case. The results are given in Table 3.4.6.

Table 3.4.6. Preliminary Cost Estimates for Makeup Fuel Fabrication and Recycle Fuel Refabrication (Constant 1980 Dollars)

Reactor/Cycle Case	Fuel	Fabrication Costs ($/kg HM)* High	Low	Most Probable
Once-Through Cases				
PBR, MO20	Makeup	1390	730	1170
HTGR, MEU(5)-Th(OT)	Makeup	1510	1070	1380
Recycle Cases				
PBR, Th/U	Makeup	770	410	640
	Recycle	2440	1530	2190
	Mid-life reload weighted average	1450	860	1270
HTGR, HEU(5)/HEU(3)-Th	Makeup	880	610	800
	Recycle	1890	1300	1740
	Mid-life reload weighted average	1510	1040	1380

*These estimates do not include the cost for uranium or thorium, since plants are assumed to operate as toll processing facilities utilizing customer-supplied heavy metals. A 1% loss of such material is assumed during processing but not included in the costs.

It should be reemphasized that each of these cost estimates is cycle specific with plant capacities defined to provide equilibrium or mid-life cycle annual reloads for 20 reactors operating on the given cycle. Thus, with the possible exception of the once-through cases where the plant capacities are nearly equivalent, direct comparisons between costs for fabrication of different types of fuel elements should not be attempted.

As indicated in the introduction to this phase of the study, the base cases were selected cycles from other studies which had utilized different economics in arriving at "optimum" cycle definitions. PBRs and HTGRs are both highly adaptable machines in that core loadings, and consequently fuel element designs, can be adjusted to adapt to varying design requirements. For example, conversion ratios can be increased by increasing the fissile-to-fertile ratio, adjusting the carbon-to-heavy-metal ratio, and/or lowering the fuel residence times to conserve fissile uranium. Thus for a true comparison of fuel fabrication costs, it is necessary to iterate fabrication cost estimates with core design calculations and specific design details for the fuel elements to derive economic optimums. Such iterations could not be done within the time and funding restrictions for this preliminary assessment.

To provide some guidance for possible future studies of this type, the effects of varying some fuel element design characteristics and of varying plant production capacities on PBR element refabrication were done. Similar work was not done for the HTGR elements because such guidance exists from previous studies.

One possible variation for PBR element designs is to increase the total heavy metal in each element. The specifics on fuel cycle mass flow and fuel performance were not examined, but the fissile-to-fertile ratios for the base recycle cases were used as a partial normalization. The most probable cost estimates, based on normal industry economics and constant 1980 dollars, are given in Table 3.4.7. It will be noted that increasing the heavy metal loadings by 22% and 73% reduces the refabrication costs by 13% and 22% respectively in terms of dollars per kilogram of heavy metal loading. If the fuel burnup potential is proportional to the heavy metal loading, then significant savings in the fuel refabrication costs are possible. However, this may not balance the additional fuel inventory charges which will be incurred.

Another variable which was investigated was a change in the coated particle designs for refabricated PBR fuel elements. This design change should not alter the core inventory or fuel burnup potential but only reduce the amount of heavy metal which would have to be remotely processed into coated particles without significantly affecting subsequent fuel fabrication process steps. In this case it was assumed that half the heavy metal loading was made up of LTI-BISO-coated* ThO_2 particles fabricated in a contact operated and maintained portion of the plant (see Fig. 3.4.1). The remainder of the thorium and all of the uranium was contained in HTI-BISO-coated* $(U,Th)O_2$ particles processed in the remotely operated and remotely maintained portion of the refabrication plant. Table 3.4.8 shows that this design modification results in an 11% reduction in the estimated costs in terms of dollars per kilogram of heavy metal. A similar cost penalty for remotely processing thorium for use in an HTGR element to provide on-site denaturing has been identified in an earlier INFCE† study. For this design modification two limitations should be noted: First, proper metering and blending of the two types of coated particles is required to assure homogeneity. While this has been demonstrated for coated particles made up into fuel rods for HTGR elements, no similar demonstration for overcoated particles for PBR elements has been reported. Second, the cost advantage for fabrication, where all processes are contact operated and maintained, will be severely reduced and because of the mixing for homogeneity may even translate into a disadvantage.

The third parameter evaluated briefly is that of the PBR fuel refabrication plant capacity, or scaling. The results are given in Table 3.4.9, where the plant capacity has been lowered or raised by a factor of two. It will be noted that lowering the plant capacity to 50% results in an estimated cost increase of 23%, while increasing the plant capacity to 200% results in an estimated cost decrease of 15%. Since the reactor mass flow is dependent on cycle design and mass flows in effect dictate plant capacities, it is apparent this can have a significant effect on fabrication costs. In general the PBR base cases in this study had lower burnup and consequently higher mass flows. This led to larger capacity plants and somewhat lower unit costs in terms of dollars per kilogram of

*LTI, HTI = low (high) temperature isotropic layer; BISO = two-layer pyrocarbon coating, low-density inner layer (buffer) and high-density isotropic outer layer.

†International Nuclear Fuel Cycle Evaluation.

Table 3.4.7. Effects of Increasing Heavy Metal Loading on PBR Fuel Refabrication Costs[a] (Constant 1980 Dollars)

Case	Pebble Loading (kg HM/Element)	Most Probable Cost Estimates	
		\$/kg HM	\$/Element
PBR, U/Th Recycle Case	16.46	2190	36.05
Higher HM Loading[b]	20	1900	38.00
Highest HM Loading[b]	25	1700	42.50

[a]All plants were designed to produce the same number of fuel elements per year.

[b]Constant fissile-to-fertile ratio.

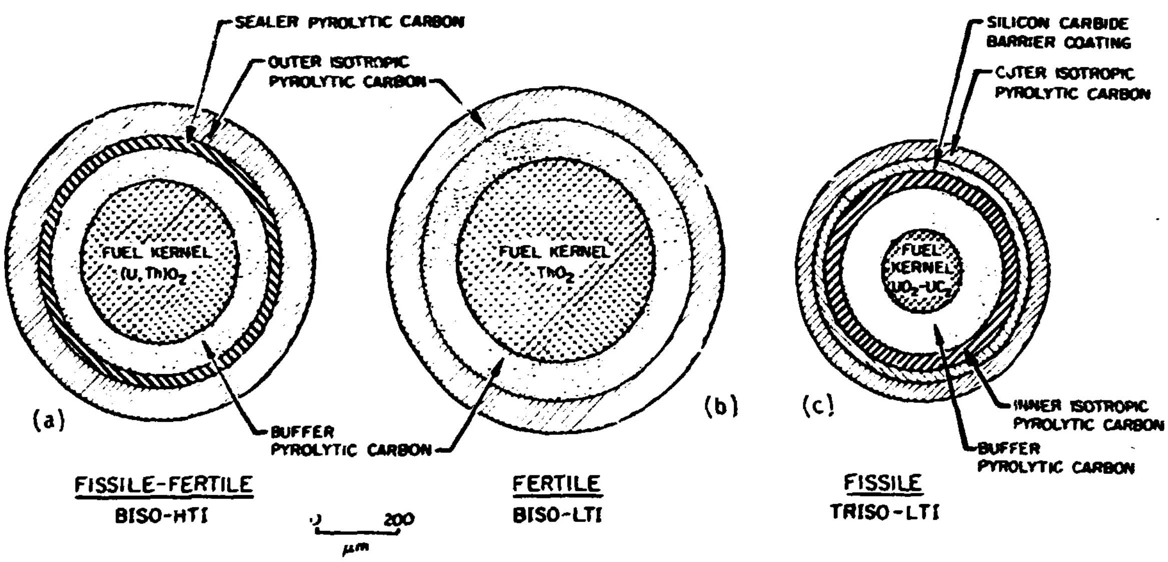

Fig. 3.4.1. Typical Coated Fuel Particles: (a) PBR fissile particles, (b) PBR fertile particles, and (c) HTGR fissile particles.

Table 3.4.8. Effects of Changing Coated Particle Design on PBR Fuel Refabrication Costs (Constant 1980 Dollars)

	Pebble Loading (g HM/Element)		Most Probable Cost Estimates	
	$(U,Th)O_2$	ThO_2	\$/kg HM	\$/Element
PBR, U/Th Recycle Case	16.46	0	2190	36.05
Modified Loading with				
50% $(Th,U)O_2$ HTI-BISO-coated and 50% ThO_2 LTI-BISO-coated	8.23	8.23	1940	31.90

Table 3.4.9. Effects of Varying Plant Capacity on PBR Fuel Refabrication Costs (Constant 1980 Dollars)

Plant Capacity		Most Probable Cost Estimates	
MT HM/yr	Pebbles/yr (10^6)	\$/kg HM	\$/Element
200.4*	12.17	2190	36.05
100.2	6.08	2690	44.30
400.8	24.35	1880	30.95

*Reference case.

heavy metal. However, when the lower power generation production per kilogram of heavy metal is considered, the fabrication cost contribution to power cost is higher for the elements.

Technology Status and R&D Requirements

The assessment of the technology status for the fabrication and refabrication of fuel elements for the PBR or the HTGR shows the technology to be fairly well advanced. All process steps have been demonstrated and much of the equipment has been demonstrated on at least an engineering scale. The research, development and demonstration (RD&D) requirements to proceed to the design of commercial-scale plants derive primarily from economic considerations and from requirements for statutory compliance to obtain a license to build and operate such plants. As is shown in Table 3.4.10, the overall RD&D cost estimates and the time required for completion depend significantly on whether or not recycle with remote refabrication is included. There is no significant difference in the estimates for the prismatic or pebble fuel developed separately and only a small incremental cost addition if they are developed together.

Table 3.4.10. Estimated Costs (in 1980 Dollars) and Time Requirements for Fuel Fabrication Research, Development, and Demonstration[a,b]

	HTGR Fuel Only		PBR Fuel Only		HTGR + PBR (Combined RD&D)	
	Cost (\$$10^3$)	Time Required (yr)	Cost (\$$10^3$)	Time Required (yr)	Cost (\$$10^3$)	Time Required (yr)
Once-Through Fuels, Fabrication	72,000	6	72,500	6	88,800	7
Recycle Fuels, Fabrication and Refabrication	180,800	11	192,100	11	215,100	11

[a]Costs do not include fuel performance verification irradiation tests, which would add \$25 to \$30 million for each fuel cycle type to assure product use licensability.

[b]Does not include consideration of detailed design and construction of a commercial-scale plant; conceptual design is included in the RD&D costs and time.

3.4.3. Reprocessing Unit Costs

(Note: The following information has been extracted by the editors from a General Atomic Company report[7] authored by L. Abraham, B. B. Haldy, and J. A. Oita.)

In estimating the costs for PBR or HTGR fuel reprocessing, the technique was again to first determine the processes and equipment required to build a plant sized to service a reactor economy of about 20 GW(e) installed capacity. The results were then used as a basis for arriving at unit costs in terms of dollars per kilogram of heavy metal processed. Common economic assumptions used for the study included consideration of common licensing regulations for construction and operation of the reprocessing plant and consistent equipment, material, and labor costs.

As has been stated earlier, both the HTGR and the PBR can be operated on a variety of fuel cycles with a variety of fuel exposures. This flexibility in fuel design and in fuel cycle variables, such as enrichment, carbon-to-heavy-metal ratios, fuel element residence time in the reactor, power density, and fuel burnup, mandates that reprocessing unit cost estimates be specific with regard to fuel cycle conditions. In the absence of such specification for the systems of interest in this comparative study, the reprocessing cost estimates given here were based on the same recycle fuels and fuel element designs as were the refabrication estimates (see Tables 3.4.1 through 3.4.5). In addition to these, incremental costs associated with a variation in the PBR fuel element design were considered.

The flow diagrams for the PBR and HTGR fuel cycles are shown in Figs. 3.4.2 and 3.4.3 respectively. The various processes and storage requirements were evaluated for both reactor concepts, including those for fuel element reduction or crushing, burning, dissolution, feed adjustment, solvent extraction, product handling and waste treatment. The plant design bases considered fuel receiving and storage, spent fuel reprocessing, product handling, waste treatment, off-gas treatment, and support facilities, including administration buildings, stores, maintenance shops, and other general services. The reprocessing plant design assumptions are summarized in Table 3.4.11.

Cost Estimates

Considering the above parameters and factors, the specific processes, and associated equipment requirements, capital costs for the reference fuel reprocessing plants were estimated on the basis of previous studies of HTGR fuel reprocessing plants. The reprocessing plant costs considered the reprocessing plant design assumptions, the fuel cycle mass flows, the process stream characterization, the reprocessing equipment and support requirements, and also the associated building and storage requirements. The resulting reprocessing plant throughputs and associated costs are summarized in Table 3.4.12. These estimates were based on BISO-coated fuel for PBRs (see Fig. 3.4.1). If the TRISO-coated fuels used in the HTGR had been assumed for the PBR, it would have increased the capital

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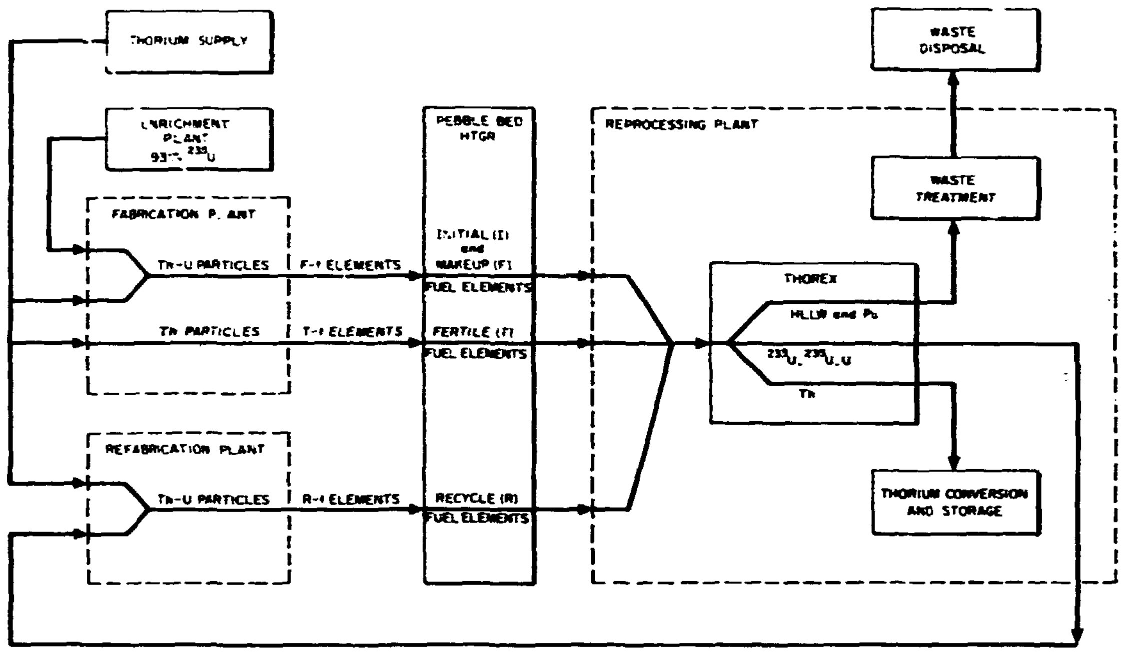

Fig. 3.4.2. Flow Diagram for Recycling PBR HEU Fuel.

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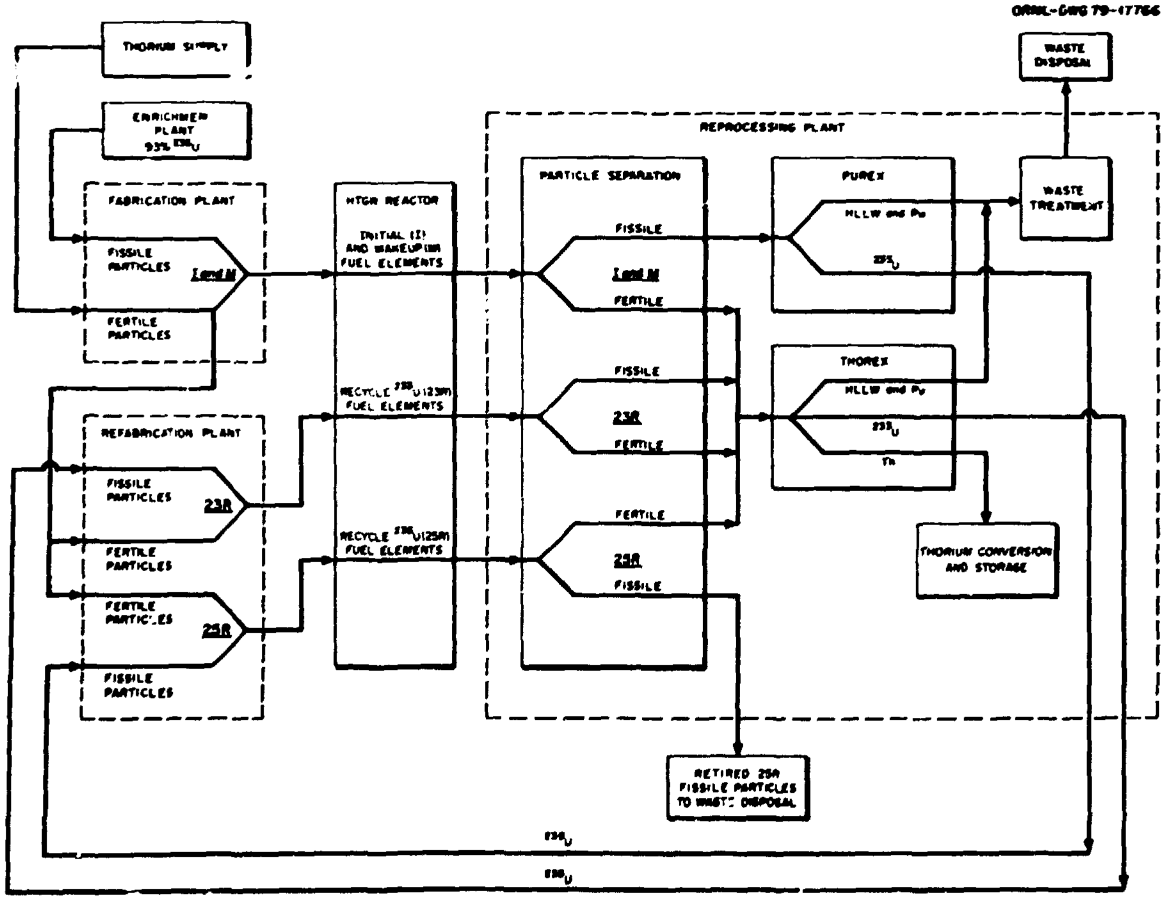

Fig. 3.4.3. Flow Diagram for Recycling HTGR HEU Fuel.

Table 3.4.11. Reprocessing Plant Design Assumptions

Design Characteristic	Assumption
Scope of facility	Self supporting
Overall design	Meets all federal, state, and local requirements for licensable commercial facility
Feed material	HTGR: Spent uranium oxycarbide, thorium oxide prismatic fuels (average 180-day cooled) PBR: Spent uranium-thorium oxide pebble bed fuels (average 180-day cooled)
Product	UO_3; ThO_2
Production capacity	HTGR: 20,000 prismatic fuel elements per year PBR: 20,000,000 pebble bed fuel elements per year
Design capacity	1.25 times average production capacity
Yield	98.5% recovery of fissile material
Surge storage	90-day spent fuel 60-day product UO_3 5-year product $Th(NO_3)_4$; 10-year solid ThO_2 5-year high-level liquid waste 1-year high-level solid waste
Maintenance philosophy	Remote replacement; out-of-cell repair varies - remote to contact
Operating life	30 years
Plant operating schedule	24 hr/day, 365 days/year
Efficiency/availability	72%
Effective operating days	263 days/year (233 days/year for head-end processing)
Safety	Administration, geometry, and/or neutron poison control for criticality Meets federal, state, and local requirements
Environment protection	No release to ground water Off-gas treatment of iodine, NO_x, tritium, radon, ^{14}C, krypton, semivolatile fission products, particulates, and combustibles Liquid wastes immobilized as insolubles for shipment Solid process wastes fixed in solid matrix for shipment
Safeguards and accountability	Near real-time accountability at sensitive process points plus semiannual inventory; supplemented by item count and weight measurements
Physical protection	Outer perimeter detection and defense system, physical barriers, and hardened defense posts.

Table 3.4.12. Reprocessing Plant Throughput Rates and Associated Costs

	HTGR	PBR
Fuel elements/yr	20,000	20,000,000
Carbon, kg/yr	1.88×10^6	3.67×10^6
Heavy metals (total), kg/yr	229.5×10^3	343.4×10^3
Reactor economy, net GW(e)	~20	~20
Capital cost (1978 \$) (\$$10^6$)		
Facility	171.5	210.6
Equipment	364.5	395.8
Total[a,b]	536.0	606.4
Operating cost (1978 \$) (\$$10^6$/yr)		
Hardware	21	24
Other	19	21
Total[c]	40	45

[a]Excluding replacement cost.

[b]The plants include 40% contingency on capital cost.

[c]The plants include 20% contingency on operating cost.

costs of the reference PBR reprocessing plant by approximately 2.6 to 4.0% and the operating costs by 1 to 2%.

Based on a capital charge rate of 20%/yr, the annual cost of operating the fuel reprocessing plants would be \$147 million/yr for the HTGR and \$166 million/yr for the PBR. With throughput rates of 230,000 kg HM/yr for the HTGR and 343,000 kg HM/yr for the PBR, the unit costs are \$639/kg HM for the HTGR and \$484/kg HM for the PBR. Adding a 4% penalty for TRISO-coated fuel would increase the PBR unit costs to \$500/kg HM.

As indicated previously, the above unit costs are based on reference fuel cycles and throughputs and they should not be interpreted to mean that unit costs for processing PBR fuel are inherently lower than those for processing HTGR fuel. For the same heavy metal loading, the same carbon content, and the same throughput rates, the unit costs of reprocessing PBR and HTGR fuel will be essentially the same. The difference in unit costs therefore largely reflect differences associated with different carbon throughputs and different heavy metal throughputs for specific fuel cycles.

The cost values given are considered to be the most likely values, with capital cost ranges of up to +10% and down to -20%; the operating costs given are also the most likely values, with an estimated range of +15% to -5%.

In addition to the reprocessing costs, transportation costs for spent fuels were estimated to be \$870,000 per GW-yr for the HTGR and \$1.32 million per GW-yr for the PBR. The higher value is associated with the higher volume of fuel that would be needed to be transported for the reference PBR fuel cycle.

Reprocessing R&D Requirements

The technical issues to be addressed in fuel reprocessing apply equally to PBR and HTGR fuels. A high degree of commonality is associated with the head-end and aqueous process systems for both reactor fuels and other technical issues are generic in nature between the two reactor fuel types and the required developmental resolution of design issues is also similar for either fuel concept.

Considering the commonalities and similarities of reprocessing development requirements for the PBR and HTGR, it is estimated that the research, development, and demonstration cost estimates for the two reactors are essentially the same.

In both cases a continuing reprocessing development program is required to demonstrate the feasibility of commercial reprocessing of high temperature reactor spent fuels. Information needed can be derived from an extensive development effort carried through appropriate laboratory, engineering prototype, and hot pilot plant stages. In the reprocessing development program for HTRs, most of the reprocessing functional areas have reached cold engineering and hot laboratory development states. R&D will be required relative to receiving and storage, head-end systems, dissolution and feed preparation systems, Thorex solvent extraction, Purex solvent extraction, gaseous effluent treatment, acid recovery and water recycle, product conversion, design studies, waste treatment, technical support, and safeguards methods. As shown in Table 3.4.13, these costs have been estimated to total $201 million, and hot pilot plant design and construction and hot pilot plant operation would increase the costs to $637 million. Specific modifications to the development program and pilot plant designs to accommodate both types of fuel elements would probably require an additional $3 million, for a total of $640 million.

Table 3.4.13. Estimated Costs and Time Requirements for Fuel Reprocessing Research, Development, and Demonstration[a]

	Cost ($\$10^3$)	Time Required (yr)
Through development[b]	201,000	11
Hot pilot plant design and construction	361,000	7
Hot pilot plant operation/demonstration	75,000	5[c]
Accommodations specific to PBR or HTGR fuel	3,000	
Total	640,000	

[a]Costs expected to be the same for HTGR and PBR fuels.

[b]Includes conceptual design of hot pilot plant.

[c]Includes cold checkout and training, plus four years of hot operations.

3.4.4. Summary and Conclusions

Consistent comparative fuel fabrication/refabrication and fuel reprocessing cost estimates for PBR and HTGR fuel elements have been made for selected base case cycles. The range of costs associated with the basic estimates have also been estimated. For both the fabrication and the reprocessing costs given, it is recognized that iterative analysis with specific modifications in fuel cycle conditions and fuel element designs could result in modifying the fabrication and reprocessing costs for the various elements With that qualification, these studies indicate that the costs of fabricating fresh or makeup fuel would be slightly less for the PBR than for the HTGR while the costs of refabricating recycle fuel would be slightly greater for the PBR. For reprocessing, it appears that the unit costs for PBR fuel would be slightly less than those for HTGR fuel, but this difference was associated with the differences in fuel loading and the heavy metal throughput of the reprocessing plants rather than with the differences in the reactor fuel concept per se. For the same carbon and heavy metal throughput, reprocessing plant unit costs would be essentially independent of reactor fuel types.

The assessment of the technology status and required research, development and demonstration of fuel fabrication and fuel reprocessing shows no significant difference between the different reactor fuels. For both reactor fuel types, processes are at a relatively high level of development and the required resources to achieve commercial status are essentially the same.

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3.5. IMPACT OF FISSION-PRODUCT RELEASES

M. F. Osborne and R. P. Wichner

(Summary of Work Performed at General Atomic Company*)

3.5.1. Introduction

The release of fission products from the reactor core into the primary coolant circuit during normal operation impacts the design of many system components in both the HTGR and the PBR, as well as plant operating and maintenance procedures. If, in addition, unexpected releases occur, an even greater impact is realized. For example, if the coolant radioactivity becomes sufficiently high that scheduled (or unscheduled) maintenance activities that normally would be performed while the reactor is operating must instead to be carried out during shutdown, then plant availability is affected. Not only are unanticipated and extended outages costly per se, but additional expenditures can accrue from the requirement for more personnel, equipment and/or shielding.

The costs associated with such releases inevitably will be determined by the fission-product radioactivities encountered, however, not necessarily in the same ratio for the two reactor types. Also, because of the differences in the HTGR and PBR cores, the releases for given outlet coolant temperatures will not be the same. Neither will they always be equivalent for given accident scenarios.

In the investigations described below, both normal and abnormal fission-product releases in the HTGR and PBR are compared, and the results are interpreted in terms of occupational and public dose-rate exposures, added downtime, and requirements for additional personnel and/or equipment, all, of which, of course, can be converted into costs.

3.5.2. Fission-Product Releases to Coolant Circuit During Normal Operation

(Summary of Work Reported by D. Hanson, General Atomic Company)

Calculational Method

Calculations were performed to determine the fission-product activities in the primary circuits of the three reactor pairs described in Table 3.5.1. It should be noted that the terms "Steam Cycle," "Gas Turbine," and "Process Heat" used in Table 3.5.1 to identify the reactor pairs refer solely to the indicated outlet coolant temperature ranges, since the procedures employed to calculate fission-product migration were not sophisticated enough to distinguish between different methods for extracting heat in the primary system. It should also be noted that the reactors in each pair are not completely equivalent. A

*Except for Section 3.5.5.

Table 3.5.1. Reactor Design Parameters for Calculations of Fission-Product Releases in Prism (HTGR) and Pebble Bed (PBR) Cores

Reactor Type	Thermal Power (MW)	Power Density (W/cc)	Fuel Lifetime (yr)	Coolant Temperature (°C) Outlet	Coolant Temperature (°C) ΔT Core	Fuel Particles[a] Fissile	Fuel Particles[a] Fertile	Fuel Element Type
Steam Cycle								
Prism	2240	7.1	4	690	382	UC_2-T	ThO_2-B	10 row (8)[b]
Pebble Bed	3000	5.5	4	700	382	UC_2-T	ThO_2-T	Ball
Gas Turbine								
Prism	2240	6.5	4	850	400	UC_2-T	ThO_2-T	10 row
Pebble Bed	3000	5.5	4	850	343	UC_2-T	ThO_2-T	Ball
Process Heat								
Prism	1530	6.5	4(3)[c]	950	500	UC_2-T	ThO_2-T	10 row
Pebble Bed	3000	5.5	4	950	650	UC_2-T	ThO_2-T	Ball

[a]T = TRISO-coated fuel; B = BISO-coated fuel.

[b]HTGR-SC calculated for both 10-row and 8-row fuel block.

[c]HTGR-PH calculated for both 4-yr and 3-yr lifetimes.

precise comparison would have required that each reactor type be optimized for the given coolant temperature range; however, this was beyond the scale of effort available for this study. In addition, reactor size was deemed not to be a sensitive parameter on a unit power basis; therefore, the higher thermal power of the PBR within each pair is not highly significant, although it does somewhat favor the PBR since smaller PBRs would have less favorable fuel temperature distributions because of the increased effect of power peaking near side reflectors.

As indicated in the table, the prism fuel element type was assumed to be a 10-row block; however, a coarser fuel dispersion (an 8-row block) was also studied for the steam cycle case. Also, a fuel lifetime of 4 yr was assumed for all the reactors with additional calculations performed for a 3-yr lifetime in the prism process heat case.

Neutron flux and power distributions were calculated for the prismatic cores with the GAUGE, FEVER, and TSORT codes and for the pebble bed concepts with the 2DB code. In general the estimates were somewhat more idealized for the pebble bed concepts in that the effects of control and shutdown rod actions were not accounted for, whereas full rod histories were incorporated in the GAUGE/FEVER calculations.

Fuel element temperature distributions and fuel failure fractions were generated for prismatic cores by the SURVEY code and for pebble bed cores by the KUGEL code.* The fuel particle assumed in all the calculations is that described in the Fuel Design Data Manual, Issue C, September 1979, which also lists the applicable fuel failure models, as well as procedures for calculating fission-product release from the core to the primary coolant system. The current fuel failure model is described in GA-LTR-15.

*KUGEL is an extension of the LASL PEBBLE code which incorporates GA's fuel failure model.

The SURVEY and KUGEL calculations yielded the release rates for gaseous fission products, but the release rates for cesium and strontium were calculated separately using TRAFIC for the prism cores and TRAMP for the pebble bed cores. The release rate formulas used for both reactor types wer those specified in GA's design data manual, with the diffusive and sorptive properties of graphite assumed to be those specified for graphite H451. (Note: At temperatures corresponding to the 700°C outlet coolant temperature, graphite properties are sensitive parameters and if some other graphite should be assumed, the calculated results for the steam cycle could be significantly in error.)

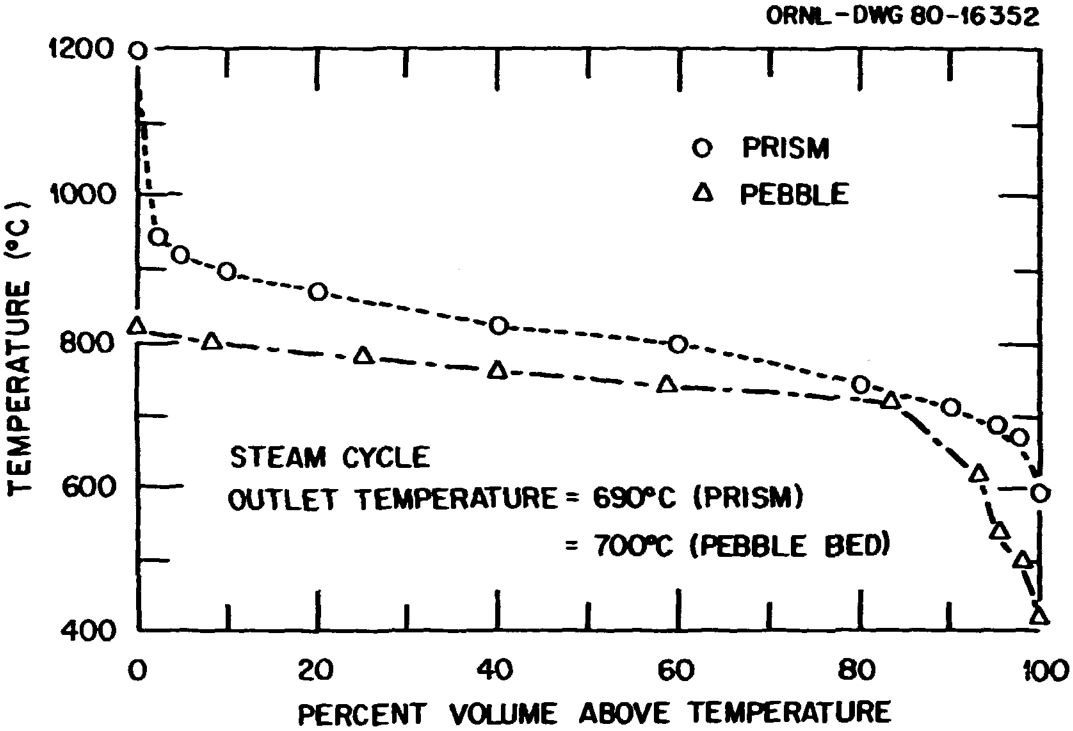
Fuel Temperature Distributions

Figure 3.5.1 presents the estimated temperature distributions in the fuel for the three reactor pairs. In each pair the fuel temperatures for the pebble bed core are significantly lower than those for the prism core. In particular, a high-temperature "tail" is shown to exist in about 5 vol% of the prismatic fuel but is absent in the pebble bed fuel. This tail is caused principally by coolant/power mismatches in the vicinity of fuel zone boundaries. To a first approximation, sharp boundaries between fuel zones do not exist for pebble bed cores; hence the high-temperature tails do not appear However, the control rod and shutdown rod actions which could cause fuel ball displacement and thereby create localized power mismatching were not considered in generating Fig. 3.5.1.

Fuel Performance and Fission-Product Releases

The fuel performance and fission-product activity releases in the primary coolant circuit for the three reactor pairs are compared in Table 3.5.2, where the categories under the fissile and fertile particle failures refer to failures due to manufacturing defects, failures due to gas pressure buildup within the particles, failures due to migration of the fuel inside the particle, and failures due to corrosion of the SiC layer by palladium attack.

Table 3.5.2 shows that the fissile failures for the 700°C (steam cycle) and 850°C (gas turbine) systems are low for both the pebble bed and prism cores. The significantly higher cesium and strontium releases for the HTGR-SC over the PBR-SC are due to lower retentions in the graphite at the higher prismatic core temperatures. (As noted earlier, graphite diffusion and sorption properties are highly sensitive parameters in the 700°C outlet temperature range.) The percent of fertile particle failures is approximately the same for all systems except for the HTGR-SC, which uses a BISO-coated particle as opposed to TRISO-coated particles in the other systems.



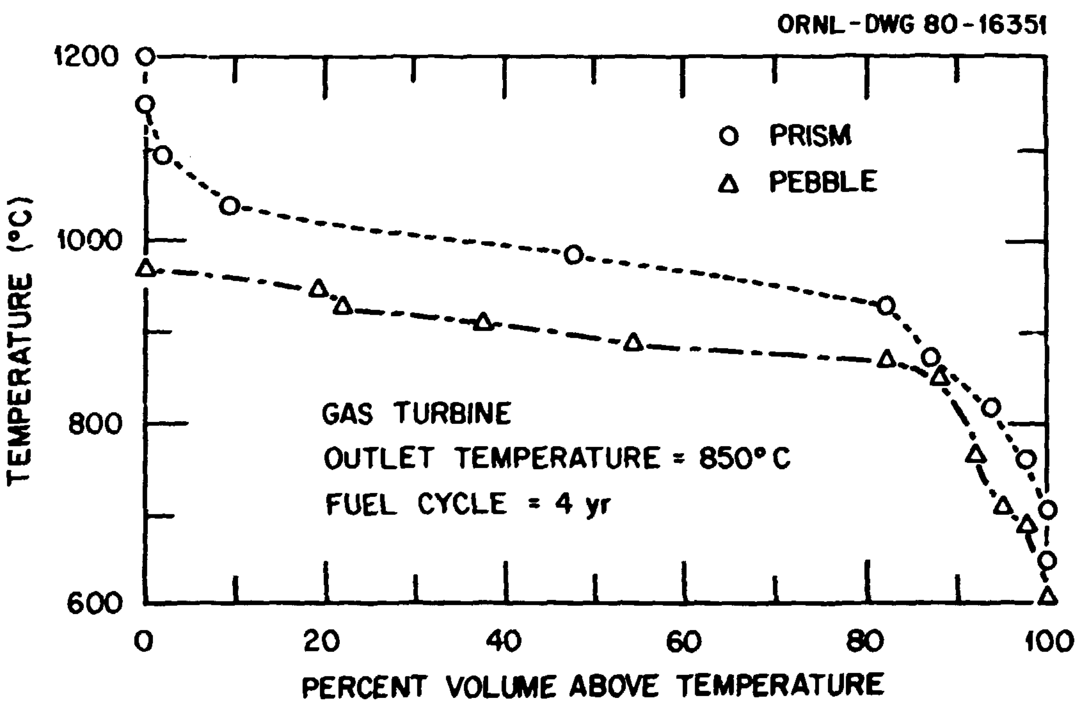


Fig. 3.5.1. Comparison of Fuel Temperature Distributions in PBR (Pebble Bed Core) and HTGR (Prism Core). All prism cases for 10-row fuel block (see Table 3.5.1).

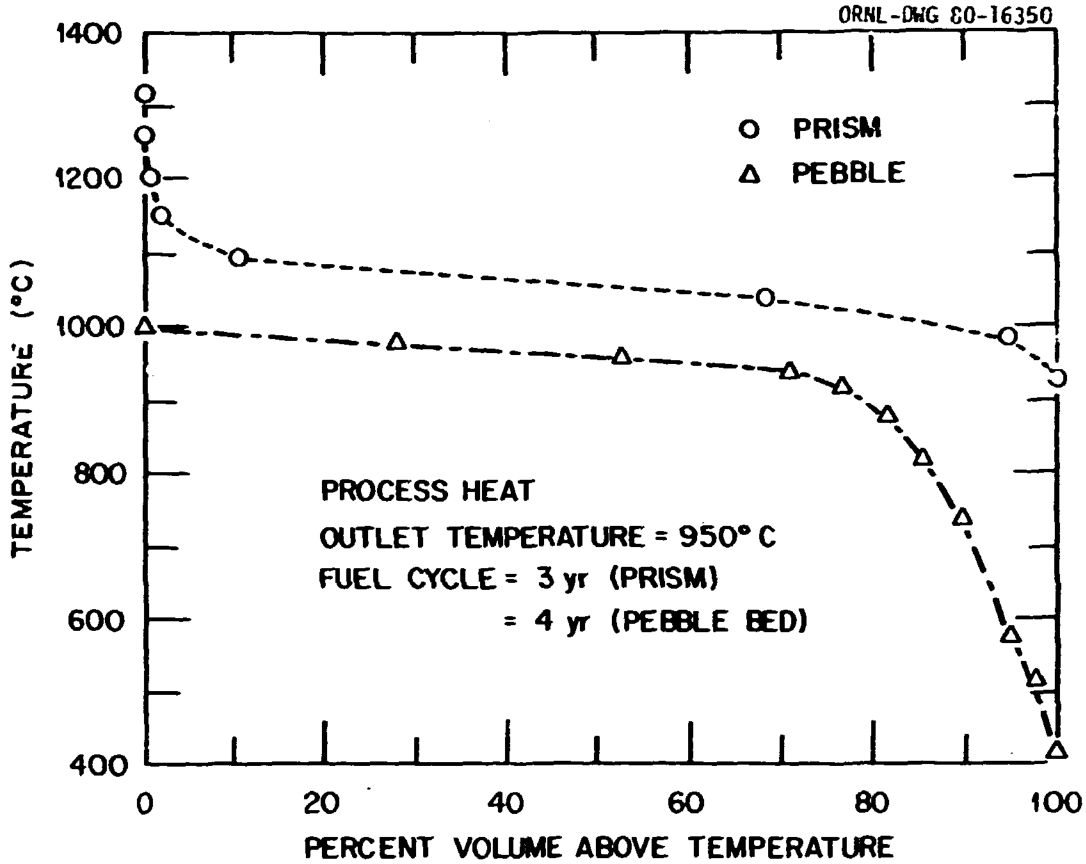


Fig. 3.5.1. (cont.)

At the 950°C outlet coolant temperature, significantly increased fuel failures appear for the prismatic fuel relative to the pebble bed fuel. The higher fuel temperatures induce increased SiC corrosion by the fission-product palladium and chemically similar metals. The term "Old Pd" in the table refers to estimates of the palladium attack rate which varies linearly with time as based on short-term test data and published in GA's design data manual. More recent test data based on longer term experiments indicate that the palladium attack rate may actually vary with the $\sqrt{time}$, which, if true, would yield fewer fuel failures for the high-temperature prism core as indicated in the columns labeled "Rev Pd."

3.5.3. Operational and Public Exposures Due to Coolant Radioactivities During Normal Operation

(Summary of Work Reported by A. Barsell, General Atomic Company)

The fission products circulating in the primary coolant during normal operation are a potential source of exposure both to reactor operating personnel, particularly those performing duties inside the containment building, and to the public outside the Exclusion Area Boundary (EAB). Fission-product dose rates were calculated parametrically and compared with established ALARA* limits for occupational exposure (100 mrem/week) and

*ALARA = as low as reasonably achievable.

Table 3.5.2. Comparison of Fuel Failures and Fission-Product Releases in PBR (Pebble Bed Core) and HTGR (Prism Core)

Parameter	Steam Cycle (T=700°C)[a]			Gas Turbine (T=850°C)		Process Heat (T=950°C)				
		Prism				Pebble Bed, 4-yr Lifetime,	Prism			
							4-yr Lifetime		3-yr Lifetime	
	Pebble Bed	10-row Fuel Block	8-row Fuel Block	Pebble Bed	Prism	Old Pd	Old Pd	Rev Pd	Old Pd	Rev Pd
Fissile particle failure (%)										
Defects	0.070	0.072	0.072	0.071	0.072	0.071	0.073	0.073	0.073	0.073
Pressure vessel	0.007	0.008	0.012	0.011	0.015	0.016	0.035	0.035	0.035	0.035
Kernel migration	—	—	—	—	—	—	—	—	—	
SiC corrosion	—	—	—	—	0.004	—	1.412	0.027	0.379	0.020
Total	0.077	0.080	0.085	0.082	0.091	0.087	1.52	0.135	0.488	0.128
Fertile particle failure[b] (%)										
Defects	0.061	0.023	0.023	0.061	0.060	0.061	0.062	0.062	0.058	0.058
Pressure vessel	—	0.088	0.090	0.001	0.002	0.001	0.011	0.011	0.004	0.004
Kernel migration	—	—	—	—	—	—	—	—	—	—
SiC corrosion	—	—	—	—	—	—	—	—	—	—
Total	0.061	0.111	0.113	0.062	0.062	0.062	0.073	0.073	0.062	0.062
Kr-88 circulating (Ci/MW)										
No hydrolysis of failed fuel	0.15	0.25	0.38	0.28	0.41	0.34	?	?	0.68	0.57
100% hydrolysis of failed fuel	0.38	1.22	1.85	0.73	1.34	1.14	19.1	2.3	4.25	1.61
40-yr plateout (Ci/MW)										
Cs-137	0.06	1.16	3.4	3.3	14.7	7.0	484	39.9	78.4	20.9
Sr-90	0.01	0.04	—	0.02	0.04	0.03	?	?	?	?

[a] T=Coolant outlet temperature.

[b] BISO-coated fertile particles assumed for HTGR-SC; all other cases assumed TRISO-coated particles.

for offsite exposures (5 mrem/yr to the whole body or 15 mrem/yr to the thyroid by inhalation). The assumptions and conditions used in these calculations were:

(1) The PCRV leakage rate was assumed to be 3.65%/yr.

(2) The analysis considered both an open containment (continuously purged up to 1 vol/hr) and a closed containment (semiannual purge).

(3) The containment was assumed to have a volume of 8×10^{10} cm^3 and a closed leak rate of 0.1%/day.

(4) On the basis of the expected absorption by concrete, an iodine decontamination factor of 100 was assumed for leakage through the PCRV. (I-131 is the nuclide controlling the thyroid dose rate.)

(5) The site boundaries and meteorology corresponded to those of a hypothetical plant whose site was less favorable than 90% of the sites of existing LWRs in the USA (so-called reference site).

(6) Kr-88 was assumed to be the dominant radionuclide contributor for both public and occupational exposures; thus, the analysis was keyed to Kr-88, with other nuclides contributing always in the same fixed proportion.

(7) The levels of circulating activity were designated as Level A (expected value) and Level B (design value), with Level B being about 4 times higher than Level A.

The relative importance of the offsite whole-body and thyroid dose rates was determined in a preliminary calculation using the Fulton HTGR as a base case. The circulating inventories (all nuclides) were increased proportionally until the offsite limit of a 5-mrem/yr whole-body dose rate or a 15-mrem/yr thyroid inhalation dose rate was reached. It was found that the whole-body limit was always reached first, regardless of the containment purge rate.

Subsequent calculations for a Level A circulating activity in the Fulton HTGR and the 3.65%/yr PCRV leak rate showed that the major radionuclide class contributing both to operating personnel exposures and to the airborne whole-body offsite exposures was the circulating noble gases (see Table 3.5.3). Kr-88 clearly dominates in all cases except for the occupational exposure with the closed containment, in which case Xe-133 and Kr-88 contribute approximately equally.

Examination of the results in Table 3.5.3 and the corresponding dose rates revealed that in most cases the inventory of circulating noble gases could be substantially increased without exceeding the maximum permissible dose rates. Table 3.5.4 shows, for example, that the Kr inventory could be increased by a factor of 1000 if containment access were not a consideration and the maximum offsite whole-body dose rate were the limiting factor. The factor of 1000 would correspond to an upper bound Level A circulating noble gas activity of 10^7 Ci, of which 2×10^6 Ci would be Kr-88. The corresponding upper bound of circulating I-131 would be 2200 Ci. However, to enable 40-hr/week containment access, the Kr-88 inventory would have to be limited to 3900 Ci and the I-131 inventory to 1.4 Ci, in which case the allowable increase factor (for a closed containment)

Table 3.5.3. Noble Gas Isotopes Contributing to Occupational and Public Dose Rates*

	Closed Containment (Purged Twice per Year)		Continuously Vented Containment (Vented at 0.5 vol/hr)	
	Containment Access Dose Rate	Offsite Dose Rate	Containment Access Dose Rate	Offsite Dose Rate
Kr-83M	7.3%	0.3%	15.0%	0.5%
Kr-85M	3.5	3.7	3.8	3.8
Kr-87	6.8	6.8	17.8	11.6
Kr-88	29.8	50.8	46.2	67.0
Rb-88	3.3	5.1	5.1	5.5
Xe-133	30.4	15.6	1.4	0.8
Xe-135	9.5	12.1	5.5	7.2
Xe-138	-	1.3	2.1	1.5
	90.6%	97.3%	96.9%	97.9%

*Level A activity, 3.65%/yr PCRV leak rate.

Table 3.5.4. Allowable Increase Factors for Kr Circulating Inventory* (Base Case = Fulton HTGR)

Containment purge rate (vol/hr)	0	0.2	0.5	1.0
Allowable Kr increase factors				
For offsite whole-body dose rate	1000	10	7.4	5.7
For occupational dose rate (at 40 hr/week)	1.5	4	7.4	11.0

*The allowable increase factors for I-131 are at least 10 times greater.

would be only 1.5. But under the condition of an open containment with a purge rate of 0.5 vol/hr, the noble gas inventory could be increased to 74,000 Ci (14,000 Ci Kr-88 and about 6.8 Ci I-131) without exceeding either the occupational dose rate (for 40-hr/week access) or the offsite whole-body dose rate. These results demonstrate that the noble gas component of the fission products, especially Kr-88, is the controlling factor both for offsite dose rates and for allowable containment access times. Offsite dose rates are directly proportional to circulating activity and increase with an increasing purge rate.

In light of the above, the Level B (design) circulating inventories of Kr-88 were calculated for each of the HTGR and PBR pairs, using the circulating levels given in Table 3.5.2 and assuming 100% hydrolysis of the failed fuel and an open containment purge rate of 0.5 vol/hr. The results, together with the corresponding offsite dose rates and containment access times, are presented in Table 3.5.5. In all cases, the dose rates for the PBR are lower than those for the HTGR (by factors of about 2 to 4) and are well below the maximum allowable dose rates. Also, even at the Level B inventories, a 40-hr/week containment access is permissible for all the PBRs and also for the steam cycle HTGR under the conditions of an open containment. Even with a closed containment, the access times are acceptable for these reactors. In the case of the gas turbine HTGR and the process heat HTGR, the offsite dose rates somewhat exceed the permissible limits, as would be expected from their Kr-88 inventories. It should be noted, however, that had Level A inventories been used as a basis for calculating the offsite dose rates, which has been allowed in past licensing of reactors, then all the reactors would be well below the maximum allowed dose rates. Even so, to meet ALARA limits for occupational exposure, some rotation of operational personnel may be required for the closed containment condition for the HTGR-GT and the HTGR-PH. At the same time the above results are based upon the assumption that the palladium attack of SiC varies linearly with time ("Old Pd" designation), which may not be valid. Using the "Rev. Pd" relationship appears to eliminate the above requirement.

Table 3.5.5. Comparison of Design (Level B) Offsite Dose Rates and Containment Access Times During Normal Operations of PBR and HTGR

Plant	Core Outlet Temperature (°C)	Design Kr-88 Circulating Activity[a] (Ci)	Offsite Dose (mrem/yr)	Containment Access Time (hr/week)	
				Open Containment	Closed Containment
Steam Cycle					
1170-MW(t) HTGR	700	5,710	2.0	>40	27
1170-MW(t) PBR	700	1,780	0.6	>40	>40
Gas Turbine					
3000-MW(t) HTGR	850	16,000	5.7[b]	35	9[c]
3000-MW(t) PBR	850	8,760	3.1	>40	17
Process Heat					
1170-MW(t) HTGR	950	19,900	7.2[b]	28	8[c]
1170-MW(t) PBR	950	5,340	1.9	>40	29

[a]Level B inventory, which is four times higher than Level A (expected) inventory.

[b]Exceeds ALARA limits slightly for design level.

[c]Personnel rotation may be required.

3.5.4. Impact of Coolant Radioactivity on Scheduled Maintenance Activities (HTGR-SC)

(Summary of Work Reported by J. N. Sharmahd and D. D. Orvis, General Atomic Company)

A study was performed[1] to determine whether the amount of fission-product activity in the primary coolant circuit would have a significant impact on plant availability by necessitating that new procedures and techniques be adopted for scheduled maintenance activities. As a reference case, the study focused on the impact that a range of "clean to dirty" coolant would have on scheduled maintenance and inspection activities for the reference 900-MW(e) steam cycle HTGR. The coolant circuit containment level of this base case was specified as "x," and the effect of relative circuit cleanliness was examined by varying the circuit circulating and plateout inventories to 0.1x, 10x, and 100x.

In order to characterize the effects of primary circuit fission-product activity inventory on availability over a range from "clean" fuels to levels exceeding typical design levels, a number of sources were studied and questionnaires were prepared. Several meetings were then held to identify the nature of the limitations on maintenance and in-service inspection (ISI) activities and on the shielding and remote-handling requirements for each of 52 activities. In the discussions, each activity was examined to determine (a) where in the plant the activity would be performed and whether personnel radiation exposure would be from gasborne contaminants, plateout, fuel element activity or neutron activation sources within the PCRV, (b) whether the scheduled operations in the reference case would be conducted by contact or by remote maintenance, (c) whether variation in the primary circuit contamination levels would require more or less shielding and/or remote operations, and (d) whether such differences would significantly change the times required to perform the activity. The subsequent analysis was divided into two major parts: (1) scheduled inspection and testing (ISI) of components inside the containment area during normal operation and (2) planned maintenance of major pieces of hardware, such as the steam generator, core auxiliary heat exchanger (CAHE), and instrumentation, during shutdown.

Of the 52 activities considered, six (12%) were scheduled to be performed during operation, and for these the analysis showed that the controlling source of radiation was gasborne activity plus direct radiation from the core. For 80% of the activities, all scheduled as shutdown maintenance, the controlling source was plateout, and for 6% it was fuel activity and plateout. Only in 2% of the cases was neutron activation and plateout the controlling source; however, neutron activation is a second-order effect.

Examination of the base dose rate (for a contamination level of x) versus the number of maintenance procedures for five categories of dose rates (<1 mrem/hr, 1-10 mrem/hr, 10-20 mrem/hr, 20-50 mrem/hr, and 100-300 mrem/hr) showed that in most cases the shielding provided by the reference design holds the dose rate at or below 10 mrem/hr. In 6% of the cases, the dose rate was below 1 mrem/hr, which is too low to have any impact on maintenance. In 82% of the cases, the maintenance operations were performed totally by remote means, so that increasing or decreasing the level of contamination by a factor of 10 would have no significant impact on maintenance. In the remaining 12% of the cases, maintenance could be performed only when extra shielding was provided.

The impact on the scheduled activities of decreasing and increasing the base case is summarized in Table 3.5.6. As would be expected, decreasing the contamination level to 0.1x would have essentially no effect since the base case already provides for high accessibility and low dose rates to personnel for the majority of the planned operations.

Increasing the contamination level by a factor of 10, however, would require that extra shielding be installed for most of the activities. In nearly all cases, the basic design provides ample space for the shielding to be included and thus no time penalties would be incurred. For some activities, temporary shielding would have to be installed during each shutdown. These activities are conservatively estimated to add about 30-56 hr/yr to the average planned maintenance, but may not add to the overall shutdown duration because of other controlling operations.

For the six activities inside the containment during normal operation, the controlling source of radiation is direct radiation from the core at the refueling floor and gasborne activity elsewhere. With the reference 900-MW(e) HTGR-SC assumed to have a PCRV leak rate of 3.65%/yr and containment venting at 0.5 vol/hr, personnel exposure limits inside the containment would be reached if the circulating activity and the consequent containment gasborne activity were to be increased by a factor of about 2. Therefore, increasing the contaminant level to 10 times the reference would require that ISI personnel be rotated and/or that the containment vent rate be increased.

When the level of contamination in dirty fuel approaches 100 times that of nominal fuel, then access to the containment during operation becomes impractical. Thus the activities inside the containment would have to be performed during shutdown and would add about 123 hr/yr to the planned outage activities. The requirements for the shutdown activities would be approximately the same as for the 10x case.

Even though there appears to be an increase of up to 56 or 179 hr/yr in planned outage activities for the 10x and 100x cases, respectively, the net impact on plant availability of increasing the primary circuit contamination cannot be quantitatively evaluated from this study since many of these tasks are performed concurrently with others in the group, as well as concurrently with refueling and turbogenerator overhaul. Quantitatively, the estimated net impact on plant availability of varying the circuit contamination level could be quite small. At the same time, the maximum increase of scheduled maintenance downtime caused by a hundred-fold increase in contamination levels above base levels would be 15% to 18%.

Table 3.5.6. Effect of Fission-Product Releases in Primary Coolant Circuit on Scheduled Maintenance Activities for Representative HTGR-SC

ISI, Test and Maintenance Activity	Number of Specific ISI/ Maintenance Activities	Approximate Time Required (hr/yr)	Controlling Source of Radiation	Effect of Change in Circuit Contamination (x = Base Case) Method:Time Impact[a]		
				0.1x	10x	100x
			Maintenance During Normal Operation			
Inside containment	6	123	Gasborne and direct from core	None:none[b]	Personnel rotation:none	No access:123 hr/yr
			Shutdown Maintenance			
At PCRV tophead	9	76	Plateout, fuel activity	None:none	Increased permanent shield, remote operation:none	Same as 10x
At PCRV bottomhead	10	115	Plateout, neutron activation	None:none	Increased permanent shield, remote operations, increased temporary shield:14-24 hr/yr	Same as 10x
Steam generator	6	220	Plateout	None:none	Increased temporary shield:8 hr/yr	Same as 10x
Core auxiliary heat exchanger	4	100	Plateout	None:none	Increased temporary shield:8-16 hr/yr	Same as 10x
Instrumentation at various locations, inside containment, control room, etc.	17	466	Plateout	None:none	Increased permanent shield, remote operation, some temporary shield:small	Same as 10x plus personnel rotation
	52	123 + 977			30-56 hr/yr	153-179 hr/yr

[a]Time impact means increase in time to accomplish group of tasks; it does not necessarily mean an extension at annual scheduled downtime.

[b]Would allow more frequent access.

Inasmuch as the fission-product release characteristics of PBR fuel under conditions comparable to those used in this HTGR evaluation are not presently available, this study was unable to establish any firm comparison of the relative effect of fuel type on radiation constraints on maintainability and availability. However, since the primary circuit contamination for the HTGR does not strongly influence plant availability for scheduled maintenance activities, it is assumed that a similar conclusion would be reached for the PBR.

3.5.5. Impact of Coolant Radioactivity on Unscheduled Maintenance Costs

P. R. Kasten

If unscheduled replacement or repair of primary coolant circuit components or of secondary systems is required, insofar as possible the same procedures, equipment, and shielding would be used as for scheduled maintenance activities. By its very nature, however, unscheduled maintenance can require activities that are not foreseen, and these could have a major impact on plant shutdown because of the times involved to make appropriate jigs, fixtures, and replacement equipment. Also, as indicated in Section 3.5.4, increasing the radioactivity of reactor systems tends to increase the requirements even for planned and scheduled maintenance, and it could have a much larger effect on unscheduled maintenance.

Since there was insufficient information available to perform a precise analysis on the effect of circuit activity on unscheduled maintenance requirements, initially a gross analysis was made in which it was assumed, first, that unscheduled maintenance normally would require about the same plant outage as scheduled maintenance, which appears to be the case in general reactor experience, and, second, that the magnitude of the coolant circuit activity could have a large influence on the time required to carry out unscheduled maintenance operations. (Recent studies[2] of personnel exposure indicate reactor downtime is proportional to personnel exposure.) Under these assumptions, unscheduled maintenance operations could lead to a mean plant outage of about four weeks per year for relatively low-level circuit activity and to about eight weeks per year for a substantially increased circuit activity. This would, in turn, lower reactor availability by about 10%, which is very significant. Thus it is important that the relationship between circuit activity and unscheduled maintenance operations be known much better than it is known today. In an attempt to examine this relationship in more detail, the approach given below was used for this comparative evaluation of the PBR and HTGR.

As is discussed in Section 3.5.2, fission-product release into the coolant loop from high-temperature reactor fuels is relatively low at the lower outlet coolant temperatures, but increases with increasing temperature (see Table 3.5.2) due to decreased fuel performance at the higher temperatures. At the lower outlet coolant temperatures, e.g., 700°C and even up to 850°C, fuel matrix contamination and manufacturing defects in particle coatings control circuit activity. However, at high outlet coolant temperatures, i.e., 950°C, palladium attack of silicon carbide coatings controls fuel failures and the associated fission-product release. For example, Table 3.5.2 shows that the 40-yr plateout values for ^{137}Cs activity in the coolant circuit of an HTGR varies from a relative value of 1 at a 700°C outlet coolant temperature up to 20 to 400 at a 950°C outlet coolant temperature, the increase at the higher temperature being influenced strongly by the mechanism of palladium attack and to a lesser extent by the fuel irradiation exposure. The 40-yr ^{137}Cs plateout activity for the PBR is much less than that for the HTGR under corresponding reactor conditions; however, as pointed out in Section 3.5.2, the PBR values are based on an "idealized" reactor in which the interactions of control rods with the fuel are not consisereo. Conceivably, there could be breakage of fuel particles from the control rod interactions, with a concomitant increase in the fission-product release, but in this study such breakage was not considered. Because of the protection given coated particles themselves, it was assumed that the primary effect of the interaction would be on the pebbles and not on the fuel particles.

In view of the above discussion, ^{137}Cs activity was used as the basis for estimating the effect of coolant circuit activity on unscheduled maintenance time, the first step being to convert the activities given in Table 3.5.2 to the relative activities shown in Table 3.5.7. Next, it was assumed that above a certain person-rem exposure level, unscheduled maintenance time would increase in proportion to the exposure. Finally, the activities in Table 3.5.7 were converted to person-rems by normalizing the "Old Pd" loop activity for the HTGR at an 850°C outlet temperature and a 4-yr fuel exposure to 400 person-rems/GW(e)-yr. The result was the family of curves presented in Fig. 3.5.2.

While the normalization to 400 person-rems/GW(e)-yr is somewhat arbitrary, it was done on the following bases: (1) At 850°C outlet temperature, the circuit activity would be significantly above that associated with FSVR operation, such that a personnel exposure of 400 person-rem per GW(e)-yr appears reasonable, particularly at the higher fission-product releases associated with the "Old Pd" correlation. (2) Above ~850°C, the palladium-induced SiC corrosion effect on fission-product release starts to become significant relative to other causes of fission-product release.

It is apparent from Fig. 3.5.2 that the manner in which the palladium attack mechanism is treated is an important factor in predicting fission-product releases. As indicated previously, GA has obtained experimental results which indicate that the palladium attack rate varies as the square root of time; however, other information available supports a linear relation.[3] Since it is still unclear as to which relation is correct, for this evaluation a value was chosen that was between the extremes, but favored the square-root-of-time relation. Thus, for the HTGR an adjusted value of 2800 person-rem/GW(e)-yr at 950°C was chosen for a four-year fuel life and an adjusted value of 1000 person-rem/GW-yr at 950°C was chosen for a three-year fuel life. And since the values chosen for the HTGR were below those indicated by the "Old Pd" linear relationship, the "Old Pd" value for the PBR was similarly lowered (see open triangle in Fig. 3.5.2). Finally, a second adjustment of the

Table 3.5.7. Relative ^{137}Cs Activity in Coolant Loop After 40 Years[a]

Reactor	Fuel Exposure (yr)	Relative Activity T=700°C	T=850°C	T=950°C
HTGR	4	1	12.7	34* — 417
	3	<1		18* — 67
PBR	4	<<1	2.8	6

[a]All activities calculated under assumption that the palladium attack rate varies linearly with time (see "Old Pd" in Section 3.5.4) except for asterisked values, which were calculated under the assumption that the palladium attack rate varies with $\sqrt{time}$ (see "Rev. Pd").

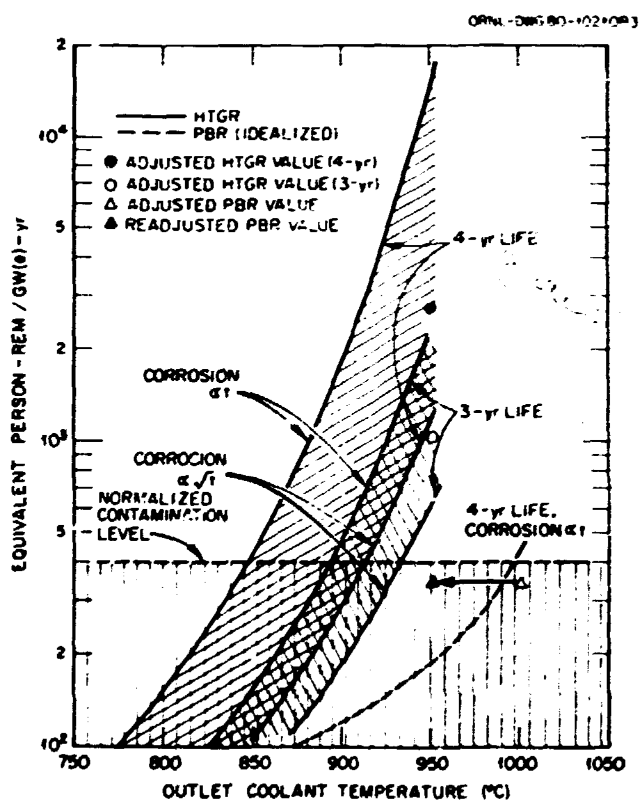


Fig. 3.5.2. Variation of Equivalent Exposure for Given Maintenance Procedures as a Function of Outlet Coolant Temperautre. Curves normalized to 400 person-rem/GW(e)-yr for HTGR operation at 850°C coolant outlet temperature and at 4-yr fuel exposure and corrosion ∝ t.

PBR value was made to account for the fact that the PBR value was for an "idealized" reactor that neglected the control rod actions. Information from FRG* had suggested that neglecting the effect of control rods on fuel movement and fuel temperatures would lead to an effective decrease of 50°C in the outlet coolant temperature for the same peak fuel temperature; that is, the PBR exposure would actually occur at a lower outlet coolant temperature. Accordingly, the PBR point was shifted 50°C to the left.

To calculate the costs of circuit activity on reactor operation from the above information, the permissible person-rem/GW(e)-yr is needed as well as the cost of a person-rem/GW(e)-yr. In this analysis, it was assumed that up to 800 person-rem/GW(e)-yr can be accumulated by reactor plant operations without exposure penalty. (This leads to no penalty for the 750 and 850°C outlet coolant temperatures). Above that exposure, a penalty of $25,000 per person-rem is imposed. This penalty includes not only the direct person-rem cost in terms of additional required manpower, but also the cost associated with the plant downtime due to the unscheduled maintenance itself. There have been a number of estimates made relative to the cost of a person-rem in terms of the personnel costs themselves, and they range from $1000 to $10,000 per person-rem and greater. Here it is considered that the direct personnel costs would be about $2500/person-rem, but that the cost of the associated plant downtime would be such that the equivalent cost of personnel plus plant downtime would be $25,000/person-rem.

Using a cost of $25,000/person-rem for exposures above 800 person-rem/GW(e)-yr, and the circuit activity and reference points given in Fig. 3.5.2, the effective cost of unscheduled maintenance for the HTGR at 950°C outlet coolant temperature would be between $5 million and $50 million per year; at 850°C outlet coolant temperature, the effective cost would be zero. For the PBR, the effective unscheduled maintenance cost would be zero for all the outlet temperatures.

It should, of course, be remembered that this calculational procedure is offered only to estimate costs. In practice, the actual person-rems would be limited to about 800 person-rems/GW(e)-yr by utilizing appropriate shielding and maintenance procedures. To estimate costs, the method basically assumes that the cost of doing unscheduled maintenance activities with unlimited manpower resources is equivalent to the cost based on limited exposure and more lengthy maintenance times involving expensive equipment.

It should also be emphasized that these estimates of the effects of circuit activity levels on plant operating costs are very uncertain. If the circuit activity due to matrix contamination and initial broken fuel particles were normalized to 100 person-rem/GW(e)-yr rather than to 400 person-rem/GW(e)-yr, all the curves in Fig. 3.5.2 would be shifted down by a factor of 4, and the HTGR would have zero penalty at 950°C outlet coolant temperature. Further, these analyses were based on relative plateout of radioactive ^{137}Cs in the primary circuit alone. The importance of this activity on the time to

*Personal communication from GHT staff during visit of U.S. team to FRG, March 1979.

carry out maintenance operations is not well known. Another factor which influences personnel exposure is gaseous activity leakage from the PCRV itself. This aspect was not considered. However, since gaseous release is less temperature-dependent than fission-product plateout, differences between the reactor types due to gaseous activity release would probably not be as significant as those considered above. Based on Section 3.5.6, there could be a factor of about four in gaseous fission-product release - PBR advantage - based on the "old Pd" relation.

The estimated annual cost penalty of $5 million to $50 million can be compared with costs presented in the previous section on the effect of increased radioactivity on scheduled maintenance times. The results given there indicate that a factor of 100 in increased activity leads to a 15 to 18% increase in scheduled maintenance time. Considering the normal scheduled maintenance time to require 1000 hours leads to an additional time of 150 to 180 hours to carry out scheduled maintenance when activities are a factor of 100 greater than normal. If this "extra" time reduced plant availability, it would mean about an additional week of plant downtime. Assuming that the effective cost of replacement power is 3¢/KWh, a week of downtime would lead to $5 million in extra costs for a 1-GW(e) plant on an annual basis.

Overall, an HTGR penalty of $5 million per year for a 1-GW(e) plant appears to be reasonable when the outlet coolant temperature is 950°C, recognizing, however, that the penalty could be zero, or as high as $50 million per year. Thus, in this study, the mean cost of unscheduled maintenance activities for a 3 GW(t) HTGR at 950°C outlet coolant temperature was taken to be $5 million per year, and the probabilistic values varied from zero to $50 million per year.

For a 1-GW(t) reactor with 950°C outlet temperatures, the HTGR penalty on a unit power basis would be the same as for the 3-GW(t) plant, i.e., $1.67 million/GW(t)-yr. For HTGR outlet coolant temperatures of 750°C and 850°C, the relative maintenance penalty would be zero, and for all PBR outlet coolant temperatures, the penalty would be zero.

3.5.6. Relative Risks of Accidents Releasing Fission Products

(Summary of Work Reported by A. Barsell, General Atomic Company)

The various design bases and Class 9 accidents* that are important in the HTGR and PBR concepts were analyzed as summarized in Table 3.5.8. Available information and/or the scope of the effort was insufficient to evaluate relative safety performance in a completely quantitative or probabilistic manner. Thus, we were not able to define a PBR risk curve for comparison with the HTGR risk curve as established by the AIPA (accident initiation and progression analysis) Phase II analysis. However, the relative safety of

*Class 9 accidents are those having very low probability but very high consequences.

Table 3.5.8. Comparison of PBR and HTGR Core Accident Analyses

Safety Issue	Pebble Bed/Prismatic Core Comparison	Required Detailed Analysis
Depressurization accidents	HTGR risk dominated by spurious lifting of PCRV relief values or small pipe leaks. For these, PBR risk should be similar; however, PBR has additional possibility of depressurization from fuel loading/removal system, unquantifiable at this time. Even so, consequences for PBR would be 2 to 4 times lower.	Reliability analysis of fuel loading/removal systems for depressurization.
Earthquake effects on reactor internals – flow blockage and local overheating	Fuel element cracking a concern for HTGR, top reflector collapse for PBR. Differences in core support structural behavior not quantifiable at this time.	Translation of earthquake intensity into loading on internals components. Probabilistic analysis of structural reliability, including oxidation effects.
Spent-fuel handling accidents	Fuel element drop by fuel-handling machine on unloading specific to HTGR. Spillage of fuel balls on transfer to transport cart specific to PBR. Drop or impact of fuel transfer cask or cart similar to both concepts.	Reliability analysis of fuel unloading or removal; consequence analysis.
Water ingress accidents	Occurrence probabilities should be similar. HTGR release paths for fission products, in order of importance, are (1) dump lines, (2) steam relief valve, and (3) PCRV relief valve per AIPA Phase II; these should be similar for PBR. Graphite oxidation should be higher for PBR by about a factor of 2 because of corresponding higher surface area of balls compared to coolant channels in the block. Steam diffusion to fuel should be similar. Hydrolysis and attendant release of noble gases should be similar for 700°C and 850°C outlet temperature designs because failed fuel fraction is similar. PBR release should be a factor of 5 lower for 950°C outlet temperature due to a lower failed fuel fraction.	Application of HRB version of OXIDE-3 code for PBR oxidation, hydrolysis, pressure.
Core heatup		
Probabilities*		
LOSP leading to LOFC	Should be same (3×10^{-6}) for both concepts.	CACS reliability analysis for PBR.
LOSP + LMLC leading to LOFC	Should be the same (3×10^{-5}) depending on CACS.	
Control rod insertion	Prismatic is 10^{-5}; PBR should be notably higher.	CRD reliability for PBR.
Reserve shutdown rod insertion	Should be similar ($\sim 10^{-3}$).	

Liner cooling failure	May be similar (10^{-1}) depending on effect of top reflector collapse in PBR.	Analysis of thermal barrier failure after top reflector collapse.
Containment failure	Corresponds directly to liner cooling system failure (10^{-1}).	
Consequences		
Afterheat function	PBR is 15% less due to lower core residence time of fuel.	
Heat capacity	Similar for active core; insufficient information for reflectors, tentatively assumed to be similar.	Reflector heat capacity analysis.
Axial heat transfer	Effective conductivities similar. PBR heat transfer area almost twice greater due to larger diameter core.	
Radial heat transfer	Significantly higher for PBR due to radiation heat transfer.	
Initial temperatures	Negligible effect of differences in initial temperature distribution.	
Fission-product release	Should be slower in PBR due to slower heatup of core.	
Containment failure time	Should be longer for PBR in sequences with liner cooling failure due to lower afterheat (higher heat transfer works opposite).	PBR concrete degradation analysis.
Boron carbide slumping (reactivity poisons)	Dead-end channels limit slumping in HTGR to ~4 blocks voided for CRD and 3 blocks voided for RSS. PBR case should be distinctly worse due to no constraints.	Need experimental data.
Boron carbide vapor diffusion in graphite	Should occur later in PBR due to lower temperatures; diffusion enhanced by higher power factors at top of core and greater surface area of balls in PBR.	Need experimental data.
Maintenance of shutdown margin	Tradeoff between lower temperatures in PBR and enhanced B_4C diffusion and limited slumping in HTGR.	SORS code analysis of B_4C behavior and corresponding reactivity analysis.

*LOSP = loss of site power; LOFC = loss of forced cooling; LMLC = loss of main loop cooling.

the two concepts based on existing information is summarized, and additional detailed analyses required for quantification of the relative risks are identified.

Design Basis Depressurization Accidents (DBDAs)

Data obtained from the "liftoff" studies in the GAIL Loop have shown that the radiation dose released during DBDAs are, in order of importance, due to

(1) Sr-90 liftoff of plated-out activity,

(2) iodine liftoff, chiefly I-131 and I-133, and

(3) noble gas circulating activity, mainly Kr-88 and Xe-133.

Using the Fort St. Vrain NRC assumption of 5% liftoff, the reference Level B plateout inventory of Sr-90* can be increased by a factor of 880 and the iodine plateout inventory by a factor of 2500 before the dose-rate limits are reached. Likewise, the noble gas circulating inventory can be increased by a factor of 20,000. These factors indicate the enormous safety margins which exist for DBDAs; therefore the DBDA should not be a constraint for setting limits of primary circuit inventories, nor a major consideration for licensing comparisons of PBR and HTGR concepts.

While for licensing analysis the Sr-90 and I-131 contributions to inhalation doses are considered to be the most important, the AIPA Phase I study showed that the major dose to the public due to depressurization is the external whole-body gamma-ray dose resulting from noble gas activity released to the circulating primary coolant. Core performance analysis (Table 3.5.5) indicates that circulating noble gas activity is 2 to 4 times lower for the PBR. Thus, it may be concluded that consequences of depressurization will be 2 to 4 times lower for the PBR.

Regarding probability and risk of depressurization, it should be noted that the events of highest risk in the AIPA study were (a) spurious lifting of PCRV relief valves, and (b) rupture of small instrument or pipe lines penetrating the primary coolant boundary. Large penetration failure was a lower risk due to low probability. The PBR should be similar in these regards; however, the PBR fuel loading and removal systems present unique possibilities for additional paths of depressurizations. Lacking a comprehensive reliability analysis for the PBR, it is not known whether depressurizations in these systems could constitute a significant risk contribution, but the preliminary availability analysis indicates a relatively low probability of failure. Therefore, the preliminary indications are that the probabilistic risk of depressurization accidents is a factor of 2 to 4 lower for the PBR. This would lead to lower reactivity in the containment vessel, which could influence maintenance activities. However, that factor is not considered to be significant.

*7890 Ci of Sr-90 in primary circuit [2240 MW(t) HTGR-SC].

Earthquake Effects Comparison*

The reactor core and PCRV internals are designed to industry code and regulatory standards which provide wide margins of safety for loadings imposed by earthquakes up to a maximum Safe Shutdown Earthquake (SSE). Thus, structural failure of these components is not within design basis. However, for purposes of this safety comparison, the structural concerns, during earthquakes in excess of SSE (class 9 events) for example, are listed. In the absence of a detailed seismic analysis of PBR and HTGR cores, it is expected that the fuel balls will withstand the seismic forces much better than the HTGR fuel blocks. Fuel element cracking could cause partial flow blockage and local over-heating in the core. On the other hand, the PBR top cover reflector is supported and suspended form the top; its failure could cause overheating locally in the PBR core if parts collapse on top of the bed. Based on existing information, such as the Mechanics Research Institute study, failure of the HTGR core support structure is an extremely low probability event, even considering graphite oxidation effects over the life of the plant. Although the PBR core support design is different, its structural reliability may be high enough that any differences with HTGRs may be unimportant. However, reliability analysis of the PBR core support design is needed for verification. Also, the reliability analysis of the HTGR core support needs to be updated for the recent design.

In summary, the concept comparison for earthquake safety seems to be governed by relative structural reliability of HTGR fuel blocks and the PBR top cover reflector. Relative reliabilities of the core support structures are not expected to be important due to extremely low failure probability of both concepts.

Spent Fuel Handling Accidents

This class of accidents does not contribute significantly to the overall risk envelope. However, there are safety implications in the HTGR concept for (a) postulated drop of a spent fuel element block on unloading the core or (b) drop of a shipping container loaded with six fuel blocks as analyzed in AIPA Phase I, Volume III. Equivalent PBR events can occur on transfer of spent fuel element balls to the fuel transport cart. Lacking a reliability analysis of the fuel removal system, there is no indication at present that the PBR risk will be different from that for HTGRs.

Water Ingress

Occurrence probabilities for water ingress (analyzed in detail in AIPA, Phase II for steam generators) should be independent of core design since they are dominated by random defects or failures in the secondary coolant boundary. For a steam cycle plant, the major risk release pathways are through the secondary system and to the atmosphere via failed-open steam relief valves or failed-open dump valves (SG-1 and SG-2 release categories, respectively). Again, this should be similar for the PBR. Regarding consequences, it is concluded that:

*See also Section 3.9.

(1) Graphite oxidation in the PBR would be about a factor of 2 higher because the surface area of the fuel pebble is about twice the surface area of the coolant channel surface area in prismatic blocks. (Most oxidation occurs near the surface when temperatures are still hot.)

(2) Steam would diffuse to the fuel when temperatures begin to cool and all fuel particles with failed coatings would hydrolyze with attendant release of noble gases. Failed fuel fractions for the PBR are similar to those for the HTGR for core outlet temperatures of 700°C and 850°C; thus, there should be no difference in fission gas release for these cases. For the 950°C case, however, the failed fuel fraction and fission gas release of the PBR would be about a factor of 5 lower than for the HTGR due to the hydrolysis.

(3) On the assumption that the PBR has about twice the core surface area of the HTGR, nuclides of cesium and strontium sorbed in graphite would undergo twice as much release in the PBR due to twice the graphite oxidation. However, these nuclides are not major contributors to health effects for this event.

In summary, the risk of water ingress appears to be similar for the PBR and HTGR with outlet core temperatures up to 850°C; at a 950°C outlet temperature, the PBR fission-product release appears to be 5 times smaller due to the lower fuel failure fractions in the PBR. These conclusions should be confirmed, however, through an analysis with the OXIDE-3 code. Even so, this difference in risk is not considered to be significant in this study.

Core Heatup

The probabilities of initiating events leading to core heatup in PBRs are expected to be similar to the probabilities previously analyzed in AIPA Phase I for HTGRs. This is because the key faults initiated are either in the cooling systems (main loop and CACS*), which are assumed to be similar for both concepts, or in electrical supply systems. However, the events that occur following heatup may differ for the two reactors, especially if governed by core response. One such event is rapid insertion of the control rods (SCRAM). Since the HTGR control rod drive (CRD) is largely passive (gravity insertion), and the PBR control rods require forced insertion into the pebble bed, the HTGR reliability (10^{-5} per ref. 4) seems inherently better. However, sequences in which

*CACS = Core auxiliary cooling system.

control rods failed to shut down the reactor did not contribute to the overall core heat-up risk in the AIPA study, and it appears doubtful that a licensed PBR control rod system could have low enough reliability that such sequences could become important for the PBR risk curve. Detailed reliability analyses are required for verification.

The PBR reserve shutdown system (RSS), being similar in concept to the HTGR system, is judged to have a similar reliability (10^{-3} per demand).

Analyses of convective, radiative and conductive heat transfer form the core to the thermal barrier, liner and liner cooling system for the PBR design are not available. Consequently, the change in overall probability of liner cooling system failure (10^{-1}) resulting in concrete degradation and subsequent containment failure cannot be assessed. One possible difference could be the high-temperature failure of metal supports for the PBR top cover reflector and the subsequent collapse of the top reflector on the pebble bed core. Based on existing information, it is not clear whether this would greatly affect liner cooling system failure.

Overall, it is judged that probabilities of core heatup initiation and important subsequent events are similar for both concepts. In considering the consequences of core heatup, it is noted that because of the lower fuel residence time in its core, the PBR has an afterheat function which is about 15% lower than that for the HTGR. Along with core heat capacity and thermal conduction out of the core, the afterheat governs the rate of temperature risk and the maximum temperature attained. Initial temperatures or temperature distributions have been found to exert little influence on the heatup transient and resulting fission product release.

Post-Accident Containment Access

Access by operating personnel to the containment building following an accident may be an important factor, both in restoring control and in mitigating the consequences. A survey of HTGR accidents involving the release of radioactivity indicated that primary coolant loop depressurizations are the most significant. Analysis of both slow and rapid depressurizations show that the containment building should be accessible within 1-4 days. However, no significant difference in containment dose rates for PBR and HTGR concepts is apparent.

References

1. "Pebble Bed Reactor Fiscal Year 1980 Review Summary Report," GA-A15956, pp 7-1 ff, July 1980.

2. J. L. Helm, as reported at GCRA Conference, San Diego, CA, October 1, 1980.

3. Personal communication from T. E. Lindemer to P. R. Kasten, September 1980. Also, see R. J. Lauf, "The Interaction of Silver and Palladium with SiC in HTGR Fuel Particles - Preliminary Report," ORNL/TM-7393 (July 1980).

4. "HTGR Accident Initiation and Progression Analysis Status Report: Phase II Risk Assessment," GA-A15000, General Atomic Company, May 1978.

3.6. HEAVY METAL LOADINGS IN PBR AND HTGR CORES

F. J. Homan

3.6.1. Introduction

Flexibility of the conversion ratio and the specific power density in a reactor is directly related to the amount of heavy metal that can be loaded into a fuel element. Higher conversion ratios require higher metal loadings, and the loading capabilities of the fuel element may be limiting. More metal could be loaded into the core by making the core larger (i.e., lowering the specific power density), but this would impact the capital costs because it would require a larger pressure vessel and a greater number of fuel elements. Thus it is important to know the extent to which the heavy metal loadings in PBR and HTGR fuel elements could be increased.

In this section heavy metal loadings that have been achieved in PBR and HTGR fuel elements are reported and the potential for increasing the loadings is discussed, together with the development work that would be required. As part of this comparative analysis of the two types of reactors, calculations of heavy metal loadings were performed for both PBR elements and HTGR elements, assuming in all cases that the reactors would be operating on highly enriched fuel. It was felt that the incentive for higher metal loadings (i.e., higher conversion ratios) is exclusively associated with recycle cases, and recycle is most promising when uranium of high enrichment is used. In the case of the PBR, consistent data sets[1-3] were available on which to base the calculations, but no such consistent sets were found for the HTGR calculations. As a result, old information from GA (as much as five years old) had to be used. The availability of more recent HTGR data probably would not have changed the final conclusion of the study, however.

3.6.2. PBR Heavy Metal Loading Capability

PBR Fuel Particles and Potential Fuel Element Loadings

The PBR core design currently favored by FRG features a spherical fuel element composed of overcoated particles similar to the particles currently being developed in the U.S. for an HTGR MEU feed/breed cycle.* Features of three variants of the FRG particles, together with those of the U.S. design, are given in Table 3.6.1. (The FRG data are taken from ref. 4.)

In this study potential heavy metal loadings were calculated for all four particle variants under the assumption that the overcoated particles comprised 65 vol.% of the fueled matrix of the element sphere. A comparison of the results (see last entry in Table 3.6.1) reveals the inefficiency of the FRG feed/breed design (Variant 3) when compared with the U.S. design (Variant 4). This is primarily due to the very small HEU

*See Fig. 3.4.1 for sketch of typical coated particles.

Table 3.6.1. Features of PBR Fuel Elements and Coated Particle Variants Under Development

	One-Particle Designs		Feed/Breed Designs	
	Var. 1, FRG	Var.2, FRG	Var. 3, FRG	Var. 4, U.S. (HTGR)
Kernel composition	$(Th,U)O_2$	$(Th,U)O_2$	UC_2/ThO_2	UCO/ThO_2
Coating type	HTI-BISO	LTI-TRISO	TRISO/TRISO	TRISO/TRISO
Kernel diameter (μm)	400	500	200/500	350/600
Coating thickness (μm)				
Buffer	80	90	100/90	115/110
Sealer + LTI	105	40	40/40	35/35
SiC		35	35/35	35/35
OLTI		35	35/35	40/40
Overcoating	100	200	250/150	250/150
Volumes (10^{-11} m^3)				
Kernels	3.351	6.545	0.4189/6.545	2.245/11.310
Coated particles	47.787	115.030	73.562/90.478	115/125
Fuel element (sphere) geometry				
Diameter (cm)	6.0	6.0	6.0	6.0
Unfueled region thickness (outer shell) (cm)	1.0	1.0	1.0	1.0
Volume (10^{-6} m^3)	113.04	113.04	113.04	113.04
Fueled region volume (10^{-6} m^3)	65.45	65.45	65.45	65.45
Densities (g/cc)				
Kernel	9.50	9.50	10.0/9.50	10.0/9.50
Heavy metal	8.35	8.35	9.10/8.35	9.2/8.35
Particles/sphere	89,020	36,982	57,829/47,017[a]	36,982/33,763[a]
Potential heavy metal loadings				
In particles (10^{-5} g/particle)	27.98	54.65	3.82/54.65	20.654/94.438
In sphere (g/sphere)	24.91	20.21	2.204/25.70[a]	7.638/31.890[a]
			1.543/7.714[b]	3.48/17.39[b]
			1.187/11.862[c]	2.25/22.50[c]
			0.812/16.233[d]	1.32/26.38[d]

[a]Assumes that spheres contain only fissile or only fertile particles; fuel elements of this type not expected to be fabricated and calculation shown only for comparison purposes.

[b]Assumes that Th/U = 5; total heavy metal loading (sum of U + Th) is 9.26 g/sphere for Variant 3 and 20.9 g/sphere for Variant 4.

[c]Assumes that Th/U = 10; total heavy metal loading is 13.1 g/sphere for Variant 3, 24.8 g/sphere for Variant 4.

[d]Assumes that Th/U = 20; total heavy metal loading is 17.0 g/sphere for Variant 3, 27.7 g/sphere for Variant 4.

fissile kernel (200-μm diameter) used in a particle with a thick (250-μm) overcoating. Variant 4 has a dense uranium oxide/uranium carbide (UCO) fissile kernel that should give superior performance compared to the dense UC_2 kernel of Variant 3. Variant 4 should also have the added advantage of a greater loading efficiency because of its larger size. The loading of Variant 4 compares very well with the FRG one-particle variants (Variants 1 and 2).

PBR Achieved and Tested Loadings

During fabrication of the PBR spherical fuel elements, the overcoated particles described in Table 3.6.1 are mixed with matrix material (composed of natural flake graphite, electrographite flour, phenolformaldehyde resin and methanol) and loaded into a rubber mold. The rubber mold is transferred to a press and the contents are compressed isostatically. This produces a fueled core, which is then removed from the mold and forwarded to the next stage of the fabrication process. There, a fuel-free zone (a 0.5-cm thick graphite layer) is added, and the element is cold-pressed to achieve high density. In this process, the heavy metal loading is determined by the relative quantities of overcoated particles and matrix material blended together for loading into the rubber mold.

Loadings that have been achieved and tested for the spherical fuel elements used in the PBR core are shown in Tables 3.6.2 and 3.6.3. The data in Table 3.6.2 are for the reference spherical fuel element design for the PNP (nuclear process heat) and HHT (gas turbine HTR) programs in Germany,[4] the particle variants being the same as described in Table 3.6.1. As shown, the heavy metal loading for the reference fuel element is 11.24 g per sphere.

Table 3.6.3 summarizes FRG's experience with fuel elements having heavy metal loadings greater than 11.24 g per sphere (up to 25 g per sphere). Comparison of the results of these experiments with the results of experiments with lower heavy metal loadings have led to two major conclusions:

(1) Volume loading and particle type influence the neutron-induced shrinkage of a spherical fuel element under irradiation. There is also a temperature dependence. As shown in Fig. 3.6.1, greater fuel element shrinkage under irradiation is associated with high volume loadings as compared to low loadings, and with BISO-coated particles as compared to TRISO-coated particles.[5]

(2) There was a high failure rate in the particles irradiated in the AVR VI experiment. There was an inhomogeneous distribution of fuel particles within the fuel element, which may have contributed to the high failure fraction. The failure mechanism is unknown at this time.

Table 3.6.2. Heavy Metal Loadings in Reference FRG PNP[a] and HHT[b] Fuel Elements ($0.96\ g\ ^{235}U$ and $10.21\ g\ ^{232}Th$ per Sphere; 93% Enriched U)

Particle Variant[c]	Heavy Metal Loading (g/sphere)	Number of Particles per Sphere	Percent Volume of Sphere Matrix: Particles Only	Percent Volume of Sphere Matrix: Particles with Overcoating
1	11.24	37,000	15	27
2	11.24	19,000	13	32
3	11.24	44,000	13	53

[a]FRG nuclear process heat system.

[b]FRG gas turbine HTR.

[c]See Table 3.6.1.

Table 3.6.3. Summary of FRG Experience with PBR Fuel Element Heavy Metal Loadings > 11 g per Sphere

	Expt. DR-K5, Spheres 1-27	Expt. FRJ2-K8: Sphere 5	Expt. FRJ2-K8: Sphere 6	Expt. AVR-VI
Heavy metal loadings (g/sphere)				
Total HM	25.0	20.0	20.0	20.0
Total U	25.0	20.0	20.0	20.0
^{235}U	2.4	1.4	1.4	1.4
Particle content of sphere matrix (vol.%)	14.3	12.4	12.4	12.4
Number of particles per sphere	23,000	9400;[a] 8200[b]	9400;[a] 8200[b]	9400;[a] 8200[b]
Kernel diameter[c] (μm)	602	608;[a] 618[b]	608;[a] 618[b]	608;[a] 618[b]
EFPD	351	137	137	1233
ϕt, $E > 0.1$ MeV ($\times 10^{21}$)	2.9	0.014	0.014	2.0
Maximum surface temperature (°C)	1200	1247	1140	980
% FIMA	4.6	3.2	3.3	7.4
Fuel element power (kW/fuel element)				
Minimum	2	5.9	5.4	1.4
Maximum	4	9.2	8.5	2.1
Particle batch	E0391	E0414-428;[a] E0431-441[b]	E0414-428;[a] E0431-441[b]	E0414-428;[a] E0431-441[b]

[a]Natural uranium kernels.

[b]Low-enriched uranium kernels.

[c]UO_2 kernels.

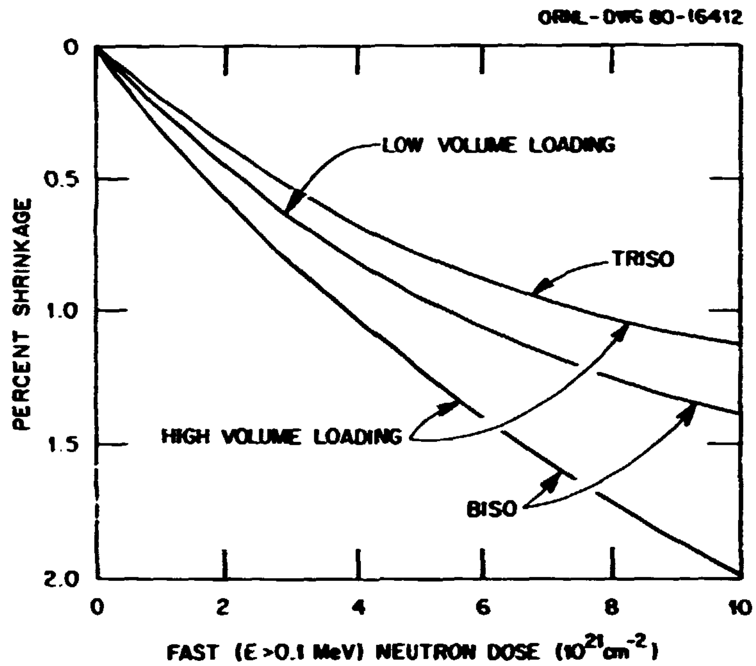


Fig. 3.6.1. Shrinkage of Spherical Fuel Elements Versus Fast Neutron Dose. (From ref. 5.)

PBR Heavy Metal Loading Limits

Loading limits are influenced by the particle design (particularly the thickness of the overcoating), the allowable reject fraction (zero under the current specification), and the required crushing strength of the fabricated fuel element (greater than or equal to 22 kN under the current specification). The overcoating thicknesses currently used for variants 1-3 are included in Table 3.6.1. Experience within the FRG program has shown that thicknesses of 150 to 250 μm are required on TRISO-coated particles to avoid particle breakage during element fabrication (presumably in the cold-pressing step). For the BISO-coated particle, an overcoating thickness of 100 μm is sufficient to ensure an as-fabricated particle breakage fraction of less than 10^{-4}. Some characterization results on fuel elements with different particle loadings are shown in Table 3.6.4.

Based on their current experience, the Germans have identified the loading limits listed below:[4]

Particle Variant	Heavy Metal Loading (g/sphere)
1	15-20
2	20
3	15

Table 3.6.4. Characterization of PBR Fuel Elements with 11 to 25 g of Heavy Metal per Sphere

	Matrix Type A3-3					Matrix Type A3-27			
Heavy metal loading (g/sphere)	11	15	18	20	25	11	15	20	25
Particle Variant 1									
Percent volume of sphere matrix									
Particles only	16	21		23	35	16	21	28	35
Particles with overcoating	27	37		50	62	27	37	50	62
Overcoating thickness (μm)	100	100		100	100	100	100	100	100
Particles/sphere (10^3)	34	47		62	78	34	47	62	78
Percent rejects after carbonization	0	0		0	0	0	0	0	0
Crushing strength (kN)									
Perpendicular	25	24		22	19	25	25	24	22
Parallel	26	23		22	18	23	22	22	21
Matrix contamination (Exposed HM/total HM × 10^{-6})	280	3[illegible]		290	280	240	230	210	170
Broken fissile particle fraction (10^{-6})	<30	100		<30	50	<20	<20	<30	<20
Particle Variant 2									
Percent volume of sphere matrix									
Particles only	13	18		24	30	13	18	24	30
Particles with overcoating	32	44		58	73	32	44	58	73
Overcoating thickness (μm)	200	200		200	200	200	200	200	200
Particles/sphere (10^3)	20	28		37	46	20	28	37	46
Percent rejects after carbonization	0	0		0	0	0	0	0	67
Crushing strength (kN)									
Perpendicular	23	23		23	20	26	24	24	NM[a]
Parallel	23	23		22	22	26	24	24	NM
Matrix contamination (Exposed HM/total HM × 10^{-6})	<30	<30		<30	<30	<30	<30	<30	NM
Broken fissile particle fraction (10^{-6})	150	170		100	160	50	60	70	NM
Particle Variant 3									
Percent volume of sphere matrix									
Particles only	18	22	26	28	34	18	22	28	34
Particles with overcoating	53	63	71	76	89	53	63	76	89
Overcoat thickness (μm)	250	250	250	1[illegible]0	150	250	250	150	150
Particles/sphere (10^3)	46	53	58	62	71	46	53	62	71
Percent rejects after carbonization	0	0	0	5	90	0	0	100	100
Crushing strength (kN)									
Perpendicular	24	24	22	19	11	25	23	NM	NM
Parallel	24	23	21	19	11	24	24	NM	NM
Matrix contamination (Exposed HM/total HM × 10^{-6})	<30	30	<30	<30	NM	<30	<30	NM	NM
Broken particle fraction (· 10^{-6})									
Fissile	160	140	350	410	NM	240	300	NM	NM
Fertile	150	530	170	1720	NM	230	250	NM	NM

[a]NM = not measured.

They feel that the absolute upper limits that could be achieved using the cold isostatic pressing fabrication technique would be the loadings associated with 65 vol.% overcoated particles, and, with the zero percent reject requirement, to attain such high volume loadings would require several years of development. If 65 vol.% of overcoated particles in the fuel matrix could be achieved, however, the heavy metal loadings could be increased to the following:

Particle Variant	Heavy Metal Loading (g/sphere)
1	27
2	23
3	18

An attempt has been made in this study to duplicate these numbers, as shown by the calculated potential loadings presented in Table 3.6.1. The calculated values shown are reasonably close to those shown above. Variant 3 provides a problem in that the Th/U ratio must be known to calculate the grams of U and Th present in the sphere. The calculation has been done for Th/U ratios of 5, 10, and 20, and also under the assumption that the spheres contain only U or only Th.

Hot pressing is a fabrication alternative being developed by FRG to increase the heavy metal loading capabilities of the spherical fuel elements. Through the use of hot pressing, they hope to: (1) influence the mix of raw materials used in fuel element fabrication (specifically to avoid the use of natural graphite and use only German electrographite), (2) use steel dies rather than rubber molds, and (3) avoid the use of overcoatings while at the same time reducing the beginning-of-life broken particle fraction to zero.

Good results have been achieved to date with the hot-pressing development effort; however, some problems still persist, including a nonuniform matrix density and the tendency for the overcoated particles to concentrate in the bottom half of the ball. At this time attempts at hot pressing particles without overcoating still result in broken particles, even when low pressures (100 kg/cm^2 or less) are used. (The pressure used for cold pressing is on the order of 3000 kg/cm^2.) It is expected that these problems will be overcome through the use of additives in the matrix and the improvement of the die shapes. Irradiation data on matrix performance will be obtained from a test series in the HFR-Petten (HFR-GM1).

Previous work on hot pressing at Dragon, CERCE (France), and UKAEA has shown that an upper limit of 35-40 vol.% of particles in the fueled matrix can be reached when working with overcoated particles. In this work TRISO-coated particles were used and excellent results were achieved relative to mechanical strength of the fuel elements and to low broken-particle fractions. The Germans expect that fuel elements containing $(Th,U)O_2$ and UO_2 TRISO-coated particles can be loaded to 35 vol.%. Particle loadings above 35 vol.% should be possible through the use of hot pressing on particles without overcoatings. However, significant development work will be required to achieve this goal.

PBR Fuel Element Development Status

The development status of the PBR spherical fuel elements is summarized in Table 3.6.5 (from ref. 1, pages 2-8). As shown, the AVR and THTR fuel is fully developed and features heavy metal loadings of 6 and 11 g per sphere. Higher conversion ratios will require higher metal loadings, in the range of 16 to 25 g of heavy metal per sphere. Loadings in this range can be reached through the cold-pressing fabrication techniques currently in use, but some additional development work is needed to reach the fuel quality required. Breeding and near-breeding will require heavy metal loadings in the range of 30 to 45 g of heavy metal per sphere. A fabrication technology featuring hot pressing must be developed and qualified to produce loadings in this range.

The heavy metal loadings for various PBR fuel cycles are summarized in more detail in Tables 3.6.6 and 3.6.7 (from ref. 1). Additional information on the various PBR fuel cycles and conversion ratio options is given in refs. 2 and 3.

Table 3.6.6 shows the relationship between the various fuel cycles, heavy metal loading requirements for the fuel cycles, and the conversion ratios associated with those fuel cycles. Table 3.6.7 shows the relationship between conversion ratio and a number of operating parameters, including the heavy metal loadings. It can be seen that for a burnup of ~100 MW(t)-d/kg HM, current technology limits the conversion ratio to about 0.71. Higher conversion ratios at this burnup will require success in the effort to develop fabrication techniques that permit higher heavy metal loadings in the spherical fuel elements.

One advantage of on-line refueling in the PBR concept is that the burnup can be decreased without shutting down the reactor for refueling. Thus, burnups well below 100 MW(t)-d/kg HM are economically feasible for high ore costs and fuel recycle. A summary of conversion ratio versus burnup for two feasible heavy metal loadings is shown in Table 3.6.8 (from ref. 2).

Table 3.6.5. PBR Fuel Element Development Status

Fuel Description	Heavy Metal Loading (g/sphere)	Fraction of Overcoated Particles in Matrix (vol. %)	Comments
AVR fuel	6	~10	Off-the-shelf; fully qualified; 3.5 years with 950°C exit gas.
THTR fuel	11	10	Developed; tested in the AVR, qualified for use.
Developed fuel	16-25	12-21	Developed; under test in AVR; not fully qualified.
Projected fuel	30-45	25-40	Needs manufacturing development (now under way).

Table 3.6.6. Descriptions of PBR Fuel Cycles

Fuel Cycle	Designation	Description	Fuel Type	Loading (g HM/Sphere)	Moderation Ratio C/HM	Burnup (MW·d/kg)	Conversion Ratio
Mixed Oxide, 93% Enriched	M093 (TOT)	*Ref. cycle (THTR fuel)*	*U-Th mixed oxide*	11.1	325	100	0.60
Mixed Oxide, Low Enriched	1113	LEU cycle (9 MW/m³)	U mixed oxide	10.4	363	70	0.58
	1013	LEU cycle (9 MW/m³)	U mixed oxide	10.4	363	100	0.55
	1213	LEU cycle (9 MW/m³)	U mixed oxide	10.4	363	130	0.52
UO_2, Low Enriched	UOT	LEU cycle (5 MW/m³)	U mixed oxide	10.2	357	100	0.58
Mixed Oxide, 20% Enriched	M020	20% U-Th fuel	U-Th mixed oxide	8.1	453	100	0.58
Seed/Breed, 20% Enriched	SB20	Like M020, except separate fertile spheres	U oxide, Th oxide	6.0/16.5	458	101	0.56
Once-Through, Mixed Oxide	4011	Once-through cycle (9 MW/m³)	U-Th mixed oxide	15	244	101	0.58
	4021	Like 4011 (5 MW/m³)	U-Th mixed oxide	15	244	102	0.62
Seed/Breed, 93% Enriched	SB93	Like M093, with separate fertile spheres	U oxide, Th oxide	6.0/20.1	355	100	0.58
Mixed Oxide, Recycle	MOR	Recycling reactor	U-Th oxide	11.2	325	100	0.57
	RSE	More complex recycling	U-Th oxide	1.7/15.0	242/2110	100	0.62
	321	Variation of MOR, RSE	U-Th oxide	16.5	220	93.6	0.71
	SFB	Even more complex recycling	U-Th oxide	2.0-20.0 (4 streams)	180-1879	100	0.65
Prebreeder	PB	Prebreeder	U-Th oxide	1.4-32.4	198	23.2	0.74
Near Breeder	NB	Near breeder	U-Th oxide	32.4	110	24.0	0.97
Net Breeder, C/HM = 110, 20% FIMA	110/20	Net breeder with radial blanket	U-Th oxide	32	110	20	1.01
C/HM = 110, 10% FIMA	110/10	Net breeder with radial blanket	U-Th oxide	32	110	10	1.04
C/HM = 80, 20% FIMA	80/20	Net breeder with radial blanket	U-Th oxide	44	80	20	1.03
C/HM = 80, 10% FIMA	80/10	Net breeder with radial blanket	U-Th oxide	44	80	10	1.05
Pebble Bed Thermal Breeder Reactor	PBTBR	Net breeder with decoupled fissile and fertile fuel flow	U-Th oxide	32	110	24	1.1

Table 3.6.7. Relationship Between Increased PBR Conversion Ratios and Operating Parameters

Fuel Cycle	Conditions	Conversion Ratio	Remarks
1. Normal Converter Reactor	5 MW/m^3, 100,000 MW-d/t HM, 11.2 g HM/sphere, C/HM = 325	0.6	• Reference cycle (THTR fuel) • No problem
2. Recycle Reactor	5 MW/m^3, 93,600 MW-d/t HM, 16.5 g HM/sphere, C/HM = 200	0.71	• Heavy metal loading increased (new THTR fuel) • No problem
3. Recycle Reactor	5 MW/m^3, ~100,000 MW-d/t HM, ~30 g HM/sphere, C/HM = 120	0.8	• Heavy metal loading increased further • Separate feed-breed balls
4. Near Breeder	5 MW/m^3, 24,000 MW-d/t HM, ^{233}U fuel, 32.4 g HM/sphere C/HM = 110	0.97	• Continued increase in heavy metal loading • ^{233}U fuel needed
5. Near Breeder	2.5 MW/m^3, 24,000 MW-d/t HM, ^{233}U fuel, 35 g HM/sphere, C/HM = 110	1.0	• Continued increase in heavy metal loading • ^{233}U fuel needed • Reduces power density
6. Breeder	5 MW/m^3, 10,000 MW-d/t HM, ^{233}U fuel, 35 g HM/sphere, C/HM = 80	1.015	• Continued increase in heavy metal loading • Reduces burnup
7. Breeder	4.5 MW/m^3, 10,000 MW-d/t HM, ^{233}U fuel, 45 g HM/sphere, Radial blanket, C/HM = 80	1.05	• Radial blanket added • Cold streaks to be looked at
8. Breeder	5 MW/m^3, 10,000 MW-d/t HM, ^{233}U fuel, 45 g HM/sphere, Radial blanket, Top thorium blanket, C/HM = 80	1.06	• Top blanket added • Top blanket to be removed periodically
9. Breeder	5 MW/m^3, 24,000 MW-d/t HM, 32.4 g HM/ball, C/HM = 110, Decoupled fertile throughput	1.1	• Uses new technology for core (spheres, flow of spheres, control rods) • Has high pressure drop, lower efficiency, greater blower power

Table 3.6.8. Summary of Several PBR Mixed Oxide Recycle Cases with High Conversion Ratios and Low Burnup*

C/HM	Heavy Metal Loading (g/sphere)	Burnup (MW-d/kg HM)	Conversion Ratio
325	11.24	23.7	0.78
		31.8	0.77
		43.3	0.75
		59.8	0.70
		100.0	0.57
180	20.0	24.0	0.87
		32.0	0.85
		43.0	0.82
		60.0	0.78

*From ref. 2.

PBR Fuel with U.S. Particle Designs

The heavy metal loadings in PBR fuel elements fabricated by the cold-pressing techniques can be increased by using the U.S. feed/breed particle design shown as Variant 4 in Table 3.6.1. (This is a likely particle concept for the U.S. HTGR operating on HEU fuel with recycle.)

As noted earlier, PBR loading calculations performed for this study included Variant 4, assuming the same overcoating thicknesses specified for FRG Variant 3. The results, summarized in Table 3.6.9, show that with respect to heavy metal loadings, the U.S. feed/breed particle concept is more efficient than either the FRG one-particle designs or the FRG feed/breed design (Variant 3). Variant 4 could be loaded to 25 to 28 g/sphere in the Th/U ratio range of interest, whereas Variant 3 could be loaded only to 13 to 17 g/sphere. It is to be remembered, however, that the calculations assume the maximum theoretically possible volume loading (65 vol.%), and the loadings achieved to date in the FRG program are well below this.

Table 3.6.9. Calculated Maximum Heavy Metal Loadings for PBR Fuel Elements Fabricated by Cold-Pressing Techniques*
(Overcoated particles occupy 65 vol.% of sphere matrix.)

	Heavy Metal Loading (g/sphere)			
Th/U	Var. 1	Var. 2	Var. 3	U.S. Var. 4
5	25	20	9	21
10	25	20	13	25
20	25	20	17	28

*See Table 3.6.1 for descriptions of coated particle variants.

3.6.3. HTGR Heavy Metal Loading Capability

HTGR Fuel Particles and Fuel Elements

The heavy metal loading capabilities of HTGR fuel elements are somewhat more difficult to assess. While the detailed studies of the various PBR fuel cycle options included calculations of the grams of heavy metal required per fuel element, corresponding data are not available for the HTGR options. In particular, two parameters of great importance — the so-called "zoning factor" and the isotopic concentration of the fissile species in recycle fuel elements — were not included in published reports and therefore were not available to us for this study.

The fuel particle and fuel element designs currently being developed within the U.S. program are summarized in Tables 3.6.10a and b. One fissile and three fertile particle designs are being considered. The HEU fuel particle design for the steam-cycle HTGR (the design for which most of the fuel cycle calculations have been done) called for TRISO-coated fissile and BISO-coated fertile particles. The TRISO-coated fertile particle designs shown in Table 3.6.10a are being pursued for the higher fission-product retention capability thought to be necessary for advanced concepts (direct cycle and process heat). The larger TRISO-coated fertile particle (with the 600-μm kernel) is being developed to provide a separable system for recycle.

Table 3.6.10a. Features of HTGR Coated Particles

	Dense UCO	ThO_2 BISO	ThO_2 TRISO	ThO_2 TRISO
Kernel diameter (μm)	350	500	600	450
Coating thickness (μm)				
Buffer	115	95	110	70
ICTI	35		35	35
SiC	35		35	35
OLTI	40	85	40	40
Volumes (10^{-11} m^3)				
Kernels	2.245	6.545	11.310	4.771
Coated particles	26.808	33.304	58.898	27.826
Kernel oxide content (g/cc)	10.5	9.5	9.5	9.5
Heavy metal content				
Kernel density (g/cc)	9.2	8.3	8.3	8.3
Particle loading (10^{-5} g/particle)	20.65	54.32	93.87	39.60
Particle density (10^{-6} g/m^3 of particle)	0.770	1.632	1.593	1.432

Table 3.6.10b. Features of HTGR Fuel Elements

	10-Row Block	8-Row Block
Fuel holes/block	216	132
Rods/hole	14	14
Fuel rod volume (cm^3)	6.435	10.055
Rod volume per block (cm^3)	19,459	18,581
Volume available to particles (cm^3)	12,064	11,521

Two points are significant from the data included in Table 3.6.10: (1) the large TRISO-coated fertile particle provides 98% as much heavy metal loading capability as the BISO-coated particle; and the 10-row block design provides higher heavy metal loading capability than the 8-row block design.

Table 3.6.11 contains the heavy metal loading requirements for several HTGR concepts. (from refs. 6, 7). Knowing the amount of heavy metal required in the core, together with the number of fuel elements, the loading calculations should be fairly straightforward on an average basis. However, the loadings are not constant throughout the core. Due to considerations such as axial and radial flux flattening, age peaking, etc., the loadings are higher at the top and edges of the core than at the center and bottom. The term "zoning factor" is the ratio of the local heavy metal content per element to the average element loading. Figure 3.6.2 (from ref. 8) shows the way in which the U and Th zoning is influenced by power density. As power density increases, the ratio of peak U loading to average U loading increases much more rapidly than the same ratio for Th. Zoning factors calculated for various HTGR cases are summarized in Table 3.6.12 (from refs. 8-10).

Another factor which influences the loading calculation is that the recycle fuel elements have a higher concentration of parasitic isotopes, and must therefore have higher concentrations of fissile isotopes to maintain the same level of reactivity as fresh and makeup fuel. Table 3.6.13 contains some representative fissile loadings for several steam cycle reactor designs operating on the HEU fuel cycle (from refs. 7, 11). It should be noted that the recycle ^{235}U loadings are decidedly higher than the ^{235}U loadings in initial and makeup blocks. The ^{233}U recycle (23R) block loadings are about the same as the initial and makeup block loadings.

Isotopic concentration of ^{233}U and ^{235}U in recycle fuel elements (23R and 25R blocks) is also a factor in the calculation of heavy metal loadings. As shown in Table 3.6.14 (from refs. 11-14), the concentration of fissile uranium decreases from the beginning of recycle to equilibrium recycle. The data listed under "Ref. 11" in Table 3.6.14 shows the ^{233}U concentration changing from 92% at the beginning of recycle to 61% at equilibrium recycle. For the 25R blocks, the change is 73% ^{235}U to 30%.

Table 3.6.11. Initial Core Loadings for Various HTGR Fuel Cycle Cases

	Reference Steam Cycle Plant[a]	Lead Plant[b]	High Conversion Ratio Steam Cycle Plant[a]	Breeder[b]
Conversion ratio	0.66	0.76	0.82	1.0
kg Th/MW(e)	32.3	38.0	49.4	128.3
kg ^{235}U/MW(e)	1.4	1.57	1.89	
kg ^{233}U/MW(e)				4.91
Th/U (93% enriched)	21	23	24	24
C/Th ratio	214	220		70
Power density (W/cm^3)	8.4	7.0	6.0	

[a]From ref. 6.

[b]From ref. 7.

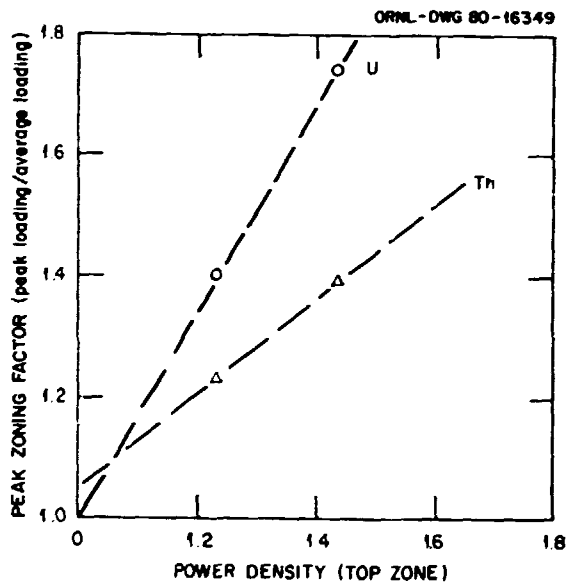


Fig. 3.6.2. Approximately Maximum Fuel Zoning Factors for a Four-Zone HTGR vs Zone Power Density.

Table 3.6.12. Heavy Metal Zoning Factors for Various HTGR Cases

	Zoning Factor[a]	
	Th	U
HTGR lead plant[b]	1.30	1.68
4000-MW(t) initial core[c] (5384 blocks)	1.45	1.63
4000-MW(t) reload (1376 blocks)	1.26	1.65
2000-MW(t) initial core[c] (2744 blocks)	1.32	1.50
2000-MW(t) reload[c] (704 blocks)	1.23	1.53
3000-MW(t) initial core[c] (3944 blocks)	1.21	1.43
3000-MW(t) reload (1064 blocks)	1.24	1.47
3000-MW(t) initial core[d]		
Layer 1	1.21	1.48
Layer 2	1.07	1.12
Layer 3	0.94	0.90
Layer 4	0.81	0.79
3000-MW(t) reload		
Layer 1	1.08	1.47
Layer 2	1.04	1.16
Layer 3	0.91	0.87
Layer 4	0.74	0.68

[a]Zoning factor = (peak heavy metal loading) : (average loading).

[b]From ref. 8.

[c]From ref. 9.

[d]From ref. 10.

HTGR Heavy Metal Loading Limits

Loading requirements for the HTGR are discussed in terms of several example cases presented in Tables 3.6.15 and 3.6.16. These cases were made up from data taken from the references cited. Complete data sets were not available for any of the cases, so assumptions had to be made on the basis of information presented in earlier tables from this section. The intent of the calculations presented in Table 3.6.16 was to establish the percent of available space in the fuel element required to achieve the heavy metal loadings necessary. The following comments and conclusions apply to the calculations shown in Table 3.6.16.

(1) It was assumed that maximum zoning factors of 1.48 for the fissile isotopes and 1.21 for the fertile isotopes applied to all cases. This seems reasonable in view of the data presented in Table 3.6.12, although for the lead plant it might have been more appropriate to use zoning factors of 1.68 and 1.30 respectively (also based on data presented in Table 3.6.12).

Table 3.6.13. Heavy Metal Loadings in Typical HTGR Fuel Elements

	Heavy Metal Loading (kg/Block)					
	Initial and Makeup ^{235}U Blocks		^{233}U Recycle Blocks		^{235}U Recycle Blocks	
	U	Th	U	Th	U	Th
HTGR Lead Plant[a]						
Initial core	0.381	9.19				
Transition reload	0.73	11.39				
Equilibrium recycle	0.79	11.39	0.63	11.39	1.31	11.39
Equilibrium HTGR[a]						
Initial core	0.394	9.19				
Transition reload	0.75	11.39				
Equilibrium recycle	0.81	11.39	0.65	11.39	1.31	11.39
Commercial HTGR[b]						
Block with max. U	0.87	7.4	0.72	12.1	1.46	8.9
Block with max. Th	0.49	8.4				
Block with min. U	0.36	5.0	0.31	6.0	0.48	6.0
FSVR[b]						
Block with max. U	0.32	11.63	0.99	11.63	1.23	11.63
Block with max. Th	0.15	12.11	0.45	12.11	0.61	12.11
Block with min. U + Th	0.11	7.84	0.37	7.84	0.49	7.84

[a]From ref. 7.

[b]From ref. 11.

Table 3.6.14. Concentrations of Uranium Isotopes in Typical HTGR Fuel Elements

Source	Fuel Cycle Step		Concentration (wt. %)					
			^{232}U	^{233}U	^{234}U	^{235}U	^{236}U	^{238}U
Ref. 11	Beginning of recycle[a]	23R		92.1	7.35	0.568	0.0245	0.0126
	Residual recycle	25R			1.40	73.0	15.7	9.88
	Fresh	IM[b]			0.79	93.0	0.22	5.81
	Equilibrium recycle	23R		61.4	24.3	8.02	6.30	0.0362
		25R			1.67	30.08	49.70	17.7
		IM			0.97	93.0	0.22	5.81
Ref. 12	Beginning of recycle	23R	0.022	44.6	9.62	14.9	23.4	7.40
	Equilibrium recycle	23R	0.024	40.1	12.7	12.3	26.9	7.95
Ref. 13	FSVR after 6 full power years	23R	0.032	78.4	17.3	3.7	0.62	
Ref. 14	Near equilibrium recycle, Conversion ratio = ~0.75	23R		75.38		4.62		
		25R				42.74		
		Makeup				93.24		

[a]Most reactive fuel.

[b]IM = Initial and makeup.

Table 3.6.15. Definitions of Cases for HTGR-SC Loading Calculations

	1, Reference[a,b]		2, Lead Plant[c]		3, Higher Conversion Ratio[b]	
	Initial Core	Reload	Initial Core	Reload	Initial Core	Reload
Power, MW(t)	3000	3000	3200	3200	3200	3200
Power, MW(e)	1200	1200	1280	1280	1280	1280
Number of fuel blocks	3944	1064	5288	1322	5288	1322
Power density (W/cm^3)	8.4	8.4	7.0	7.0	6.0	6.0
Conversion ratio	0.66	0.66	0.76	0.76	0.82	0.82
Heavy metal loading [kg/MW(e)]						
^{235}U	1.4	0.63	1.57		1.89	
Th	32.3	8.1	37.97		49.4	
Average loadings per block (kg)						
^{235}U (IM)	0.458	0.764	0.331	0.79	0.46	
23R				0.63		
25R				1.31		
Th	9.83	9.12	9.19	11.39	11.96	
Atom Density Ratios						
Th/C	21	12	22		24	
C/Th	214	214	220	180		

[a] From ref. 9.

[b] From ref. 6.

[c] From ref. 7.

Table 3.6.16. Summary of Loading Requirement Calculations for HTGR-SC Cases 1, 2 and 3[a]

Case[b]	Zoning Factor		Peak Heavy Metal Loading per Block (kg)				Particles per Block (10^{+6})		Particle Volume (cc) Per Block			Volume Percent of Fuel Space Needed For Particles	
	U	Th	^{235}U (IM)	23R	25R	Th	Fissile	Fertile	Fissile	Fertile	Total	10-Row Block	8-Row Block
1, Initial core	1.48	1.21	0.729			11.89	35.3	213.9	946	7289	8235	68	71
1, Reload			1.216			11.03	58.9	203.1	1578	6762	8340	69	72
2, Initial core	1.48	1.21	0.564			11.2	27.3	204.7	732	6317	7548	63	66
2, Recycle	1.48	1.21	1.169			13.78	56.6	253.7	1517	8448	9965	83	56
2, Recycle	1.48	1.21		0.922		13.78	45.1	253.7	1209	8448	9767	81	35
2, Recycle	1.48	1.21			1.949	13.78	94.4	253.7	2509	8448	10977	91	95
3, Initial core	1.48	1.21	0.732			14.47	35.4	266.4	950	8870	9820	81	85
3, Recycle	1.48	1.21	1.517			17.89	73.5	329.4	1964	10967	12936	107	112
3, Recycle	1.48	1.21		1.210		17.89	58.6	329.4	1570	10967	12537	104	109
3, Recycle	1.48	1.21			2.530	17.89	122.5	329.4	3283	10967	14250	118	124
2, Initial core	1.68	1.30							842	7324	8116	67	71
2, Recycle	1.68	1.30							1745	9076	10821	90	94
2, Recycle	1.68	1.30							1391	9076	10467	87	91
2, Recycle	1.68	1.30							2810	9076	11936	99	104

[a]See Table 3.6.15 for case definitions.

[b]Assumes that Case 3 ratio of fissile material in recycle blocks and makeup blocks to fissile material in initial core blocks is same as for case 2 (lead plant).

(2) It was assumed that 62% of the space in fuel rods would be available for particles. This is certainly an upper limit. GA has used values of 56% (ref. 15) in their calculations. As shown in Table 3.6.10, the 10-row block design has somewhat more space available (about 4.7% more) than the 8-row block design.

(3) For case 1 (reference steam-cycle HTGR, 0.66 conversion ratio, 8.4-W/cm^3 power density), there is no loading problem. Only about 70% of the space available for fuel is needed. No recycle was considered for this case, because loading data for the 23R and 25R blocks was not available.

(4) For case 2 (lead plant design, 0.76 conversion ratio, 7.0-W/cm^3 power density), essentially all of the space was needed in the recycle fuel elements, especially the 25R blocks (recycle of ^{235}U). If the higher zone factor values were used and the 56% particle volume fraction value, the 25R blocks could not pack in enough fuel. However, it could be argued that the 25R blocks could be used in regions of the core associated with lower zoning factors, but this would reduce flexibility in fuel management schemes. The loading requirements for the initial core (IC) for case 2 can be met with less space than for case 1, even though the heavy metal loadings for case 2 are higher. This is because the power density is lower for case 2.

(5) For case 3 (higher conversion ratio plant), the recycle loading requirements could not be met under the assumptions used in the calculation. The assumptions were that the maximum zoning factor values were 1.48 (fissile) and 1.21 (fertile), and the heavy metal loadings in the 23R, 25R and makeup fuel elements were increased by the same ratios (over case 2 values) as the ratios for heavy metal loadings in the initial core elements. This second assumption was required because no mass balance data were available for this case. While the fresh fuel loadings can be met for case 3, this is of little comfort. There is little logic associated with striving for higher conversion ratios unless the bred fuel can be recycled.

The conclusion reached in this analysis of the HTGR fuel loading requirements is that current technology will limit the conversion ratio to about 0.76 for a burnup of approximately 65 MW(t)-d/kg HM. Higher loadings are possible with a larger core (lower specific power). Higher conversion ratios can be achieved with the current design (7.0 W/cc) if the recycle fuel is used only in those regions of the core where low zoning factors are required. Data relative to options in fuel management schemes to qualify this possibility were not available for this study.

3.6.4. Comparison of PBR and HTGR Fuel Element Heavy Metal Loading Capabilities

The following conclusions have been reached in this comparison.

Higher conversion ratios are possible by modified designs or fuel cycle assumptions. An economic penalty is associated with either option. Lower specific power would require a larger core and higher capital cost. Lower fuel burnup would result in higher fuel cycle costs. Lower burnup is a more feasible possibility with the PBR because of its on-line refueling feature. Lower burnup for the HTGR would mean more frequent shutdowns and possibly lower reactor availability.

The PBR fuel element has greater potential for improved loading capability through advanced technology. Thus, the potential higher conversion ratio is greater in the PBR because of higher possible loadings and the economic feasibility of low burnups due to on-line refueling. If the overcoating could be made thinner, or eliminated through the use of the hot-pressing fabrication technology, higher heavy metal loadings could be achieved. Less flexibility exists with regards to high fuel loadings for HTGR fuel. Coating thicknesses substantially reduced in thickness relative to current designs are unlikely to meet the performance requirements currently in place.

There is considerable uncertainty associated with this comparison, because of the lack of specific designs and d tailed calculations of zoning factors, heavy metal loading requirements for each type of prismatic fuel element (initial core, makeup, 23R, 25R) etc. The quality of information on the PBR system, relative to heavy metal loading requirements, was superior to what was available for the HTGR system.

3.6.5. Cost Estimate for Fuel Development for PBRs

A cost estimate was prepared for u › with the PBR vs. HTGR assessment. This estimate represents the cost increment associated with development of PBR fuel within the U.S. program. The following assumptions were used:

(1) No costs would be associated with fuel particle development. The particles would be developed under the HTGR program.

(2) Costs associated with the Fort St. Vrain reactor surveillance are also covered under the HTGR fuel development program.

(3) All HOBEG fabrication technology would be available to GA at no cost. Some development costs would be incurred by GA in setting up the equipment and processes in San Diego, and in making a product using a somewhat different particle concept and somewhat different requirements (i.e., different broken particle fraction during fabrication and during irradiation).

A program to develop and qualify PBR fuel is outlined in Fig. 3.6.3. This program calls for five HRB capsules, two HT capsules, and proof testing in the Fort St. Vrain reactor. There does not currently exist a capability in the U.S. program for testing spherical fuel elements. Some work would be necessary to modify test rigs and perhaps develop a miniature fuel element for testing in HRG capsules.

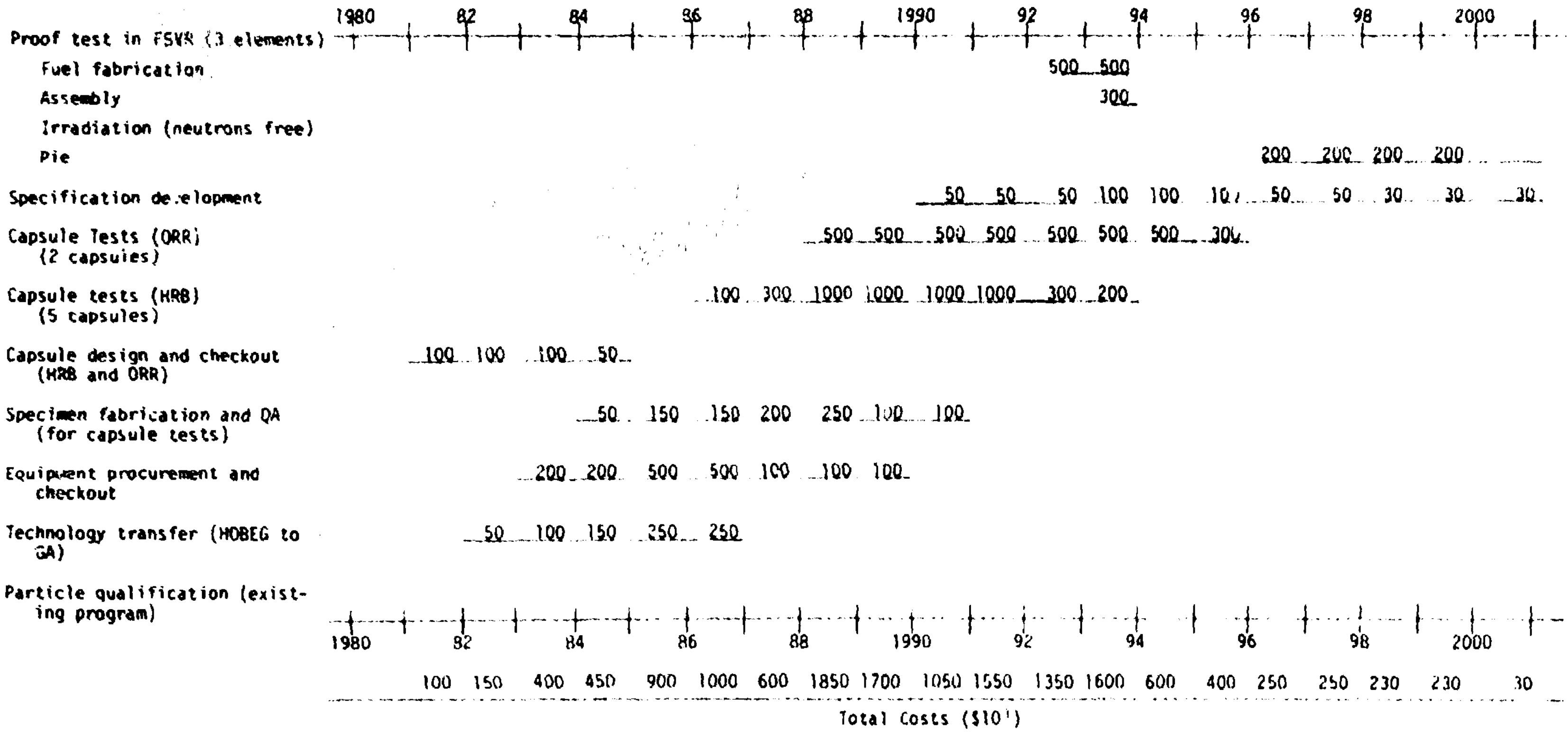


Fig. 3.6.3. Cost Estimate for PBR Fuel Development Qualification and Licensing Within the U.S. Program (1980 Constant Dollars).

As design work proceeds on the reactors, there may be changes in the fuel performance requirements. A decision relative to final application (direct cycle, steam cycle, or process heat) and specific design features (intermediate heat exchanger or no intermediate heat exchanger, for example) will strongly influence fuel performance requirements. The qualification program may need to be stretched out if the performance requirements become more strict.

The incremental cost increase associated with PBR fuel development in the U.S. relative to HTGR fuel development totals approximately $15 million.

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3. J. J. Rütten et al., "The Pebble-Bed HTR as a Net-Breeding Reactor System," Jül-1521 (July 1978).
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5. Based on internal KFA-Jülich reports.
6. P. R. Kasten, F. J. Homan et al., "Assessment of Thorium Fuel Cycle in Power Reactors," ORNL/TM-5565 (January 1977).
7. Memo dated February 3, 1977 from M. Whitner (GA) to R. K. Lane (GA). Subject: Heavy Metal Loadings for Lead Plant HTGR.
8. Memo dated February 7, 1977 from A. M. Baxter (GA) to R. F. Turner (GA. Subject: Maximum Packing Fractions in the Lead Plant HTGR.
9. Memo dated October 16, 1975 from S. Karlin/J. Redding (GA) to C. Smith (GA). Subject: Distribution of Required Packing Fractions in Large HTGR.
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11. Memo dated June 21, 1974 from A. Bagierer (GA) to S. Strausberg (GA). Subject: FSV Fuel Block Loadings.
12. Memo dated June 17, 1974 from K. R. Schultz (GA) to Sid Strausberg (GA). Subject: Further Information on Recycle Fuel Block Loadings for ORNL.
13. Memo dated March 18, 1975 from K. R. Schultz (GA) to Sid Strausberg (GA). Subject: FSVR Fuel Loadings.
14. A. R. Olsen, "Pebble and Prismatic Fuel Element Fabrication and Refabrication Cost Estimates," ORNL GCR report in preparation.
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3.7. REACTOR AVAILABILITY

V. H. Guthrie

(Note: Most of the information in this section has been extracted from a General Electric Company report[1] authored by C. R. Davis and W. B. Scott.)

3.7.1. Introduction

Reactor availability directly influences the capital cost component of energy since an increase in availability permits more energy to be obtained without any significant increase in capital investment. A comparison of availability between the HTGR and the PBR is particularly important because the PBR feature of continuous refueling during operation would seem to indicate a higher availability of the PBR than that of the HTGR, which must be shut down for refueling. However, both reactors must be shut down for (1) periodic servicing of the core equipment, (2) maintenance and inspection of the turbine-generator, (3) in-service inspection of primary coolant boundaries and reactor internals, and (4) maintenance of primary and secondary heat transfer system components. If the time required to carry out each of these tasks is the same for the two reactor types, and if the HTGR is never refueled more frequently than the scheduled shutdown for maintenance, and if the refueling can be done in parallel with the scheduled maintenance and within the same time frame, then the total reactor unavailability due to refueling and scheduled maintenance would be the same for the PBR and the HTGR.

The objective here is to compare the plant outages of the PBR and the HTGR resulting from scheduled shutdowns for refueling, inspection, and maintenance. This is done by reviewing the HTGR work flow charts for scheduled shutdowns and determining whether there will be any differences in the tasks to be carried out for the PBR or in the times required to perform the tasks. Although an annual refueling of the HTGR is most likely, this comparison also covers 2- and 3-year shutdown intervals for the two types of reactors. In considering this comparison, it should be noted that shutdowns for unscheduled maintenance can also cause a loss of reactor availability. While this aspect of reactor availability is not treated here, it is discussed in Section 3.5.5.

3.7.2. Assumptions

This study of availability concentrates on the differences between the services required for the spherical fuel of the PBR and the prismatic fuel of the HTGR. The differences in the two types of fuels result in different types of refueling systems, different designs and quantities of control rod and control rod drive assemblies, and different designs and quantities of reflector elements to be replaced. Likewise, the handling equipment for fuel, for control rods and drives, and for reflectors are different. Ex-core systems such as the turbine, heat transfer systems, etc. are assumed to be similar for the two concepts. With these points in mind, the following assumptions are made:

(1) The differences in the availability of two equal size plants (PBR vs. HTGR) used for the same purpose (generation of electricity or process heat) will be

almost totally dependent upon the differences in the time required to service the cores and other reactor components.

(2) PBR refueling is accomplished during power operation.

(3) HTGR refueling requires reactor shutdown and depressurization of the primary circuit.

(4) Outage requirements to maintain the components of the primary and secondary heat transfer systems (PHTS and SHTS) are the same for each plant. This could change if one plant has more radioactivity plated-out in the PHTS than the other (see Section 3.5).

(5) Maintenance of all HTGR core-servicing equipment is accomplished between refueling activities without the need for reactor shutdown.

(6) Maintenance of all PBR core-servicing equipment, except for safety valves, can be accomplished during power operation. Replacement of safety valves (49 in the refueling system) requires reactor shutdown and depressurization.

(7) The reference reactors were an HTGR-SC rated at 900-MW(e) (ref. 1) and a PBR-SC rated at 3000-MW(t) (ref. 2).

(8) The time required for in-service inspection (ISI) of the primary coolant boundary and reactor internals is assumed to be the same for both reactors.

3.7.3. Evaluation of HTGR and PBR Availability

The work flow chart for a scheduled annual refueling shutdown for a steam-cycle 900-MW(e) HTGR is shown in Fig. 3.7.1. A similar flow chart can be determined for a PBR by projecting the times required for control rod and control rod drive removal and replacement, maintenance of the core-servicing equipment, and reflector removal and replacement. Recall that these tasks are the only ones assumed to impact differences in availability between the concepts. Each task discussed below is followed by an analysis of the effect of increasing the servicing intervals and refueling period of the HTGR to two or three years. The time required for maintenance of the turbine-generator may control the shutdown time interval and is discussed separately.

Fuel Element, Control Rod, and Control Rod Drive Replacement

HTGR. Refueling of the HTGR is accomplished on an annual basis to replace one-fourth of the fuel assemblies. During the same outage (reactor shutdown and depressurization), an average of one-eighth of the control rods and control rod drive assemblies are also replaced. Removal and replacement of control rods and drive assemblies are accomplished in parallel with refueling, so that only 5 hr is added to the refueling time, i.e., 2 hr to remove the first control rod and drive assembly before start of refueling, and 3 hr to replace the last control rod and drive assembly after the end of refueling. [This time was confirmed at a recent refueling outage of the Fort St. Vrain Reactor (FSVR).[3]]

According to GA, replacement of fuel assemblies requires 8.7 min each, and this was used here. This time may be optimistic when compared to observed and estimated times for other plants; however, the GA refueling system is not available for evaluation. Some other plant refueling times are given in Table 3.7.1; it is estimated that 10 min per assembly is achievable in HTGRs.

Table 3.7.1. Refueling Times for Several Reactors

Plant	Time Required to Replace One Fuel Assembly (min)
Fort St. Vrain Reactor (HTGR)	126
Prototype Fast Reactor (LMFBR)	36
Conceptual Design Study (LMFBR)	32
Brown's Ferry Reactor (BWR)	10

The annual refueling time required for the 900-MW(e) HTGR is 156 hr (6.5 days). This includes handling 936 fuel assemblies and 14 control rods and drive assemblies. [It also includes 104 reflector elements, whose replacement times are the same as the fuel element replacement times (see further discussion below).]

PBR. Although not required to refuel the PBR, reactor shutdown and depressurization will be necessary to remove and replace control rods and drive assemblies (on an annual basis or on some other predetermined schedule). Since the PBR control rods and drive assemblies are similar in design and installation to those used in LMFBRs, it is expected that handling equipment and removal and replacement times will be the same as for the Clinch River Breeder Reactor (CRBR) or the Conceptual Design Study (CDS) plant. The equipment for the CRBR has been designed and built and will be performance tested. The time is predicted to be 7.2 hr per control rod and drive assembly, which is the same (7 to 8 hr) as is estimated for the THTR in West Germany.

In the reference PBR, there are 151 control rod and drive assemblies; 51 are estimated to have a 4-yr life and 100 are estimated to have indefinite life. Therefore, on an annual basis, one-fourth of the 51 (13) and one-tenth of the 100 (10) will be removed and replaced. Removal, inspection and replacement of the 100 control rc. and drive assemblies at the rate of 10 per year will satisfy in-service inspection (ISI) requirements. Section XI, Division 2 of the ASME Code requires that components traversing the primary coolant boundary be inspected 100% within a 10-yr period.

Removal and replacement of control rods and drive assemblies requires also the removal and replacement of some (estimated as one-fourth of 43) of the fuel feeding tubes with their valves, hoppers, distributors, and other components. On an annual basis, it is estimated that removal of the fuel feeding tubes will require 66 hr, that removal and

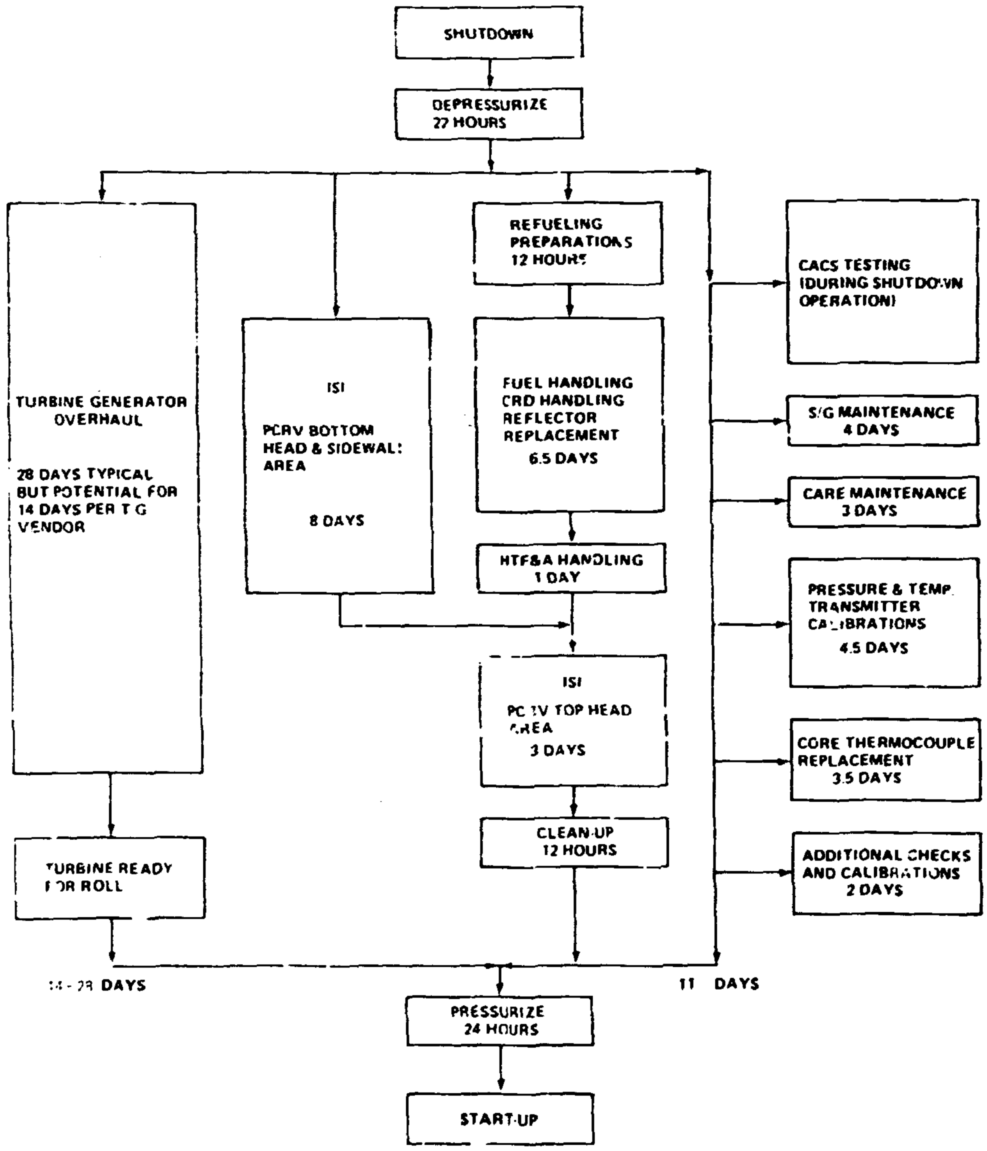

Fig. 3.7.1. Typical Outage Flow Chart for a 900-MW(e) Steam Cycle HTGR (1-year Interval).

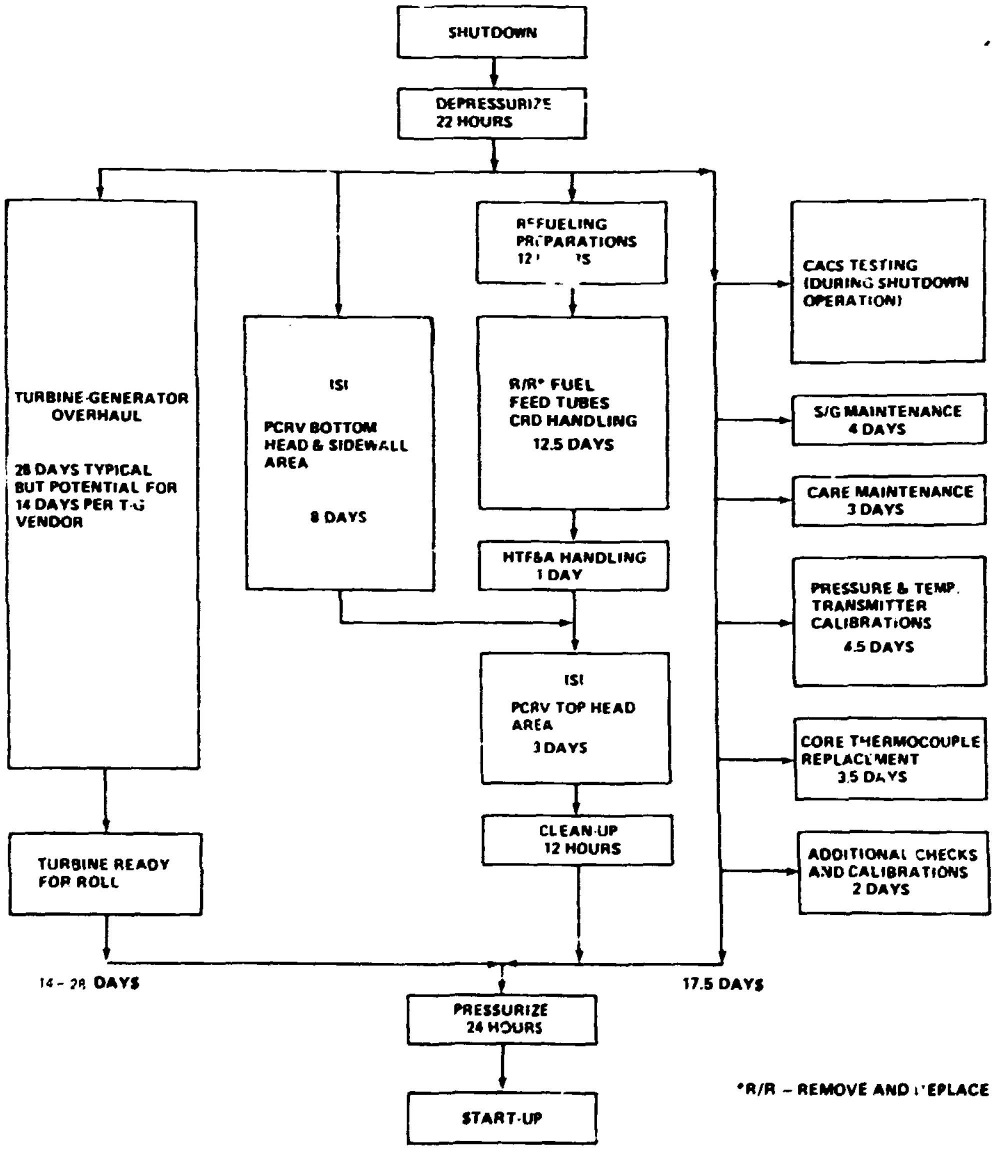

Fig. 3.7.2. Typical Outage Flow Chart for a 3000-MW(t) PBR (1-year Interval).

replacement of the control rods and drive assemblies will require 166 hr, and that replacement of the fuel feeding tubes will require 66 hr. The total time would be 298 hr. The resulting flow chart for a 3000-MW(t) PBR plant is shown in Fig. 3.7.2.

The reference control rod lifetime used in this study is 4 yr. If the lifetime could be extended to 8 yr as estimated by KFA, the annual core-servicing outage time would be reduced by two days.

Maintenance of Core-Servicing Equipment

HTGR. Maintenance of all core-servicing equipment used for the HTGR can be accomplished between refueling operations and hence no outage times for this activity are required.

PBR. Maintenance of all core-servicing equipment used for the PBR, except the equipment required for the removal and replacement of safety valves, can be accomplished during power operation.

It is estimated that during the 40-yr life of the plant, there will be seven safety-valve failures (ref. 1, App. B). Six of the 43 safety valves located in the fuel feed tubes and one of the six safety valves used in the spent fuel exit tubes will need to be replaced. To replace a safety valve requires that the reactor be shut down and depressurized. An air-lock (or glove box) must be built about the valve and the valve removed and replaced by working through glove ports in the air-lock. For replacement of the safety valve in a spent fuel exit tube, the air-lock must include gamma shielding and, if necessary, be large enough for the use of long-handle tools in order to protect personnel from excessive exposure to radiation (ref. 1, App. C). Table 3.7.2 presents the times required to remove and replace the seven safety valves and the resultant effect on plant availability.

Table 3.7.2. Times Required for Removal and Replacement of Fuel Feed and Spent Fuel Exit Safety Valves in a PBR

Action	Time (hr)	
	Fuel Feed	Fuel Exit
Shut down and depressurize	24	24
Install air-lock	8	24
Remove/replace safety valve	4	12
Remove air-lock	8	24
Pressurize and start up	24	24
Total hours per valve	68	108
Total number of valves	6	1
Total hours	408	108
Average hours per year	13	

Reflector Removal and Replacement

HTGR. The top, bottom, and radial side reflectors for the HTGR are removed and replaced in the same manner as the fuel assemblies. Assuming a 10-yr life for reflectors, one-tenth of them are replaced each year (104) as a part of the refueling operation. The time required is 8.7 min per reflector, which, as noted above, is included in the refueling time given in Fig. 3.7.1.

PBR. Only the top and upper one-third of the reflectors in the PBR require removal and replacement.* This replacement has been determined to be necessary only once during the life of the plant.[5] It is assumed that this activity can be accomplished, if not in parallel with, then in series with an annual outage for control rod and drive assembly replacement. For this reason, no additional time is required for reactor shutdown, depressurization, pressurization, and startup. Two concepts of equipment designs and procedures have been developed for reflector removal and replacement, one by Novatome of France and the other by KFA of West Germany. Both concepts are described in ref. 5. Table 3.7.3 gives the times required for replacement of the top and upper one-third radial sidewall reflectors.

Table 3.7.3. Times Required for Removal and Replacement of Top Reflector and Upper One-Third of Radial Reflector in a 3000-MW(t) PBR

Action	Time (hr) Concept 1	Time (hr) Concept 2
Unload one-third of fuel elements (477)[a,b]	179	477, 179[b]
Remove and replace reflectors	900	520
Reload one-third fuel elements[a]	577	577
Total Hours	1,656	1,276
Average Hours per Year	41.4	31.9

[a]Unloading is at a rate of 2,100 fuel spheres per hour, and reloading is at a rate of 1,733 fuel spheres per hour (see Sections 5.2.2 and 5.2.3 of ref. 2).

[b]Unloading of one-third of the fuel spheres in the core requires 477 hr; however, unloading can be accomplished in parallel with the 298 hr required to remove and replace control rods and drive assemblies (see Fig. 3.7.2).

*See discussion in Section 3.8 concerning possible development of reflector graphites to last throughout plant lifetime.

Alternative Core-Servicing Intervals

HTGR. For the 2-yr refueling interval for the HTGR, the following assumptions are made:

(1) the life of the fuel assemblies is 4 yr;
(2) the life of the reflectors is 10 yr;
(3) the life of the control rods is 8 yr;
(4) the time required for in-service inspection is reduced by 5%, since more inspection is accomplished during the extended refueling time; and
(5) the time required for turbine-generator maintenance is reduced by 10% over that required for annual outages.

The resultant outage time is given in Fig. 3.7.3. On an annual basis, this is an increase of 1% in the maximum theoretical availability.

For the 3-yr refueling interval, the assumptions are revised as follows:

(1) the life of fuel assemblies is 6 years (the same as for the FSVR);
(2) the life of the reflectors is 12 yr;
(3) the life of the control rods is 8 yr;
(4) the time required for in-service inspection is reduced by 10% since more inspection is accomplished during the extended refueling time; and
(5) the time required for turbine-generator maintenance is reduced by 20% over that required for annual outages.

The resultant outage time is given in Fig. 3.7.4. On an annual basis, this is an increase of 1% in the maximum theoretical availability.

PBR. For the 2-yr core-servicing interval for the PBR, the following assumptions are made:

(1) the life of fuel is 3 to 4 yr (same as for annual core-servicing outage);
(2) the life of the reflectors is about 20 yr (same as for an annual core-servicing outage);
(3) the life of the control rods is 4 yr for one-third (51) and 10 yr for two-thirds (100);
(4) the time required for in-service inspection is reduced by 5% since more inspection is accomplished during the increased core servicing time; and
(5) the time required for turbine-generator maintenance is reduced by 10% over that required for annual outages.

The resultant outage time is given in Fig. 3.7.5. On an annual basis, this is an increase of 1% in the maximum theoretical availability.

For the 3-yr interval the following assumptions are made:

(1) the life of the fuel is 3 to 4 yr;

(2) the life of the reflectors is about 20 yr;

(3) the life of the control rods is 6 yr for one-third (51) and 10 yr for two-thirds (100);

(4) the time required for in-service inspection is reduced by 10% since more inspection is accomplished during the increased core servicing time; and

(5) the time required for turbine-generator maintenance is reduced by 20% over that required for annual outages.

The resultant outage time is given in Fig. 3.7.6. On an annual basis, this is an increase of 2% in the maximum theoretical availability.

Steam Turbine-Generator Maintenance

It is interesting to note that for all the outage intervals cited above for the HTGR, the turbine-generator is the "critical path" item. For the PBR, the critical path item may not be the turbine-generator servicing, depending upon the time required.

The vendor recommends[6] that the turbine-generator be totally disassembled and rebuilt, with appropriate repairs and replacements, at the end of 1 yr of operation and every 5 yr thereafter. This procedure requires 8 to 10 weeks each time the turbine-generator is disassembled and rebuilt. For a 40-yr life plant, there would be eight turbine-generator outages (years 1, 6, 11...36). On an annual basis this would be equal to 11.2 to 14.0 days per year. The vendor also reports that turbine-generators being supplied for nuclear plants can be disassembled and rebuilt in sections. Dependent upon the number of sections, the required outage time can be reduced to 2 to 3 weeks a year, or 3 to 5 weeks on a 2- to 3-yr interval.

HTGR. As indicated in Fig. 3.7.1, the HTGR-SC plant may employ a turbine-generator outage of 14 to 28 days annually. For 14 days of turbine-generator servicing, the time required for refueling and other core-servicing activities is about the same since the two activities could be performed in parallel. Therefore, the resultant maximum theoretical plant availability could be 96%.

PBR. For the PBR, 14 days annual outage for the steam turbine-generator servicing is less than the 19.5 days of outage required for core servicing. Therefore, 19.5 days would result in a lower theoretical plant availability of 94%.

Projected Plant Outages

The projected scheduled plant outages of the HTGR and PBR are summarized in Table 3.7.4. Note that the time required for reflector and safety valve replacements for the PBR have been annualized and 2 days/yr added to the scheduled outage on the average. For all scenarios, the outage for the PBR is longer than that for the HTGR.

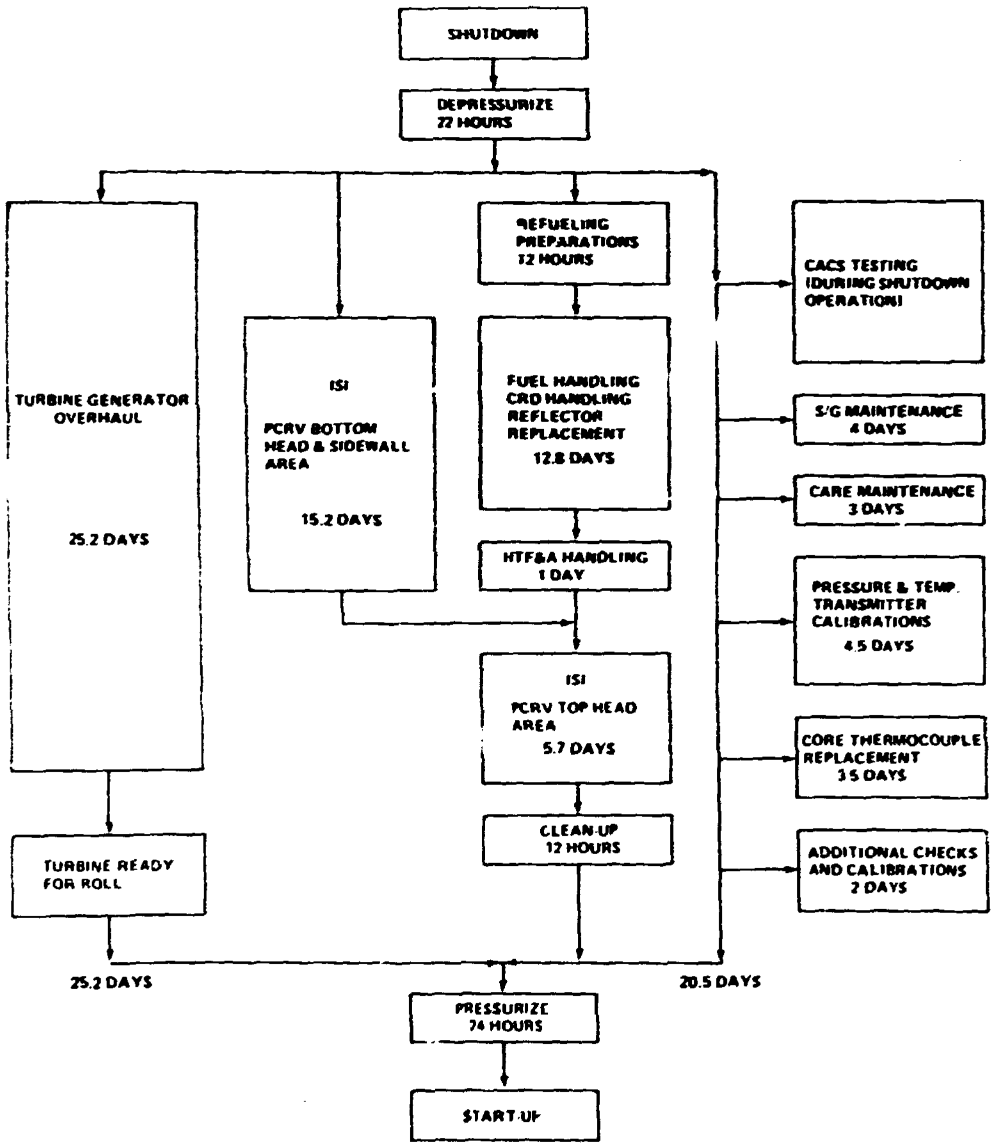

Fig. 3.7.3. Typical Outage Flow Chart for 900-MW(e) Steam Cycle HTGR (2-year Interval).

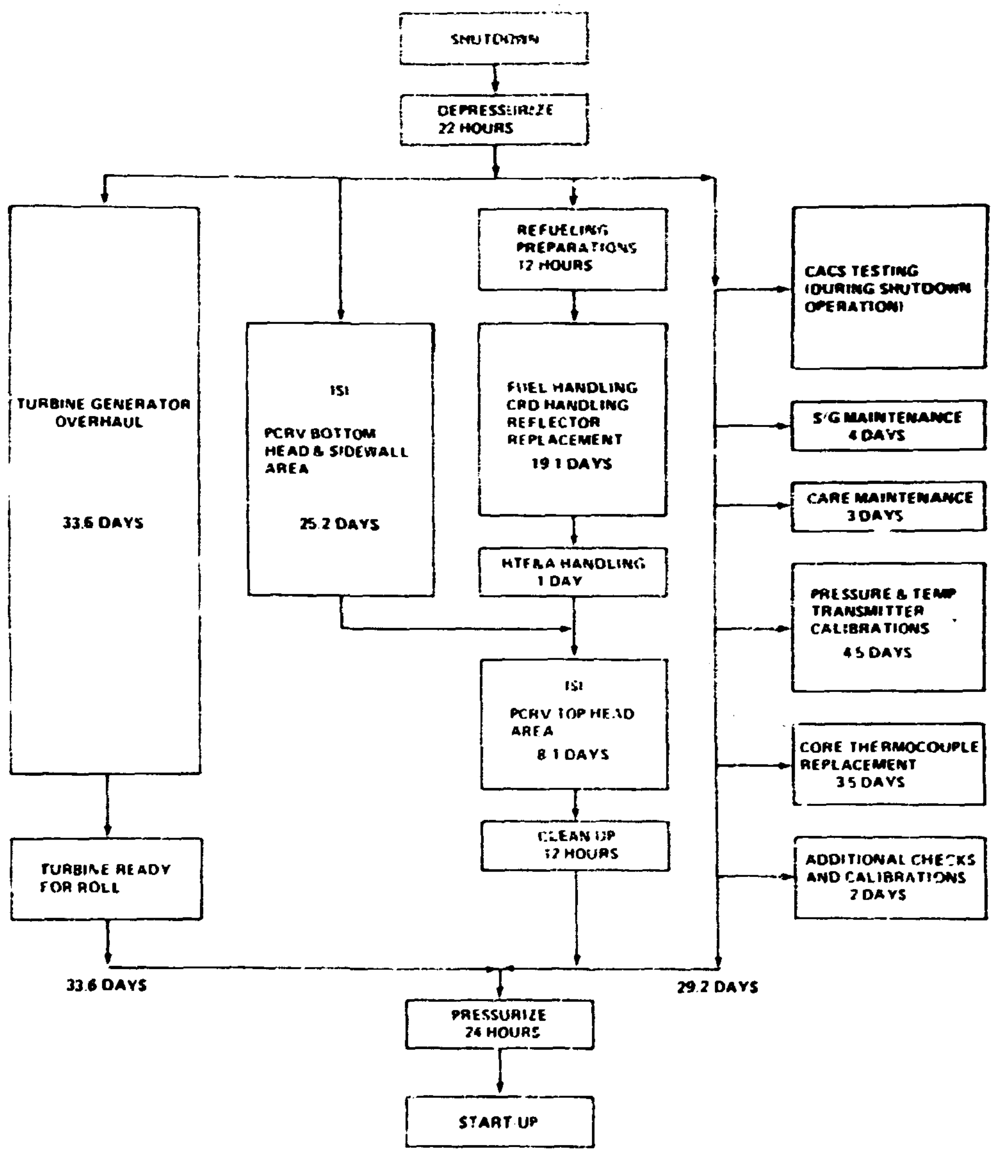

Fig. 3.7.4. Typical Outage Flow Chart for 900-MW(e) Steam Cycle HTGR (3-year Interval).

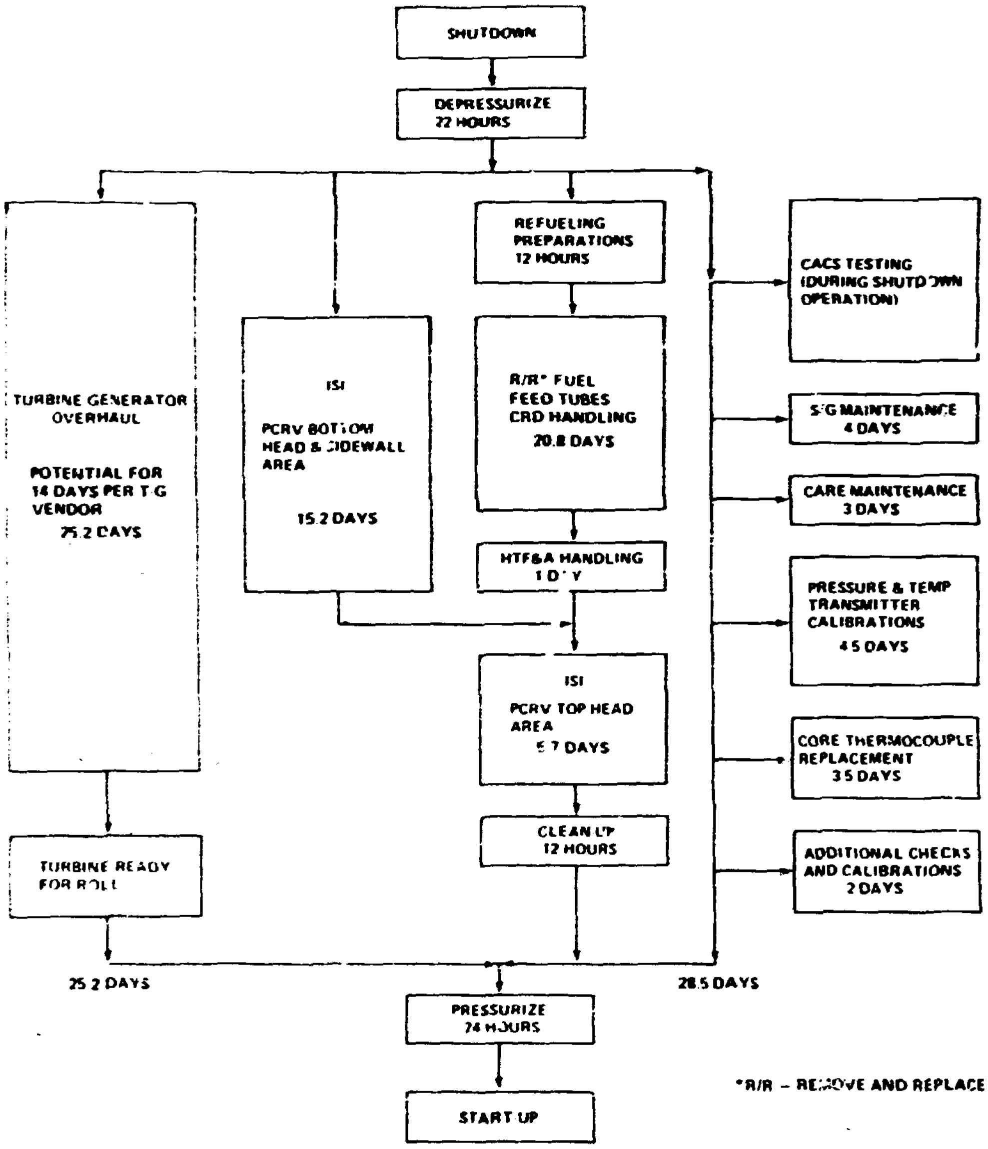

Fig. 3.7.5. Typical Outage Flow Chart for 3000-MW(t) PBR (2-year Interval).

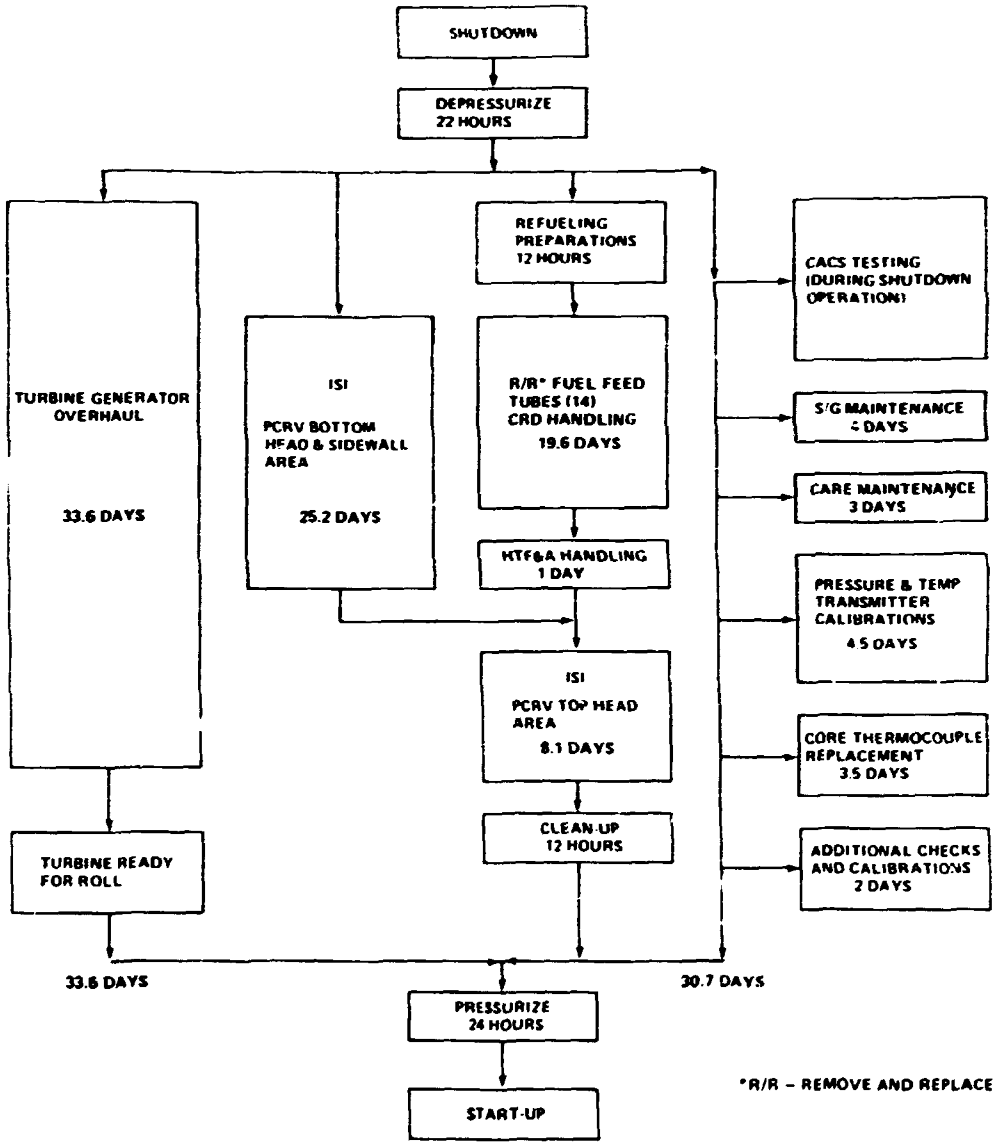

Fig. 3.7.6. Typical Outage Flow Chart for 3000-MW(t) PBR (3-year Interval).

Table 3.7.4. Comparison of HTGR and PBR Outage Times for Refueling, Inspection and Maintenance

Activity	Time (Days) Per Scheduled Shutdown: One-Year Interval		Two-Year Interval		Three-Year Interval	
	HTGR	PBR	HTGR	PBR	HTGR	PBR
Shutdown and depressurization	1.0	1.0	1.0	1.0	1.0	1.0
Equipment preparation	0.5	0.5	0.5	0.5	0.5	0.5
Refueling	6.5	NA[a]	13.0	NA	19.5	NA
Removal and replacement of control rods and drives	[b]	12.5	[b]	21.0	[b]	19.5
Removal and replacement of top and radial reflectors	[b]	1.5[c]	[b]	3.0[c]	[b]	4.5[c]
Maintaining serving equipment	NA	0.5[d]	NA	1.0[d]	NA	1.5[d]
In-service inspection[e]	11,3.0	11,3.0	21,5.5	21,5.5	33,8.0	33,8.0
Removal and replacement of high-temperature filter and absorber	1.0	1.0	1.0	1.0	1.0	1.0
Cleanup	0.5	0.5	0.5	0.5	0.5	0.5
Steam turbine-generator maintenance[e]	14,2.5 [28,16.5][f]	14,0 [28,10.5][f]	25,4.5	25,0	33.5,4.5	33.5,3.5
Pressurize and startup	1.0	1.0	1.0	1.0	1.0	1.0
Total outage	16.0 [30.0][f]	21.5 [32.0][f]	27.0	34.5	36.0	41.0
Availability ( )	96 [92][f]	94 [92][f]	96	95	97	96

[a]NA = Not applicable.

[b]Included in refueling time for HTGR.

[c]Occurs only once in the life of the plant.

[d]Maintenance of fuel feed and spent fuel exit safety valves requires seven outages during plant life.

[e]First number in each column represents total time for activity; second number represents time required for part of activity that cannot be accomplished in parallel with other activities.

[f]Assumes turbine-generator maintenance requires 28 days.

In terms of normal scheduled outages (additional 2 days/yr not included for the PBR), the following observations are made:

(1) The PBR core-servicing outage is longer than the HTGR annual refueling outage; i.e., 19.5 days versus 13.5 days (5.3% vs. 3.7%). The PBR core-servicing outage is used to service the control rods and drives. The assumed life of one-third (51) of the control rods is only 4 yr. However, KFA is striving to lengthen control rod life to 8 yr, and, if successful, the annual PBR core outage time would be reduced to 17.5 days. HTGR control rods and drives are serviced in parallel with refueling.

(2) The outage time required to service an HTGR turbine-generator on an annualized basis exceeds the HTGR refueling time; i.e., 14.0 days versus 11.5 days. (Outages caused by maintenance of other major components, such as heat exchangers, helium circulators, recuperators, etc., were not considered.)

(3) The outage time required to service the PBR turbine-generator on an annualized basis is less than the PBR core-servicing outage, i.e., 14.0 days versus 17.5 days. (Outages caused by maintenance of other major components were not considered.)

(4) Plant availabilities can be theoretically increased by 1% to 2% by allowing 2- or 3-yr intervals between core-servicing and turbine-generator maintenance outages (see Table 3.7.4). This is desirable if core servicing, in-service inspection and other major component maintenance can be accomplished in parallel.

3.7.4. Research and Development Costs for Core-Servicing Equipment

R&D costs for HTGR core-servicing equipment will be minimal. GAC has completed the design and engineering of the prismatic-fuel handling equipment, which also removes and replaces reflector elements. The cost of building and testing prototype refueling equipment should be minimal since the engineering principles employed are the same as are used for the FSVR, which has a proven performance record. Handling of control rods and drives can also use proven FSVR type equipment.

Most of the handling equipment for PBR new and spent fuel has been developed and operated in West Germany in the AVR, and in the near future it will be operated in the THTR. If sufficient information is available to satisfy U.S. licensing requirements, no additional R&D should be required. Since the control rods and drives for the PBR are similar in design and installation to those used in CRBR, the CRBR equipment could be adaptable to the PBR. The CRBR equipment has been built and is scheduled for performance testing in the near future. Removal and replacement equipment for reflector elements has been only cursorily addressed in Europe so far. Two conceptual designs have been prepared with the objective of proving to European utilities that the operation can be performed, if necessary. Continuation of the development of either design on a schedule for final design, building, and testing has not been established. If a decision is made to build a PBR, it would be prudent for the U.S. to plan on developing and performance testing the reflector handling equipment. It is estimated that such an R&D project would cost about $10 million, including a full-size mockup of the reactor for testing purposes.

3.7.5. Recommendations

It is recommended that a more in-depth study of plant maintenance and availability be carried out when GA releases the overall design for a large HTGR and when a comparable design of a PBR is available. It is also recommended that the decision to build either the HTGR or the PBR be based on criteria other than core-servicing or steam turbine-generator outage time requirements, since differences obtained are well within the uncertainties of such estimates.

The major item in the PBR core-servicing time is the removal and replacement time for control rods and drive assemblies. This time depends on control rod life and inspection requirements. PBR development efforts should be directed toward extending control rod life and reducing the total number of control rods required.

The spent fuel unloading system for the PBR should be evaluated for redesign so that the principal components would be more like the new fuel feeding system. This would include transfer of spent fuel by pipe(s) through the reactor containment to storage, thereby eliminating spent fuel container transporters and decreasing radiation exposure to personnel.

If the PBR plant is selected for construction in the United States, it is recommended that its design be carried out in parallel with the design of the necessary equipment for reflector element removal and replacement. Likewise, a facility for development and performance testing of this equipment needs to be designed and built.

References

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2. "Germany Pebble-Bed Reactor Design and Technology Review," Gas Reactor International Cooperative Program, Report No. COO-4057-6, Prepared by GE-ARSD for US-DOE, September 1978.

3. Telecon GE No. XP-886-CRD00006, Fred Swartz, Public Service of Colorado to Chuck Davis, GE-ARSD, January 30, 1980.

4. Telecon, GE No. YP-848-00067, Dr. Kraus Peterson and Dr. Singh, KFA, West Germany, to A. J. Lipps, GE-ARSD, February 21, 1980.

5. Dr. Neis, KFA, West Germany, Unpublished Report.

6. Telecon, GE No. YP-271-00012, Bob Mills, Product Services, Steam Trubines, GE Schenectady, N.Y., to Don E Lutz, GE-ARSD, February 15, 1980.

3.8. GRAPHITE REFLECTOR DAMAGE

W. P. Eatherly

3.8.1. Introduction

The graphite reflectors for both the HTGR and the PBR are subject to high temperatures and neutron fluences, and the resulting stresses and dimensional instability are a major problem. The damage to the graphite reflectors is a much more important problem for the PBR than for the HTGR for two reasons:

(1) The front face of the side and bottom reflectors in the HTGR are composed of graphite blocks similar to the hexagonal fuel element blocks and can be replaced periodically during refueling with the same equipment used for replacing the fueled blocks. The PBR side reflectors, however, cannot be so easily replaced since the fueled pebbles in the core must be emptied from the core at least to the depth to which the reflector must be replaced. This would be a lengthy and expensive process that should be avoided if at all possible.

(2) The side reflectors in the PBR must serve the additional function of providing lateral containment of the core. Thus, maintaining structural integrity in the graphite reflectors takes on an obvious added significance for the PBR.

The preferred solution for the PBR is to design both the top and side reflectors such that they will not need to be replaced during the lifetime of the reactor. However, to ensure that the graphite will withstand the high temperature and neutron fluences over the reactor lifetime will require detailed analyses of the expected damage rates of the graphite grades being considered as potential candidates for use in the reflector. The objective here is to provide an estimate of reflector lifetimes that might be expected for the PBR.

Preliminary indications from results of ORNL irradiation experiments on German-grade graphite[1] are that for the large PBR, the established reference grades of graphite being considered will not last much longer than perhaps one-half of a 30-year reactor lifetime (see below). Certain specialty grades of graphite, such as POCO AXF, might provide a viable solution if it can be verified that their use substantially increases reflector lifetimes. However, experimental results on the damage rates of these specialty graphites are fragmentary with regard to both fluence and temperature and therefore accurate estimates of their expected lifetimes cannot be made at this time. A description of the PBR reflectors and a more detailed discussion of potential problems are provided in refs. 2 and 3.

3.8.2. Experimental Studies of Graphite Damage

The development of reflector and support structure graphites in the Federal Republic of Germany (FRG), which must consider the demanding requirements of systems for process heat as well as electricity generation, has advanced to the stage where several candidate materials have been fabricated on full-scale production runs. The previously established reference grade graphite ATR-2E is now considered too anisotropic to withstand the severe neutron irradiation exposure without excessive dimensional instability; therefore, new isotropic graphites with greater dimensional stability are needed for use in those regions of more severe exposure in the reactor. These graphites are primarily based upon domestically (German) available fillers derived from coal tar coke, in contrast to the U.S.-developed petroleum-derived filler-cokes.

FRG graphites that have been examined in irradiation experiments in the High-Flux Isotope Reactor (HFIR) at ORNL are listed in Table 3.8.1. (Their properties have been given previously by Haag et al.[4]) The graphites were irradiated to a maximum EDN fluence* of 2.0×10^{22} neutrons/cm^2, and dimensional changes, electrical resistivity by eddy-current measurement, elastic constants by sonic measurements, brittle-ring strengths, and the 500°C coefficient of thermal expansion (CTE) were measured. The irradiation was performed at 600 to 620°C as indicated by SiC thermal monitors.

Table 3.8.1. Graphites Irradiated in HFIR

Grade	Filler	Type of Fabrication
V483	Pitch coke	Isostatic molding
V356	Petroleum coke	Isostatic molding
ATR-2R	Semi-isotropic pitch coke	Vibrational molding
ATR-2E	Semi-isotropic pitch coke	Extrusion
ASR-1R	Pitch coke	Vibrational molding
ASR-2R	Pitch coke	Vibrational molding
UKAEA-11*	Pitch coke	Extrusion

*An experimental UKAEA grade run for comparison.

The time at which a particular graphite returns to its original bulk density after contraction is defined as its "lifetime." At this point, the physical properties are degrading rapidly and this definition thus closely approximates the true useful life. As a part of this investigation of reflector graphites, a significant body of data on the thermomechanical properties has permitted the conclusion that this definition of lifetime is conservative. The volume changes of several of these graphites are shown in Fig. 3.8.1, and the corresponding lifetimes in Table 3.8.2. Thus, if graphite grade ATR-2E were to be used for the reflector material, it could be used with confidence to an EDN fluence of 1.7×10^{22} neutrons/cm^2 or slightly greater.

*EDN = Fluence (EDN) ≈ Fluence (E > 0.18 MeV)/1.8.

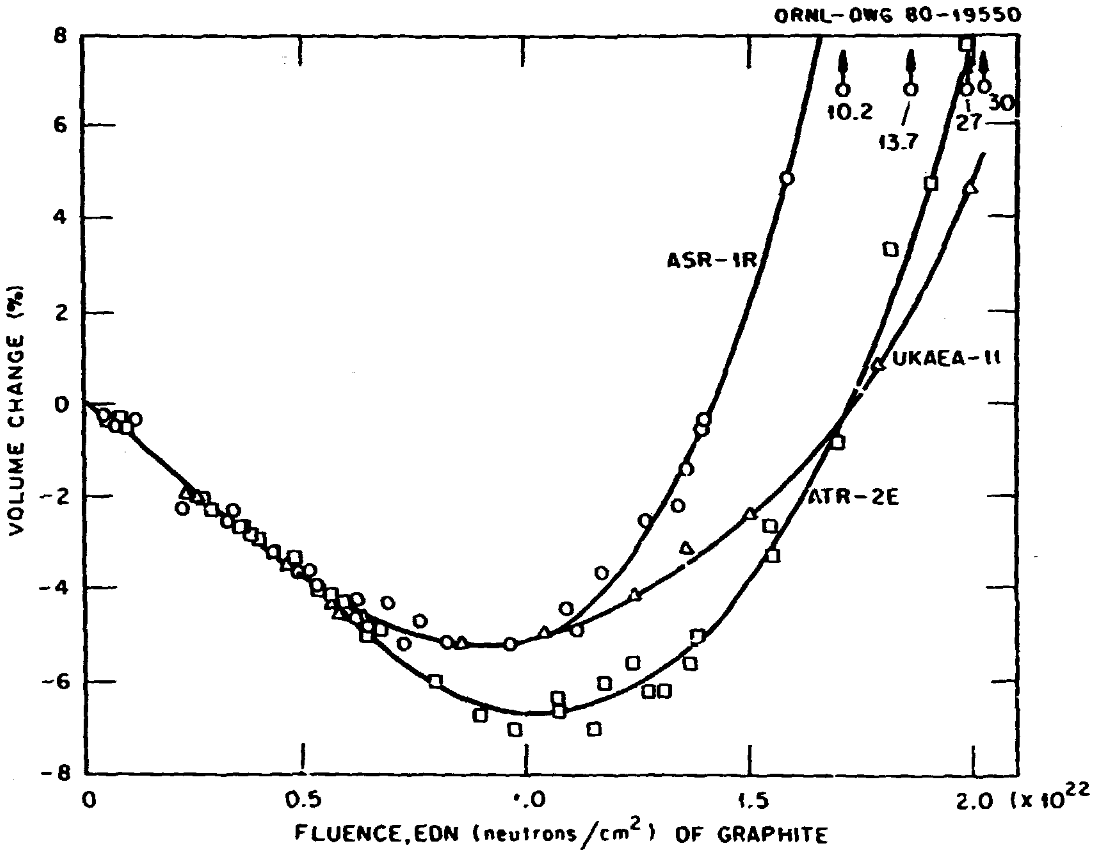

Fig. 3.8.1. Percent Volume Change of Graphite as a Function of Fluence.

Table 3.8.2. "Lifetimes" of Graphites Tested in HFIR

Grade	Graphite Lifetime ($\Delta V/V_0 = 0$) Fluence, EDN (10^{22})	Type of Fabrication
V483	1.55	Isostatic molding
V356	1.6	Isostatic molding
ATR-2R	1.4	Vibrational molding
ATR-2E	1.7	Extrusion
ASR-1R	1.45	Vibrational molding
ASR-2R	1.35	Vibrational molding
UKAEA-11	1.7	Extrusion

Microscopic damage to the graphite increases rapidly as the volume of the graphite undergoes a net volumetric expansion (see Figs. 3.8.2 and 3.8.3). As illustrated in Fig. 3.8.2, the moduli of elasticity increase in two stages before the maximum densification of the graphite is achieved. As the graphites achieve maximum density and begin to expand, the structure degradation is evidenced by a decrease both in Young's modulus and in Poisson's ratio. This loss in structural integrity is also observed by a decrease in the strength of the graphite (see Fig. 3.8.3).

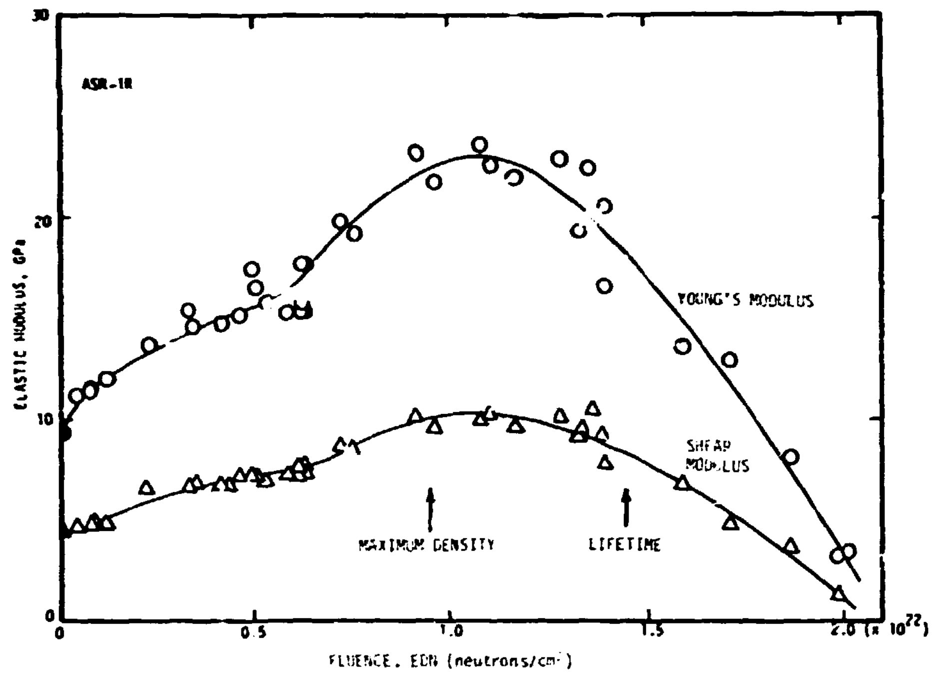

Fig. 3.8.2. Young's Modulus and Shear Modulus for Graphite as a Function of Fluence.

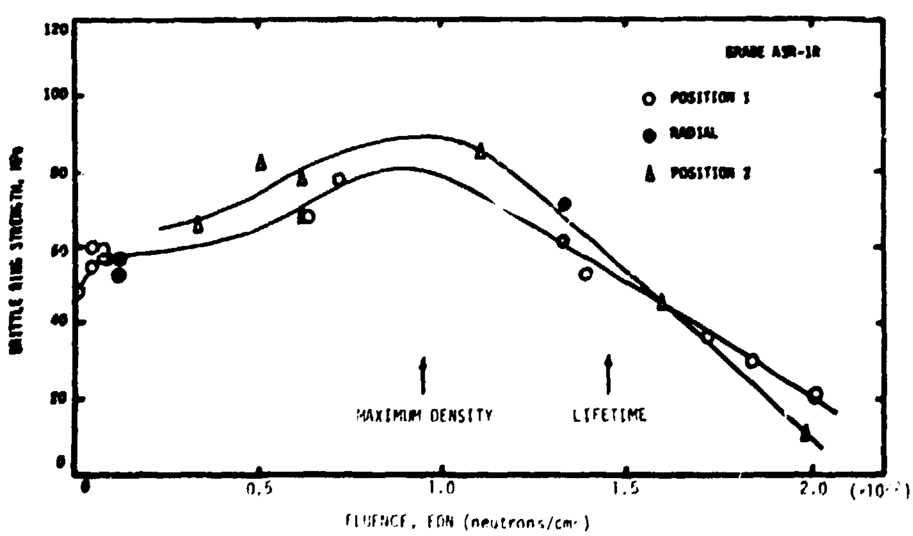

Fig. 3.8.3. Brittle Ring Strength of Graphite as a Function of Fluence.

In general, the linear growth of the isotropic German graphites is dominated by the rate of densification. The maximum density for all grades was obtained at an EDN fluence of from 0.9 to 1×10^{22} neutrons/cm^2 and the overall lifetime depended upon the subsequent rate of volume expansion. The German grades ATR-2E and V356 were found to be equivalent to or slightly longer lived than UKAEA No. 11. The actual choice of reference grades for the reflector and core support blocks will obviously be based upon subsequent evaluations of compatibility or performance with design and economic considerations. But if a graphite reflector is required to last a full reactor lifetime, on the order of 30 years, then with the high flux levels in a large PBR, the improved graphites under development must be shown to have projected lifetimes greater by a factor of approximately two than those studied here. Such improvements appear feasible.

3.8.3. Cost of Improved Graphite

A graphite reflector capable of maintaining its structural integrity throughout a reactor's lifetime will require the development and use of a superior grade of graphite. This implies an increase in the capital cost of the reflectors above that for the current reference graphites. It is estimated the cost of improved graphite (not including development) would be about a factor of ten greater than present type graphite ($20-30/lb versus $2-3/lb).

References

1. Results presented by C. R. Kennedy at the American Nuclear Society Winter Meeting held in San Francisco, 1979.

2. "Gas Reactor International Cooperative Program Interim Report," Volume titled "German Pebble Bed Reactor Design and Technology Review," General Electric Report COO-4057-6, September 1978.

3. "Pebble Bed Reactor Review Update," Fiscal Year 1979 Annual Report, General Atomic Company, January 1980.

4 G. Haag, W. Hammer, and M. F. O'Connor, "Preliminary Results from the FRG Graphite Programme for the Process Heat HTR," Proceedings of the Fifth London International Carbon and Graphite Conference, London, 1978.

3.9. SEISMIC EFFECTS

G. A. Aramayo

3.9.1. Introduction

The purpose of this review is to assess the state of knowledge of seismic effects on the core regions of HTGRs and PBRs. The review is primarily concerned with the safety considerations as related to licensing requirements. The safety issues under consideration pertain to the ability of the reactor to operate safely when subjected to low-level seismic excitations and, in the limit, to the ability of the reactor to achieve a safe shutdown when subjected to a higher level excitation.

Extensive work has been performed in the U.S. to verify the integrity of the HTGR core when subjected to seismic excitations of the range of levels expected in the U.S. Reference 1 presents a review of the work performed at GA to verify the methodology used to assess the seismic issues related to licensing requirements of HTGRs. This work covers extensive and expensive development of computer programs that study the core behavior. Most of these programs are mathematical simplifications based on observations and results obtained from a broad experimental program. The assignment of numerical values to the parameters involved in the analytical work is also a result of the experimental work.

By contrast, it is not clear that similarly detailed work has been performed to assess the core behavior of PBRs in response to a seismic excitation. Reference 2 presents a cursory review of work associated with the PBR seismic issue. Some analytical work has been performed at General Electric Advanced Reactor Systems Division, but the results do not permit an adequate engineering assessment. Neither have the results of an experimental program conducted in West Germany in support of the PBR system been made available.

The cost of obtaining the knowledge required to address the seismic question is a function of the number of problem areas needing further work. These areas and their current status are discussed below.

3.9.2. Current Results

The safety-related areas of major concern with respect to seismic effects are:

(1) Core Disarray. - Is there any possibility that as a consequence of a seismic excitation the core will undergo a disarray that will cause blockage of the coolant and prevent insertion of control rods?

(2) Core Support. - What is the probability of failure of the core support structure as a consequence of a seismic event?

(3) Core Lateral Restraints. - What would be the effects of failure or loss of support of the core due to structural failure of the side reflectors?

(4) Top Reflector Response. - How would the top reflector structure respond relative to the reactor core? (It is clear that if large relative motions exist, the ability for insertion of control rods could be impaired.)

The above issues have been addressed very specifically for the HTGR cores.[1] To look at the problem areas, GA has developed a number of good programs that are based on the observation that in the scale model testing of the HTGR core, the core behaves as a single unit. It is then possible to uncouple the core so that by analyzing a horizontal layer and a column, conclusions can be drawn on the core behavior. It appears that the analysis addresses the issues in detail, although there is some question as to the validity of the data used in the impact problem (impact between the core blocks and between core and reflector blocks); the concern is with regard to scaling issues. The question of core disarray seems to be addressed correctly on the basis of the experimental work; additional problems are to be considered if higher temperatures in the core are used, since the ability of the dowels and core support parts at this higher temperature might be limited.

The state of knowledge on these issues is less well established for the PBR. The available sources have been considered, but most of the conclusions presented below are primarily subjective.

One of the sources in ref. 2, which presents a brief summary of work conducted at GA on seismic effects on the PBR. However, this work does not really address any of the areas that could be considered as safety items. Moreover, it is quite questionable whether the mathematical representations of the core in the analysis are correct. Observation of the PBR core suggests that it would behave like a highly viscous and highly damped fluid and not a collection of blocks similar to the HTGR core. In any event, the scope and content of the results of this analysis are very limited.

Finally, on the basis of observation of a scaled-down model of the PBR core, HRB in Germany has concluded that there is no problem associated with blockage of the cooling passages caused by the accumulation of smaller pebble particles and KLAK at the chutes which might occur as a consequence of a seismic event. Traditionally in the case of granular material there is a segregation by size of the granular components with the smaller size particles going to the bottom of the container. Observations of the experiment indicated that only the top two or three layers of pebbles suffer any significant disarray; thus the bulk of the core would not undergo any considerable disarray. The core behaves as a single unit with no significant relative motion between core barrel and pebbles; this is also based on experimental observation. Control-rod insertion during a simulated seismic event presented no problem on the basis of experimental observation.

3.9.3. Conclusions

At the present time the issue of core integrity subject to seismic effects is not totally resolved for either the HTGR or the PBR, although in the case of the HTGR, an extensive R&D effort has been conducted. While most of the work has been documented, there is a need for a report that summarizes the past work and indicates how the various concerns related to seismic issues are addressed. For the PBR, significant additional research, both analytical and experimental, is still needed. On the other hand, it appears that either concept is capable of being developed into a seismically safe system without unreasonable effort.

References

1. General Atomic Company, "Core Seismic Methods Verification Report," GAA14812 V.1 (December 1979).

2. General Atomic Company, "Pebble Bed Reactor - Review Update," GAA15687 (January 1980).

3.10. TEMPERATURE/FLOW OSCILLATIONS IN HTGRs

P. R. Kasten

One technical issue that has not been addressed in this comparative analysis of the PBR and HTGR, but one that would be thoroughly analyzed preliminary to the selection of the HTGR for commercialization, is the temperature/flow oscillations that have been observed in the Fort St. Vrain HTGR (FSVR) as the power has increased above certain levels. These oscillations apparently have been due to periodic tilting of fuel elements within the core. The fuel block movements open up alternate flow paths, leading to oscillations in coolant flow through a specific region of the reactor and causing substantial and varying changes in the core outlet coolant temperature at a given position. A similar type of temperature oscillation would not occur in a PBR since no significant changes in coolant flow paths could take place in a pebble bed system.

The temperature oscillation problem became evident during the operation of the FSVR at power levels of about 60 to 70% of the design level, and, under certain conditions, at lower power levels. Significant temperature variations in the coolant leaving a specific region of the FSVR core were observed, with the period of an oscillation being about 10 minutes. Investigations of the phenomenon included measurements of neutron fluxes in the region of temperature change, and the flux readings exhibited marked step changes in amplitudes with time. Since physical movement of the graphite blocks could open up pathways for neutrons to more easily stream to the neutron detection instruments, and thereby increase the flux readings, it was inferred that block movement had occurred.[1] In addition, out-of-reactor tests were also carried out – at General Atomic (GA) and at the nuclear research center near Jülich, Federal Republic of Germany (FRG) – which indicated that block movement can cause variations in cooling in specific regions. (The FRG results were reported by H. G. Groehn.[2])

In the FRG studies, bypass flow was introduced by causing a wedge-shaped gap (1.85 to 6 mm) to be located between two adjacent graphite blocks. Coolant velocity distributions and pressure losses were measured in pertinent regions. From the results it was inferred that bypass flow through gaps between stacks of fuel blocks could cause variations in coolant temperature at specific locations.

The GA explanation of the mechanisms responsible for the FSVR oscillations is known as the "Jaws theory."[3] This theory postulates that periodic tilting of fuel elements near the top of the core will open up alternate coolant flow paths through the "jaws" so formed, and that the resulting flow changes through a region's coolant channels could cause substantial and rapid changes in coolant outlet temperature. GA postulated that the largest temperature fluctuation observed could have been caused by a 38% change in region flow, and that such a flow change was feasible based on the Jaws model.

To investigate the validity of the Jaws model, ORNL[4] performed modeling studies and calculated the regional flow variations required to provide the observed outlet temperature perturbations. The calculated flow variations appear larger than could reasonably be expected from Jaws-type bypass flow leakage; however, it is quite possible that the measured temperatures of the outlet coolant were incorrect, since bypass flow leakage into the thermocouple assembly sleeve itself could have affected the temperature readings. Thus, the uncertainties associated with region outlet thermocouple readings makes it difficult to prove or disprove the Jaws theory.

Since the neutron flux measurements at FSVR definitely indicate physical motion of the blocks, and since the temperature measurements are probably not accurate because of fluid flow bypasses affecting the temperature readings, it is reasonable that the temperature fluctuations in the FSVR are indeed due to block motion, with this motion probably being due to pressure differentials across blocks and to temperature gradients in the core support structure. By reducing the gaps between blocks in new core designs, control of the temperature oscillations to tolerable levels should be possible.

Another factor which can influence coolant bypass flow is the effect of irradiation exposure on graphite dimensions. Particularly for the higher temperature operating systems, the effect of reactor irradiations on graphite dimensional changes can be significant over a period of time (see Section 3.8). Thus, the design of an HTGR, particularly for the higher outlet coolant temperatures, has to be done with care so that changes in block dimensions during reactor operation do not lead to bypass flows that can cause regional variations in coolant temperature.

In summary, while temperature oscillations in HTGR systems should be controllable by proper core design, it is important that all the factors which can influence flow oscillations be carefully considered, particularly for the higher outlet coolant temperature systems.

References

1. Private communication from Walter Simon, General Atomic Company, to Paul R. Kasten, Spring 1980.

2. H. G. Groehn, "Disturbance of the Cooling Gas Distribution in HTGR Fuel Blocks Due to Bypass Flow," KFA-Jülich, Institute for Reactor Components, presented at the Topical Meeting on Nuclear Reactor Thermal Hydraulics, Saratoga, New York, October 6-8, 1980.

3. G. Kuzmycz, NRC Memorandum, "Summary of Meeting Held on December 14, 1978 to Discuss FSVR Fluctuations," (January 12, 1979).

4. S. J. Ball, J. C. Cleveland, and J. C. Conklin, "High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research, Quarterly Progress Report April 1 - June 30, 1979," ORNL/NUREG/TM-356 (NUREG/CR-1091).

3.11. PLANT CAPITAL COSTS

J. G. Delene and M. L. Myers

3.11.1 Introduction

As has been explained in Section 2.9, the capital costs of the HTGR and PBR were compared by estimating the costs for the HTGR systems and then estimating the change in costs in various capital cost categories for the same plant with a pebble-bed core. This section explains how capital cost estimates, in 1979 dollars, were obtained for HTGRs based on all three concepts (steam cycle, gas turbine, and process heat). In each case a 3000-MW(t) plant was assumed, and for the HTGR-PH, estimates were also obtained for a 1000-MW(t) plant. The estimates include both direct and indirect costs and a contingency allowance but exclude costs for interest during construction (IDC) and costs for escalation during construction.

Because the work scope for this comparative assessment did not provide for a detailed cost analysis of the individual systems, the general procedure was to use available information to obtain an internally consistent set of cost estimates for the three concepts in as much detail as possible. The available information largely consists of capital investment cost estimates by General Atomics (GA) and United Engineers and Constructors (UE&C), beginning with 1975 dollar estimates for a 3000-MW(t) HTGR-SC and a 3000-MW(t) HTGR-GT [1160-MW(e)] (ERDA-109, ref. 1) and for a 3000-MW(t) HTGR-PH, both with and without an intermediate heat transfer loop (IHL) (ORNL/TM-5409, ref. 2). Later, GA and UE&C issued companion reports giving estimates in 1979 dollars for a somewhat smaller HTGR-SC system, GA providing the NSSS costs[3] and UE&C the balance of plant (BOP) costs.[4] In the analysis presented here, data from these later GA and UE&C reports were used as a basis for scaling and otherwise adjusting the 1975 dollar estimates for the 3000-MW(t) plants to 1979 estimates. In addition, costs for the PCRV structure, liners, and penetrations and for the thermal barrier and containment annulus were estimated from unit costs and quantities of material. These estimates are discussed in Section 3.3.

3.11.2. Technique for Estimating HTGR-SC and HTGR-GT Costs

The 1975 dollar estimates presented in ERDA-109 for the 3000-MW(t) HTGRs are actually adjusted costs based on the breakdown of a steam cycle NSSS* bid package for a 770-MW(e) plant in January 1973 dollars. The 1973 estimates for the HTGR-SC were adjusted by UE&C to a 1160-MW(e) plant [$\sim$3000 MW(t)] in mid-1974 dollars. They then assumed a further escalation of 5% to obtain a total estimate for the NSSS bid package of $\$126 \times 10^6$ in January 1975 dollars.

*In nuclear terminology NSSS (originally coined as the acronym for Nuclear Steam Supply Systems) is commonly used to cover the reactor-related components in all nuclear plant concepts.

The cost breakdown of the 1975 estimate, together with that for the original 1973 bid package, is shown in Table 3.11.1. Here the PCRV liner and penetration cost was taken directly from ERDA-109 (Table 6.6-1), and the costs for the remainder of the reactor equipment items were assumed to increase [from the 770-MW(e) 1973 values] in proportion to the increase in the balance of the totals for the two estimates. In addition, the reactor internals category was assumed to include the permanent side reflector, hexagonal reflector blocks, and the PCRV pressure relief systems cost items.

Estimates of NSSS costs for the HTGR-GT in January 1975 dollars are also shown in Table 3.11.1. These costs, which are based on Tables 6.6-2, 6.6-3, 6.1-1 and 6.7-2 of ERDA-109, total $121 million.

To the NSSS costs were added the reactor equipment balance of plant (BOP) costs to obtain a total cost summary by cost category for the 3000-MW(t) HTGR-SC and HTGR-GT in 1975 dollars. The reactor equipment BOP costs, presented in Table 3.11.2, were estimated from Tables 6.7-2 and 6.1-1 in ERDA-109, using the cost breakdowns given in Table 3.11.1.

The January 1975 cost breakdowns of the non-reactor plant accounts were assumed to be those in Table 6.7-2 of ERDA-109.

The next step was to estimate the 3000-MW(t) HTGR-SC costs in January 1979 dollars. The general procedure was to take the unit 1, equilibrium NSSS base scope price estimated by GA for the 900-MW(e) HTGR-SC [2240 MW(t)] (Table 4-2, ref. 3) and add to it the BOP costs estimated by UE&C (Table 5.3-1, ref. 4). These costs were scaled to 3000 MW(t) using scale factors for each account. These scale factors, given in Table 3.11.3, are based on ORNL estimates and are employed in ORNL code CONCEPT.[5] The capital cost for the heat reject system for the HTGR-SC plant was obtained by escalating the $\$40.8 \times 10^6$ for a dry cooling system in 1975 dollars (given Table 6.4-2 of ERDA-109) for 4 years at 8%.

The 1979 capital investment costs for the HTGR-GT plant were obtained as follows:

(1) The 1975 cost ratios between the HTGR-SC and the HTGR-GT were assumed to hold to 1979, so that, except as noted, all costs were obtained by applying these ratios to the 1979 HTGR-SC costs.
(2) As in the case of the HTGR-SC plant, the costs for PCRV structure, liners and penetrations, and for the thermal barrier and the containment annulus were calculated from weights, volumes and unit costs. This analysis is discussed elsewhere in this report.
(3) The equipment portion of the instrumentation and control was assumed to be the same as for the HTGR-SC, although in ERDA-109 this ratio is 0.43 (GT/SC).
(4) The circulating water system items (232) and intake structure (214) were taken as part of the heat reject system (Item 26).

Table 3.11.1. 1975 NSSS Cost Estimates[a] for 1160-MW(e) HTGR-SC and HTGR-GT [3000-MW(t)]

Item	Estimated Costs ($10⁶) HTGR-SC 770-MW(e), Jan 1973	HTGR-SC 1160-MW(e), Jan 1975	HTGR-GT, 1160-MW(e), Jan 1975
PCRV Liner and Penetration	6.32	21.22	30.09
Thermal Barrier	3.01	9.79	12.79
Reactor Internals	5.57	18.12	18.12
Reactor Control System	3.10	10.08	10.08
Total Reactor Equipment	18.00	59.21	71.08
Reactor Coolant System	12.00	35.18	33.00
Safeguards Cooling System	1.32	3.89	4.06
Rad Waste System	0.32	1.22	1.26
Fuel Handling Equipment	3.12	9.14	9.14
Helium Service System	1.48	4.31	2.50
Instrumentation and Control	3.77	13.02	b
Total NSSS	40.01	126.0	121.04
PCRV Construction		11.5	22.8
Total, including PCRV		137.5	143.8

[a]From ERDA-109 (ref. 1).

[b]UE&C includes all I&C in BOP costs (see Table 3.11.2).

Table 3.11.2. 1975 Reactor Equipment Balance of Plant (BOP) Costs* for 1160-MW(e) HTGR-SC and HTGR-GT [3000 MW(t)]

Item	Estimated Costs ($10⁶) HTGR-SC	HTGR-GT
PCRV Construction	2.454	2.503
Main Heat Transport System	1.770	7.987
Safeguards Cooling System	0.414	0.414
Rad Waste System	2.496	2.496
Fuel Handling System	0.808	0.808
Other Reactor Plant Equipment	16.790	20.583
Instrumentation and Control	3.191	7.181

*Based on Tables 6.1-1 and 6.7-2 in ERDA-109 (ref. 1).

Table 3.11.3. Capital Investment Cost Scale Factors

Account Number	Account Name	Scale Factors
21	Structures and Improvements	0.5
22	Reactor Plant Equipment	0.6
23	Turbine Plant Equipment	0.8
24	Electric Plant Equipment	0.4
25	Miscellaneous Plant Equipment	0.3
26	Main Condenser Heat Reject Equipment	0.8
91	Construction Services	0.43
92	Home Office Engineering	0.21
93	Field Office Engineering	0.41
94	Owners Costs	0.40

(5) $10 x 10^6 was added to the 1975 natural draft dry cooling tower costs of $17.2 x 10^6 (Table 6.4-2, ERDA-109) to obtain agreement with the circulating water system cost of $27.2 x 10^6 from Table 6.7-2 of ERDA-109. This cost was escalated at 8% to obtain the $37.0 x 10^6 cost in 1979 dollars.

(6) The ratio of indirect costs between the HTGR-SC and the HTGR-GT was assumed to be that given in Table 2.3-6 of ERDA-109.

3.11.3. Technique for Estimating HTGR-PH Costs

The 1975 costs for the 3000-MW(t) HTGR-PH systems were estimated on the basis of information contained in ORNL/TM-5409 (ref. 2). Table 22 of ref. 2 gives reactor plant differential costs for the process heat reactor (without an intermediate heat transfer loop) relative to an HTGR-SC in mid-1974 dollars. These differentials were escalated by 5% to obtain January 1975 dollars and then added to the corresponding numbers (including reactor plant BOP costs) for the HTGR-SC which were derived from ERDA-109. The BOP costs (excluding reactor plant) were taken from Table 23 of TM-5409 and escalated by 5% to January 1975 dollars. In several cases, where there was no apparent reason for a difference, the costs were assumed to be the same or nearly the same as similar costs for the steam cycle. In addition the $3 × 10^6 differential cost between the HTGR-SC and the HTGR-GT for the thermal barrier (see Table 3.11.1) was doubled to give a $6 × 10^6 differential cost between the HTGR-SC and HTGR-PH. Also, the $21.3 × 10^6 (reactor plant differential) in 1975 dollars was split as follows: thermal barrier, $6.0 × 10^6; other internals, $1.6 × 10^6; PCRV structure, $6.9 × 10^6; and PCRV liners and penetration, $6.8 × 10^6.

January 1, 1979 dollar estimates for the HTGR-PH system (w/o IHL) were obtained by assuming that the 1975 cost ratio between the HTGR-SC and the HTGR-PH persists to 1979.

Table 24 of TM-5409 gives cost adjustments in July 1974 dollars for the inclusion of an intermediate heat transfer loop. These differential costs were escalated by 5% in order to obtain January 1975 dollars and then escalated by 8% per year for 4 years in order to obtain January 1, 1979 dollars. This January 1, 1979 differential was then added to the capital costs for the HTGR-PH system without an IHL to obtain the costs for the HTGR-PH with an IHL.

The capital investment costs for 1000-MW(t) HTGR-PH were estimated by scaling the cost estimates for the 3000-MW(t) plant to 1000 MW(t) using the scale factors given in Table 3.11.3.

3.11.4. Summary of Costs

The reference capital cost estimates in January 1979 dollars for the 3000-MW(t) HTGR-SC, HTGR-GT and HTGR-PH (with and without an IHL) are given in Tables 3.11.4 and 3.11.5. Table 3.11.4 summarizes the costs to the two-digit level and includes the indirect and contingency costs. Table 3.11.5 breaks the direct cost estimates down to the three-digit level.

Table 3.11.4. Summary of Capital Cost Estimates for 3000-MW(t) HTGRs in January 1979 Dollars

		Estimated Costs ($10^6)			
				Process Heat	
		Steam Cycle	Gas Turbine	W/O IHL	With IHL
20	Land and Land Rights	2	2	2	2
21	Structures and Improvements	135	147	135	144
22	Reactor Plant Equipment	282	335	396	514
23	Turbine Plant Equipment	105	78	0	0
24	Electric Plant Equipment	48	37	42	43
25	Miscellaneous Plant Equipment	12	10	14	14
26	Heat Reject System	56	37	0	0
2	Total Direct Costs	640	646	589*	717*
91	Construction Services	83	69	83	92
92	Home Office Engineering	109	90	109	120
93	Field Office Engineering	34	28	34	37
94	Owners Costs	47	39	47	52
9	Total Indirect Costs	273	226	273	301
Contingency (10%)		91	87	86	102
	Total	1,004	959	948	1,120

*This value is based on no-turbine plant equipment. Since a process heat plant will undoubtedly also generate some electricity, a nominal value of $50 million should be added to this sum to give relative costs.

Table 3.11.5. Reference Cost Estimates for 3000-MW(t) HTGRs in January 1979 Dollars

			Estimated Costs (10^6)			
					Process Heat	
			Steam Cycle	Direct Cycle	W/O IHL	With IHL
20	Land and Land Rights		2	2	2	2
21	Structures and Improvements					
	211	Yardwork	8.4	8.9	8.6	8.7
	212	Containment Building	46.7	66.5	54.9	54.9
	213	Turbine Building	9.2	---	---	---
	214	Security Building	0.4	0.4	0.4	0.4
	215	Reactor Service Building	10.8	10.8	10.8	10.8
	216	Main Circ. Control Building	0.6	0.6	0.6	0.6
	217	Fuel Storage Building	10.8	10.8	10.8	10.8
	218	Other Structures	48.5	48.5	48.5	58.0
		Total	135.	147.	135.	144.
22	Reactor Plant					
	221A	PCRV Structure	37.5	69.3	49.5	47.6
	221B	Liners and Penetrations	28.8	39.6	39.0	39.0
	221C	Reactor Control	10.3	10.3	10.3	10.3
	221D	Reactor Internals	39.4	43.4	52.4	52.4
	222	Main Heat Transfer and Transport	60.8	67.4	128.3	246.6
	223	Core Auxiliary Cooling System	22.5	23.0	24.0	24.0
	224	Rad Waste System	4.1	4.2	4.9	4.9
	225	Fuel Handling System	42.9	42.9	45.2	45.2
	226	Other Reactor Plant Equipment	22.0	24.0	28.6	29.2
	227	Instrumentation and Control	11.4	8.7	11.5	[illegible]
	228	Reactor Plant Miscellaneous Equipment	2.4	2.6	2.2	[illegible]
		Total	282.	335.	396.	514
23	Turbine Plant Equipment					
	231	Turbine Generator	60.2	69.6		
	233	Condensing System	11.7	---		
	234	Feedwater System	14.1	---		
	235	Other Turbine Plant Equipment	15.5	3.5		
	236	Instrumentation and Control	1.9	3.0		
	237	Turbine Plant Miscellaneous Equipment	1.8	1.8		
		Total	105.	78.	0	0
24	Electric Plant Equipment					
	241	Switchgear	7.0	5.6	6.0	6.2
	242	Station Service Equipment	11.0	8.7	11.0	11.0
	243	Switchboards	0.8	0.4	0.3	0.3
	244	Protective Equipment	1.9	1.9	1.9	1.9
	245	Elect. Struct. + Wiring Centers	10.7	6.2	10.7	11.1
	246	Power and Control Wiring	16.6	14.4	11.9	12.0
		Total	48.0	37.0	42.0	43.0
25	Miscellaneous Plant Equipment					
	251	Trans + Lifting Equipment	1.8	4.1	1.7	2.3
	252	Air, Water and Steam Services	6.9	3.3	9.1	9.1
	253	Communications Equipment	1.7	1.7	1.7	1.7
	254	Furnishing and Fixtures	1.3	1.3	1.3	1.3
		Total	12.0	10.0	14.0	14.0
26	Heat Reject System		56	37	0	0
		Total Direct Costs	640.	646.	589. *	717. *

*This value is based on no-turbine plant equipment. Since a process heat plant will undoubtedly also generate some electricity, a nominal value of $50 million should be added to this sum to give relative costs.

The range of uncertainty in the direct investment costs was also estimated for the various concepts. These cost ranges, shown in Table 3.11.6, include a basic uncertainty in equipment and labor costs. Additional uncertainty was added for cost increments from the reference steam cycle.

Table 3.11.6. Capital Cost Range Estimates for 3000-MW(t) HTGRs in January 1979 Dollars

		Estimated Costs ($10^6^)			
				Process Heat	
		Steam Cycle	Direct Cycle	W/O IHL	With IHL
20	Land and Land Rights	2-3	2-3	2-3	2-3
21	Structures and Improvements				
211	Yardwork	6-9	7-10	7-10	7-10
212	Containment Building	32-55	40-70	35-65	35-65
213	Turbine Building	8-12	---	---	---
214	Security Building	0.3-0.5	0.3-0.5	0.3-0.5	0.3-0.5
215	Reactor Service Building	8-12	8-12	8-12	8-12
216	Main Circ. Control Building	0.4-0.7	0.4-0.7	0.4-0.7	0.4-0.7
217	Fuel Storage Building	2-12	2-12	2-12	2-12
218	Other Structures	30-50	30-50	30-50	30-60
	Total	85-150	85-155	80-150	80-160
22	Reactor Plant				
221A	PCRV Structure	30-50	60-80	40-70	40-80
221B	Liners and Penetrations	20-35	30-50	30-55	30-55
221C	Reactor Control	5-20	5-20	5-20	5-20
221D	Reactor Internals	30-55	35-60	40-70	40-70
222	Main Heat Transfer and Transport	50-80	50-90	100-160	200-300
223	Core Auxiliary Cooling System	15-30	15-30	15-30	15-30
224	Rad Waste System	3-8	3-8	3-8	3-8
225	Fuel Handling System	30-50	30-50	30-50	30-50
226	Other Reactor Plant Equipment	15-30	15-30	20-40	20-40
227	Instrumentation and Control	10-20	8-20	10-20	10-20
228	Reactor Plant Miscellaneous Equipment	2-3	2-3	2-3	2-3
	Total	210-380	250-440	295-530	395-680
23	Turbine Plant Equipment				
231	Turbine Generator	50-70	60-80		
233	Condensing System	8-14	---		
234	Feedwater System	12-16	---		
235	Other Turbine Plant Equipment	12-14	2-6		
236	Instrumentation and Control	1-3	2-4		
237	Turbine Plant Miscellaneous Equipment	1-2	1-2		
	Total	85-145	65-90	0	0
24	Electric Plant Equipment	40-55	30-45	35-50	35-50
25	Miscellaneous Plant Equipment	10-15	8-12	10-18	10-18
26	Heat Reject System	50-80	30-50	0	0
	Total Direct Costs	480-825	470-790	420-750*	510-900*

*This value is based on no-turbine plant equipment. Since a process heat plant will undoubtedly also generate some electricity, a nominal value of $50 million should be added to this sum to give relative costs.

The costs for the 1000-MW(t) HTGR-PH (with and without an IHL) are given to the two-digit level in Table 3.11.7. A breakdown of the direct costs to the three-digit level and ranges of uncertainty are given in Table 3.11.8.

It is emphasized t' in many instances the cost estimates are based on extrapolation of costs which themsel were extrapolated. It is our feeling that the cost information is crude, especially for the HTGR-GT and HTGR-PH systems. Also, the PH system costs did not consider that a turbine generator would be associated with the plant, although it is likely that all PH systems will generate electricity in addition to process heat. However, the cost estimates were the best that could be obtained within the time limits and with the limited information available.

Table 3.11.7. Summary of Capital Cost Estimates for 1000-MW(t) HTGR-PH in January 1979 Dollars

		Estimated Costs (10^6)	
		Without IHL	With IHL
20	Land and Land Rights	2	2
21	Structures and Improvements	78	83
22	Reactor Plant Equipment	205	266
23	Turbine Plant Equipment	0	0
24	Electric Plant Equipment	27	27
25	Miscellaneous Plant Equipment	10	10
2	Total Direct Costs	322*	388*
91	Construction Services	52	57
92	Home Office Engineering	87	95
93	Field Office Engineering	22	24
94	Owners Cost	30	34
9	Total Indirect Costs	191	210
Contingency (10%)		51	60
	Total	564	658

*This value is based on no-turbine plant equipment. Since a process heat plant will undoubtedly also generate some electricity, a nominal value of about $32 million should be added to this sum to give relative costs. Indirect costs would have to be adjusted accordingly.

Table 3.11.8. Reference Cost Estimates and Cost Range Estimates for 1000-MW(t) HTGR-PH in January 1979 Dollars

		Estimated Costs ($10^6)			
		Without IHL		With IHL	
		Ref.	Range	Ref.	Range
20	Land and Land Rights	2.	1-3	2.	1-3
21	Structures and Improvements				
211	Yardwork	5.0	4-6	5.0	4-6
212	Containment Building	31.7	20-40	31.7	20-40
213	Turbine Building	---	---	---	---
214	Security Building	0.2	0.2-0.3	0.2	0.2-0.3
215	Reactor Service Building	6.2	5-8	6.2	5-8
216	Main Circ. Control Building	0.3	0.2-0.5	0.3	0.2-0.5
217	Fuel Storage Building	6.2	2-8	6.2	2-8
218	Other Structures	28.0	20-35	33.5	25-40
	Total	78.	50-100	83.	55-105
22	Reactor Plant				
221A	PCRV Structure	25.6	20-35	24.6	20-40
221B	Liners and Penetrations	20.2	15-30	20.2	15-30
221C	Reactor Control	5.3	3-10	5.3	3-10
221D	Reactor Internals	27.1	20-35	27.1	20-35
222	Main Heat Transfer and Transport	66.4	50-80	127.6	100-150
223	Core Auxiliary Cooling System	12.4	10-20	12.4	10-20
224	Rad Waste System	2.5	2-5	2.5	2-5
225	Fuel Handling System	23.4	15-30	23.4	15-30
226	Other Reactor Plant Equipment	14.8	10-20	15.1	10-20
227	Instrumentation and Control	5.9	5-10	6.5	5-10
228	Reactor Plant Miscellaneous Equipment	1.1	1-2	1.1	1-2
	Total	205.	150-250	266.	200-350
24	Electric Plant Equipment				
241	Switchgear	3.9		4.0	
242	Station Service Equipment	7.1		7.1	
243	Switchboards	0.2		0.2	
244	Protective Equipment	1.2		1.2	
245	Elect. Struct. + Wiring Centers	6.9		7.2	
246	Power and Control Wiring	7.7		7.7	
	Total	27.	25-35	27.	25-35
25	Miscellaneous Plant Equipment				
251	Trans + Lifting Equipment	1.2		1.7	
252	Air, Water and Steam Services	6.5		6.5	
253	Communications Equipment	1.2		1.2	
254	Furnishing and Fixtures	0.9		0.9	
	Total	10.	8-14	10.	8-14
	Total Direct Costs	322.*	230-430	388.*	290-510

*This value is based on no-turbine plant equipment. Since a process heat plant will undoubtedly also generate some electricity, a nominal value of about $32 million should be added to this sum to give relative costs. Indirect costs would have to be adjusted accordingly.

Based on the above, on similar information for 1000-MW(t) HTGR-SC and HTGR-GT plants, and on differential costs between HTGR and PBR direct costs, the relative capital costs (total direct costs) for HTGRs and PBRs developed in this study are as given in Table 3.11.9.

Table 3.11.9. Comparison of Direct Construction Costs for Various Applications of HTGRs and PBRs[a] (1979 Dollars)

Outlet Temperature (°C)	Type System	HTGR Cost ($10^6)				PBR Cost ($10^6)			
		3000 MW(t)		1000 MW(t)		3000 MW(t)		1000 MW(t)	
		Ref.	Range[b]	Ref.	Range[b]	Ref.	Range[b]	Ref.	Range[b]
750	SC	640	608-704	346	329-381	714	657-816	374	344-427
850	GT	646	614-711	350	333-385	720	663-823	378	348-431
950	PH	767	729-844	420	400-462	841	778-956	448	415-508

[a]Excludes inflation, scheduling delays, and regulatory impacts.

[b]This range covers -5% to +10% for HTGR; the range for the PBR is based on the HTGR range plus the uncertainty in PBR relative costs.

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3.12. REACTOR RESEARCH AND DEVELOPMENT COSTS

P. R. Kasten

3.12.1. Introduction

Estimates are given here of the research and development (R&D) costs that would be necessary to bring the HTGR and PBR to the stage of commercialization. In the comparative cost evaluation comprising this report, these R&D costs were not used as data input. Rather, it was assumed that the R&D had been completed and, moreover, that the operation of demonstration plants and lead commercial plants had been accomplished. As a practical matter, the R&D costs must, of course, be considered in any selection of a reactor system. Therefore, estimates of these R&D costs are given below.

The R&D expenditures necessary for commercialization of a reactor system fall under two categories: (1) base technology that is largely generic to a number of applications, and (2) equipment technology development related to a specific demonstration plant. Commercialization costs would also include first-of-a-kind costs for construction of the early plants, corresponding to the penalty above costs of commercial plants. The first-of-a-kind costs for lead commercial units are not well known and are not estimated here; however, such costs could be substantial.

For a given reactor, the R&D costs will vary with the reactor's projected application [i.e., electricity production by the steam cycle or the direct cycle (gas turbine), or for high temperature process heat production] and with the outlet temperature of the coolant. The systems covered here are listed in Table 3.12.1, along with their estimated relative introduction schedule in the U.S.

The procedure for arriving at the R&D costs was first to estimate the HTGR R&D costs and then, on the basis of discussions in the preceding sections, to project the additional expenditures that would be required for a PBR. The additional costs for an R&D effort to provide fuel recycle capability were also estimated.

Table 3.12.1. Reactor Systems for Which R&D Costs are Estimated

Reactor	Power [MW(t)]	Application	Outlet Coolant Temperature (°C)	Introduction Date for Lead Plant
HTGR	1000, 3000	Steam cycle	750	1993
		Gas turbine	850	2002
		Process heat	950	2010
PBR	1000, 3000	Steam cycle	750	1997
		Gas turbine	850	2004
		Process heat	950	2010

3.12.2. Expenditures Required to Develop HTGR Technology

The R&D cost estimates for development of HTGRs with outlet coolant temperatures of 750°C (SC), 850°C (GT), and 950°C (PH) are summarized in Table 3.12.2. The base R&D includes work required for development of fuels, structural materials, graphite, and containment vessels, and for development of information about fission-product behavior in reactor systems under various conditions. Reactor equipment R&D includes work on equipment design, development, fabrication and testing and on associated systems. The costs given are those considered to be above those associated with vendor/utility commercial investments. The variation of R&D costs with outlet coolant temperature reflect the equipment and material differences associated with the various applications (i.e., steam cycle at 750°C, gas turbine at 850°C, and high-temperature process heat at 950°C).

Reactor performance is enhanced by reprocessing and refabrication of spent fuel. Thus, Table 3.12.2 also estimates R&D expenditures required to develop HTGR fuel recycle technology to the point that commercialization appears practical.[1] The recycle base R&D includes work on head-end reprocessing (operations involving fuel crushing, burning, dissolution, and fuel recovery) and on fuel refabrication (kernel preparation, coating, rod fabrication and assembly operations). The pilot plant costs include the design, construction and startup of a hot pilot plant to demonstrate fuel recycle equipment and systems. Initial operation will undoubtedly require interactions with the base recycle work, and so 5 to 8 years of pilot plant operating costs/interactions are included in the recycle base R&D costs.

Table 3.12.2. Estimates of HTGR R&D Costs for the Reactor Systems and for Fuel Recycle

Reactor Costs			
Outlet coolant temperature, °C	750	850	950
Reactor base R&D, $\$10^6$	200-250	250-400	400-600
Reactor equipment R&D, $\$10^6$	100-150	200-400	200-400
Total reactor R&D, $\$10^6$	300-400	450-800	600-1000
Fuel Recycle Costs			
Recycle base R&D,* $\$10^6$		500-800	
Recycle pilot plant, $\$10^6$		900-1300	
Total recycle R&D, $\$10^6$		1400-2100	

*Includes 5 to 8 years operating costs/interactions with pilot plant.

3.12.3. Incremental Expenditures to Develop PBR Technology in the U.S.

In estimating R&D costs for PBR development, the HTGR R&D estimates are considered to be base values and estimated incremental costs for developing the PBR in the U.S. are added. (Note: The increments would be different in the FRG where the emphasis has been on PBR development). In doing this, it is assumed that the U.S. effort would focus either on the HTGR or the PBR, but not on both. (There is, of course, much common technology development, but is was not identified here).

Table 3.12.3 lists the incremental PBR R&D costs on the further bases that the technology developed in the FRG is available to the U.S., and that U.S vendors are to furnish the PBRs. Thus, the incremental costs consider the need to develop the pertinent PBR technology in the U.S. The resulting R&D costs of the PBR would be the HTGR costs plus about $145 million (estimated range of incremental costs is $100 million to $200 million). This increased investment would effectively increase the cost of power from PBRs, but the effect would be small over a long time period and with wide application of PBRs.

Table 3.12.2. Estimated Incremental PBR R&D Costs

	$\$10^6$
Fuel process development	6
Fuel development and qualification	15
Graphite development and qualification	7
Fission product behavior	3
Control rod materials development	5
Safety and reliability	2
Control and safety instrumentation systems development and qualification	50
Design methods development (codes)	2
Component design (seismic design studies, fuel handling, graphite handling, reactor internals, core. PCRV)	28
Fuel recycle development	12
THTR surveillance (U.S. participation)	5
Program coordination and management	10
Total incremental R&D	145
Estimated range	100-200

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